WHC-EP-0240-2

NUSAR: N Reactor Updated Safety Analysis Report Amendment 21

Prepared for the U.S. Department of Energy
Assistant Secretary for Defense Programs

Received by OSTI

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Westinghouse Hanford Company

Richland, Washington

Hanford Operations and Engineering Contractor for the
U.S. Department of Energy under Contract DE-AC06-87RL10930

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NUSAR: N Reactor Updated Safety Analysis Report Amendment 21

G. L. Smith

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Prepared for the U.S. Department of Energy
Assistant Secretary for Defense Programs

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WHC-EP-0240-2
NUSAR AMENDMENT 21

G. L. Smith
Cognizant Manager

PREFACE

The enclosed pages are Amendment 21 of the N Reactor Updated Safety Analysis Report (NUSAR). NUSAR, formerly UNI-M-90, was revised by 18 amendments that were issued by UNC Nuclear Industries, the contractor previously responsible for N Reactor operations. As of June 1987, Westinghouse Hanford Company (WHC) acquired the operations and engineering contract for N Reactor and other facilities at Hanford. The document number for NUSAR then became WHC-SP-0297. The first revision was issued by WHC as Amendment 19, prepared originally by UNC. Summaries of each of the amendments are included in NUSAR Section 1.1.

A Manual Revision Instruction (MRI) form for all of WHC-EP-0240-2, Amendment 21, is in front of the cover. Please return the first signed MRI form to Document Control to confirm your receipt of this amendment and subsequent revision of your manual.

Additionally, a detailed MRI form starts the amendment pages for each of the seven volumes. The MRI summarizes the pages of Amendment 21 that are to be replaced or added by the manual holders to revise each NUSAR volume.

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This tabulation provides the complete listing of all effective NUSAR pages, tables and figures, through Amendments 21.

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Date: December 1989

To: Custodian

Document No.: WHC-SP-0297 VOL 2

Title: N REACTOR UPDATED SAFETY ANALYSIS REPORT

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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with NRC General Design Criteria

SUMMARY

Criterion	Title	Degree of Conformance	Page
1	Quality Standards and Records	Meets	3.1-6
2	Design Bases for Protection Against Natural Phenomena	Meets	3.1-11
3	Fire Protection	Meets	3.1-14
4	Environmental and Missile Design Bases	Meets	3.1-16
5	Sharing of Structures, Systems, and Components	Meets	3.1-18
*			
10	Reactor Design	Meets	3.1-18
11	Reactor Inherent Protection	Meets	3.1-19
12	Suppression of Reactor Power Oscillations	Meets	3.1-19
13	Instrumentation and Control	Meets	3.1-20
14	Reactor Coolant Pressure Boundary	Meets Intent	3.1-21
15	Reactor Coolant System Design	Meets	3.1-22
16	Containment Design	Meets Intent	3.1-23
17	Electric Power Systems	Exceeds	3.1-25
18	Inspection and Testing of Electrical Power Systems	Meets	3.1-28
19	Control Room	Meets	3.1-29

*Numbers missing - General Design Criteria not issued by NRC.

Criterion	Title	Degree of Conformance	Page
20	Protection System Functions	Meets	3.1-30
21	Protection System Reliability and Testability	Meets Intent	3.1-31
22	Protection System Independence	Does not Meet	3.1-33
23	Protection System Failure Modes	Does not Meet	3.1-35
24	Separation of Protection and Control Systems	Meets	3.1-37
25	Protection System Requirements for Reactivity Control Malfunctions	Exceeds	3.1-38
26	Reactivity Control System Redundancy and Capability	Exceeds	3.1-39
27	Combined Reactivity Control Systems Capability	Exceeds	3.1-40
28	Reactivity Limits	Exceeds	3.1-41
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31	Fracture Prevention of Reactor Coolant Pressure Boundary	Meets	3.1-45
32	Inspection of Reactor Coolant Pressure Boundary	Meets	3.1-45
33	Reactor Coolant Makeup	Exceeds	3.1-46
34	Residual Heat Removal	Meets	3.1-48
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Criterion	Title	Degree of Conformance	Page
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39	Inspection of Containment (Confinement) Heat Removal Systems	Meets	3.1-54
40	Testing of Containment (Confinement) Heat Removal Systems	Meets	3.1-55
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52	Capability for Containment (Confinement) Leakage Rate Testing	Meets	3.1-66
53	Provisions for Containment (Confinement) Testing and Inspection	Meets Intent	3.1-66
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55	Reactor Coolant Pressure Boundary Penetrating Containment (Confinement)	Does not Meet	3.1-68
56	Primary Containment (Confinement) Isolation	Does not Meet	3.1-69

*Numbers missing - General Design Criteria not issued by NRC.

Criterion	Title	Degree of Conformance	Page
57 *	Closed System Isolation Valves	Does not Meet	3.1-70
60	Control of Releases of Radioactive Materials to the Environment	Meets Intent	3.1-71
61	Fuel Storage and Handling and Radioactivity Control	Meets Intent	3.1-72
62	Prevention of Criticality in Fuel Storage and Handling	Meets	3.1-74
63	Monitoring Fuel and Waste Storage	Meets	3.1-74
64	Monitoring Radioactivity Releases	Meets	3.1-75

3.1.1 Introduction

This section discusses the extent to which the N Reactor Plant, which started operation in 1963, conforms to the present NRC "General Design Criteria for Nuclear Power Plants", specified in Appendix A to 10 CFR 50, and how that conformance is achieved. The evident intent of some of the criteria is addressed rather than the literal wording, since the criteria were written with the assumption that certain major design features would be used, and N Plant differs significantly from those assumed design features. The following are examples of major differences:

- o 1003 pressure tubes instead of a single large reactor vessel;
- o Two reactivity control systems, each individually capable of rapidly shutting the reactor down safely and maintaining it subcritical in the cold shutdown state (instead of one system with that capacity, plus another system capable of controlling reactivity changes resulting from "planned normal power changes");

*Numbers missing - General Design Criteria not issued by NRC.

- o Multiple diesel-driven emergency cooling pumps instead of motor-driven emergency cooling pumps supplied by diesel-driven or gas turbine-driven electrical generators;
- o A confinement system instead of a containment system;
- o Metallic uranium fuel instead of uranium oxide fuel;
- o A moderator cooling system, separate from the reactor coolant system;
- o Significantly shorter operating cycle, lower fuel burn-up, more frequent tests and inspections. 21

Those and other design differences that are described in other sections of this SAR necessitate the evaluation and presentation of conformance to some of the general design criteria on the basis of comparability. The integrated plant is of unique design when compared to typical light water reactors (LWR) and has the unique dual capability of producing special nuclear materials for the Department of Energy (DOE) and, as a by-product, producing steam for generating electricity. The extent to which some of the features exceed criteria and increase reactor safety and the extent to which other features fall short of meeting the criteria are discussed so that a balanced comparison can be made between systems important to reactor safety at this plant and those of the typical LWR on which the criteria were based.

Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Response

Since 1963, numerous design changes have been made under authorization Projects, Equipment Modification Procedures and Design Changes. Records of these changes, their quality requirements and acceptance test results have been retained. The quality assurance program for operation, maintenance, and modifications of the reactor is under the direction of the Quality Assurance organization and is described in Chapter 17.

The design, construction, inspection, and testing of N Reactor were performed in accordance with applicable, nationally recognized codes and standards. These were supplemented in a number of important areas by standards developed for N Reactor to improve the quality, commensurate with the importance of the safety functions. These additional standards and the quality assurance methods that were used were in advance of the state-of-the-art for such components as the Zircaloy pressure tubes and the carbon steel piping for nuclear service. These codes, standards, and special quality control requirements were implemented as follows:

1. Technical Criteria

   A series of 37 technical criteria were developed and used (Ref. 1), that defined nuclear safety, reactor life, and operability requirements. These criteria identified systems essential to the prevention or mitigation of accidents and specified applicable seismic design requirements, safety systems design requirements, safety system performance requirements, coolant system independence requirements, and many other important features of the plant design. References to applicable codes are given in several criteria, and requirements were provided for special equipment not covered by the codes, such as the Zircaloy-2 pressure tubes and the Grayloc pipe fittings.

2. Design Bases

   A series of ten design bases was developed and used which elaborated on the functional requirements for instrumentation and controls, reactor coolant system (RCS) pressure relief, confinement system and reactivity controls (Ref. 2).

3. Design Criteria

A series of design criteria documents (Ref. 2) was developed and applied, based on the technical criteria and design bases. These provided the detailed design criteria governing preparation of engineering specifications. They provided descriptions of the systems and their functional requirements. Suggested design was shown by schematic diagrams, but no attempt was made to establish the detailed design in these criteria. In certain areas, the applicable codes to be followed were given.

4. Hanford Works Specifications (HWS Documents)

Based on the guidance from the design criteria documents, detailed specifications were prepared to define the detailed design requirements. These documents covered the material grades to be used, fabrication techniques, testing and acceptance requirements and all applicable code requirements, referring to codes by paragraph numbers wherever necessary.

5. Codes and Standards

The main national codes used were:

- o ASME Boiler and Pressure Vessel Code, Section I - Power Boilers 1956 and 1959.
- o ASME Section VIII - Unfired Pressure Vessels 1956 and 1959.
- o ASA B.31.1.0, 1955 - American Standards Association Power Piping Code.

Other codes and standards specified in design requirements were national codes such as ASME, ASA, ASTM, AWWA, Military Standards, and Federal Specifications. Local standards, such as the Hanford Plant Standards and special N Reactor Standards (N-Standards), were specified. The Hanford Plant Standards and special N-Standards generally started with the applicable national codes as a minimum and imposed additional requirements. The 21 N-Standards were developed by Burns and Roe and reviewed and approved by General Electric Co. They covered such details

as RCS pipe welding (N-7) and fabrication of steel forgings (N-21). These N-Standards exceeded national codes in many aspects, such as follows:

- o The N-7 welding standard required inert gas purging during welding and stress relieving of RCS piping, while the ASAB.31.1 pressure piping code recommended it, but did not require it.
- o The N-7 standard required 100% stress relieving of all RCS pipe welds, whereas, ASAB.31.1 (now ANSI B.31.1) only required stress relief of welds in pipe 3/4-in. or greater in thickness.
- o ASA B.31.1 also was exceeded in the areas of welder qualification and records requirements.

In designing the plant it was found that a few exceptions to the applicable codes were needed. These exceptions were carefully considered, found acceptable, and documented. A complete list of the code deviations is given in Section 5.2. Specific code variations include:

- o Zirconium alloy pressure tubes
  The use of zirconium alloy for pressure tubing was not covered by the ASME B&PV codes or the ASAB.31.1 piping code. The tubes were designed on a very conservative basis.

- o Reactor coolant system pressure relief
  The ASME B&PV Code required that pressure relief be capable of accommodating the maximum possible thermal energy generating rate. This rate cannot be defined for a nuclear reactor. The reactor coolant system pressure relief devices were sized on the basis that any significant power excursion would be terminated by an automatic reactor scram. On this basis, the installed pressure relief capacity is adequate. (The Code later gave recognition to the dependence on reactor protective systems to limit the energy output.)

- o Instrument take-off connections
  Take-off connections are smaller in some instances than was permitted by the B&PV Code. An example is the sensing lines for the pressure tube flow monitoring system which are constructed of small diameter tubing of stainless steel or Inconel. Use of the larger lines required by the Code for the 1003 sensing lines would have been impractical on the basis of space considerations. In addition, the failure of a small diameter sensing line, while it will cause a spurious scram, will not be a serious reactor safety event.

- o Use of flexible cooling line connectors for horizontal control rods, shield and graphite cooling tubes
  The use of thick wall tubes in this service would have been impractical in that it would not provide for necessary movement. In addition, the design of these systems is such that failure of a flexible connector would not be a nuclear safety hazard.

The current general program for use of codes is discussed in Section 5.2.1. Those codes applying to the reactor coolant system, and their application, are discussed in Section 5.2.

6. Research and Development

Comprehensive R&D programs were performed to support the design and construction of certain critical components of the N Reactor. In the years 1958-1964, about 25 million dollars (Ref. 3) were spent in R&D work specifically in support of N Reactor design. The work included: (1) lattice physics tests in two test reactors, (2) analog and digital simulation of plant control, (3) thermal hydraulic tests in the laboratory and in-reactor test facilities, (4) Zircaloy-2 tube irradiations, (5) carbon steel pipe tests, (6) graphite sample irradiation, (7) Zircaloy-2 pressure tube break testing, (8) reactor coolant system component thermal cycling testing, (9) pressure tube to nozzle joint tests, (10) ball safety hopper tests, (11) horizontal control rod tests, and many more. Descriptions of these programs are

given in Reference 3. These provided valuable inputs to the design of the plant and tested the final design of some of the equipment.

7. Equipment Development and Testing

Approximately 200 design tests were performed on various components and equipment to prove their adequacy of design or to determine areas of improvement needed before actual application. This testing is described in more detail in Reference 3.

8. Consultation and Design Reviews

Extensive use of experts and special service consultants was made during the course of the project to ensure that all key design decisions were well-founded. This involved consulting contracts with national experts as well as consultants within the parent General Electric Co. A list of 16 key consultants is given in Section 1.4. In addition, laboratory and industrial services were obtained from 21 organizations which included 3 in the parent company.

Numerous technical audits were made of the design, construction, and preparation for operation of the plant. The main nuclear safety reviews were performed by AEC (Advisory Committee for Reactor Safeguards, Division of Reactor Licensing and Reactor Operations Office) and the General Electric Technological Hazards Council (GETHC). The GETHC consisted of senior technical managers from the local and parent General Electric Co. organization. The Council provided multi-disciplined reviews of safety-related design features through all stages of design, construction, testing, and operation of the reactor until 1967 when the AEC contractor responsibility for N Reactor was transferred from General Electric Co. to Douglas United Nuclear Inc., which later became United Nuclear Industries Inc. The technological reviews were then performed by the UNI Company Nuclear Review Board which consisted of the Level 2 Department Directors. Currently technological reviews are performed within Westinghouse Hanford Company by the Reactor Subcouncil of the Safety and Environmental Advisory Council (SEAC).

9. Quality Assurance Program

The quality assurance (QA) program for design and construction of the plant consisted of using applicable codes, supplementing them with design specifications, performing inspections, and extensively testing the installed systems. This was part of project management rather than a defined separate QA program. Records of these specifications, inspection results, and test results are retained in files. Examples of records include test results on materials such as pipe sample destructive tests, radiographs, other nondestructive test results, and operational performance tests. The Quality Group classification of components is discussed in Section 3.2.2. | 21 | 21

The quality assurance program for operation, maintenance, and modifications of the reactor including In-service Inspection and Testing is now under the direction of the Quality Assurance organization and is described in Chapter 17. | 21

The plant design conforms to Criterion No. 1.

Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Response

Chapter 2 discusses natural phenomena for this site. These were considered in the design of structures, systems and components important to safety.

Winds, floods, and earthquakes are the primary phenomena potentially affecting the plant. Precipitation in the form of rain, hail, sleet, or snow required no special design since this is a semi-arid site with a moderate climate. Tsunami and seiches do not apply to this inland site, with no adjacent large body of water.

The plant is designed to accommodate, without loss of important functions, the maximum flood for which there is historical evidence. The flood peaked at approximately 800,000 ft^3/s on June 7, 1894. The highest flood in the history of the Hanford Project, in 1948, peaked at 690,000 ft^3/s. The 800,000-ft^3/s flood has been calculated to have produced a water level at N Reactor of 415 ft mean sea level (MSL). The 181-N river pump house floor is located at 421 ft MSL. Diesel-driven pumps at 181-N, which are part of the ECCS system, are pedestal-mounted and have an operating capability up to 426 ft MSL water level. This also provides sufficient freeboard to ensure their operation in event of a Probable Maximum Flood (PMF) which represents flood discharge that may be expected from the most severe combination of meteorological and hydrological conditions that are reasonably possible in the region. The PMF, with some regulation by upstream dams, is calculated to produce a peak flow of 1,440,000 ft^3/s and a 423 ft MSL water level at 181-N. All other important plant buildings are located on higher ground at 454.5 ft MSL. See Section 3.4.

The plant was not built to present day seismic standards. However, because of the degree of conservatism used in the design of safety-related structures and systems, it has been possible over the years with new analyses, modifications, and upgrades to bring the plant into compliance with current seismic Category I requirements. The Category I structures and systems are identified in Section 3.2.2. Discussion of current seismic design criteria, methods of analyses, and modifications made to assure compliance of all identified seismic Category I structures, systems, and components to the SSE requirments for the Hanford Site is presented in Sections 3.7 and 3.8.

As an additional precautionary feature related to natural phenomena, the N Reactor design includes three seismoscopes that are set (on a 2/3 logic) to

scram the reactor and actuate the ball safety system on occurrence of seismic intensity IV on the Modified Mercalli (MM) scale or greater.

Buildings were designed in accordance with the Uniform Building Code (1961) specifications for wind loads, as follows:

Under 20 ft	15 lbs/ft^2
20 to 50 ft	20 lbs/ft^2
50 to 100 ft	25 lbs/ft^2
Over 100 ft	30 lbs/ft^2

These values of pressure correspond to a maximum wind velocity of about 70 mi/h at 30 ft above the ground.

The ventilation stack was designed for the following wind loads which were obtained from "ACI Standard 505, Specification for the Design and Construction of Reinforced Concrete Chimneys:"

0 to 50 ft	20 lbs/ft^2
50 to 100 ft	24 lbs/ft^2
100 to 200 ft	28 lbs/ft^2

These values correspond to a maximum wind velocity of about 100 mi/h at 30 ft above the ground.

No special design was provided for hurricanes, which historically do not occur in the Pacific Northwest, nor for tornadoes which (at that time) were considered to be of such low probability of occurrence that additional expenditures to protect the plant against a tornado impacting it were not considered warranted. However, Hanford Plant Standards subsequently were revised to include a criterion (Ref. 5) for design to withstand a design bases tornado (DBT) windspeed of 165 mi/h at the Hanford site. See Section 3.3. Although the plant was not designed to withstand a wind of that velocity, it has been evaluated, and required modifications have been made, or will be completed either prior to restart or installed within 24 months

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following restart as part of the Long Term Service (LTS) program, such that all systems required for safe shutdown on the emergency core cooling system (ECCS) would survive a tornado with windspeeds up to 165 mi/h. See Section 3.3.

The plant design meets Criterion No. 2.

Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Response

The plant is designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used, to the extent practical, in the confinement, in control rooms, in components of systems important to safety and throughout the plant.

Fire detection, alarm, and extinguishment equipment are provided, along with appropriate use of fire stops and isolation, to protect the plant equipment and personnel. The fire protection systems include wet pipe sprinklers, dry pipe deluge systems, foam extinguishing systems, a carbon-dioxide system, hose reels, portable extinguishers, and outside fire hydrants as appropriate for the types of extinguishment needed in particular locations. The systems are installed in accordance with National Fire Protection Association (NFPA) codes and are tested and inspected in accordance with those codes and DOE requirements. See Section 9.5.1.

Fire protection systems are designed to assure that their operation (including inadvertent operation) will not significantly impair systems important to safety. In each location where sprinklers or deluge sprays are used, the floor drains have adequate capacity to accept the maximum flows that can occur by actuation of those systems.

Improvements have been made in the fire protection systems and methods almost continuously since the plant started up in 1963, and more improvements are scheduled for installation in the near-future. See Section 9.5.1. One of the most significant previous improvements in fire protection was the application of a Flamemastic coating to power and instrument cables. A description and criteria for the scheduled improvements are given in Reference 6. The proposed improvements are the result of continual evaluations of N Reactor fire protection, by Westinghouse Hanford and DOE-RL personnel and by Factory Mutual Engineering Association personnel, acting as independent inspectors and consultants to DOE-RL. 21

The plant design meets Criterion No. 3. 21

Criterion 4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Response

Systems important to safety were designed to accommodate the effects of environmental conditions resulting from normal operation and postulated accidents. Effects that were considered in the design included:

- o Pipe whip on an assumed complete break of a reactor supply or return main coolant line;
- o Pipe whip on an assumed complete break of an individual pressure tube connector at a location between a supply riser and a pressure tube;
- o Stack lift and forces exerted on adjacent pressure tubes in event of a complete break of a pressure tube in the moderator of the reactor; See Section 4.5.3. (These experimental evaluations also considered the extent of damage that would occur in the graphite moderator, the amount of steam venting area needed in the reactor core design to prevent over-pressurization of shields, and the force exerted on horizontal control rods.)
- o The steam, temperature and pressure conditions that would occur in the confinement structure upon a pipe break.

The extent to which the plant is resistant to missiles from failure of major rotating machinery is discussed in Section 3.5 and is as follows:

1. The turbine generators for commercial power production are housed in a separate building (185-N), located approximately 650 ft from the 105/109-N building, and are oriented such that the turbine blades rotate in a plane parallel to the 105/109-N building rather than in a plane intersecting the building.

2. The onsite Turbine Generator (TG) set (for onsite power) is housed in 184-N and is oriented such that missiles from its turbine blades cannot impact the 105/109-N building and missiles from turbines in 109-N cannot impact the TG set.

3. The confinement system structure consists mainly of massive reinforced concrete walls, floors, and roofs and is not vulnerable to damage by missiles of the type that could be generated by failure of pressurized or rotating equipment.

4. The RCS pumps are located in individual cells with reinforced concrete walls between cells. The pumps are driven by through-wall shafts from turbine and pony motor sets located in the 109-N turbine bay outside of the confiner.

5. The RCS pumps drive turbines and attached pony motors are mounted in a single row in the turbine bay with no missile barriers between them. Failure of a drive turbine could potentially cause failure of one or more other drive turbine-pony motor sets. If more than two drive turbines failed, the reactor would automatically be scrammed and placed on once-through emergency cooling. The RCS drive turbines, pony motors and pumps are not used for the emergency cooling function. Failure of any one or two of the five normally operating drive turbines would cause a reactor scram but would not cause a need for emergency cooling. The drive turbine blades rotate in a plane parallel to the confinement structure rather than intersecting it. 21

   The rod safety system, ball safety system and ECCS system are located such that they are not potentially exposed to missiles from failure of high-pressure piping nor failure of rotating machinery. The RCS and ECCS share a common location at their junction in the inlet pipe space, but analyses in Sections 3.6 and 3.5 indicate that redundancy in the systems provides adequate protection against the effects of missiles or pipe whip from RCS breaks.

The plant design meets Criterion No. 4.

Criterion 5 - Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not

significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Response

N Reactor does not share structures, systems, or components important to safety with any other nuclear power unit.

The plant design meets Criterion No. 5.

Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Response

The reactor core, coolant, controls, and protective systems are designed with appropriate margins to assure that fuel cladding failure will not be caused by excessive power, inadequate heat removal, or abnormal corrosion of cladding during any condition of normal operation including the effects of anticipated operational occurrences. The core thermal hydraulics design is based on preventing departure from nucleate boiling during normal operation and design operational transients. The fuel element design is such that, under design power conditions or anticipated operational occurrences, fuel dimensional stability is maintained. Trip settings for protective systems are established on the basis of thermal hydraulic experimental data and analyses and are set to trip the reactor prior to departure from nucleate boiling. In more than 22 yr of operation, these protective actions have proven to be sufficient to prevent fuel damage during the events that have occurred. These have involved RCS coolant flow reductions and pressure reductions. See Chapter 15 for evaluation of power increases, power maldistribution, etc. which show that the protective instrumentation and reactivity control systems are capable of preventing fuel failure from power-related transients.

The plant design meets Criterion No. 10.

Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Response

The N Reactor core has a large negative feedback characteristic to compensate for a rapid increase in reactivity. The Doppler coefficient is approximately -0.004 mk per degree F increase in fuel temperature. The coolant temperature increase (over the inlet temperature) that would result from a power increase also has a negative feedback coefficient, approximately -0.0014 mk per MWt, and, in addition, further power increase (or system depressurization) resulting in increased void fraction is offset by a negative void coefficient (see 4.3.3.2). The net effect of all changes resulting from a rapid reactivity increase is a large negative feedback that rapidly suppresses any power increase. See Chapter 4. 21

The plant design meets Criterion No. 11.

Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Response

N Reactor is designed to be inherently very stable with respect to power oscillations. There have been no significant power oscillations of either the short-term feedback-induced type or the longer-term xenon-induced type of oscillation in more than 22 yr of plant operation. The large number (84) of horizontal control rods entering from both sides of the reactor provide more than adequate flexibility in reactivity control. Protection 21

systems are provided to assure that fuel design limits are not exceeded. See Chapter 4.

The plant design meets Criterion No. 12.

Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Response

Instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the confinement system and its associated systems. Appropriate controls are provided to maintain these variables within prescribed operating ranges. See Chapter 7 for description and discussion of instrument and control systems. This instrumentation includes a 1003-channel pressure tube flow monitoring system, a 109-channel pressure tube outlet coolant zone temperature monitoring (ZTM) system and a 1003-channel all tube temperature monitoring system (additional features, when compared to the typical LWR plant) which are used to ensure adequate protection to each individual fuel charge in the reactor.

With the recognition that a confinement system rather than a containment system is monitored, the plant meets Criterion No. 13.

Criterion 14 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Response

The reactor coolant pressure boundary is designed, fabricated, erected and tested such that it has an extremely low probability of rapidly propagating failure and of gross rupture. Furthermore, the use of high-quality carbon steel piping and Zircaloy-2 pressure tubes provides superior qualities of resistance to thermal shock and to stress corrosion cracking, compared to the use of stainless steel.

The N Reactor mission of nuclear material production, using metallic fuel, necessitated a design that provided for frequent and economical replacement of fuel columns. This resulted in using a large number of pressure tube caps and system valves, including many valves that connect to drain systems. The system design involves thousands of locations where leaks can develop at packing glands, past valve seats, and through flexible gaskets. Consequently an accumulation of many small leaks is "normal" even though a vigorous leak repair program is pursued. Small leaks at these locations do not present a potential for rapidly propagating failure of the reactor coolant pressure boundary.

The reactor design includes bleed streams from the reactor coolant system to provide control of coolant quality and inventory, and for water samples for quality analyses and fuel cladding failure detection. These routine bleed streams from the reactor coolant system total about 300 gal/min. 21

Although the total bleed-off and leakage from the reactor coolant system is high compared to a typical LWR, the makeup capacity from the high-pressure injection system at N Reactor is correspondingly high (about 6000 gal/min at full pressure), so that coolant system inventory losses can be replaced at rates well beyond those normally experienced. Any abnormal leak that would indicate a pressure boundary failure, such as a steam generator tube break,

would be promptly identified by an increase in the makeup rate to the reactor coolant system.

Several means, discussed and evaluated in Section 5.2.5, are used to ensure that a progressive crack in the reactor coolant system piping would be detected at an early stage, thus ensuring an extremely low probability of a rapidly propagating rupture and of gross rupture of the reactor coolant pressure boundary. If leaks are detected by these procedures, actions are taken to either prevent the reactor from being started up (in the case of certain kinds of leaks being found during an inspection) or to shut the reactor down.

The plant design meets the intent of Criterion No. 14. The plant design directly conforms to the part of the criterion that specifies a design for low probability of rapidly propagating failure or gross rupture.

The intent of the criterion is met with respect to the reactor coolant pressure boundary having an extremely low probability of abnormal leakage, in that the many small leaks that do occur are of the type that do not present a risk of rapid or gross failure of the pressure boundary.

Criterion 15 - Reactor Coolant System Design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Response

The reactor coolant system and associated auxiliary, control, and protective systems are designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. See Section 5.2 for design details. Since initial operation of N Reactor there have been no safety problems related to design of the reactor coolant system or the associated equipment and controls that would prevent maintaining the system within design conditions.

A program of reactor coolant system inservice inspection is in effect. See Section 5.2.4. In summary, the results from the inservice inspection program show that no condition of inadequate strength will be reached within the foreseeable operating life of the reactor. Gradual deterioration from such mechanisms as corrosion, hydriding, and irradiation is occurring at a much slower rate than was allowed for in the design. 21

The burst-strength of the pressure tubes is at least a factor of five greater than the normal operating stress, as shown by actual tests of pressure tubes removed from the reactor coolant system.

The plant design meets Criterion No. 15.

Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Response

Because of the unique design features and the large (5 1/2-mi) exclusion zone boundary distance, N Reactor is able to conform to 10 CFR 100 guidelines using a confinement system rather than a containment system. The confinement concept for the hypothetical large coolant pipe break is to release the initial burst of steam to the atmosphere, and then when the confiner pressure subsides, to close the steam vents and reopen ventilation exhaust valves to the filters to establish a filtered release of the confinement atmosphere. Fuel failure is not expected in this case. For postulated pressure tube breaks with fuel failure, fission products will be released to the confiner concurrently with the steam from the tube break. The confiner pressure will increase, actuating the confinement sequence, but not enough to open the steam vent caps, and the coolant release will be contained within the confiner, except for confiner leaks. The unfiltered leak rate from the confiner will increase in proportion to the pressure increase. Accident

analysis for these cases show that the dose rates outside the exclusion zone boundaries are well within the 10 CFR 100 limits. See Section 6.2 for further details on the confinement system design and Chapter 15 for accident analyses. The confinement concepts depend on the following features, which differ from the typical PWR, for their adequacy to control releases within guidelines given in 10 CFR 100:

- Approximately 366 metric tons of fuel used to generate the rated 4000 MWt of power, which gives specific power approximately the same as commercial light water reactors, a comparatively long time (420 s) to reach fuel cladding failure temperature in the unlikely event of the loss of all cooling;
- Maintenance of low activity in the reactor coolant by promptly shutting the reactor down and discharging any fuel element that experiences fuel clad failure;
- A separate moderator cooling system that limits the potential for fuel melting even if all pressure tube direct cooling water supplies are lost;
- An exclusion zone boundary distance of 5 1/2 mi.

Leak tightness, which is emphasized in the criterion, is not nearly as significant to this confinement system as it would be for a containment system. The confinement system pressure is very low, 0.1-in. w.g., during the filtered release mode. At this pressure a 200 ft^3/min unfiltered leakage rate is assumed in calculating doses outside of the exclusion zone. The system is routinely tested for total leakage to assure conformance to the assumed maximum leakage rate of 200 ft^3/min at 0.1-in. w.g. See Section 6.2.6. At higher confiner pressures, correspondingly higher leak rates are used to calculate the doses outside of the exclusion zone. In all cases, these doses are well within the 10 CFR 100 guidelines.

Based on conformance to guidelines in 10 CFR 100, the reactor, on this site, meets the intent of Criterion No. 16 by not releasing unacceptable radioactivity products beyond the site boundary. It does not fully conform in that the confiner is not designed to be "essentially leaktight."

Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Response

N Reactor has an offsite ac power system and an onsite ac power system that are electrically independent of each other. The onsite power system is supplied by the 184-N turbine-generator (TG) set which supplies power to approximately half of the electrically driven equipment as an inservice supply during reactor operation. The steam supply for the TG set and the primary pump drive turbines is provided by the reactor or by standby oil-fired boilers. The offsite system is supplied by the BPA power grid.

Either the offsite power system or the onsite power system alone will provide for the orderly safe shutdown of the reactor and will maintain the reactor and engineered safety features in a safe shutdown mode. In the event of postulated accidents there is sufficient capacity in either system to provide for the reactor scram transient and for the shutdown condition without exceeding fuel design limits or reactor coolant system pressure boundary design limits, and without loss of recirculation core cooling, confinement operability or other vital functions. Neither individual system provides for continuing reactor operation on failure of the other system. A reactor scram is initiated by undervoltage relays associated with each power source.

In addition to the ac power sources, N Reactor has dual 125 Vdc systems. These dc systems are designed to function in the event of an SSE or in the event of single component failure and will furnish power to actuate certain non-failsafe components in the emergency core cooling system, the confinement system, the graphite and shield cooling system, and the fog spray system.

The 125 Vdc systems are continuously tested for grounds by sensitive ground detectors. These dc electrical systems are continuously in service during the power mode of operation.

Offsite electrical power is supplied to the N Reactor onsite electrical distribution system by two physically and electrically independent taps (N-Tap and KW-N Tie Line) from the Hanford 230 kV Loop. The Hanford 230 kV Loop is normally powered from the BPA transmission network through two independent lines from the Midway Substation. The Ashe 230 kV line is

normally powered from the Hanford 230 kV Loop, but in the event that both Midway lines to the Loop are interrupted, the Ashe bus will supply sufficient power to the Hanford Loop (from the White Bluffs substation) to maintain power or shutdown operation of N Reactor. See Figure 8.2-1 in Chapter 8. The N-Tap and the KW-N Tie Line are each powered from electrically isolated sections of the Hanford 230 kV Loop. Electrical isolation is provided by redundant zones of relaying and 230 kV breakers such that one fault will not cause loss of both circuits to N Reactor. The N-Tap is the primary supply and meets the criteria for an offsite power system. The KW-N Tie Line provides an additional source of electrical power and can be used to (1) supply the onsite power system in parallel with the 184-N TG set or to (2) supply either the offsite or onsite power system on failure of the normal primary supply. The latter capability requires manual switching to supply offsite power. As noted above, neither offsite nor onsite electrical power is needed for a safe shutdown and emergency cooling of the reactor.

Provision was made to minimize the probability of losing any of the remaining supplies of electric power as a result of, or coincident with the loss of power generated by the 185-N Hanford Generating Project, the loss of power from the offsite electrical supply, or the loss of power from the onsite electric power supply. Power generated by the Hanford Generating Project is fed into the BPA system at 500 kV at a distance remote from the location where the offsite supply from the BPA power is supplied to N Reactor at 230 kV. Experience with interruptions of power generated by the Hanford Generating Project have demonstrated that no significant feedback effect occurs on the BPA system to the location where N Reactor gets its offsite supply.

See Chapter 8 for further description and evaluation of electrical power supplies.

The plant design exceeds Criterion No. 17 in that: (1) the onsite electrical generating equipment (the TG set and steam supplies) are operating during the power mode of operation, (2) the electrical supplies for systems that provide for safe shutdown and emergency cooling are in service and backed by

battery supplies, and (3) no sequencing or transfer of electrical supplies is required during emergency conditions.

The plant design exceeds Criterion No. 17.

Criterion 18 - Inspection and Testing of Electrical Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Response

The electrical systems important to safety are designed such that they can be periodically inspected. This inspection and testing is done in accordance with Equipment Maintenance Standards (Ref. 7). Relays, breakers, buses, cables, batteries, inverters, etc. are checked during scheduled outages. Functional testing of an operational sequence that brings an emergency power supply system into service, as required by the criterion, is not applicable to N Reactor. The emergency electrical supplies are continuously in operation during reactor operation. The ECCS functions are provided direct-drive diesel-powered pumps, which are actuated and monitored by battery-powered dc supplies. The diesel-driven pumps in the emergency core cooling system are automatically started each time the reactor scrams. Those and the diesel-driven pumps in the RWS-2 fog spray supply system are functionally tested just prior to each reactor startup and are given additional tests and inspections as required by the Equipment Maintenance Standards. There are

redundant diesel pumping units for each emergency function, and they have demonstrated a continuing high degree of reliability. See Section 9.5.6.

The plant design meets Criterion No. 18.

Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Response

N Reactor has one main control room (105-N control room), four auxiliary control rooms (at 109-N, 182-N, 163-183-N and 184-N) and the Emergency Shutdown Panel (109-N). Activities in the auxiliary control rooms support, and are under instruction from the 105-N control room. The main parameters of systems important to reactor safety are controlled and monitored from the 105-N control room. The Emergency Shutdown Panel allows trained operating personnel to effect rod and ball safety system actuation and where emergency cooling could be actuated if access to the 105-N control room were made impossible. In addition there are locations outside of the 105-N main control room from which, through procedural actions, the reactor can be shut down (scrammed) and supplied with emergency cooling (see Section 7.4). 19 19 21

For postulated loss of coolant accidents involving release of fission products, as analyzed in Section 15.6.4, the 105-N control room is habitable for the duration of the accident, with a whole body radiation exposure of less than 5 rem or its equivalent to any part of the body. Replacement of operating personnel would assure a relatively low dose to any individual. See Section 6.4 for further discussion on control room habitability.

The plant design meets Criterion No. 19.

Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Response

The protection system for N Reactor consists of:

- Rod safety circuits and instrumentation (see Section 5.6)
- Ball safety circuits and instrumentation (see Section 6.7)
- Emergency core cooling system (ECCS) actuation circuits and instrumentation (see Section 6.3)
- Confinement system actuation circuits and instrumentation (see Section 6.2.1)
- High Pressure Injection System (HPI) actuation circuits and instrumentation (see Sections 9.3.4 and 15.6.6)
- Graphite and Shield Coolant System once-through cooling actuation circuits and instrumentation (see Sections 5.5, 6.3, and 15.6.4).

The protection system design provides for automatic operation of the appropriate systems, including reactivity control systems, to assure that protective action will be initiated as necessary to prevent fuel cladding failure from occurring as a result of anticipated operational occurrences. It also senses accident conditions and initiates the operation of systems, and components important to safety that prevent or minimize damage to reactor fuel and vital plant equipment.

Refer to Chapters 7.2 and 7.3 for detailed descriptions and evaluations of the circuits and instrumentation listed above. Also refer to accident analyses in Chapter 15, Sections 15.1 through 15.6 and 15.8. In summary, they show that the combined protection system, reactivity control systems and emergency core cooling system are adequately designed and set to provide protection against fuel failure occurring as a result of any one of a wide spectrum of postulated transients and accidents. | 2

The plant design meets Criterion No. 20.

Criterion 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Response

The protection system for N Reactor, as defined in Response to Criterion 20, was designed and placed into operation prior to the existence of the General Design Criteria (GDC), IEEE Standards, and other safety criteria in use today. | 2

The system was designed for high reliability and testability based on the design criteria and bases developed from actual design and operating experience from eight Hanford special nuclear materials (SNM) production reactors and from design considerations current in the nuclear industry at the time of design. See Section 7.2.2.

Modifications to the system have occurred from time to time. These modifications met the intent of the applicable codes and standards in effect at the time of the change. See Sections 7.2.2, 7.3, and 7.4 for discussion of modifications and upgrades performed that have enhanced the protection system reliability.

The protection system provides for automatic operation of the appropriate systems, including reactivity control systems, to assure that protective actions will be initiated as necessary to prevent fuel cladding failure from occurring as a result of anticipated operational occurrences. It also senses accident conditions and initiates operation of systems and components that prevent or minimize damage to reactor fuel and vital plant equipment and mitigate the impact of accidents on the environment or surrounding population.

The first four subsystems of the protection system, i.e., the actuation circuits and instrumentation for the rod system, ball safety system, ECCS and the confinement system, correspond to the reactor trip system (RTS) and the engineered safety feature actuation system (ESFAS) as discussed in Sections 7.1, 7.2, and 7.3. The analysis of the RTS (Section 7.2.2) shows that these systems generally meet the requirements of IEEE-279-1971. Where specific components may not meet the IEEE-279 standards, there is sufficient redundancy in the overall protection system to provide reliability and redundancy equivalent to the level specified in IEEE-279. The rod safety system and the ball safety system, for example, are each separately capable of total reactor shutdown and, in combination, provide a functional reliability and redundancy equivalent to or exceeding Criterion 21. These two systems are fail-safe on loss of power to the system circuits.

The emergency core coolant system (ECCS) instrumentation is fail-safe except for the reactor coolant bulk outlet temperature monitors which use power from the 125 Vdc/120 Vac inverter power supplies. The ECCS actuation circuits were deliberately designed to fail negative on loss of power to avoid the potential for spuriously dumping the reactor coolant system on power loss to the circuits. In order to compensate for the attendant reduction in design reliability, two separate control circuits were provided, each supplied by a separate and independent 125 Vdc power source. Operation of either circuit will actuate the system. Frequent tests of the individual control circuits guard against any undetected failure. These tests are performed during reactor shutdown and include cycling each V-3 and V-4 valve. Tests also are performed to check all combinations of inputs to each master circuit. 21

Mechanical actuation of the ECCS valves depends on compressed air, which is supplied to each valve by two independent sources. One of these is a local accumulator for each individual valve that is capable of operating the valve several times without recharging.

The circuits are powered from the highly reliable 125 Vdc power source.

The confinement actuation system utilized multiple trip inputs from diverse systems to ensure that no single failure could prevent the actuation system from performing its protective function. The redundancy of all other portions of the actuation system ensures that no single failure will affect more than one channel in the confinement actuation circuits. 21

The graphite and shield cooling system (GSCS) provides a supplementary core cooling feature to supplement the ECCS in the long term. The actuation circuits and instrumentation for the GSCS are fail-safe, putting the GSCS on once-through cooling on loss of power.

The analysis in Section 7.3.2 shows that no single failure will cause the loss of these protection functions. Removal from service of any single channel does not result in loss of the required protection, since the

remaining channels will reliably complete the protection function. Further details on the protection system are given in Section 7.2.

The N Reactor protection system is not designed to permit functional testing of the channels during reactor operation since there are no bypasses for the scram breakers. System reliability is therefore assured by routine functional tests (EMSs) of the protection systems during reactor shutdowns. Individual inputs to the trip channels can be tested during reactor operation if bypassed. Bypassing of the input circuits is under strict administrative control. See Sections 7.2.2 and 7.2.2.5.

The plant design meets the intent of Criterion 21 by providing an equivalent degree of protection system reliability and testability.

Criterion 22 - Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Response

The N Reactor protection system can be subjected to extremes of heat or cold, steam, water, pressure, and radiation from the environment in which the components are located, and though highly unlikely, to accidental fires. These environmental conditions were considered in the design of the system components and were factored into the material specifications for the components. Redundance of instrument channels protects against spurious actions by failure of individual components. If a protection system circuit is adversely affected by environmental conditions, an electrical ground or short usually results and is registered and annunciated on the 125 Vdc ground detector. Similarly, electrical faults from operational activities, maintenance, testing, or accident conditions can be detected and corrective

action taken to assure that the protection function of the system is not lost. System redundancy gives further assurance that the protection function can be maintained.

Multiple, diverse inputs (pressure, temperature, flow, etc.) are provided to the protection systems so that any of several accident-caused anomalies in reactor functions can actuate the protection function. The redundant instrument channels are physically separated in most locations. The cable-spreading room (room 6) below the control room is a common location for most of the channels, so special precautions have been taken in that area as follows:

1. Cable loads and spacing are designed to eliminate abnormal self-heating.
2. The cables have been liberally coated with Flamemastic.
3. A firefighting system is installed in the room.
4. Combustibles in the room are minimized.
5. Room 6 is a controlled access area.
6. No large mechanical equipment is permitted near the critical cable trays.

In other areas where instrument channels share a common location, cable loads and spacing have been reviewed to preclude abnormal self-heating. The cables have also been liberally coated with Flamemastic.

In addition, the rod and ball safety systems are each separately capable of total reactor shutdown under all normal and accident conditions (see Section 4.6). Each system features redundant, diverse, fail-safe trip functions. The two systems can interact only at the trip relay for actuation of one system by the other. As an example of mutual interactions, selected channels of balls will enter the core to provide a reactivity curtain if three or more rods fail to enter the core within the specified time limit on a rod trip. Further details are given in Chapter 7. A number of reviews, analyses, modifications, and upgrades have been performed to improve protection systems independence and to provide enhanced assurance for system operation. 21

It has been concluded that acceptable levels of safety have been attained (see Section 7.2.2).

The plant design does not fully meet Criterion 22. The criterion is not fully met in that protection system channels share common locations.

Criterion 23 - Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Response

Of the five parts of the protection system listed in Response to Criteria 20, three are designed to fail into a safe state if they lose their power sources or the circuits are interrupted. These are listed below.

- Rod safety circuits and instrumentation.
- Ball safety circuits and instrumentation.
- Graphite and shield coolant system once-through-cooling actuation circuits and instrumentation.

For the remaining two parts, the designs and design considerations are as follows:

1. The ECCS instrumentation is fail-safe except for the reactor coolant bulk outlet temperature monitors which require power from the highly reliable (and redundant) 125 Vdc system. The ECCS control circuits are deliberately designed to be fail-negative on loss of power to avoid spurious dumps of the reactor coolant. Two separate control channels are provided, each powered by an independent 125 Vdc power source. Either circuit will actuate the system.

2. The confinement system actuation circuits and instrumentation must have power to initiate the sequential actions that provide for initial release of steam and then, after pressure subsides, close some valves and open other valves to establish the filtered release mode. The actuation system utilizes multiple trip inputs from diverse systems to ensure that no single failure can prevent the actuation system from performing its protective function. The redundancy of all other portions of the actuation system ensures that no single failure will affect more than one channel in the confinement actuation circuits. 21

Potential adverse environments, resulting from normal operation and from accident conditions (e.g., extreme heat or cold, fire, pressure, steam, water, and irradiation), were considered in the design of the protection system components. The common location of equipment and protection system channels in the cable spreading room (room 6) presents a remote possibility for adverse environmental effects on protection systems reliability. Means were provided to minimize the potential for failure of individual components from environmental conditions. However, if an instrumentation component is caused to fail, it generally will fail to the safe state, as discussed above. Redundance of instrument channels protects against spurious actions by failure of individual components. If a protection system circuit is affected by an adverse environment, grounds will register and alarm on the 125 Vdc ground detectors and shorts or open circuits will, in most cases, be annunciated in the control room. Since the protection system circuits and their dc power supplies operate as an ungrounded system, ground detection is a very sensitive means of monitoring for adverse conditions in the locations where equipment important to safety is located; especially that involving steam, water, or any local loss of wiring insulation. Administrative and physical protection is provided in room 6 to minimize the potential for adverse environmental effects. A number of modifications and upgrades have been made to provide enhanced assurance for fail-safe modes. Recent evaluations have concluded the safety levels are acceptable. See Section 7.2.2. 21

The plant design does not fully meet Criterion 23. The criterion is not met in that protection system channels share a common location that has a remote

possibility of adverse environmental effects. However, a high degree of protection system reliability is maintained through administrative and physical protection areas where protection system channels are located in close proximity.

Criterion 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundant, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Response

The protection system consists of five subsystems, as listed in Response to Criterion 20. They sense selected process variables and, if needed, actuate: (1) rod scram, (2) ball drop, (3) emergency core cooling, (4) confinement, and (5) graphite and shield once-through cooling. They are designed to be independent of normal operating controls in that they have independent sensors, transmitters, circuits, and final action relays, with the exception that the pressurizer level transmitters provide signals for both control and safety functions. However, by having redundant level transmitters and using separate re-transmitters for control and safety respectively, the failure of a single component or channel in the control system will not cause the protection system to fail.

One single-failure potential interconnection between the rod safety circuits and rod control assignment circuits physically exists but is controlled by keeping that control circuit locked open (de-energized) during reactor operation and by making checks that there is no voltage on the control circuit. The control circuit was provided for taking an unserviceable rod out of service and out of the "failed rod ball backup" counting circuit. Other means are used to secure an unserviceable rod. The potential path for

interconnection by component failure is thus limited and safety is not impaired. See Chapter 7.2.

The plant design meets Criterion No. 24.

Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Response

The protection system is designed to assure that in the event of a rod withdrawal accident or a major rod swap accident that distorts flux distribution, a reactor trip will be initiated prior to departure from nucleate boiling (DNB). The reactor trip will interrupt the normal control function of the rods to shut the reactor down by full length rod insertions. If three or more rods (or two rods in any one column) fail to scram into the reactor within the required time, the independent ball safety system will be actuated, partially or fully as needed, to provide compensating reactivity control. Either the rod or the ball safety system provides sufficient reactivity control to assure that the fuel design limits are not exceeded. See Sections 4.6 and 15.4 for further details. 2

The plant design exceeds Criterion No. 25 in that two separate reactivity control systems are provided, each capable of assuring that fuel design limits are not exceeded.

Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with

appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Response

Two independent reactivity control systems of different design were provided. These are the rod safety system and ball safety system. Either system is capable of shutting the reactor down and holding the reactor core subcritical under cold, xenon-free conditions.

The rod safety system consists of 84 horizontal control rods of which 45 enter the core from the right side of the reactor and 39 enter from the left side. Separate hydraulic systems are provided for each side. The two rod rooms are physically isolated from each other; this provides some built-in protection against a common-mode event, such as a fire, disabling all rods. The rod safety circuits are redundant and fail-safe in that interruption of power to either of the two circuits will scram all operable rods. This includes the control rods that are selected for use in control of power level and flux distribution. The rod system total reactivity control strength is such that it can provide "speed-of control" (insert fast enough to prevent fuel cladding failure for all accidents) and "total control" (shut the reactor down and maintain it subcritical indefinitely). A reserve of three full-strength rods or the equivalent strength in a larger number of spare rods is provided in the system and amply covers stuck-rod considerations. In the event that more than three rods fail to insert within the required interval, ball columns would automatically be inserted.

The ball safety system is designed to provide a semi-fluid, gravity-inserted backup reactivity control system. It is particularly well designed to function in the event that an earthquake or other emergency condition were to prevent insertion of the rods when needed. (Note: the reactor moderator

and rod safety system are designed to function correctly during the OBE and SSE.) There are 107 active ball hoppers and ball channels. Enough hoppers are kept in service during reactor operation to provide speed of control backup to the rods and total control backup even if up to five of the highest reactivity worth hoppers fail. The ball system serves a safety system function rather than an operational control function.

See Sections 4.3 and 4.6 for further description and evaluation of the reactivity control systems.

The plant design exceeds the requirements of Criterion No. 26. This criterion is exceeded in that two systems are provided, each capable of meeting total control and speed of control criteria.

Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Response

The criterion assumes the use of soluble reactivity control material, inserted into the coolant as a means of achieving total control after control rod insertion. The N Reactor design provides two independent, fully-capable, automatically actuated reactivity control systems. Each system has the capacity to shut the reactor down and maintain it subcritical for an indefinite period of time. Only in the case where two or more rods fail to insert on opening of the safety circuit will supplemental action occur to establish total control by automatically dropping ball columns or manually inserting rods. Automatic ball column insertion will occur on two (or more) rods in one vertical column failing to reach the 75% insertion point within the specified time or on three rods failing anywhere in the system.

Technical Specifications (Ref. 8) specify the minimum number of rods that must be scrammable during power operation. Conformance to these requirements allows a few "extra" rods of known location to be out of service, if needed, for maintenance, without reducing total system strength below the value needed to provide speed of control and total control. The requirements include three rods that are maintained to cover the possibility of stuck rods. Additional margin for stuck rods is provided by backup from the ball safety system.

See Sections 4.3 and 4.6 for further description and evaluation of the reactivity control systems.

The plant design exceeds Criterion No. 27. The criterion is exceeded in that two reactivity control systems are provided to shut the reactor down and reactor shutdown can be maintained with either system.

Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure and cold water addition.

Response

The maximum rate of reactivity insertion that can be induced by misoperation of the rod system is about 0.54 mk/s. Analyses given in Chapter 15 show that this maximum reactivity insertion by the rod system will not: result in damage to the reactor coolant pressure boundary, disturb the core physical geometries, cause fuel cladding failure, or impair the capability to cool the

core. See Sections 3.9.4, 4.3, 4.6, and 5.6 for a discussion of the horizontal control rod system. 2

The rods do not penetrate the reactor coolant pressure boundary, and are not vertically mounted. Therefore, rod ejection and rod dropout are not potential accidents. Pressure tube breaks within the graphite stack will not generate enough pressure to cause outward movement of the rods (Ref. 9).

Other accidents, some not involving rod withdrawals, are evaluated in Chapter 15 and show that postulated reactivity accidents would not damage the coolant pressure boundary, cause fuel melting, disrupt the core, or interfere with the core cooling capability.

The ball safety system is normally operated with no columns of balls in the reactor core. Reactor transients associated with ball system use are evaluated in Chapter 15. No transients were identified whose effects would damage the coolant pressure boundary, melt fuel, disrupt the core, or interfere with the core cooling capability. See Sections 3.9.7 and 4.6.

The plant design exceeds Criterion No. 28.

The Criterion is exceeded in that: (1) two full-capacity reactivity control systems are provided, (2) rod ejection or dropout is not a possible accident-initiating event, and (3) transients involving reactivity increases will be terminated without fuel cladding failure and without any disturbance of the pressure tubes or moderator.

Criterion 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Response

The protection and reactivity control systems are designed to ensure an extremely high probability of accomplishing their safety function in the

event of anticipated operational occurrences. The same protection instrumentation and reactivity control systems that are provided primarily for sensing and controlling accident conditions, also establish suitable boundaries for changes in operating conditions that may occur as a result of operational occurrences.

Such occurrences would be detected by two or more independent protective instrumentation systems when the operating boundaries established by the trip settings were reached. For example, a decrease in the reactor coolant bulk flow rate would be sensed by the following instrumentation:

- o The 1003 pressure tube flow monitors;
- o The pressure tube outlet water zone temperature monitors;
- o The outlet riser boiling point suppression monitors;

Upon initiation of a reactor scram, the response of either one of the two full-capacity reactivity control systems would be adequate to ensure that fuel failure would not occur as a result of an anticipated operational occurrence. See discussion of reactivity control system redundancy and capabilities in Response to Criterions 26 and 27. The assurance that the instrumentation and reactivity control systems will properly respond to such occurrences is maintained by the inspection, testing, and calibrations that are performed as required by Equipment Maintenance Standards (Ref. 7).

The plant design meets Criterion No. 29.

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Response

Components which are part of the reactor coolant pressure boundary were designed, fabricated, erected, and tested to the highest quality standards. See Sections 3.2.2 and 5.2.3 and Response to Criterion 1. 2

Means have been developed for detecting abnormal leakage in the reactor coolant pressure boundary using the following:

- o Frequent (approximately every 3 mo) visual leak inspection of the reactor coolant system external to the reactor, with the coolant system pressurized;
- o Monitoring for abnormal moisture in the moderator atmosphere;
- o Deviation analysis of pressure tube flow data;
- o Monitoring for abnormal radioactivity or temperature in some coolant systems that interface with the reactor coolant system;
- o Monitoring for abnormal increase in reactor coolant system makeup.

The frequent visual leak inspections identify the specific location and type of each leak detected. (Startup is not permitted with a through-wall leak on reactor coolant system piping or valves, regardless of how small the leak rate.) Monitoring for abnormal radioactivity or temperature in selected interfacing systems identifies a leak by general location. A leak on a pressure tube, nozzle, or connector would be identified with the individual pressure tube. When injection rate limits are exceeded as a result of a leak, the leak would be located by inspection during the required shutdown. See Sections 5.2.5 and Response to Criterion 14 for further discussion on leak detection. 2

The plant design meets Criterion No. 30.

Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steadystate, and transient stresses; and (4) size of flaws.

Response

The reactor coolant pressure boundary was designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The reactor coolant system materials are identified and evaluated in Section 5.2.3. The design was in accordance with applicable codes, as supplemented by specially developed standards to cover some materials and quality assurance controls that were not adequately covered in the codes. See Sections 3.2.2 and 5.2 and Response to Criterion 1. Extensive testing of materials demonstrated that they would behave in a ductile manner throughout the expected ranges of temperature and pressure during normal and postulated accident conditions. See Section 5.2.3.

The plant design meets Criterion No. 31.

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Response

Provisions were made in the design for internal inspection of major components that form part of the reactor coolant pressure boundary. These include access hatches in steam generators, the pressurizer, and recuperative heat exchanger. In addition, the pressure tubes and nozzles can be inspected internally, with access by removing nozzle caps. Access ports for valve

inspections have been installed on the left and right sides of the emergency core cooling system dump line just downstream of the V-4 valves. 21

External surfaces of the reactor coolant system are accessible, except for the outside of the pressure tubes. An appropriate surveillance program has been established for the pressure tubes which includes periodic removal of a pressure tube for direct examination and testing. These include destructive tests which provide a direct evaluation of the effects of irradiation and operating conditions on the tubes rather than on indirect evaluation which is typical for LWR pressure vessels.

The plant design meets Criterion No. 32.

Criterion 33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Response

Reactor coolant system makeup is provided by the high pressure injection (HPI) system. Capacity of this system is 6000 gal/min at normal reactor coolant system operating pressures, with both the offsite and onsite power supplies in service and 3000 gal/min with either the offsite or onsite power supply in service. Consequently, the N Reactor design provides unusual capabilities of maintaining reactor coolant system pressure and coolant inventory in the event of small breaks.

The HPI makeup can readily accommodate the loss of coolant from a double-ended break of a reactor pressure tube (about 2.7-in. diameter) in addition to the normal requirement for makeup of coolant spill and leaks. The makeup system is therefore more than adequate to offset coolant losses from breaks in small piping such as sample lines or steam generator tubes or small components of comparable size.

Another safety function of the HPI system is to provide rapid makeup during a reactor scram transient in order to minimize the outsurge of water from the pressurizer that otherwise would result from coolant cooldown and shrinkage. The system is capable of performing this function using two of four pumps in the event that either offsite or onsite power is concurrently lost to two HPI pumps. Loss of power to two HPI pumps has been simulation-tested during a planned reactor scram. The system can provide the scram transient makeup requirements using either the offsite power supply or the onsite power supply. Purpose of this function is to preclude any boiling in the outlet piping of the reactor and thus maintain a large margin of boiling suppression at the reactor coolant pumps, which are downstream of the steam generators on the cold legs of the loops.

The HPI system provides protection against exceeding fuel design limits for potential small break cases and for scram transients, including any scram resulting from a power supply system failure. The same HPI pumps used for routine makeup to the reactor coolant system are the ones that would supply the abnormal demands. No transfer of pumps, water sources, power supplies, or injection control valves are needed. See Sections 9.3.4 and 15.6.

The plant design exceeds the requirements of Criterion No. 33. The criterion is exceeded in that the HPI system is capable of maintaining the reactor coolant inventory with leaks substantially exceeding those postulated in the criterion.

Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual

heat from the reactor core at a rate such that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

Unlike the typical PWR, the residual heat removal equipment at N Reactor is not required to operate as a part of the emergency core cooling function. It has no borating function and is not needed to remove heat during major breaks in the reactor coolant pressure boundary. Instead, the capability of afterheat removal is provided by the reactor coolant loop for those shutdown conditions where reactor loop recirculation is maintained and, therefore, emergency cooling is not needed. This is not a safety function. The ECCS provides residual heat removal during emergency conditions.

The afterheat removal mode is established each time the reactor shuts down, after the reactor generated steam becomes insufficient for powering the drive turbines and the 184-N turbine-generator set (TG). Portions of the secondary cooling system that normally contain steam are flooded to establish a water-water recirculation heat transfer between the in-service steam generators and the dump condensers, using the Afterheat Removal (AHR) fill pumps, which draw from the demineralized water storage tank or the AHR storage tank. Circulating raw water from the CRW system is maintained on enough dump condensers to transfer the heat to the river. Recirculation of the coolant within the secondary coolant system is performed by some of the 16 condensate return pumps as needed.

The graphite and shield cooling system (GSCS) removes residual heat from the graphite moderator and from the reactor shield. In the normal recirculating

mode, the coolant passes through heat exchangers to transfer the heat to the river. On loss of recirculation, the GSCS transfers automatically to single pass cooling, with the effluent directed to the 1325-N Facility.

Normal shutdown heat removal from the reactor can be accomplished either by equipment powered by the offsite power supplies or the onsite electrical supply plus steam from the standby boilers.

Redundant equipment is provided in both the offsite and the onsite powered cooling equipment, such that with the exception of the single TG set for the onsite power, either system can remove residual heat assuming a single active component failure. If the TG set fails while the reactor is being cooled solely by onsite power, reactor cooling will be provided by the ECCS, which does not depend on onsite or offsite ac power for its operability.

The plant design meets Criterion No. 34. The system differs in concept and operation from the typical PWR, but provides adequate redundancy in the three cooling systems to assure that fuel design limits and coolant pressure boundary limits are not exceeded.

Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

Response

The N Reactor emergency core cooling system (ECCS) was designed to automatically provide an abundant supply of water to the reactor in the event that any condition in the reactor coolant system should prevent adequate heat removal by the normal recirculation cooling mode of heat transfer from the core to the steam system and thence to the river. Refer to Chapter 6.3 for evaluation of the adequacy of ECCS. This evaluation shows that the emergency cooling is adequate to prevent cladding failure except for certain single tube events and can limit cladding metal-water reactions to negligible amounts.

The ECCS pumps are driven directly by diesel engines, the inlet and outlet valves are operated by compressed air and the protective instrumentation, trip circuits, breakers, solenoids and monitoring instruments are powered by 125 Vdc batteries or 125 Vdc/120 Vac inverter supplies from the batteries. Thus, the system will function in the absence of offsite power and onsite power.

The ECCS is designed to meet its minimum required level of performance in the event of a failure of any single active component in the short term or any single active component failure or any limited-leakage passive-component failure in the long term. Under some accident conditions, the GSCS provides an independent path for indirect cooling of the fuel and preventing fuel failure in the fuel area covered should complete failure of the ECCS occur after 2 hr of operation. 21

The plant design meets Criterion No. 35.

Criterion 36 - Inspection of Emergency Core Cooling System. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Response

The emergency core cooling system (ECCS) is designed such that all major components can be inspected during the scheduled outages. These outages occur each 3 or 4 mo. Many of the major components can be visually inspected during reactor operation, and some, such as the ECCS pumps, can be functionally checked without reactor shutdown. The scheduled inspections are performed in accordance with Equipment Maintenance Standards (Ref. 7) and Pressure Systems Instruction (Ref. 10). The inspections are adequate to assure the integrity and capability of the system.

Inspections of spray rings and of water injection nozzles in the reactor pressure vessel are not applicable to this plant since the reactor has 1003 pressure tubes instead of one reactor pressure vessel. The ECCS system provides emergency cooling by establishing a once-through cooling mode which provides water to all supply risers and distributes flow to the pressure tubes through the individual tube inlet connector pipes. Internal sprays are not required.

The plant design meets Criterion No. 36.

Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system (see Section 6.3.4, Tests and Inspections).

Response

The emergency core cooling system (ECCS) is functionally tested as follows:

- o The diesel-driven ECCS pumps are automatically started and accelerated to full speed each time the reactor scrams. This action is initiated by the rod safety system trip circuits rather than by the ECCS trip circuits.
- o Each diesel-driven pump is started prior to each reactor startup.
- o Each diesel-driven pump is tested and inspected in accordance with Equipment Maintenance Standards, which includes, in part, routine inspections of the diesels' internal components.
- o A periodic ECCS flow check is conducted. 21

Total ECCS diesel starts are about 250/yr on the five diesels.

The V-3 (ECCS supply) and V-4 (ECCS dump) valves are functionally tested approximately once per quarter.

The ECCS protective instrumentation and trip circuits are individually tested in accordance with the Equipment Maintenance Standards (Ref. 7).

In addition to these tests, the system is normally used once each year to provide actual once-through cooling of the reactor core during the annual extended outage, for reactor coolant system piping chemical-decontamination, and also, whenever work is performed on parts of the reactor coolant system. During these modes, the only parts of the system that are not used are the low lift diesel-driven pumps, the LERF and the ECCS supply (demineralized water storage) tank. 21

These uses and the periodic ECCS flow check provide a test of system performance as closely as is practicable to the full operational sequence that brings the system into operation. These functional tests also serve to check the system components for structural integrity and leak-tightness up to the system conditions encountered if the systems were actuated, except

for the V-3 and V-4 valves. It is not practical to routinely test these valves against reactor operating temperatures and pressures.

Testing of transfer between normal and emergency electrical supplies is not applicable for this emergency core cooling system, since it is not dependent on maintaining the onsite ac electrical supply. The redundant 125 Vdc instrument and actuation power supplies are individually tested in accordance with Equipment Maintenance Standards (Ref. 7).

The plant design meets Criterion No. 37.

Criterion 38 - Containment (Confinement) Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

Response

The N Reactor confinement concept and design includes two provisions for heat removal from the confinement system in 105-N and 109-N. For LOCA involving large pipe breaks, a large fraction of the sensible heat released from the RCS is vented to the atmosphere by the steam vents. For LOCA involving the reactor pressure tubes, the steam release is slower and the steam vent caps are not opened, except for the special steam vent in one case. In both types of LOCA, the confiner pressure rises enough to actuate the fog spray system to further cool the confiner atmosphere and condense steam released from the reactor coolant. Analyses in Chapter 6 and Chapter 15

show that the confiner pressure is controlled below the design rating of the confiner structure in all cases, and the confiner atmosphere temperature is controlled to acceptable levels. The confinement system routes the fog spray water, with subsequent disposal of the water, to the Liquid Effluent Retention Facility (LERF), 1312-N. See Section 6.2.2 for further discussion and evaluation of the confinement spray systems. 21

A separate cooling spray system is provided for the confinement halogen and particulate filters in the 117-N filter building. This system would automatically start spraying if temperature elements on the filters sense a temperature of about 240 F, and shut off automatically when the temperature drops to 215 F. The 117-N building and the associated ducts provide adequate containment for the filter spray water.

As with the ECCS system, the confinement heat removal systems do not depend upon either the offsite or the onsite ac electrical supplies.

Control power is provided by redundant 125 Vdc systems and 125 Vdc/120 Vac inverter supplies. See Section 8.3. Pumping is by direct diesel drive. However, if offsite power or onsite power is retained, separate motor-driven pumps can also supply water to these spray systems. 2

The plant design meets the intent of Criterion No. 38. The confinement heat removal system can perform its function without onsite or offsite power. Also, with few exceptions, the system can perform with single active component failure. The 105-N and the 109-N pipe gallery fog spray systems have been upgraded to meet seismic Category I requirements. 15

Criterion 39 - Inspection of Containment (Confinement) Heat Removal Systems
The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Response

The confinement heat removal spray system (steam vents, fog sprays, and filter sprays) are designed to permit appropriate periodic inspection of important components with the exception of the Zone I fog spray nozzles which, in some cases, are located near the ceiling in 75-ft-high pipe spaces and are nearly inaccessible because of the height from the floor. In 1984, Project H-522 replaced the original nozzles in the 109-N pipe gallery with spiral non-clogging nozzles. The 105-N spray nozzles are also spiral, non-clogging types. Further, the 109-N cell spray nozzles have a spiral configuration which provides similar large-particle clearance capability. The 117-N filter sprays are fully accessible for inspection.

The overhead fog sprays, though largely inaccessible for inspection, are considered to be in good, reliable condition based on the results of a test of spray the system, which used water from the RWS-2 fog spray supply system and showed no significant nozzle plugging. A functional spray test of the 105-N spray nozzles was performed during the 1977 Summer Outage. No nozzles plugged during the test. Another test was performed during the 1987-88 outage and some plugging of lines to nozzles due to rust accumulation was observed and corrected.

The heat removal systems use a river water source and do not take suction from a sump or torus. The resulting spray water would be collected and adequately contained in the Zone I pipe spaces and 117-N building exhaust air tunnel and filter spaces.

The plant design meets Criterion No. 39.

Criterion 40 - Testing of Containment (Confinement) Heat Removal Systems

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into

operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Response

The confinement heat removal systems consist of: (1) the steam vents, (2) the 105-N and 109-N fog spray system, and (3) the 117-N confinement filter spray system.

The confinement heat removal systems are tested in accordance with the Equipment Maintenance Standards. These tests do not include actual discharge through the 105-N and 109-N nozzles. Actual spray tests were conducted during the 1977 Summer Outage and during the 1987-88 outage (see Section 6.2.1.6). The 117-N filter sprays are flow-tested through the spray nozzles. These tests are considered to be as close as practical to the performance of the full operational sequences. Pressurization of the confinement envelope to functionally test the release of the steam vent weather caps is not practical. In lieu of the functional tests, certified shear pins are installed according to EMS procedures. 15 21

See Section 6.2 for further descriptions and evaluations of confinement heat removal systems.

The plant design meets Criterion No. 40.

Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality (quantity) of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available), and for offsite electric power system operation (assuming onsite power is not available), its safety function can be accomplished, assuming a single failure.

Response

In event of a loss-of-coolant accident (LOCA), the confinement fog spray system sprays raw river water into the 105-N pipe spaces and the 109-N pipe gallery to condense steam and remove fission products (primarily iodine and particulates). Experiments have demonstrated that raw water sprays are effective in removing elemental iodine and are better for that purpose than demineralized water (Ref. 11). After passing through sprays, the confiner atmosphere is routed through particulate and halogen filters in the 117-N building and then released through the 200-ft high ventilation stack. Refer to Sections 6.2 and 6.5 for further discussion of the confinement system and control of fission products.

Offsite and onsite ac electric power are not needed for these confiner-atmosphere cleanup systems to work. The fog spray and filter spray water can be supplied by direct-diesel-driven pumps, and the controls for both systems can be powered by either offsite ac power or the onsite power 125 Vdc battery systems. Spray control valves for both air cleanup systems are spring-operated to open on being de-energized.

On emergency actuation, the confinement filter system will automatically operate in a single-pass mode during a LOCA utilizing Cell D filters (see Section 6.2.4). At least one spare filter cell is maintained that can be switched into service if needed and a defective cell isolated by operation of controls from the reactor control room. See Section 6.5.1 for a description of in-place filter tests and inspections.

The fog spray system and filter spray system are independent of each other. In addition, the fog spray systems in 105-N and the 109-N pipe gallery meet

the single failure criterion (Project H-522 upgraded the fog spray system in the 109-N pipe gallery). 15

The amount of hydrogen generated in accident situations has been shown by analyses to be insignificant. A hydrogen monitoring system and a hydrogen mitigation system have been added as part of the Safety Enhancement Program (SEP) to ensure the ability of the N Reactor to safely accommodate hydrogen generated from postulated severe accidents consistent with requirements of 10 CFR 50.44 for commercial LWRs. The combustible gas monitoring system was installed under Project H-798 to provide control room operators the capability to determine post accident hydrogen and oxygen concentrations in the confinement atmosphere in the event of a LOCA. Two independent and redundant measuring systems are provided. The system is not required for safe shutdown or to initiate automatic hydrogen mitigation activity. The hydrogen mitigation system is being installed under Project H-791 and will provide 40,000 cfm of nitrogen flow from two redundant trains to inert the confinement atmosphere within 15 minutes after the accident. See Section 6.2.5, and Chapter 15 for discussion of hydrogen generation and control. 21

The plant design meets the intent of Criterion No. 41.

Criterion 42 - Inspection of Containment (Confinement) Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Response

The confinement fog spray system serves the dual functions of a heat removal system and an atmosphere cleanup system. See Response to Criterion 39 for discussion of the inspection of that spray system. In summary it provides for inspection except for the partially inaccessible overhead spray nozzles which are inferred to be in good condition on the basis of testing of 2

identical nozzles in a test station and a functional test performed during the 1977 and 1987 outages.

The confinement filters are accessible during reactor shutdown for direct visual inspection and instrumented testing of efficiency. This includes the ducts, filters, frames, seals, cooling spray piping, spray nozzles, filter temperature monitors, and spray valves. See Section 6.5.1 for a description of in-place filter tests and inspections.

Inspections of the atmosphere cleanup systems are performed in accordance with Equipment Maintenance Standards (Ref. 7).

The plant design meets Criterion No. 42.

Criterion 43 - Testing of Containment Confinement Atmosphere Cleanup Systems
The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Response
The confinement atmosphere cleanup systems consist of the fog spray system and the 117-N confinement filter system. Refer to Response to Criterion 39 for testing of the fog spray system that performs the dual function of

confinement heat removal and atmosphere cleanup. The 117-N filter system is tested as follows:

- o The particulate filters are tested for total efficiency (including effects of leakage around the filters), using the dioctyl phthalate (DOP) method.
- o The halogen filters are tested for total efficiency, using radioactive elemental iodine, and results are corrected to moisture and temperature conditions that would exist during the LOCA by applying laboratory test results of samples taken from the filters to the in-place test results.

The filter efficiency tests are performed during reactor shutdowns. See Section 6.5.1 for a description of the in-place filter tests and inspections.

The plant design permits testing of the atmosphere cleanup systems, under conditions as close to design as practical, for their performance during postulated LOCA conditions. These systems do not include fans and dampers and do not use ac power sources as inferred by the criterion. The water can be supplied by diesel-driven pumps and all actuating circuits are powered by redundant 125 Vdc battery systems.

The plant design meets Criterion No. 43.

Criterion No. 44 - Cooling Water

A system to transfer heat from structures, systems and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response

N Reactor design differentiates between: (1) the normal operating and shutdown cooling heat removal function and (2) the functions of emergency core cooling and confinement heat removal during accident conditions.

Normal operating heat removal is provided by recirculating the reactor coolant through steam generators. The steam produced from the reactor heat by the steam generators is partially used to drive the RCS pump drive turbines and the WPPSS electrical generator drive turbines. The remainder of the heat is transferred to the river via condensers. See Section 5.4. The graphite and shield cooling system (GSCS) removes heat from the reactor shield and the graphite stack by recirculating coolant through raw-water-cooled heat exchangers which transfer the heat to the river. See Section 5.5.

Normal shutdown residual heat removal is accomplished by recirculations of the reactor coolant to one or more steam generators and transfer through the secondary coolant system from steam generators to dump condensers in a flooded afterheat removal recirculation mode. The heat is then transferred to the river by the circulating raw water (CRW) systems on a single pass cooling of the dump condensers. The normal shutdown residual heat removal functions can be accomplished using either the offsite or the onsite power supply. If both power supplies are lost, the ECCS would be actuated to remove the residual heat. See Section 6.3.

Residual heat removal from the reactor core during emergency conditions is provided by the emergency core cooling system (ECCS) operating in a single pass mode using river water to transfer heat to a dump tank and thence to the Liquid Effluent Retention Facility (LERF), 1312-N, and after the LERF is full, to the 1325-N crib which dissipates the heat to the ground and to the atmosphere. The emergency cooling water supplies for core cooling, graphite cooling, fog sprays and filter sprays are provided by diesel-driven pumps. The diesels provide their own jacket and pump bearing cooling water supplies that transfer the heat to the river. The diesels are maintained prepared for at least 5 days operation. See Section 6.3. The GSCS provides additional

core cooling in emergency conditions also. If the accident includes total loss of power, the GSCS automatically transfers to once-through cooling, transferring the heat to the 1312-N/1325-N facilities. | 21

Confinement heat removal is accomplished by a combination of controlled release of energy to the environs and by fog sprays and filter sprays using cool river water. The heated water from the fog sprays and filter sprays is collected in Zone I and is then pumped to the 1312-N LERF. See Section 6.2.1. | 21

The plant design meets Criterion No. 44.

Criterion No. 45 - Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Response

The N Reactor design uses separate equipment for: (1) residual heat removal during normal shutdown conditions and (2) heat removal from the reactor core and from the confinement system during accident conditions. See discussion in Response to Criterion 44. Each of these heat transfer systems has redundant pumps, which provide the capability for inspections during reactor operations, if needed, in addition to the inspections normally made during reactor shutdowns. Steam generators and heat exchangers are accessible for inspection during reactor shutdown and, by proper isolation, during reactor operation. | 2

The plant design meets Criterion No. 45.

Criterion No. 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance | 2

of the full operational sequence that brings the system into the operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Response

The N Reactor design uses separate equipment for: (1) residual heat removal during shutdown conditions and (2) heat removal from the reactor core and from the confinement system during accident conditions. See Response to Criterion 44.

The residual heat removal equipment is used each shutdown to transfer heat from the reactor coolant system. Most of the time it operates on the offsite power supply; however, it operates on the onsite power supply during maintenance or testing of the offsite system. This usage tests the capabilities of this equipment for the normal shutdown heat removal function. Since it makes use of equipment that is already in service at time of shutdown, no equipment starts nor automatic transfer to an emergency power supply is needed.

The emergency core cooling system diesels provide their own jacket and pump bearing cooling water. The ECCS system is tested routinely in accordance with Equipment Maintenance Standards (Ref. 7). In addition, the system has been used to actually cool the reactor core about once per year, during maintenance on the reactor coolant system. The graphite and shield cooling system is an additional, partial-capacity, core-cooling system as discussed in Response to Criterion 35. It is tested in accordance with Equipment Maintenance Standards. The diesel jacket and pump bearing cooling systems are functionally tested, together with the routine diesel tests.

The confinement fog spray and filter spray systems do not use heat exchangers to remove heat from the confiner. The heated spray water would collect in pipe spaces and the exhaust air tunnel, respectively. The RWS-2 system diesel-driven pumps that supply fog spray water, provide cooling water for

their jacket cooling and pump bearing cooling systems. These are tested, together with the routine diesel tests.

Visual inspections for structural integrity and leak-tightness are performed during functional tests of the systems.

The plant design meets Criterion No. 46.

Criterion No. 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Response

The design of the confinement structure was based on a double-ended break of the reactor coolant outlet manifold with concurrent failure of the ECCS. In this hypothetical accident, steam would be initially vented to the atmosphere from the confiner through normally open steam vent valves (low-pressure weather caps are released) in the interval before fission products could be released from fuel if the fuel were not cooled by the ECCS system. After a timed interval and after subsidence of reactor coolant pressure and confiner pressure, the steam vent valves would close and the exhaust confinement valves open to establish a filtered release of the confiner gases to the ventilation stack. In postulated accidents involving pressure tube breaks with fuel failure, the timed confinement system sequence would

be actuated, as in the large pipe break case, but the confiner pressure would not increase enough to open the steam vent caps. Fission products would be released to the confiner concurrently with the steam in these cases, but the steam and fission products would be contained within the confiner, except for an increased leak rate from the confiner due to the confiner pressure buildup. After the confiner pressure subsided, the confinement sequence would be completed to establish a filtered release of gases from the confiner. Analyses in Chapter 15 show that, for all cases, the dose rates outside the exclusion zone boundaries would be well within the 10 CFR 100 guidelines. See 6.2 for a description of the confinement equipment actuation sequences. The confinement structure is designed for 5 lb/in^2 (gage) internal pressure and -2 lb/in^2 (gage) vacuum [12 lb/in^2 (gage) and -2 lb/in^2 (gage) in the 109-N cells]. The peak pressure that would be produced in the confiner by the maximum size pipe break has been calculated to be approximately 4.0 lb/in^2 (gage). Similarly, the potential negative pressure has been calculated to be approximately -0.5 lb/in^2 (gage). See Section 6.2.

The plant design meets Criterion No. 50.

Criterion No. 51 - Fracture prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties; (2) residual, steady-state, and transient stresses; and (3) size of flaws.

Response

The confinement structure is primarily made of reinforced concrete. The components in the confinement system which are made of ferritic materials include:

- o Rebar in concrete
- o Confinement valves
- o Steam vents
- o Vacuum relief valves
- o ECCS dump tank shell and vent line connecting the tank to confinement Zone I
- o Zone I exhaust fan housings and associated ducts.

The ferritic materials were selected to behave in a nonbrittle manner under normal and accident conditions to minimize the probability of a rapidly propagating failure. The design reflects consideration for service temperature during normal operation and postulated pipe-break accident conditions.

The plant design meets Criterion No. 51.

Criterion No. 52 - Capability for Containment Leakage rate testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Response

The confinement structure is routinely given total leak rate tests which include the confinement Zone I structure, the exhaust tunnels to the 117-N filter building and the filters to the downstream filter gates. The test data are corrected to the pressure conditions which would exist during an accident. See Section 6.2.6.

The plant design meets Criterion No. 52.

Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Response

The confiner leak rate test is performed at the pressure anticipated during release of the confiner atmosphere through the 117-N filters, and includes a check of the leak-tightness of penetrations of the confinement boundary. The tests results are normalized to 0.1-in. w.g. for comparison with the specified leak rate limit of 200 ft^3/min at that pressure. At higher confiner pressures anticipated during some postulated accidents, the confiner leak rate is extrapolated from the test results for use in calculating dose rates outside of the exclusion zone. See Section 6.2.6. No leak tests are performed at confiner design pressures because it is not practical to pressurize the entire confiner to these pressures.

A surveillance program of confinement valve tests and inspections is required by Equipment Maintenance Standards (Ref. 7) and is an essential part in the program for maintaining leakage within the limit specified by Technical Specifications (Ref. 8). The limit is based on the allowable leakage assumed during calculation of exposures offsite during accidents analyzed in Chapter 15.

The plant design meets the intent of Criterion No. 53.

Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to periodically test the operability of

the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Response

Piping systems that penetrate the Zone I confinement structure were not provided with leak detection, isolation, and containment capabilities to the extent that is typical for a containment system today. The degree of isolation capability that was provided during construction is discussed in Section 6.2.4. The design is considered to be adequate, since accident analyses indicate that breaks in the piping system that penetrate the confiner will not result in fuel damage or significant radiological releases. See Section 15.6.2. The confinement leak rate test periodically checks the integrity of penetrations through the confiner boundary. o 21

The plant design does not meet Criterion No. 54. 21

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment Confinement

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and, on loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Response

Reactor coolant piping systems that penetrate the Zone I confinement structure were not provided with leak detection, isolation, and containment capabilities to the extent that is typical for a containment system today. The degree of isolation capability that was provided during construction is discussed in Section 6.2.4. The design is considered to be adequate, since accident analyses indicate that breaks in the reactor coolant piping that penetrate the confiner will not result in fuel damage or significant radiological releases with operator action. See Section 15.6.2. The confinement leak rate test periodically checks the integrity of penetrations through the confiner boundary. Also, N-Reactor has methods to rapidly detect fuel failure resulting in immediate shutdown.

Since fission products are not available for release, the plant design does not need to meet Criterion No. 55.

Criterion 56 - Primary Containment Confinement Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment

isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and, on loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Response

The confinement penetrations that connect directly to the confiner atmosphere are provided in most cases (but not all) with a primary closure and a backup closure, but not in the manner indicated in Criterion 56. These closures are routinely tested for leak tightness during the confiner leak rate test. See Section 6.2.6 for a discussion of confiner isolation. The degree of isolation provided is considered adequate, since the confinement system does not depend on a high degree of leak tightness to perform its function and fission products are not available for release (see above GDC 55). 2 21

The plant design does not meet Criterion No. 56. 2

Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve

which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Response

Piping systems that penetrate the Zone I confinement structure were not provided with leak detection, isolation, and containment capabilities to the extent that is typical for a containment system today. The degree of isolation capability that was provided during construction in 1961 and 1963 is discussed in Section 6.2.4. The design is considered to be adequate, since the confinement system does not depend on a high degree of leak tightness to perform its safety function. The confinement leak rate test periodically checks the integrity of penetrations through the confiner boundary.

The plant design does not meet Criterion No. 57.

Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Response

The N Reactor design includes means to control releases of radioactive gaseous, liquid, and solid wastes to levels well within the requirement of DOE Orders 5820.2A and 5480.1B. The above is achieved without the plant design including special holdup facilities. The site environmental conditions are favorable, the exclusion area is unusually large, reactor coolant activity

level is low (any fuel element that experiences clad failure is promptly removed from the reactor), radioactive releases to the river are minimized, liquid wastes are discharged to an engineered soil column treatment facility and most of the confinement structure ventilation air is passed through filters prior to release from the ventilation stack. These are all factors in the achievement of good control over offsite releases. See Chapter 11.

The plant design conforms to the intent of Criterion No. 60, as demonstrated by conformance to DOE 5820.2A and 5480.1B containing requirements on (1) Radioactive Waste Management and (2) Standard for Radiation Protection. (These correspond to parts of 10 CFR 20.) | 21

The plant design meets the intent of Criterion No. 60. The criterion is not fully met in that holdup facilities are not provided for gaseous effluents. Other means are provided to control offsite releases within acceptable limits.

Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Response

The fuel storage, fuel handling, radioactive waste facilities, and other systems which may contain radioactivity from N Reactor operations are designed

to assure adequate safety under normal and postulated accident conditions. Specifically:

1. The spent fuel storage basin is monitored continuously for coolant level and has adequate shielding for radiation protection. The ventilation air can be diverted to full filtered exhaust. The basin has more than sufficient heat removal capability and is designed to prevent significant reduction in coolant under accident conditions. See Section 9.1.2. The storage basin conforms to Criterion No. 61.
2. The 1310-N chemical waste storage tank is inspected prior to each use, is diked to provide adequate shielding for radiation protection, fully confines the liquid effluents but allows some atmospheric release of radionuclides, has no need for residual heat removal and is so located that no direct release to the river will result on catastrophic failure of the tank. The chemical waste storage tank conforms to the intent of the criterion except that it is not provided with confinement filters. However, no confinement or filtration is needed for this tank, as shown by measurement of gases released from the tank. See Section 11.3.
3. The 1312-N/1325-N soil liquid waste disposal facility is not designed to permit below grade periodic inspection, but the surface components are periodically surveyed; it provides suitable shielding for radiation protection, confines most of the radionuclides so treated but does allow small unfiltered releases of radionuclides to the atmosphere. See Section 11.3. The soil column treatment facility does not conform to the literal wording of the criterion.
4. There are three used fuel spacer temporary storage silos. They are cylindrical structures (one reinforced concrete, two metal) that are installed vertically underground with open bottoms and concrete capped tops that are approximately at grade. Dimensions are 11-ft diameter by 25-ft high for one, and the other two are 17-ft by 37.5-ft high. Transfer of the irradiated spacers from the 105-N fuel storage basin to the silo currently in use is by mechanical displacement through a 3-in pipe. The storage basin end of the pipe is 10-ft under the water surface and it inclines upward to just under the concrete cap at the silo. No water from the storage basin enters the silo other than that wetting

the spacer surfaces. Shielding is provided by the 10-ft of water, the ground and the 1-ft thick concrete cap. (There is a ball valve on the storage basin end of the line.) No confinement filtration is needed on these storage silos. The tops are sealed, except when an inspection plug is removed.

The used fuel spacer silos provide for adequate inspection, have suitable shielding, do not require confinement filtration and do not require residual heat removal capability. The fuel spacer silos conform to the intent of Criterion No. 61.

The plant design meets the intent of Criterion No. 61. The literal requirements of the criterion are not met since the 1310-N chemical waste storage tank is not equipped with confinement filters, and no confinement is provided for the 1312-N/1325-N soil column treatment facility. The radionuclide releases from these facilities, however, are well within boundary dose requirements in all cases. 21

Criterion 62 - Prevention of Criticality in Fuel Storage and Handling
Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Response
The ^{235}U enrichment levels in N Reactor fuel are 0.95% for the standard fuel elements and 1.25% for the spike fuel elements. These low enrichments make it comparatively easy to maintain criticality safety in handling and storing the fuel. Unirradiated fuel storage pallets, charging monotubes, monotube racks, irradiated fuel discharge carts, irradiated fuel storage baskets and storage cubicles are designed to provide nonoptimum physical spacing. These physical controls, together with procedural limits on the quantity of fuel per container, provide assurance that no single action or failure of controls can cause an array of fuel that would go critical.

Criticality prevention is discussed further in Section 9.1.

The plant design meets Criterion No. 62.

Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Response

Monitoring instrumentation is provided to alarm in the event that an excessive radiation level or a low water level occured in the irradiated fuel storage basins. Appropriate safety actions are initiated by operator action.

In discharging the irradiated fuel from the reactor, the fuel elements drop directly into a water-filled discharge chute from the horizontal pressure tubes. During the fuel discharge, radiation monitoring equipment is in service to alarm in the unlikely event that a fuel element would fail to enter the discharge chute pool and in the further unlikely event that the fuel element would ignite. The ventilation air from the discharge area routinely goes through confinement halogen and particulate filters, and there are fog sprays available by manual control to cool the fuel and scrub fission products from the discharge area air, if ever needed. The radiation monitoring equipment is described in Section 12.1.5. 2

Level monitoring instrumentation is provided for the 1310-N waste tank.

The plant design meets Criterion No. 63.

Criterion 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Response

The confinement atmosphere is sampled routinely for iodine, noble gases, tritium, and particulates at the 116-N ventilation stack, the 109-N ventilation exhaust air duct, and at the Cell 6 exhaust air duct. The 116-N stack air monitoring equipment also includes three gamma radiation monitors arranged to provide for a range of monitoring from normal activity levels through accident activity levels that would result from releases during the postulated accident.

Radioactivity levels in liquid effluents paths are sampled by three methods:

- The effluent to the 1312-N/1325-N liquid waste disposal facility is automatically sampled by equipment that takes samples at preset intervals and composites them. A similar sampling system provides composite samples of riverbank springs and a sampling well for effluent seepage from the retired 1301-N soil column treatment facility. 21 15 21
- The main circulating raw water return line (102 in.) is sampled by equipment that concentrates any traces of radioactive material, in particulate filters and ion exchange resin beds for periodic analysis.
- Grab samples are taken from other streams returning to the river such as the 66-in. line and the 182-N water storage tanks overflow line.

Radiation gamma monitors are installed in several locations: 105-N, 109-N, 1312-N/1325-N facility, on top of the ECCS dump tank, at the fence line between the reactor plant and the WPPSS Hanford Generating Project and at the reactor main gate. These monitoring devices would be used to monitor radioactivity levels that would result from accidents which release fission products. The four environmental monitors that are located at the 1325-N facility, Westinghouse Hanford-WPPSS fence line, top of the ECCS dump tank and area main gate transmit their readings via telemetry to the Central Fire Station located approximately three miles south of the 100-N Area. All the space radiation monitor readings are locally recorded in the 105-N control room. The environment radiation monitors and the 116-N ventilation stack gamma monitors provide the capability of monitoring radioactivity levels under postulated accident conditions. 21 15 21

See Chapter 11 for additional monitoring details. Data from the installed monitors and samplers and from the site surveillance program are used to make routine calculations of dose commitments to the public for a distance of 50-mi radius around the reactor.

The plant radioactivity monitoring design meets Criterion No. 64.

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

GDC 2 of Appendix A, 10 CFR 50, requires that structures, systems, and components important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions. Appendix A of 10 CFR 100 defines a Safe Shutdown Earthquake (SSE) and an Operating Basis earthquake (OBE) and further requires that if the SSE occurs that certain structures, systems and components of nuclear plants be designed to remain functional. These plant features are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary (RCPB).

2. The capability to shut down the reactor and maintain it in a safe shutdown condition. or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100.

Regulatory Guide 1.29, Revision 3, describes an acceptable method for identifying and classifying those plant features that should be designed to withstand the effects of an SSE. Those structures, systems, and components, including their foundations and supports, so identified have been designated as Sesimic Category I. The Guide also requires that the interfaces between Seismic Category I and non-Seismic Category I be designed to Seismic Category I requirements. No other Seismic Categories are identified in the NRC Regulations and Regulatory Guides.

Table 3.2-1 lists the structures, systems and components designated as Seismic Category I which have been analyzed under the loading conditions of the Safe Shutdown Earthquake (SSE) with appropriate margins of safety to assure that their design functions are not jeopardized in the event of an SSE. Further

details of seismic design criteria, methods of seismic analysis and qualification are presented in Section 3 as follows:

Seismic Design	3.7
Design of Category I Structures	3.8
Systems and Components	3.9
Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10

Those structures, systems, and components not analyzed under the loading conditions of the SSE are classified non-seismic Category I. For these, the applicable seismic loading conditions as determined from the Uniform Building Code was used for design. The Operating Basis Earthquake, as defined in 10 CFR 100, Appendix A, is not incorporated as part of the seismic classification scheme.

The seismic classification indicated in Table 3.2-1 is in accordance with NRC Regulatory Guide 1.29, Revision 3, except as otherwise noted in the table. Also listed in Table 3.2-1 are the basic methods of validation, i.e., static analyses, dynamic analyses, III/I walkdown and seismic testing (Ref. 1 and 2).

The column headed "Seismic Test" represents, for the most part, testing performed prior to the 1987-88 period. Results from the seismic and EQ program currently in progress are not totally reflected in Table 3.2-1 at this time, but will be entered at time of restart.

Figures 3.2-1 through 3.2-7, and 3.2-13 are provided to show the Category I parts of the emergency core cooling system, reactor coolant system, and fog spray system. Category I parts are differentiated from those having lesser

seismic capability by heavy lines or other notations. Figures 3.2-8 through 3.2-12 are provided to show the reactor structures that are Category I. See Section 3.8 for figures of other Category I structures. 15 021

3.2.2 System Quality Group Classifications

3.2.2.1 Original Design

Quality classifications as they exist today were not in use during the period of 1958 through 1963 when N Reactor was being designed, procured, constructed, tested, and placed into operation. Methods used to obtain adequate quality for systems important to reactor safety were as follows: 021

1. Pressure-retaining components of the RCS and associated high-pressure systems were procured, fabricated, and constructed in accordance with Hanford Works Specification HWS-6897, "Furnishing, Delivery and Erection of Primary Coolant Piping in 105-N and 109-N buildings", dated April 1, 1960, and Amendments 1 (July 1960) and 2 (August 1961). Basic to this specification were the following sections of the ASME Boiler and Pressure Vessel Code and the Power Piping Code:

   - Section I--Power Boilers, 1956 and 1959.
   - Section VIII--Unfired Pressure Vessels, 1956 and 1959.
   - ASA B31.1.0, 1955--American Standards Association Power Piping Code. 2

   In 1959, as today, the ASME Boiler and Pressure Vessel Codes and ASA B31.1 (currently identified as ANSI B31.1) represented minimum industrial standards for design, fabrication, and erection of pressure systems. HWS-6897 greatly surpassed the minimum code guidance in effect and, for that time, represented the most detailed and highest quality requirement specifications in existence for a steam generating plant. 021

2. HWS - 6897 is representative of the system of HWS-xxxx engineering specifications, which were the primary means of establishing quality requirements for the plant. These specifications totaled more than 400 and covered such equipment as control rods, electrical systems, etc., 21

in addition to pressure-retention systems. Purchase orders, blueprints, letters, etc., also transmitted information regarding the required quality.

3. Where codes did not apply properly to the materials or where better quality than required by the code was desired, special standards were developed and used. These were designated Section N-1 through Section N-21 and covered such things as:

   - Section N-7, "Welding Specifications for Radioactive System Service"
   - Section N-14, "Nondestructive Testing Requirements for Metals-Technical Sections"

   These special standards were required, where appropriate, in HWS-6897, the specification for the RCS and associated pressure systems and in specifications for other pressure-retention systems.

4. Project inspectors were extensively used to ensure quality materials and fabrication at the vendors' plants. Many of these were "resident" inspectors who made daily inspections at vendor plants and were empowered to interpret the engineering specifications to the vendors. Other inspectors, based at Hanford, made numerous trips to vendors for in-process and preshipment inspections. A system of inspection records was employed. In the case of instrument systems, for example, skilled instrument engineers were used as inspectors, and such contacts with vendors often resulted in equipment design improvements being made during fabrication.

5. Project engineers at N Reactor closely followed the placement, welding, mounting, testing, etc. of all equipment.

6. Acceptance testing included a Title-III engineering inspection (such as checking for proper rotation direction on a pump), followed by functional tests which were performed on the completed individual

systems under instructions described in Acceptance Test Procedures (ATP) and/or Operation Test Procedures (OTP). An example of each would be:

- o ATP No. 2452, Pressure Tube Flow Monitoring System
- o OTP T0017, Secondary Coolant System

In 1969, in light of advancing materials technology, HWS-8732, "Seamless Carbon Steel Pipe for Critical N Reactor Piping Systems," was issued to provide for the use of improved materials in the maintenance and repair of the reactor pressure systems. The specification included requirements for seamless pipe in what Section III of the code currently identifies as Class 1 systems. Adoption of this specification required that upgraded materials with improved properties be used in repair or replacement work in the plant. Quality requirements imposed by this specification exceeded the industry code requirements then in effect and in fact compare favorably with current codes. Repairs or modifications also remained subject to the provisions of HWS-6897. 21

The general result of the quality assurance methods described above was to obtain systems, components, and structures that equaled or exceeded the state-of-the-art at that time. These efforts also advanced the state of the art in several areas: notably, carbon steel piping quality, electronic instruments, and welding techniques. The materials, procurement, fabrication and erection of the RCS and associated pressure system components, under HWS-6897, compares very closely with present day code Section III provisions. The quality was reviewed and attested to in 1976 by ERDA-RL consultants (Ref. 3).

For purposes of quality control over design changes, component replacements and maintenance activities, the systems were classified as "critical" (C) or "noncritical" (NC). Further classifications of the degree of complexity and

uniqueness were established for maintenance work on critical systems, as follows:

1. C-1 Complex and nonroutine
2. C-2 Complex and routine
3. C-3 Simple tasks with no significant chance of error

Chapter 17 describes the quality assurance controls and evaluations that are applied to activities affecting the critical systems. These implement DOE Order 5700.6B, Quality Assurance. A list of the critical systems is given in Section 17.2.2.2 and includes 39 of the total 58 systems. In assigning the classifications, the conservative approach was taken that a system would be classed as critical if failure of any part could adversely affect nuclear safety, personnel safety, reactor operating reliability, long-term operability of the reactor, or environmental safety. Provisions were made within the quality assurance program for lowering the classification of a noncritical component within a critical system in accordance with guidance given in the control manual, WHC-CM-4.28.

3.2.2.2 Reclassification Program

Upgrading and reclassification of N Reactor began within a year following attainment of full power operation. In March 1965 the AEC requested production reactors to determine necessary plant modifications to comply with licensed power reactor safety criteria. Over the next year and a half, proposals for improvement in waste management, radioactive effluent control and control of environmental consequences of the releases of radioactive materials were submitted to the AEC DRL.

The effluent control program was favorably accepted by the DRL and ACRS in 1968 as being necessary to attain "Licensability" status. As part of this program, long-term upgrading of the ECCS system was initiated. In 1969 ECCS supply tank bypass lines and lines connecting the demineralized water and filtered water storage tanks to the emergency cooling high-lift pump suction were installed. A new ECCS line, between the high-lift pumps and the reactor ECCS supply manifold in 105-N was installed in 1970. Also, by 1970 the DOE

was supporting a concentrated effort at N Reactor on "Comparabililty Studies" and an organized requirements study program was in place with teams of key engineers assigned to key safety-related systems.

By the late 1970's, the existing sections of the ASME B&PV code had materially changed (Section III had been expanded to address nuclear power plants and Section XI Inservice Inspection and Testing added). Also the NRC had issued Regulatory Guide 1.26 which established quality group classifications for nuclear reactors which were generally related to ASME Section III code classifications and the related requirements for inservice inspection. Programs were initiated by the NRC and the DOE to review the older operating plants for changes that had occured in regulatory and code requirements and reconcile any issues which could affect safety.

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In 1977, for the older vintage commercial plants, the NRC established the Systematic Evaluation Program to address a large number of generic and plant specific safety concerns. Many of these issues, and later, issues which arose from the TMI accident involved a consideration of the original design criteria versus the current Regulatory and Codes and Standards requirements. As part of this review process, the Franklin Research Center performed for the NRC a generic review and comparison of the codes and standards in effect during the mid 1950s and early 1960s with the 1977 ASME Boiler and Pressure Vessel (B&PV) Code as supplemented through the Summer 1978 Addenda (Ref. 4-6). The purpose of the review was to identify any significant differences in codes that could affect structural integrity, and to assess the effect of these differences on the safety margins of the systems and components.

The following areas were identified where the requirements had changed:

1. Fracture Toughness
2. Quality Group Classification
3. Code Stress Limits
4. Radiography Requirements
5. Fatigue Analysis of Piping Systems.

Of these it was further determined that changes in areas 2, 3, and 5 had not significantly affected the safety functions of the systems and components reviewed. It was recommended that reanalysis efforts be directed to the areas of Fracture Toughness and Radiography Requirements.

At N Reactor these two areas in code differences did not become a major safety issue because HWS-6897, the specification for primary piping in buildings 105 and 109, and supplemented by the special Section N-1 through N-21 standards, had imposed stringent requirements. In fact, as the plant was going into operation it was determined that a number of the V-notch Charpy tests for fracture toughness on the ECCS piping did not meet specifications. In order to assure that required margins of safety would be maintained, a program to temper the ECCS piping was established with AEC DRL concurrence, and later, as discussed above, in 1968 under the N Reactor Effluent Control Project GCP 411 (Ref. 7) a long-term program to replace all high-pressure ECCS piping was initiated (low-pressure piping to the ECCS high-lift and FS pumps was determined to meet SSE criteria and verified again in the 1987-88 seismic reanalysis program). The final replacements under this program were completed during the 1987 shutdown.

In the area of equipment and component classification, Regulatory Guide 1.26 had established "quality groups" for determining acceptable quality standards for safety-related fluid-containing components (i.e., pressure class boundaries). Components were assigned to quality groups, which were in turn related to ASME Section III established equipment code class requirements. The code recognized different levels of importance associated with the function of each item and allows a choice of rules that provide assurance of structural integrity and qualilty commensurate with the relative importance assigned. The relation between the ASME construction code

classifications and the NRC quality group classifications in accordance with 10 CFR 50 and Reg. Guide 1.26 is as follows:

Quality group	Construction Code Class
A	1
B	2
C	3
D	(B31.1 - Section VIII)

For example, the reactor coolant system (RCS) would be classified as a Qualilty Group A (Class 1 - Sect. III) system, which if it were under construction, would require ASME Section III requirements, and in addition, the requirements of Section XI "Rules for In Service Inspection and Testing" would be invoked. Section XI also addresses the applicability of revised and updated code editions to existing plants. Z1

For N Reactor, the 1977 Edition - W79 addendum of Division I of Section XI was adopted, with DOE concurrence as the basis for reclassifing the systems and components of the N Reactor. Reclassification efforts to provide a tabulation of pressure systems and components versus quality groups were initiated in 1977 (Ref. 8). As plant upgrades and modifications were performed to the systems and structures, the code classifications were also periodically re-evaluated, most recently in 1988, (Ref. 9 and 10). Additional discussion is provided in section 3.9.6 where Tables 3.9-1 through 3.9-4 list the systems and components that have been assigned to each of the Quality Groups A, B, C, and D.

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
1. REACTOR SYSTEM				
.1 Fuel Assemblies including fuel element support system	X	X	X	X
.2 Control rod assemblies	X	X		
.3 Control rod drives	X		X	X
2. REACTOR COOLANT SYSTEM				
.1 Primary Coolant pumps (6) includes shaft penetration seals	X	X	X*	
.2 24" pipe from pump discharge and 24" check valve PCSV-202 to cell wall (6)	X	X	X*	
.3 24" pipes (6) from cells to 24" supply headers (2) one each side	X	X	X*	
.4 24" supply headers (2)	X	X	X*	
.5 24" PCSV-206 diversion valves (2)	X	X	X*	
.6 18" inlet risers, crossovers and downcomers including check valves CV-5 (16)	X	X	X	
.7 4" extensions including check valves CV-2R, flushing valves V-9 and shutoff valves V-1 to 8" HPFL manifold and ECCS connectors (16)	X	X	X	

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*Items to be qualified as part of return-to-service (RTS) or long-term-service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
2. REACTOR COOLANT SYSTEM (cont.)				
.8 2.625" OD connector piping including control valves V-11s and venturis (54-66 max. per each riser)	X	X	X	
.9 3.25" OD nom. pressure tubes, caps, connectors and nozzles (1003)	X	X	X	
.10 2.625" outlet connector piping to 18"outlet risers (16)	X	X	X	
.11 6" extension from 18" outlet riser to 10" HPFL manifold (purity blowdown) including angle valves V-14, V-26 and BDV-201 valves, throttling valves V-17 and V-8 flush valves, V-2 maintenance valves (16)	X	X	X	
.12 18" outlet crossovers and downcomers through check valves CV-3 (16) to 24" outlet header	X	X	X	
.13 8" extension from outlet downcomers (16) to 10" HPFL manifold connected to 16" discharge lines (2) through 8 10" dump valves V-4 and 2 4" valves RV 2-1 and RV 2-2 and 3-way valves V-23 (2)	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
2. REACTOR COOLANT SYSTEM (cont.)				
.14 26" pipe from header to cells 1-6 including PSCV-202 valves in tie-line to 24" supply	X	X	X*	
.15 18" pipes from 24" lines to steam generators (2 per cell) through remote control valves PCRV-203	X	X	X*	
.16 Steam generators (12)	X	X	X*	
.17 18" bypass pipe including normally closed valve PCRV-204 to 26" pipe at pump suction (6)	X	X	X*	
.18 16" tieline and valve PCSV-203 26" outlet and 24" supply lines	X	X	X*	
.19 6" pipe and valve IWV-252 to injection makeup system	X	X	X*	
.20 6" pipe and valve PCSV-210 to decon flush system	X	X	X*	
.21 6" pipe and valves PCRV-206 and PCRV-214 to decon feed and flush	X	X	X*	
.22 6" pipe and valve PCRV-210 to sump	X	X	X*	

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*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
2. REACTOR COOLANT SYSTEM (cont.)				
.23 Pressurizer vessel, 10" surge line from 26" pipe including block valves PRV-205 (2)	X	X	X*	
.24 4" spray line from 24" supply pipe with control valves PRV-202 (2) and block valves PRV-207 (4)	X	X	X*	
.25 2" pipe and valve PRV-215 to decon and PRV-212 to floor drain	X	X	X*	
.26 2" pipe and PRV-210 vent and RVTV-202 relief valves and injection water valve PRV-219	X	X	X*	
.27 RCS Radioactive Drain Piping	X	X	X*	
3. SECONDARY COOLING SYSTEM				
.1 Main Steam Restrictors	N/A			
.2 Main steam line isolation	N/A			
.3 Turbine Generator	N/A			
.4 Dump Condenser	N/A			
.5 Afterheat Removal (met with non-safety water-water mode)	N/A			

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
4. EMERGENCY CORE COOLING SYSTEM				
.1 Intake structure and piping	X	X	X	
.2 Low-lift pumps, diesel drivers, 2 @ 35,000 gpm, 1 @ 20,000 gpm 30" discharge pipes through check valves RWV 504 (2) and check valves RWV 518, RWV 519	X	X	X	
.3 30" discharge pipes through check valves RWV 504 (2) and check valves RWV 518, RWV 519	X	X	X	
.4 6" diesel cooling pipe through valves RWV-509 (2) and RWV-515	X	X	X	
.5 2 30" pipes to ECCS bypass(silo) and pipe to RWS storage tank	X	X	X	
.6 ECCS bypass (Silo) and overflow pipe	X	X	X	
.7 DW storage tank including:				
a. 12" DW-3 overflow & 6"branch pipe	X	X	X	
b. 6" IW-3 supply line	X	X	X	
c. 4"-MP, 4"-MPr and 3" nozzle connection	X	X	X	
d. 18" DW-1 line, 14"and 8" branches up to DWV-804 and DWV-805	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
4. EMERGENCY CORE COOLING SYSTEM (cont.)				
e. 12" DW-11 and DWV-801-2 and 10" branch up to DWV-812-1	X	X	X	
f. 1-1/2" lines nozzle connections	X	X	X	
.8 ECCS 36" and 42" pipes, valve DWV-816, check valve DWV-820 from DW tank to 182-N HL pumps (3)	X	X	X	
.9 42" suction header through valves RWSV 552-5R,6R,7R to High Lift Pumps (3) @ 13,750 gpm and diesel drivers including FWV-817.	X	X	X	
.10 Pump discharge pipe, RWS-620 valve, check valve RWSV 555-1 6" (and smaller) bypass piping through valves RWSV-577-1,2,3,4,5,6 and tees on 6" pipe RWSV-577-7	X	X	X	
.11 High-lift pump test line, RWRV-554-1R valve and pipe to ERW tank	X	X	X	
.12 30" supply line 182-N to 109-N roof, 30"-20" tee reducer to 20" vertical down to cor. 5 (2), 20"-16" reducers and 16" supply headers (2)	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
4. EMERGENCY CORE COOLING SYSTEM (cont.)				
.13 36" underground reserve line, valve RWSV 619-1, 36"-30" reducer, 30" pipe, 30"-24" reducer, pipe to 16"/24" manifold including 2" bypass line with valve V62-3	X	X	X	
.14 8" distribution pipes from 16"/24" Corr. 5 manifold, 8" V-3 valves, and 8" CV-1 check valves (8 ea)	X	X	X(air control valve)	
.15 24" bounded by valves CV-61-1, V 62-2 on East, and 16" V-19, 1" V-15, and relief valve RV-1 on West	X	X	X	
.16 6" FL(LP) pipe and flush valves V-18 (2)	X	X	X	
.17 4" supply pipes from 8" headers 18" RCS inlet risers with valves 4" V-1, 1.5" V-9, 4" CV-2 and CV-2R (16 ea)	X	X	X	
.18 18" RCS inlet risers, 18" CV-5 check valves and 1" V-38 vent valves (16 ea)	X	X	X	
.19 2 5/8" connectors, V-11 valves, venturi flow meters, Graylok connectors and pressure tube inlet nozzles (1003 ea)	X	X	X	
.20 Pressure tubes (1003)	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
4. EMERGENCY CORE COOLING SYSTEM (cont.)				
.21 2 5/8" pressure tube outlet nozzles, Graylok connectors, outlet piping (1003) to 18" outlet RCS downcomers	X	X	X	
.22 18" outlet downcomers and piping to the CV-3 check valves, 6" V-2 valves, drain valves V-8 and vent valves V-21 and V-33 (16 ea)	X	X	X	
.23 6" FL(HP) pipes (16) to 10" FL(HP) headers (2), the 4" V-43 valves (2) and RV2-1,2 relief valves (2)	X	X	X	
.24 10" FL(HP) headers (2), 16" flush valves V-5 (2), blocking valve V-36 and 6" PBD pipes to V-14 (2)	X	X	X	
.25 Fuel Rupture Monitor piping with valves RMRV13-7 and RMRV4-7	X	X	X	
.26 10" dump valves V-4 and piping (8)	X	X	X	X
.27 16" FL(HP) pipe, valve V-23 (2) and check valve CV-14, exp. joints (2), 16"-24" reducers, 24" drain pipe, 24" FL(HP) pipe, penetrations, 24"-30" reducer, 30" pipe to EDT and 30" sparger.	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
4. EMERGENCY CORE COOLING SYSTEM (cont.)				
.28 Emergency dump tank (1304-N)	X	X	X	
.29 EDT, FLV-858R inlet valve, FLV-860 and FLV-802-1 outlet valve and tank vent pipe to 109-N	X	X	X	
.30 EDT overflow pipe to vent header, upstream of 1312-N liquid effluent retention facility (LERF)	X	X	X	
.31 Branch pipe from zone 1 drain and 105-N lift station connecting to EDT overflow pipe	X	X	X	
.32 Emergency Core Cooling System actuation sensors, sensing lines, transmitters, relays and actuation circuits	X	X	X	
5. CONFINEMENT SYSTEM				
.1 Steam vent assemblies	X	X	X	
.2 Penetrations	X	X*	X*	
.3 Ventilation system confinement valves, hydraulic system local accumulators, pipes and valves	X	X*	X*	

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
5. CONFINEMENT SYSTEM (cont.)				
.4 Ventilation and confinement backup valves and gates	X	X*	X*	
.5 Reactor moderator gas system	X	X*	X*	
.6 Emergency drain system including pipe, valves,pump, diesel drive, diesel fuel supply, jacket cooling, starting air system and controls	X	X*	X*	
.7 Confinement filter frames	X	X	X	
.8 Fog Sprays: 105-N and 109-N Cells	X	X*	X*	
.9 Vacuum relief valves	X	X	X	
.10 Filter sprays	X	X	X	
.11 Fog spray line in valve pit from Bldg 182-N to circ. water pump and check valve RWSV-551	X	X	X	
.12 Confinement sensors, sensing lines, transmitters, pressure switches, timers, relays and circuits for actuation of confinement sequence, fog sprays and control room isolation	X	X	X	

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
5. CONFINEMENT SYSTEM (cont.)				
.13 Filter temperature switches, circuits for actuation of filter sprays	X	X*	X*	
6. GRAPHITE AND SHIELD COOLANT SYSTEM (GSCS) Category I (Backup Cooling)				
.1 Cooling tubes in reactor moderator and shields	X	X	X	
.2 GSCS pipes, and valves (RWV-234-1,2) for shield coolant supply from RWS-2	X	X*	X*	
.3 12" piping graphite supply, (RWS-2)- 109N Aux.	X	X*	X*	
.4 GCSV 316 - 1 & 2 valves and operators - 109N Aux.	X	X*	X*	
.5 12" GCS supply line - 109N Gallery	X	X*	X*	
.6 12" GCS supply & return 105N rear pipe space.	X	X*	X*	
.7 12" GCS dump 105N Pipe, orifice, and pipe in trench to lift station effluent pipe	X	X*	X*	
.8 8" GCS supply header to 10-2 1/2" pipe, valves GCSV-4, 10 - 4" pipe GCS supply - 105-N front pipe space	X	X*	X*	

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*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
6. GRAPHITE AND SHIELD COOLANT SYSTEM (GSCS) Category I (Backup Cooling) (cont.)				
.9 3/4" connector tubes - 105-N right inner rod room.	X	X*	X*	
.10 3/4" connector tubes - 105-N left inner rod room.	X	X*	X*	
.11 GSCS sensors, circuits and cabinets for actuation of once-through emergency cooling of moderator and shields	X	X*	X*	
7. RWS-2 BACKUP SUPPLY SYSTEM - CATEGORY I				
.1 Branch 30" pipes from low lift pumps to 182-N fog spray 24" suction header	X	X	X	
.2 DW tank, RWSV-810 check valves and pipes to the low-lift pump discharge pipes	X	X	X	
.3 Branch pipe from CRW system check valve RWSV-555-1 fog spray suction header	X	X	X	
.4 Fog spray ECCS isolation valve, RWV-801	X	X	X	

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
7. RWS-2 BACKUP SUPPLY SYSTEM - CATEGORY I (cont.)				
.5 Fog spray suction header to sectionalizing valve and isolation valve RWSV-552-1, between ECCS suction header and fog spray suction header	X	X	X	
.6 Remainder of suction header, pipes and valves to fog spray pumps suction nozzles and bounding valves RWSV-570-1, 555-1R, 596-1	X	X	X	
.7 Fog spray pumps and diesel drives (2) 182-N	X	X	X	
.8 Fog spray pump discharge pipes, valves, manifold, and bounding pipe and valves in 182-N; RWSV-551-1,2 and RWSV-669-1,2,3	X	X*	X*	
.9 Supply pipe from 182-N to 109-N	X	X*	X*	
.10 Supply pipe in 109-N pipe gallery and 105-N with distribution pipes and valves to fog spray networks	X	X*	X*	
.11 Fog spray networks in 109-N pipe gallery and 105-N.	X	X*	X*	
.12 Demineralized water tank piping.	X	X	X	

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
8. HIGH PRESSURE INJECTION (HPI) SYSTEM				
.1 Pipe from demineralized water tank to HPI pumps in 182-N	X	X	X*	
.2 HPI pumps, check valves, valves and pipe to 109-N	X	X	X*	
.3 HPI control valves, check valves and piping to reactor coolant system and RCS pump seals in 109-N	X	X	X*	
9. CONTROL ROOM HABITABILITY CATEGORY I				
.1 Equipment of emergency air handling unit, air cooled condensing unit, centrifugal fans, and roll filter	X	X		
.2 Control room emergency filter unit	X			X

*Items to be qualified as part of return-to-service (RTS) or long-term service (LTS) programs.

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
9. CONTROL ROOM HABITABILITY CATEGORY I (cont.)				
.3 Motor-operated butterfly valve dampers:				
a. all valve assemblies	X			
b. valve operators	X			X
c. active valves	X			X
.4 Standby diesel generator				
a. diesel generator and its associated components	X	X		X
b. skid-mounted electrical equipment, controls, panels, active equipment	X	X		X
.5 Low voltage motor control center and general purpose dry type transformers	X			X
.6 Control Room panel X-X insert	X			
.7 Annunciator, indicators, recorders, and miscellaneous control devices	X			X
.8 Toxic Gas (chlorine, ammonia) monitors	X			X
.9 Control room radiation monitoring system	X			
.10 Blast doors and embedments with hatch outside of control room and room 6	X			

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
9. CONTROL ROOM HABITABILITY CATEGORY I (cont.)				
.11 Corridor 7 wall	X			
.12 Hangers for ductwork	X			
10. AUXILIARY MECHANICAL SYSTEMS				
.1 Diesel Oil Supply Systems				
a. Supply piping from 166 oil storage	X	X	X	
b. 182 diesel oil day tanks (3)	X	X	X	
c. Oil piping 182 to 181	X	X	X	
d. Oil piping inside 181	X	X	X	
e. Oil piping inside 182	X	X	X	
f. Engine-mounted equipment (instrumentation, tachometer, dc power, starting air solenoids)	X	X	X	
.2 Low-Lift Diesel Jacket Water Systems				
a. Engine-driven jacket water pump (3)	X	X	X	
b. Jacket water expansion tank (3)	X	X	X	
c. Lube oil cooler (3)	X	X	X	
d. Jacket water cooler (3)	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
e. Jacket water 4" and smaller piping and valves from exhaust manifolds and JW pumps to valves JWV-501, 502 and 503 (3)	X	X	X	
.3 Low-Lift Diesel Lube Oil Systems				
a. Engine-Driven lube oil pump (3)	X	X	X	
b. Multibasket oil strainer (3)	X	X	X	
c. Lube oil cooler (3)	X	X	X	
d. Lube oil filter (3)	X	X	X	
e. Lube oil 3" and smaller piping from engine ports to LOV 504-2 transfer pumps, LOV 509-5 low pressure air, LOV 516-2 LO storage tank, and LO drains and drain valves (3)	X	X	X	
.4 Low-Lift Diesel Jacket Water Cooling system				
a. Jacket water cooler (3)	X	X	X	
b. Gear drive oil cooler (3)	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
c. Raw Cooling water supply 6" and smaller piping from floor penetrations pump discharges to JW and GO coolers, vent and drain valves	X	X	X	
.5 Low-Lift Diesel Starting Air Systems				
a. Engine-mounted equipment (3)	X	X	X	
b. Accumulater (3)	X	X	X	
c. Starting air valve (3)	X	X	X	
d. Pilot solenoid valve (3)	X	X	X	
e. Air filter (3)	X	X	X	
f. Air supply 1.5" and smaller piping and valves from manifolds to instruments, air bottles and accumulators	X	X	X	
.6 Diesel Air Intake and Exhaust Systems				
a. Intake air 16" pipe from intake filters and silencers to engine intake manifolds (3)	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
b. Exhaust air 14" piping from engine exhaust manifolds to atmosphere (3)	X	X	X	
.7 High-Lift Diesel Jacket Water Systems				
a. Engine-driven jacket water pump (3)	X	X	X	
b. Jacket water expansion tank (3)	X	X	X	
c. Lube oil cooler (3)	X	X	X	
d. Jacket water cooler (3)	X	X	X	
e. Jacket water 4" (and smaller) piping and valves bounded by JWV-533-6, 553-9, 553-15	X	X	X	
.8 High-Lift Diesel Lubes Oil systems				
a. Engine-driven lube oil pump (3)	X	X	X	
b. Multibasket oil strainer (3)	X	X	X	
c. Lube oil cooler (3)	X	X	X	
d. Lube oil filter (3)	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
e. Lube oil 4"(and smaller) piping and valves bounded by LO filters and sub system LO-6 (FS) connector H-1-51044 (includes LO for 2 FS diesels)	X	X	X	
f. LO 3/4" supply piping (and smaller) from valves LOV-558-1,2,3,4,5 to HLP diesels 1,2,3 and FS diesels 1,2	X	X	X	
g. LO drain piping (1" and smaller) from each engine (5) and MOV-552	X	X	X	
.9 High-Lift Diesel Jacket Water Cooling Systems				
a. Piping from low-lift diesel disc.	X	X	X	
b. Piping from low-pressure raw water pumps (3)	X	X	X	
c. Jacket water cooler (3)	X	X	X	X
d. Gear drive oil cooler (3)	X	X	X	X
e. Cooling water 4" and smaller piping and valves, bounded by Hi Lift JW coolers, FS JW coolers and valves RWSV-627-1, 628-1 and 575-1	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
.10 High-Lift Diesel Starting Air System				
a. Engine-mounted equipment (3)	X	X	X	
b. Accumlater (3)	X	X	X	
c. Starting air valve (3)	X	X	X	X
d. Pilot solenoid valve (3)	X	X	X	X
e. Air filter (3)	X	X	X	
f. Air supply piping and valves (3)	X	X	X	
.11 Diesel Air Intake and Exhaust System				
a. Intake air 16" pipe from filters and silencers to intake manifolds (3)	X	X	X	
b. Exhaust air 14" pipe from exhaust manifolds to atmosphere	X	X	X	
.12 Fog Spray Diesels Starting Air System				
a. Engine mounted equipment (2)	X	X	X	
b. Accumulator (2)	X	X	X	
c. Jacket water cooler (2)	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
d. Pilot solenoid valve (2)	X	X	X	
e. Air filter (2)	X	X	X	
f. Air supply piping and valves (2)	X	X	X	
.13 Fog Spray Diesels Air Intake and Exhaust				
a. Intake air 16" pipe from filters, silencers to intake manifolds (2)	X	X	X	
b. Exhaust air 14" pipe from exhaust manifolds to atmosphere (2)	X	X	X	
.14 V-3 Valve Operating System				
a. Operating cylinder	X		X	
b. Pressure tubing	X		X	
c. Solenoid valves	X		X	
d. Check valves	X		X	
e. Gas accumulators	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
10. AUXILIARY MECHANICAL SYSTEMS (cont.)				
.15 V-4 Valve Operating System				
a. Operating cylinder	X		X	X
b. Pressure tubing	X		X	
c. Solenoid Valves	X		X	
d. Check valves	X		X	
e. Gas accumulators	X	X	X	
.16 Diesel Fire Protection Spray System	X	X	X	
11. HORIZONTAL CONTROL ROD SYSTEM - CATEGORY I				
.1 Sensors, relays, logic and actuation circuits for the horizontal control rod safety	X	X	X	
.2 Rod assemblies, drive mechanisms, rod racks and actuators	X(Rod racks)	X	X	X(Rods and drives)
.3 Rod scram hydraulic oil system from accumulators through solenoid valves to rod drives	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
12. BALL SAFETY SYSTEM - CATEGORY I				
.1 Hopper assemblies, including holding coils and tripping mechanisms	X	X	X	
.2 Step plugs and ball channel blocks part of moderator	X	X	X	
.3 Ball drain piping to and including the normally closed drain valves	X	X	X	X (Drain valves)
.4 Ball recovery drain piping and skip hoist	X	X	X	
.5 Seismoscope assemblies and mounting	X	X	X	
.6 Sensors, relays and logic circuits	X	X	X	
13. PROTECTIVE INSTRUMENTATION				
.1 Sensors, sensing lines, transmitters, relays and circuits that actuate post-scram action functions listed in Chapter 7	X	X	X	X (Hg sw & relays)
14. PROTECTIVE ELECTRICAL SUPPLY SYSTEMS - CATEGORY I				
.1 Battery chargers	X	X	X	
.2 125 Vdc battery systems in 105-N, 181-N and 182-N excluding battery chargers	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
14. PROTECTIVE ELECTRICAL SUPPLY SYSTEMS - CATEGORY I				
.3 125 Vdc/120 Vac inverters and associated cabinets and trays	X	X	X	
.4 125 Vdc breakers, cabinets and wiring trays for circuits that actuate ECCS, confinement, GSCS backup cooling, filter cooling, and fog spray	X	X	X	
.5 Panel DA1 (109-N) 125 Vdc (BAT. A)	X	X	X	
.6 Breaker No. 5 - Relay and term CAB "S1" (Hi lift Diesel #1 182-N)	X	X	X	
.7 Breaker No. 8 - Heat & Vent Cont. Pnl. (Doors B9 through B12)	X	X	X	
.8 Panel DB1 (109N) 125 Vdc (BAT.B)	X	X	X	
.9 Breaker No. 5 - Relay & Term Cab "S1" (Hi lift Diesel #2, 182-N)	X	X	X	
.10 Breaker No. 8 - Heat & Vent Cont. Pnl (doors B8 & B103)	X	X	X	
.11 Portable emergency lights located in: 105-N Control Room, 105N Room 6, 181-N Emergency Pump House, 181-N Annex, and 182N.	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
14. PROTECTIVE ELECTRICAL SUPPLY SYSTEMS - CATEGORY I (cont.)				
.12 Cable Spreading Room 6 Equipment:	X	X	X	
a. Batteries (two racks)	X	X	X	
b. Relay cabinets (PR 21, 22, 26, 35, 45, 46, 47, 48, 49, 53)	X	X	X	
c. Relay cabinets (PR 8, 9, 10, 11, 12, 13, 23, 25, 29, 34, 35, 44)	X	X	X	
d. Relay cabinets (PR 19, 20, 30, 31, 32, 33, 36, 37, 38, 39, 40, 54),	X	X	X	
e. Relay cabinet (PR 27)	X	X	X	
f. Inverter, 10kVA	X	X	X	
g. Relay Cabinets (PR 14, 41, 42, 43, 61, 62, 65, 66),	X	X	X	
h. Relay Cabinets (PR 16, 17, 18, 25, 34, 55, 59, 60),	X	X	X	
i. Power supply, Flow monitor (G1, G2, G3)	X	X	X	
j. Flow Monitor Power Supply (PNL D),	X	X	X	
k. Flow Monitor Terminal Cabinets,	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
14. PROTECTIVE ELECTRICAL SUPPLY SYSTEMS - CATEGORY I (cont.)				
l. Power Supply (-10 Control Unit)	X	X	X	
m. Power Distribution Racks	X	X	X	
n. Bailey Model 720 (040, 041, 042, 043, 067, 068,069, 086, 087, 089, 090, 092, 093, 094, 095, 096, 108, 119, 475),	X	X	X	
o. Bailey Model F20 (061, 062)	X	X	X	
p. Bailey Model F20 (430, 431, 437, 438)	X	X	X	
q. Invertors, 1.5 kVA (A & B)	X	X	X	
r. B & R Relay Cabinets (R1, R2, R3, R4, S1, S2)	X	X	X	
s. 105N Zone 1 Exhaust Relay Cabinet	X	X	X	
t. Relay Cabinets (R50, 51)	X	X	X	
u. B & R Relay Cabinet (W1)	X	X	X	
v. Cable Trays	X	X	X	

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TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
15. STRUCTURES CATEGORY I				
.1 Reactor structure including: graphite moderator, thermal shield, primary radiation shield, moderator supports, bottom insulation layer and foundation	X	X	X	
.2 105-N and 109-N confinement building foundations, walls, roofs and tunnels that enclose seismic Cat. I systems and components	X	X	X	
.3 105-N control room suspended ceiling/room 6 - H&V duct and equipment, lighting fixtures, fire extinguishing pipe and equipment	X	X	X	
.4 Confinement exhaust tunnel to filters and ventilation stack	X	X	X	
.5 116-N Ventilation stack (chimney)	X	X	X	
.6 181-N main structure and ECCS low lift pump house	X	X	X	
.7 181-N gantry crane	X	X	X	
.8 182-N ECCS high-lift pump house	X	X	X	
.9 105-N irradiated fuel storage facility	X	X	X	
.10 117-N filter house	X	X	X	

TABLE 3.2-1

STRUCTURES, SYSTEMS AND COMPONENTS

CATEGORY I

System/Components (Shared Components Are Duplicated)	Validation Method: Static Analysis	Dynamic Analysis	III/I Program	Seismic Testing
15. STRUCTURES CATEGORY I (cont.)				
.11 Standpipe	X	X	X	
.12 1316-N Valve box	X	X	X	

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3.3 Wind, Tornado and Volcanic Ashfall Loadings

Wind pressures used in the design of N Reactor plant are described in Section 3.3.1. Tornado and volcanic ashfall loadings were not considered in the original design. However, the anticipated behavior of N Reactor and critical plant structures, when subjected to the Hanford design tornado, is discussed in Section 3.3.2. The effects of tornado-generated missiles are discussed separately in Section 3.3.3 and information on volcanic ashfall is provided in Section 3.3.4. Figure 3.3-1 provides a plan view of N Reactor and locates the critical facilities required for safe shutdown of the reactor. 13 21

3.3.1 Normal Wind Loadings

Design criteria for N Reactor specified values of wind pressures to be used in the design calculations and not wind velocities. Values of wind pressure used in the design and corresponding wind velocities are discussed below.

3.3.1.1 Design Wind Velocity

The design wind, based on wind pressures used in the design of N Reactor, has a gust velocity of 78 mph and a recurrence interval of about 25 years. The magnitude and vertical velocity distribution of the design wind are described in Section 3.3.1.2 below.

3.3.1.2 Design Wind Pressures

A peak wind (gust) velocity of 72 mph was recorded at the 50-ft level by the Hanford, 200-West Meteorological Tower on May 26, 1948 and again on June 6, 1957. This gust velocity was the greatest that had been experienced for this elevation up to the time that N Reactor was built. A gust velocity of 80 mph was subsequently recorded at the 50-ft level on January 11, 1972. This extreme wind did not damage N Reactor buildings and equipment or the offsite power source. Based on Hanford wind records, a gust velocity of 80 mph has an average recurrence span of about 33 years. See Figure 2.3-3. This recorded wind velocity represents the peak gust velocity, so that a gust factor of 1.0 applies. Additional information on severe winds experienced at Hanford is presented in Section 2.3 on meteorology.

Wind pressures specified in the original design criteria were as follows:

Vertical Location (ft above ground level)	Wind Pressure (lb/ft^2)	Equivalent Gust Velocity (mph)*
Under 20	15	70.7
20 to 50	20	77.8
50 to 100	25	87.0
Over 100	30	95.3
		*C_D = 1.3

*See Section 3.3.1.3.

These values of wind pressure conform to requirements presented in: (1) Sectio SDC-4.1, "Hanford Standard Design Criteria for Conventional Construction", dated September 19, 1957 (Ref. 2) and in (2) ASA Standard A58.1-1955, "American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures", (Ref. 3) except for pressures 20 to 30 ft above ground level In the latter case, ASA A58.1 specifies a pressure of 15 lb/ft^2, whereas, SDC-4.1 specifies a more conservative value of 20 lb/ft^2.

The above wind pressures represent minimum values required to satisfy design criteria. In a number of cases, the designer chose to use more conservative values. For example, the structural steel framework and walls of the 105-N building metal storage and transfer area (which are less than 50 ft high) were designed for 30 lb/ft^2. The pressurizer penthouse of the 109-N building was also designed for 30 lb/ft^2, although its roof is less than 85 ft above the ground. In a number of cases, earthquake design requirements are more severe than wind effects and consequently controlled the design for lateral forces. Additional details on the effect of severe winds on plant buildings are present in Section 3.3.2.

The 116-N ventilation stack, which is 200 ft high, was designed for the following wind pressures applied to its projected area:

Vertical Location (ft above ground level)	Wind Pressure (lb/ft^2)	Equivalent Gust Velocity (mph)*
0 to 50	20	104
50 to 100	24	114
100 to 200	28	123

*CD = 0.6

*See Section 3.3.1.3.

These wind pressures are specified in ACI Standard 505-54, "Specification for the Design and Construction of Reinforced Concrete Chimneys", and dated February 23, 1954. They are apparently based on a peak velocity of 100 mph at 30 ft above ground and a gust factor of 1.1 (Ref. 4). Based on Hanford Site wind records, this extreme wind has a recurrence interval of about 1,800 years. See Figure 2.3-3.

Lateral forces acting on the ventilation stack, due to the above wind pressures, are more severe than the earthquake forces specified in the design criteria (Ref. 5).

3.3.1.3 Determination of Applied Forces

N Reactor design criteria specified only lateral wind pressures at various elevations and not design wind speeds. In accordance with procedures presented in NUREG 0800, values of peak wind velocity (V mph), corresponding to the various design wind pressures (p), can be calculated from the following formula: | 21

$$p = q \times C_D = q \times 1.3 = 0.0033\ V_2\ lb/ft^2,$$

Where the dynamic wind pressure (q) is equal to 0.002558 V_2 (Ref. 6), CD is a shape or drag coefficient and equals about 1.3 for a rectangular-shaped building and 0.6 for a vertical cylinder. Design pressure (p) is the total horizontal pressure acting on the structure. In case of a typical, rectangular-shaped building, it consists of 0.8 q acting on the windward wall plus a suction pressure of 0.5 q on the leeward wall. As was standard practice at the time, exterior walls and interior steel framing were designed for the total horizontal wind pressure (p = 1.3 q). In addition, walls and roofs of the 105-N building

were designed for the following (internal) confinement pressures. See Section 3.8.1.3 and Reference 7.

	Internal Design Pressures for Confinement Structure	
	Positive Pressure lb/ft^2	Partial Vacuum lb/ft^2
Zone I	+ 720	288
Zone II*	+ 72.0	14.4
Zone III**	+ 28.8	14.4

*Includes supply and exhaust fan rooms, but not the decontamination room.
**Includes the decontamination room.

Confinement pressures were assumed not to act during periods of maximum wind. Uplift forces, due to wind passing over a rectangular building, are usually around 0.5 q at a specified elevation (Ref. 6). Thus, uplift pressures on the various roofs, due to positive confinement pressures, substantially exceed those that might be produced by wind forces.

3.3.2 Tornado Wind Loadings

Tornado wind loads were not a consideration in the original plant design. However, by 1974, in response to DOE orders, the Hanford Standard Design Criteria were modified by revision 7 to include tornado effects (Ref. 8). These criteria applied to all new construction or modifications on the Hanford reservation. Since then, in response to changing requirements, the tornado requirements have been updated three times [revision 8, January 1984 (Ref. 22), revision 9, February 1985 (Ref. 27) and revision 10, July 1988 (Ref. 28)]. The latest, most current tornado criteria (extracted from SDC 4.1 Rev. 10) are summarized in Section 3.3.2.1. It should be noted that the N Reactor design criteria specify that the tornado does not occur during the SSE nor simultaneously with its peak pressure associated with the DBA, and this is the bases for the tornado impact analyses.

During the period from 1976 to 1978, tornado analyses were performed for critical structures and equipment at the N Reactor (Ref. 1), and evaluations were made using the tornado requirements of SDC 4.1, Rev. 7 (Ref. 8) which included 175-mph winds and various missile considerations. The analyses were conservative in that current tornado maximum winds are 165 mph. In this first, early tornado analysis, it was concluded that the critical structures and equipment at N Reactor were adequately protected against tornados. However, no stress analyses for tornado forces was performed.

In 1986 following the Chernobyl accident, tornado concerns were raised during the design reviews of N Reactor and became commitments in the Accelerated Safety Enhancement Program (ASEP) and part of Task 2D Equipment Qualification in the Westinghouse Hanford Independent Safety Assessment Programs which followed. Analysis of these ASEP and Task 2D tornado issues was conducted in 1987-88 concurrently with the comprehensive and extensive Return to Service (RTS) program (Ref. 59) also established in 1987 to re-evaluate the seismic capabilities at the N Reactor site for assuring safe shutdown in event of an SSE. The analyses utilized the requirements of SDC 4.1, Rev. 9 (Ref. 27). The tornado requirements in SDC 4.1, Rev. 9 are the same as those listed in Section 3.3.2.1 except that the wind values given for non-reactor facilities are somewhat higher, i.e. more conservative. The results of these analyses and evaluations for the effects of the design basis tornado (DBT) and tornado-generated missiles on Seismic Category I structures and equipment are presented below in Sections 3.3.2.2 through 3.3.2.14. In cases where more than one tornado analysis was performed (i.e. 181-N), the results are presented in chronological order in the interests of completeness and traceability. In accordance with the prioritization guidelines for seismic and tornado upgrades (Ref. 50), modifications needed solely for compliance with tornado requirements were not needed prior to reactor restart, and were assigned to the Long Term Service Program (LTS) for completion within 24 months following restart.

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3.3.2.1 The Hanford Design Tornado - Applicable Design Parameters

Category I structures, systems and components (DOE Order 6430.1A Safety Class) are those that perform a function required for nuclear criticality safety or whose failure might result in a significant release of radioactivity, or hazardous or toxic materials as defined by DOE Order 5480.1A, Chapter 11. For nonreactor

facilities, the term Category I is equivalent to the UCRL 15910 Category IV. Included in this category are those systems and components vital for safe shutdown. These structures, systems, and components are to withstand the effect of extreme environmental loads, including Safe Shutdown Earthquake (SSE), Operating Basis Earthquake (OBE), Design Basis Tornado (DBT), Design Basis Ashfall, and Probable Maximum Flood (PMF). Where applicable, abnormal loads such as those generated by a postulated high-energy break shall be considered.

1. Design Basis Tornado

The Design Basis Tornado for the design of Category I facilities is based on a probability of 10^{-7} per year for reactor facilities. For nonreactor facilities the procedures in UCRL 15910 for High Hazard Facilities shall apply.

a. Category I structures, systems, and components shall be designed to withstand the effects of a tornado having the following maximum horizontal wind speed over the full height of the structure without loss of capability to perform their safety functions. Wall and roof pressure coefficients shall be in accordance with ANSI A58.1.

	Reactor Facility	Nonreactor Facility
Total horiz. wind velocity, mph	165	100

b. Category I structures, systems, and components shall be designed to withstand the differential pressure loading that results from the following ambient pressure decrease at the rate indicated, held for one second, and return to ambient at the same rate. Suitable allowances for venting and for structural components shielded from such ambient pressure changes may be made.

	Reactor Facility	Nonreactor Facility
Pressure drop, lb/ft^2	54	20
Rate of pressure change, $lb/ft^2/sec$	25	7

c. The effects of the following tornado-generated missiles on Category I structures, systems, and components shall be analyzed. Protection from penetration and secondary effects of scabbing shall be provided.

Missile	Weight (lb)	End Area (ft2)	Horiz. Veloc. (ft/sec)		Max. Height (ft)
			Reactor	Nonreactor	
Wood Plank 4"x12"x12'	139	0.29	114	88	200
3" Std Steel Pipe, 10' Long	76	0.067*	80	59	100

*Value given is gross cross-sectional area for use in penetration calculations.

NOTE: Vertical missile velocity is 2/3 missile horizontal velocity. The horizontal and vertical velocities should not be vectorially combined.

These DBT criteria values were established by DOE document UCRL - 53526, "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites," prepared by Lawrence Livermore National Laboratory, February 1984. Available from National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161. The document contains a tabulation of detailed tornado design parameters.

These requirements are applicable in all cases except where adjacent structures or high ground create a potential for more damaging missiles or missile impacts at higher elevations.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 River Pumphouse and Gantry Crane

The river water pumphouse consists of a reinforced concrete substructure that provides support and suction wells for seven deep-well turbine pumps. Three of the pumps, two of which are located in the central bays, and one which is locate in the 181-N Building Annex, supply river water to the Emergency Core Cooling System (ECCS). The other four pumps (located in the outer bays, two on each side) supply river water to the N Reactor during normal periods of operation. The three ECCS pumps are driven by diesel engines. The four circulating water pumps are electrically driven. Two of the ECCS pumps, diesel engines, and related auxiliary equipment are housed in a masonry/concrete superstructure located on the main floor of the River Water Pumphouse. The third ECCS pump/diesel engine is located in the separate steel-framed metal annex building which is adjacent to the masonry/concrete superstructure. In addition to the seven pumps and their drivers and auxiliary equipment, there is a gantry crane and an armored observation guard tower.

The superstructure and equipment located on the operating floor (diesels, motors, etc.) (elevation 421 ft above MSL) of the 181-N building would be exposed to tornado wind loads, especially if the tornado approaches from the west. The east bank of the Columbia River, which is higher than the 181-N superstructure, provides some protection against winds coming from the east. The large electric motors, which drive the four circulating water pumps, also provide partial wind protection for the north and south walls of the superstructure. The superstructure houses two large diesel engines that drive the two ECCS low lift pumps located in the 181-N pumphouse and do not depend on electric ac power. The rigid frame of the concrete superstructure, consisting of the roof slab, beams, girders and 12 columns, is designed to resist lateral loads independently of the concrete-block walls. These walls are non-load bearing and are constructed of hollow concrete blocks, 12 in. thick. They are reinforced vertically and horizontally and are anchored to the concrete floor with dowels and to horizontal bond beams near the roof. They are also keyed into the sides of the exterior roof columns. Construction of the river water pumphouse and its superstructure is described in Section 3.8.4.1.

The wind loads on exposed surfaces of the superstructure were assumed to be 25 lb/ft^2 and on the guard tower, 30 lb/ft_2 in the original design of the 181-N building; however, earthquake forces control the design of the building except for the guard tower. No tornado analysis was performed as part of the original design and the criteria do not meet the 165-mph tornado listed in Section 3.3.2.1. The first tornado analysis of Building 181-N was performed in 1977 and reported in Appendix B of UNI-666 (Ref. 1) wherein it was concluded that tornado wind pressure acting directly on the exposed walls of the pumphouse structure would not cause it to collapse. The analysis was conservative in that it did not count on the block-filled walls in resisting shear forces. It also used a tornado wind velocity of 175 mph rather than the current requirement of 165 mph (Refs. 27 and 28). However, when column bending stresses were calculated they were calculated on the basis of tornado loads alone. Concurrent dead and live loads were not included and the analysis did not include tornado missile loads. 21

The river pumphouse was re-analyzed for tornado effects in 1987 by Kaiser and Westinghouse Hanford and specifically concentrated on the concrete block walls in the superstructure in response to ASEP issues 12 and 72. The results are reported in U-12008 (Ref. 29) and summarized below.

The masonry walls were scanned by utilizing an ultrasonic microwave detector, the results confirmed that the walls were reinforced as shown in the design drawings.

Based on the results of calculations, it was concluded that the masonry walls enclosing the two diesel ECCS pumps have adequate capacity to withstand the DBT wind pressure. The maximum stresses in reinforcing bars and concrete masonry for the most severe combination of tornado and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 75% of the ultimate strength of concrete masonry, and thus complies with the structural acceptance criteria contained in Section 3.3.2.1. The design margin is 20%.

Either of the two design basis tornado missiles would penetrate the 181-N Building if they hit a block wall end on. Also, the missiles would penetrate the building if they hit the vent grill located on the west wall end-on or if they hit the openings in the roof after the two pump hatch covers lifted off.

The third low lift pump is located outside of the emergency pump room concrete block superstructure and is not protected from tornado wind pressure or missile The diesel engine which drives the third low lift pump is located in a metal frame, metal siding annex to the 181-N Building. Tornado missiles would penetrate the siding if it remained in place. Also, it is expected that the siding would come off the annex exposing the diesel engine to tornado wind loading as well as missile hazards.

The report (Ref. 29) from which the above analysis was summarized was reviewed as part of the Seismic, Wind, and Tornado Evaluation of the ECCS (Ref. 30). Th conclusion of this review (Ref. 30) was that the analysis in Reference 29 was in complete accordance with SDC 4.1, Rev. 9 but only covered earthquake and tornado loads normal to the concrete block walls; earthquake and tornado loads parallel to the concrete block walls, and similar loads on other building elements had not been included in the scope of the analysis. Therefore, it was recommended that a dynamic, finite element, seismic analysis of the 181-N Building be performed and a tornado analysis of the superstructure be completec to include effects on the roof and columns, and bending and shear in the plane of the walls.

As a result of this recommendation, the Building 181-N structure and gantry crane were evaluated in 1988 (Ref. 34) to verify conformance to Hanford Standa Architectural-Civil Design Criteria SDC 4.1, Rev. 9 (Ref. 27). Loads due to the SSE or DBT were combined with dead and live loads. Detailed evaluation wa performed for selected structural members. These members were the most highly loaded in comparison to their capacities, based on a review of the structural system and load distributions.

Results of the capacity evaluation were expressed in terms of factors of safet The factor of safety for a structural member is defined as the ratio of its allowable stress or load capacity to the applied stress or load. A factor of

safety greater than unity generally signifies that the member satisfies SDC 4.1 criteria, while a value less than unity implies that it does not.

SDC 4.1 specifies that load combinations and allowable stresses for Category I structures be in accordance with DOE 6430, Chapter XXI (Ref. 32). Applicable load combinations for reinforced concrete are the following:

$U = 1.1\ (D + L) + 1.0\ SSE$
$U = 0.9\ D + 1.0\ SSE$
$U = 1.1\ (D + L) + 1.0\ W_t$
$U = 0.9\ F + 1.0\ W_t$

D = Dead load
L = Live load
SSE = Safe shutdown earthquake
W_t = Design basis tornado

The three sources of tornado loads on structures, specified in SDC 4.1, are the following:

- Wind pressure
- Atmospheric pressure drop
- Impact of tornado-generated missiles

Tornado wind pressures were determined following guidelines in ANSI A58.1-1982 (Ref. 33). The maximum horizontal wind velocity of 165 mph was conservatively assumed to be invariant with radius from the tornado center for evaluation of the superstructure. The velocity pressure at height z, q_z, is given by the following equation:

$q_z = 0.00256\ K_Z\ (IV)^2$
V = Basic wind speed = 165 mph
K_Z = Exposure coefficient
I = Importance factor

The wind pressure acting on a portion of the structure, q, is found as follows:

$q = q_z G_h C_p$
C_p = Pressure coefficient
G_h = Gust response factor

ANSI A58.1 provides wind loads for the design of ordinary building structures. The tornado wind speed selected for the design of nuclear facilities is sufficiently conservative to account for gusts and turbulence (Ref. 25). Coefficients I, K_x, and G_h were taken to be unity. Wind pressures acting on the superstructure were determined by the following equation:

$q_z = 0.00256\ V^2$
$= 70\ lb/ft^2$
$q = Q_z\ C_p$

The results of the tornado analysis of the Building 181-N structure reported in Reference 34 are summarized below.

Using the ANSI A58.1 pressure coefficients, the DBT wind pressures listed in Table 3.3-1 were calculated.

Atmospheric pressure drop as a tornado passes through a structure causes negative pressure on the walls and roof. Previous analyses for Building 181-N (Appendix C of Reference 1) determined that the hatch covers on the superstructure roof would pull off and provide sufficient vent area to relieve this negative pressure. An independent review of these calculations was performed. It was noted that, in addition to the hatch openings, the fixed louvers at the east side of the superstructure will also provide venting. Even if the hatch covers do not pull off, the open louver area is sufficient for the structure to be assumed fully vented. Negative pressure due to atmospheric pressure drop was not included in this evaluation.

Tornado-generated missile impact effects are highly dependent on the specific targets exposed. The degree of protection against these missiles for safety related components is addressed in detail in Section 3.3.3

The concrete block wall panels were the only structural components found necessary to be evaluated for tornado wind pressure. Tornado loads on the concrete beams and columns were found to be much less than those due to the SSE. Only tornado wind in the N-S direction was considered. This direction governs for the following reasons:

- The greatest out-of-plane loads occur in the north and south wall panels since they are taller than the east and west panels. The N-S direction tornado windward pressure on these walls is greater than the pressures on the leeward and side walls.
- The greatest in-plane loads occur in the west wall panels due to structure loading in the N-S direction. The largest overall structure loads occur due to wind pressure in the N-S direction since the exposed surface area is the greatest.

The following two governing conditions were evaluated:

- North/south wall panels subjected to windward pressure.
- West wall panel subjected to out-of-plane loads due to side wall pressure and in-plane loads due to windward and leeward pressures on the north and south panels.

Out-of-plane load distributions within the block wall panels were obtained from static analyses of the detailed finite element models. Loads due to overall structure forces were found by static analysis with the superstructure model. Block wall and bond beam stresses were calculated using the same techniques developed for the seismic capacity evaluation. Stresses and factors of safety are listed in Table 3.3-2. All factors of safety are greater than unity.

The 181-N Annex was re-analyzed and evaluated for the SSE motions and DBT conditions (Ref. 38). The 0.25 g SSE, 165 mph DBT and the structural acceptance criteria used were as specified in SDC 4.1 (Ref. 27). The objective was to determine if the structure could safely resist the SSE and DBT.

The DBT windward, leeward, and roof uplift pressure for the annex due to windspeed were calculated according to ANSI A58.1 (Ref. 33). Static analyses were conducted for the structural frame, critical connections in the frame, plus the siding and roofing including fasteners. The analyses results were evaluated. By inspection the DBT generated missiles specified in SDC 4.1 could penetrate the annex siding and roofing.

The results of the SSE analyses for the 181-N Annex are reported in Section 3.8.4.

The results of the tornado analyses and evaluation of the 181-N Annex Building reported in Reference 38 showed some potential distress, due to the DBT. These were as follows.

- In the three structural steel E-W bents along column lines AA, AB, and AC, the column base plate concrete anchor bolt tension loads exceed the design allowable by 1.4. Also, the structural bolts through flange plates in the roof beam to column moment resisting joints are overloaded.
- Leeward wind pressures on the siding sheet metal panels could fail the fasteners to the wall girts, and carry some panels away.
- The roof purlins are inadequately fastened to the structural frame to resist the DBT uplift pressure on the roofing sheet metal panels.
- DBT-generated missiles could penetrate the siding and roofing.
- The roof-mounted exhaust system, intake air filter, and ventilator for the diesel could be carried away by the DBT, or damaged by DBT missiles.

Conclusions and recommendations listed in the report for the 181-N Annex Building based on the DBT analyses and evaluation results were as follows.

- It is believed that the tension loads in the column base plate concrete anchor bolts could be acceptable up to double the normal design allowable under DBT conditions for this type of application. Further evaluations could be made to support this, including possibly, proof load testing of the anchors.

- The DBT potentially overloads structural bolts through flange plates in the E-W roof beam to column connections. However, this is a self-limiting condition because bending stresses in the beam and column should reach yield before bolt failures occur. Also, the analysis neglected the rotational strength contribution from the 4 bolt simple connection between the beam web and column flange. However, this is standard practice, and this connection alone, based on structural joint testing results, would be expected to have sufficient rotational stiffness and ductility to preclude collapse.

The alternatives for the above DBT upgrades for the 181-N Annex Building include:

1. Provide missile protection for equipment and components inside the annex.
2. Remove the wall insulation and replace siding fasteners to wall girts with 3/8-in. bolts as necessary.
3. Install A-307, 1/2-in. bolts in the purlin to roof beam connections. One bolt through the purlin webs and standing legs of the sheet metal clips. Two bolts through the beam web and other clip leg.
4. Provide DBT wind and missile protection for the roof mounted exhaust system, intake air filter, and ventilator for the diesel.

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The report (Ref. 38) confirmed the prior analyses (Ref. 34) with regard to missile impact. The batteries, diesel fuel lines, power panelboards, diesel engines and pumps, and control circuits located within the 181-N Emergency Pump Room and the exposed fuel oil lines where they enter the Emergency Pump Room are safety-related components located at Building 181-N which may be exposed to tornado missiles due to penetration at the following locations:

- Superstructure roof hatch covers
- Superstructure louvers at the east and west walls
- Superstructure exterior doors
- Steel-framed structure housing the third diesel engine.

The structure does not meet the tornado missile requirements of SDC 4.1. In addition, the block wall panels may be vulnerable to penetration or scabbing. The hollow cores of the masonry units are enclosed by face shells about 1.5 in. thick, providing a total thickness of 3 in. It was recommended that consideratio be given on notification of restart to provide protection against missiles for these components to enhance assurance of ECCS operation in the event of a DBT.

However, as stated in the Prioritization Guidelines (Ref. 50), it has been concluded this is an acceptable risk because a 165-mph tornado on the Hanford Site has a return probability of less than 10^{-7} per year. The probability of damage to the equipment located within the building and the exposed fuel lines located outside the building from missiles is less than the probability of the tornado. This is because the major portion of the building roof would not allow missile penetration. Equipment located outside the building such as the electric pumps shield the building walls, and the missile needs to impact the structure end on. The recommended modifications have been placed in the LTS program for implementation within 24 months following restart.

Additional discussion of tornado missiles is given in Section 3.3.3.

Gantry Crane Evaluation

The gantry crane is not a safety-related component. Evaluation of the gantry crane was performed to verify if it could impact safety-related components mounted on the operating floor. An approximate comparison based on scaling of design loads reported in Section 3.8.4.2 indicated that member stresses should not exceed allowable values. Therefore, detailed stress analysis of the gantry crane was not performed. A prior evaluation in 1977 (Ref. 9) concluded the crane would not move or tip over during the postulated tornado. All surfaces were included and drag forces conservatively accounted for; so the results and conclusions were fully satisfactory from an overturning standpoint. However, no stress analysis was performed.

Overturning of the gantry crane was evaluated in 1988 (Ref. 34). Acceptable performance was defined as a calculated factor of safety against overturning greater than 1.1 similar to overall structure stability.

When not in use, the gantry crane is generally located at the west side of the 181-N Building. In this position, rail clamps and wheel stops keep the crane from moving horizontally on its tracks; whereas, tie-down brackets are installed to keep it from tipping over in case of a severe earthquake. Walkdown determined that some of the anchors for the restraints were missing and that the restraints do not extend along the entire length of the crane rail. The hold-downs were therefore considered ineffective in preventing gantry crane uplift.

For the seismic evaluation, SSE loads were obtained from the dynamic analysis. In the tornado evaluation, wind velocity variation with radius from the tornado vortex was included based on parameters recommended in Reference 45.

The following factors of safety against gantry crane overturning were calculated:

Tornado - F.S. = 1.2
Seismic - F.S. = 1.8

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These factors of safety are greater than the required value of 1.1.

Raised Observation Post - 181-N

The 181-N raised observation post is a structural steel (mostly pipe sections) space frame tower 50-ft tall with a cab on top. The tower is 8 by 17 ft in plan. The observation post, which stands on the 181-N river pumphouse substructure, was previously analyzed for the 0.25 g free field SSE ground motions (Ref. 36). The subsequent SSE analyses for the 181-N river pumphouse (Ref. 34) showed the resulting operating floor motions at the base of the observation post tower significantly exceed the SSE free field motions, due to the response of the substructure. Therefore, it was necessary to re-analyze the observation post for the SSE motions at the 181-N operating floor determined from the results of the Reference 34 analyses.

The seismic and tornado evaluation of the observation post documented in Reference 36 was reviewed. The mathematical model of the tower structure is adequate and was used for this SSE reanalysis. The identical model was

constructed for ANSYS-PC/LINEAR (Ref. 37). Dead load and modal response spectrum analyses for the SSE were conducted for the structure.

Response spectra for the 181-N operating floor for the SSE based on soil structure interaction analysis control points at elevations 421 ft and 454 ft are documented in Reference 34. The SRSS combination of response spectra peak accelerations for the 421 ft and 454 ft elevation control points are 0.64 g and 0.62 g, respectively. The response spectrum for the 421 ft elevation control point was used for the model analysis of the tower.

The results of this 1988 analysis (Ref. 40) are summarized below. The dead load analysis results for the 181-N observation post show excellent agreement with the comparable analysis reported in Reference 36.

The 181-N tower structure member stresses due to the SRSS combination of SSE loads are within the allowables specified by SDC 4.1 (Ref. 27).

The tower SSE reaction loads on the reinforced concrete haunch supporting the tower on the operating floor are within acceptable limits. These SSE loads are lower than the 175-mph DBT reaction loads reported in Reference 36.

It was concluded the raised observation post on Building 181-N can safely resis the SSE motions, with the structural stresses remaining below SDC 4.1 (Ref. 27) limits.

3.3.2.2.2 ECCS Bypass Structure and Piping

The ECCS bypass structure, commonly called the silo, is a concrete cylindrical structure having an outside diameter of 13.0 ft and an inside diameter of 11.0 ft. It rests upon a 26-ft circular foundation, 5 ft thick. The bottom of the concrete foundation is 14.5 ft below ground level; whereas, the top of the cylindrical structure is 49 ft above the ground. Since this structure is an essential part of the emergency core cooling system, it must remain in servi during and after the occurrence of the design tornado. Davis, in 1969 (Ref. 10), performed a seismic/tornado analysis of the silo and reported the results as follows:

Peak wind pressures on the bypass structure, caused by the Hanford Site design tornado, would equal:

$$p = CD\ (0.002558\ V^2) = 0.6\ (0.002558 \times 175^2) = 47\ lb/ft^2$$

along the exposed height of the structure (Ref. 6). The horizontal reaction on the projected area of the cylinder would then be 47 lb/ft^2 x 13 ft x 49 ft = 29,939 lbs. This value is about one-eighth of the total lateral force generated on the structure during the safe-shutdown earthquake (SSE), which was assumed to produce a maximum horizontal ground acceleration of 0.25 g. Davis concluded the ECCS bypass structure was designed to survive this earthquake without producing stresses greater than normal design values (Ref. 10), and therefore, because the bypass structure was designed conservatively to resist lateral forces due to the above earthquake, it was concluded that the silo would not be damaged if exposed to the 175-mph wind forces generated during the Hanford Site design tornado. The effect of a design basis tornado was re-analyzed in 1978 (Ref. 1) and the silo was again found to be adequate.

Later, in 1987, in response to issue No. 74 of the ASEP (Ref. 39) which recommended a review be conducted of the silo near the 182-N Building for strength and deformation of the pipe connection, the silo was re-analyzed (Ref. 41) for seismic, tornado wind, and tornado missile loading for a wind speed of 175 mph. It was determined that the silo can safely withstand both the wind and missile loading. Based on the results of this analysis, it was concluded that the cylindrical walls are adequately reinforced to withstand the safe shutdown earthquake (see Section 3.8.4) and the 175-mph tornado wind. The maximum stresses in the reinforcing bars and concrete for the most severe combination of seismic or tornado and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 85% of the ultimate strength of concrete, which is in accordance with the structural acceptance criteria contained in NUSAR and SDC 4.1.

Overturning of the silo under vertical and lateral seismic loading is below allowable limits and the silo will not slide.

Stresses in the large-diameter steel piping interior and exterior to the silo are within allowable limits.

These results were reviewed by the Westinghouse Independent Safety Analysis Task Group (Ref. 31) and by the ECCS Task Group (Ref. 30) in the latter half of 1987 and found acceptable.

3.3.2.2.3 Demineralized Water Tank

The Demineralized Water (DW) Tank is required for the safe shutdown of N Reacto The DW tank is a 1,000,000 gallon capacity tank 75 feet in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the demineralized water tank is to:

- o Provide a supply of 432,000 gallons of warm demineralized water for ECCS use
- o Provide a normal supply of demineralized water to the high-and low-pressur injection pumps
- o Provide a reservoir for effluent water from the demineralizer plant.

In the event of an ECCS actuation caused by an RCS pipe break, the low-lift pumps and the high-lift pumps would start. The demineralized water tank is connected to the suction of the high-lift pumps and supplies the initial supply of warm water to the reactor. Once the supply of demineralized water is exhausted, the low lift pumps supply river water to the high-lift pumps through the raw water bypass silo.

The design and installation requirements for the demineralized water tank are included in Specification HWS-6873, dated March 15, 1960. The tank has been analyzed twice, once in 1984 and again in 1985 as part of the Plant Life Extension Program for seismic considerations (see Section 3.7) but neither analysis evaluated sliding, tornado wind loadings, or tornado missiles.

The demineralized water tank was re-analyzed in 1987 in response to ASEP issue 73 (Ref. 42). In addition to the seismic analyses, reported in Section 3.8.4, the analyses utilized wind loading from a 175-mph tornado and the impact of tornado missiles on the tank. The current operating level for the tank is 27.5 ft and this level was used in the analyses.

The analyses showed that the tank would not be overstressed by DBT winds and that the tank would not be penetrated by tornado missiles.

3.3.2.2.4 Emergency Raw Water Tank

The Emergency Raw Water (ERW) tank is required for the safe shutdown of the N Reactor.

The ERW tank is a 200,000-gal. capacity tank 35 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand. 21

The main function of the ERW tank is to provide a reliable initial source of water for the fog spray pumps suction.

In the event of a primary system pipe break, the fog spray pumps will start. The ERW tank is connected to the suction of the fog spray pumps and supplies the initial supply of water until the 181-N Building low-lift pumps are in operation.

The design and installation requirements for the ERW tank are included in Specification HWS-6873, dated March 15, 1960. The tank was analyzed for SSE conditions in 1985 as part of the Plant Life Extension Program (see Section 3.8.4), but no evaluation of sliding, tornado wind loading, or tornado missiles was performed. The tank was next evaluated in 1987 as part of the ASEP (Ref. 43). The evaluation was not initially identified as part of the N Reactor ASEP, but was added because of the close proximity of the ERW tank with the ECCS.

A seismic analyses for the SSE was prepared and is reported in Section 3.8.4. In addition, the wind loading from a 175-mph tornado was analyzed, including th impact of tornado missiles on the tank. The current operating level for the tank is 17 to 20 ft and a level of 20 ft was used in the analyses.

Based on the analyses, it was concluded that the stress in the tank shell woul be substantially below yield for either the SSE or the DBT and that tornado-generated missiles will not penetrate the tank.

The results of the tornado analyses for both the demineralized water tank and the emergency raw water tank were reviewed and found acceptable by the ECCS Task Group and the Westinghouse Independent Safety Analysis Task Group (Refs 3 and 31).

3.3.2.2.5 Filtered Water Tank

The Filtered Water (FW) Tank is an 850,000-gal. capacity tank 65 ft in diamete located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the FW tank is to:

- Provide backup supply of clean water to the ECCS
- Provide normal suction supply to the filtered water (FW-1) pumps which supply all other FW loads.

The FW tank is not a part of the ECCS and is not required to function after th SSE, but if it is functional, it may be used to supply clean water to the suct header of the 182-N Building high-lift pumps by manually opening valves betwee the tank and the pump suction header.

The design and installation requirements for the FW tank are included in Specification HWS-6873, dated March 15, 1960. Prior to this evaluation, the tank has been analyzed twice for seismic conditions (see Section 3.7). Neithe analyses evaluated sliding, tornado wind loading, or tornado missiles.

The tank was evaluated in 1987 in response to ASEP issue 73 (Ref. 44) for SSE and DBT conditions. A 175-mph tornado was used in the analysis and the missiles listed in Ref. 27.

It was concluded that the stress in the tank shell is substantially below 90% of yield for the DBT and that tornado-generated missiles will not penetrate the tank.

3.3.2.2.6 After Heat Removal Tank

The After Heat Removal (AHR) Tank is a 500,000-gal. capacity tank 50 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the AHR tank is to:

- o Supply demineralized water for secondary loop makeup during normal operation
- o Serve as a receiver for secondary loop and condensate returns
- o Provide an alternate source of clean water for ECCS.

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The AHR tank is not required to function after the SSE, but is required to be operable after the OBE.

In 1987, a review of the AHR tank was requested. The evaluation (Ref. 45) was not initially identified as part of the N Reactor ASEP, but was added because of the close proximity and potential interface of the AHR tank with the ECCS.

The evaluation report noted that the design and installation requirements for the AHR tank are included in Specification HWS-6873, dated March 15, 1960. Prior to this evaluation, the tank was analyzed in UNI-3133 Rev. 1, "N Reactor Balance of Plant Evaluation," Task 7 - Water Storage Tanks, M. R. Lindquist, dated January 1985. This analysis examined the AHR tanks response to a 0.20 g earthquake. The analysis was part of a program related to plant life extension and used original plant design seismic loads instead of the current 0.25 g.

The seismic analysis calculated tank shell stresses that are essentially in agreement with the KEH analysis and that show that the tank shell stress is below 90% of yield. The analysis did not evaluate sliding, tornado wind loading or tornado missiles.

It was concluded from the analyses (Ref. 45) that the stress in the tank shell and the tank bottom is substantially below 90% yield for the DBT and that tornado-generated missiles will not penetrate the tank. Other seismic results, including sliding, are discussed in Section 3.8.4.

3.3.2.2.7 Emergency Dump Tank

The Emergency Dump Tank (EDT) and associated piping was designed and constructed under the Effluent Control Program Project No. GCP-411. It provides the following:

- Function - to dissipate primary coolant line energy prior to ECCS cooling water entering the reactor
- 615,000 gal. of quench water
- Venting of radioactive gases to 109 Zone I.

The ECCS dump tank receives the reactor coolant system blowdown associated with a transfer to ECCS coolant, and the ECCS coolant flows through the tank. The internal negative design pressure for the tank is 0.5 psi or 72 lb/ft^2. The design criteria for the tank contained in HWS-7759 did not require design to tornado winds or tornado missiles. It did require design to a 25 lb/ft^2 wind pressure, which is equivalent to a velocity of approximately 100 mph. An analysis performed in 1978 and reported in NUSAR noted that the internal negati design pressure of 72 lb/ft^2 was greater than the external pressure of 47 lb/ft that would be exerted by tornado wind forces, and, therefore, the tank would not be expected to fail by buckling. Since the horizontal shear force (158 kip and the overturning moment on the tank (5700 kip-ft) due to the Hanford design tornado were considerably less than the respective values used for earthquake design (Ref. 11), it was concluded (in 1978) that the capability of the ECCS du tank would not be impaired by wind forces acting during the Hanford Site design tornado.

The ECCS Seismic and Tornado Task Group (Ref. 30) reviewed the EDT in 1987 and observed that the 1978 evaluation was done to the requirements of SDC 4.1, Rev. 8 (Ref. 22). Revision 9 of SDC 4.1 (Ref. 27) changed tornado wind and missile requirements, but generally the values are lower. The maximum wind speed for reactor facilities went from 175 mph down to 165 mph. The missiles are generally less damaging; however, a 3-in. diameter, 76-lb pipe travelling at 80 ft/s was added to Revision 9. This compares to a 6 inch diameter, 267-lb pipe at 55 ft/s in Revision 8. The 3-in. diameter pipe at the higher velocity has significantly more penetrating power, however, calculations show it would still not penetrate the tank.

The Westinghouse Independent Safety Analysis Task review (Ref. 31) concluded that the EDT and piping downstream of valve V-23 had not been re-analyzed. Failure of that tank or section of piping could result in unacceptable offsite dose consequences. It was concluded that the risk to N Reactor from failure of essential piping or tanks due to a DBE or DBT during continued operation is acceptably low, provided the following recommendations are implemented: (a) A seismic analysis of the sections of the ECCS piping outside of 105-N/109-N be performed. The analysis to establish any upgrades required to allow the system to meet current Hanford Site seismic criteria. If upgrades are required but impractical, the analysis should at least quantify the risk involved in continued operation. (b) The EDT should be qualified to SDC 4.1, Rev. 9 (Ref. 27). 21

Because of the deviations noted above, the following analyses were performed, in 1988 (Ref. 46) to enable compliance with requirements of SDC 4.1, Rev. 9:

1. Seismic and Tornado analysis of the EDT, internal piping, external connected accessories and 24 inch vent pipe to the current loading requirements
2. Slope and Foundation stability and strength analyses.

The results of this structural evaluation of the EDT and its associated piping indicated that it can safely withstand the DBT. Four safety modifications were required to comply with the SSE requirements. The seismic disucssion is presented in Section 3.8.4.

3.3.2.2.8 Tornado Evaluation of Exposed ECCS Piping

The 30-in. active ECCS line emerges from the ground outside the 109-N Building. It goes up the side of the building to the roof, then across the roof to the 105-N Building. The 30-in. pipe is then tee reduced to two, 10-in. pipes which descend into Corridor 5 in 105-N. The aboveground ECCS and reactor cooling system piping within the 105-N/109-N complex was analyzed in 1987 by Westinghous Hanford (Ref. 47) using the current Hanford Site Design Basis Earthquake and the computer program PIPSD. Results of this analysis showed overstress conditions in some areas. Installation of the additional required supports will be completed before restart to alleviate this condition (see Section 3.8.4). No tornado wind loads or missile effects were considered. The ECCS Seismic and Tornado Task (Ref. 30) concurred with the above seismic analyses, but recommend that exposed ECCS piping be further evaluated for tornado wind and missile effects. The analyses further noted that the exposed piping from ground level up to column line D-17 is housed with structural steel framing and corrugated metal sheeting. Based on review of the drawings it was established that the exposed vertical "stub up" would be the most vulnerable and critical for missil impact loads.

As part of the Reference 30 review, the vertical section of the pipe (length 20 ft, 11 in.) above ground level was analyzed for tornado-generated missile impact load criteria defined in HPS SDC 4.1 (Ref. 27). The existing protective structural steel framing with corrugated metal sheeting around the piping was conservatively not considered in the analysis. The structural response of the target (pipe), i.e., the strain energy, was calculated to determine if the targ will remain stable during and after missile impact. Also the ductility ratio of the target, i.e., the maximum permissible deflection of the structural syste to the deflection at "effective yield," for the system was calculated. The computed value is within the allowable limits. The pipe was also checked for required wall thickness to prevent perforation during impact. The existing pipe wall thickness of 0.375 in. is adequate. No modifications are required. Therefore, it was concluded that the Emergency Raw Water Supply Line (ECCS) will remain stable and functional during and after missile impact.

Prior analyses had established that the wind pressure caused by the Hanford Site design tornado would be the same as for the bypass structure, i.e., 47 lb/ft^2. The horizontal reaction force due to wind on the projected area of the vertical pipe cylinder would be 47 lb/ft^2 x 2.5 ft diameter = 117.5 lb/ft of pipe length. The piping was designed for an equivalent lateral seismic load of 221.5 lb/ft, which is equal to 0.5 times the weight of the pipe and its contents. Therefore, it will not be damaged by the tornado forces.

The ECCS effluent piping pressure relief standpipe was also analyzed for seismic and tornado forces (Ref. 48). The standpipe was installed in the N Reactor ECCS piping just downstream of the 1316-N valve box as part of N Reactor Project H-797. The purpose of the standpipe is to provide a means for pressure relief in the event that downstream piping becomes constricted under faulted loading conditions such as the SSE. It was calculated and concluded that the subject standpipe will maintain its intended function of providing a pressure relief path under faulted load conditions. 21

3.3.2.2.9 High-Lift Pump House (182-N)

Certain components of the reactor ECCS are located in the basement of the 182-N building. Included are the diesel drive units, three high lift-pumps, two fog spray pumps, and associated piping. The ground floor of the building, which is also the ceiling over the basement, consists of a concrete slab supported by a series of integral beams and girders. Basement walls and columns are also constructed of reinforced concrete and support the ground floor slab and the superstructure. The walls of the superstructure, which are about 21 feet high, are generally made of hollow, concrete block with intermittent reinforcement, anchored to a structural steel framework. Precast roof planking is supported on a series of steel beams and covered with built-up roofing. Additional information on building construction is presented in Section 3.8.4.4. 21

The building is designed to resist lateral loads corresponding to a seismic factor of 0.2 g, which controlled the design rather than the specified wind loading of 15 lb/ft. A review of the design calculations indicates that the walls and steel superstructure are capable of resisting a tornado wind load of 70.5 lb/ft^2 without failure. However, if the bursting pressure, due to an

ambient pressure drop of 0.375 psi*, occurs simultaneously with maximum tornado suction forces, sections of the block walls and the roof may possibly burst outward. However, the large, roll-up door (located on the northeast side of the building) is likely to fail first. In this case, the area involved will be of sufficient size to equalize pressure inside and outside the building. The block walls and concrete roof of the 182-N Building control room (not required for ECCS operation) will remain intact because the bursting pressure would be expected to blow open one, or both, of the personnel access doors. This should equalize air pressure inside and outside of the control room and minimize damage to the equipment contained therein. Significant pressure differentials cannot occur between the basement and the main floor areas becaus the equipment access hatch, which is 10 by 12 ft in size, is always open.

Critical components of the ECCS, located in the basement of the 182-N Building, are beneath the ground floor slab and, consequently, won't be damaged by tornac winds or missiles. Diesel fuel oil day storage tanks (see Section 9.5.4) are located below ground level to the north of the 182-N Building. These tanks supply the ECCS low lift and high lift diesel-driven pumps and other diesels (fog spray, fire suppression, control room habitability) within the 182-N Building. Because of their buried location, they would not be subjected to tornado forces.

Following the several design reviews in 1986 due to the Chernobyl accident, the features of the 182-N structure that appeared to the independent reviewers to be the most vulnerable to damage from seismic and tornado loading were the exterior concrete masonry walls of the superstructure. The seismic loading in this part of the building is principally resisted by the structural steel fram supporting the roof. The structural steel frame of the superstructure, consisting of beams, girders and columns, is designed to resist lateral loads independently of the concrete-block filler walls. These walls are non-load bearing and are constructed of hollow concrete blocks, 12 in. thick.

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*The ambient pressure drop at the point of maximum wind speed in the tornado vortex is equal to about one-half of the maximum ambient pressure drop, or 0.75/2 psi (Ref. 15).

Issue No. 12 of the ASEP (Ref. 39) recommended that the concrete block superstructure portions of the 181-N and 182-N Buildings be re-examined for possible wind damage to the walls and the possibility of the blocks falling on vital equipment housed within these facilities.

Also, Issue No. 73 of the ASEP program requested a review be conducted to determine if concrete blocks in the superstructure of Building 182-N can fall and damage the batteries and/or damage diesel peripherals including fuel lines.

An analysis was performed in 1987 (Ref. 49) where it was concluded that the exterior masonry walls in the superstructure of the High Lift pumphouse, including the anchorage of the masonry walls to the structural steel building frame has adequate capacity to withstand the SSE and DBT wind pressure. The maximum stresses in reinforcing bars, concrete masonry, and masonry anchors for the most severe combination of seismic or tornado and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 75% of the ultimate strength of concrete masonry, and thus complies with the structural acceptance criteria contained in SDC 4.1 (Ref. 27). 21

Subsequent analyses resulted in reinforcement of the structural cross-braces which reinforce the masonry walls.

The high lift diesel pumps, the fog spray diesel pumps, the fuel supply lines for the diesel pumps, and the control panels for the diesel engines are all located in the basement of the 182-N Building and are protected from tornado wind or tornado missile damage.

The 182-N batteries are located in the northeast corner of the main floor near the north wall of the building. The DC power distribution panels are located next to the batteries on the north wall. The conduit which carries DC power runs along the north wall and then to the diesel engine control panels located in the basement.

Either of the two SDC 4.1 tornado missiles would penetrate the 182-N Building superstructure if they hit the block wall end on. These tornado missiles present a potential hazard for the batteries and for the power distribution panels and

conduit that is located on or near the north masonry wall of the building. However, a 175 mph tornado on the Hanford Site has a return probability of less than 10^{-7} per year and the probability of the missile impacting the north mason wall end on, between the vertical reinforcing bars, and in the location of the batteries or power panels is less than the probability of the tornado.

Based on the prioritization guidelines of Reference 50, the report recommended the following fix be considered:

- Within 24 months after the end of the current outage provide protection for the batteries, power panels, and conduit on the main floor of the 182-N Building.

This protection would assure operation of high-lift pumps and fog spray pumps after the DBT.

A review of this analysis was performed by the Westinghouse Hanford Independent Safety Analysis Team (Ref. 31) which concluded:

- The latest analysis conducted recently by Kaiser (Ref. 41) specifically concentrated on the abovegrade concrete block wall panels in response to a safety review comment. The analysis shows no overstress conditions for either the DBT wind or DBE loadings but recommends addition of steel plat to the inside face of the block walls to prevent tornado missile penetration.

- That the analyses and reviews documented have utilized static equivalent load methods and no dynamic analysis of the facility has been performed t date.

- That no unacceptable risk has been identified and the risk to N Reactor o operating with the 182-N Building in its present configuration is accepta low provided a dynamic, finite element, seismic analysis of the 182-N Building be performed.

Accordingly, a structural analysis of the 182-N Building with respect to the requirements of Hanford Plant Standards SDC 4.1 (Ref. 27) was performed (Ref. 51) to ensure its adequacy during a seismic event in order to preclude damage to the enclosed ECCS components. In addition to assessing structural integrity, in-structure response spectra were generated at selected locations for subsequent equipment and system evaluation. Since the engineering evaluation of the components of the ECCS was continuing on a parallel path, the analysis of certain localized structural components needed to be continually upgraded until the ECCS component configuration and support reactions were finalized. The results are discussed in Section 3.8.4. 21

Tornado loads and tornado-generated missiles were not evaluated at this time. It was determined by the project that the tornado event was not a return to service (RTS) program issue. Lateral forces resulting from 165-mph winds were calculated and found to produce frame loads that are less severe than the seismic loads. Wind loads on the siding and roof components were not evaluated. Tornado loads and missiles will be evaluated under a separate scope of work during the LTS program.

3.3.2.2.10 Concrete Confinement Structure (105-N and 109-N Buildings)

The concrete confinement structure consists of the interior portions of the reactor building (105-N) and part of the heat exchanger building (109-N). See Section 3.8.1 for details of construction. The confinement structure is designed to resist pressure loads due to the design basis accident combined with effects due to thermal, dead, and earthquake loads. The magnitudes of the internal design pressures, consisting of an internal bursting pressure of 5 psi followed by a partial vacuum of 2 psi, are more severe than any of the tornado pressure loadings on the concrete confinement structure. See Table 3.3-1. For example, the 5 psi (720 lb/ft^2) confinement pressure is greater than the maximum bursting load (101.0 lb/ft^2) acting on a side wall by a factor of 7.1. The dead weight of the roof of the confinement structure varies from 225 to 675 lb/ft^2 which is greater than the maximum uplift force on the roof (116.6 lb/ft^2). See Table 3.3-3. Thus, wind loads produced by the Hanford Site design tornado will not damage the concrete confinement structure. Likewise,

missiles associated with the specified tornado will not penetrate the thick walls of the confiner, which range from a minimum of 1.0 ft at the 109-N vent structures to over 5-ft thick for major shielding walls. See Section 3.3.3.

The Westinghouse Hanford Independent Safety Analysis Task Group reviewed the 105-N/109-N Building seismic and tornado capability in 1987 (Ref. 31). At that time the latest analysis of the 105-N/109-N complex was in June of 1986 (Ref. 52) using modern techniques involving soil structure interaction analysis and 3-D dynamic analysis and utilized the Hanford Plant Standard SDC 4.1, Rev. 9 (Ref. 27) DBE and DBT criteria to produce equipment response spectra. The Task Group concluded, based on the 1986 study, that N Reactor can be safely operated with an acceptably low risk of major damage to the 105-N/109-N structures which might result from extreme natural phenomena such as earthquake and tornados; and although a high confidence level exists that the 105-N/109-N building complex would meet the Category I requirements of HWS SDC 4.1, no analysis exists to document that fact; therefore, it was recommended that such an analysis be scheduled and completed to dispel all doubt about seismic capaci of the facility.

The analyses were performed and completed in 1988 (Ref. 53), see Section 3.8.4. The results did not change the conclusion stated above, i.e., tornado wind loac will not damage the concrete confinement structure nor would the postulated missiles penetrate the walls.

3.3.2.2.11 116-N Ventilation Stack

The 116-N stack is constructed of reinforced concrete and rises 200.5 ft above the surrounding ground. Its foundation is 22.5 ft below ground level so that the total height of the stack is 223.0 ft. The stack has an outside diameter of 15.0 ft at the top and 22.1 ft at its base. The base of the stack is tied to, and supported by, a reinforced concrete foundation. The latter is a 33.5 octagon, 4.5 ft thick below the stack and tapered to 2.5 ft at its perimeter. The design is based upon the requirements of ACI 505-54, "Design and Construct of Reinforced Concrete Chimneys," and a design wind of approximately 100-mph (Ref. 3). Wind pressures used in the original design calculations are summari in Section 3.3.1.2. Bending moments produced by the 100-mph wind are shown in Figure 3.3-1 (Ref. 12).

With respect to the Hanford Site design tornado, the 175-mph wind velocity will produce an equivalent pressure of 0.6 times 78.3, or 47.0 lb/ft^2 along the exposed height of the stack (in this case, 0.6 is the drag coefficient and 78.3 is 0.002558 V^2) (Ref. 6). Bending moments produced by the 175-mph wind are also shown in Figure 3.3-1 along with curves showing the moments along the stack corresponding to the initiation of yielding in the reinforcing steel and for the ultimate bending resistance at various elevations.

Figure 3.3-1 shows that the 175-mph wind would produce stresses in the reinforcing steel greater than 40,000 psi in a region 145 to 185 feet from the top of the stack. This stress level is equal to the specified minimum value of the yield strength for intermediate-grade, reinforcing steel; whereas, the average value for this grade of material is probably closer to 50,000 psi (Ref. 13). If yielding actually occurs, the greatest amount would occur at about 167 ft from the top. Since the ultimate bending resistance of the ventilation stack at this location is somewhat greater than the overturning effect of the wind, the stack would not fail. However, if the average yield strength of the reinforcing steel is around its specified minimum of 40,000 psi, the reserve strength of the stack is not great enough to assure that it would not fail when subjected to the 175-mph wind. If the stack should fail, the upper section would most likely break off from the lower part at about 167 ft down from the top. Most damage would result if the upper part of the stack fell on the roof of the 117-N filter building. The 105-N reactor building, which is located further to the south, would not be affected.

On the other hand, if the wind velocity does not exceed 155 mph, maximum stresses in the reinforcing steel would be less than 36,000 psi, which is 0.9 times the specified minimum yield point of 40,000 psi. These latter conditions satisfy stress requirements of the original Hanford Standard Design Criteria (SDC 4.1) and assure that the stack would not be damaged in this case. The effect on stack behavior due to vortex shedding was not considered due to the short life (about 7 seconds duration) of the specified design tornado. | 21

The 117-N filter building and the stack are not required for safe reactor shutdown. In the event that either the onsite power supply or the offsite power supply does not fail during the tornado, the reactor would remain on recirculation cooling. If both the onsite and offsite power supplies fail, a transfer to emergency core cooling would occur. The recirculation coolant would flow to and be quenched by the water in the emergency dump tank. No fission products would be released from the fuel elements. See Section 6.3 for an evaluation of the ECCS and see also Section 3.3.2.1 for plant response to a tornado.

The Westinghouse Hanford Independent Safety Analysis Task Group (Ref. 31) re-examined the 116-N Stack in 1987 and concluded that the documented analyses show that the risk of damage or failure of the 116-N Stack from a large earthquake is very low and that the N Reactor can be operated and shut down safely without undue risk from natural phenomena damage to the 116-N exhaust stack.

3.3.2.2.12 Exterior Portions of the 105-N and 109-N Buildings

Building structure, exterior to the concrete confiner and the rooms which conta the horizontal control rods, is of conventional steel framework, concrete floor and insulated metal siding. Several types of structural roof members are utilized, namely; concrete slabs at the 109-N Building and precast concrete slabs or corrugated metal decking, covered with conventional-weight or light-weight (insulating) concrete, at the 105-N Building. Pressure loadings for the various zones of the 105-N Building are discussed in Section 3.3.1.3.

Most of the exterior walls of the 105-N and 109-N Buildings are constructed of insulated-metal panels and are supported by steel framing. Wind requirements controlled the lateral-force design of these walls; whereas, earthquake requirements controlled the design of exposed concrete walls. Exterior walls of the office and service areas of the 105-N Building, including rooms which contain the intake ventilation fans, are designed for a normal wind pressure of 20 lb/ft^2. The exterior walls of the Zone I exhaust-fan room of the 105-N building are designed for an internal pressure of 0.5 lb/in.2 or about 75 lb/ft Other areas of the 105-N Building and the exterior walls of the 109-N Building are designed for a wind load of about 30 lb/ft^2.

A review of construction details indicate that the girts (horizontal support channels) will probably be overstressed first should wind forces exceed normal design values. The working stress in the girts is typically 20,000 psi at the design wind load. Thus, wall panels, designed for 20 lb/ft^2, will probably start deforming appreciably and separating from the main building structure at wind loads of around 40 to 50 lb/ft^2. Similarly, the wall panels and girts designed for 30 lb/ft^2 would probably deform and blow away at wind pressures of 60 to 75 lb/ft^2. Thus, a direct impact of the Hanford design tornado on the 100-N plant, would cause severe damage to the exterior and interior walls of the office and service area of the 105-N building and minor to moderate damage to exterior walls of the 109-N Building and those at other locations around the 105-N Building. However, sharp edges, at ridges and corners of buildings, produce large changes in local pressure. As a result, peak (negative) pressures acting on ridges, eaves and corners are likely to be 2 to 5 times greater than the dynamic wind pressure (78.3 lb/ft^2) associated with the Hanford design tornado. See Table 10 of Reference 14. Thus, roofing and metal siding located at roof lines and building corners are likely to be blown away first. Likewise, exterior metal ducting, roof ventilators and roll-up doors will probably be torn from their supports on both buildings when subjected to 175-mph winds.

When the ambient pressure drop of 0.375 psi* is added to suction pressures acting on the leeward and side walls and on light-weight roofing, a number of wall and roof panels are likely to disengage outward from the supporting steel structure, to the extent required, to equalize pressures inside and outside the building. This loading will not result in subsequent damage to the concrete confinement structure nor to the reactor components housed therein. The ventilation ducting, located on the roof of the 109-N building, has been upgraded (Ref. 56) in response to SEP issue 56 (Ref. 39) and the ventilation ducting as discussed in Section 3.3.3.1 was designed and installed to withstand the tornado effects of SDC 4.1 Rev. 10 (Ref. 28) and in addition met the more conservative criteria discussed in Reference 55.

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*The ambient pressure drop at the point of maximum wind speed in the tornado vortex is equal to about one-half of the maximum ambient pressure drop, or 0.75/2 psi (Ref. 15).

3.3.2.2.13 Reactor Control Room

The reactor control room is housed in a rectangular-shaped structure located on the southeast corner of the 105-N building. The structure is constructed of reinforced concrete. The roof slab and walls are typically 1-foot thick. The concrete floor of the control room, which is at ground level, is supported on steel girders, beams and columns and on the outer basement walls. The control room floor is also the ceiling of a basement which houses reactor instrumentati and switch gear. The combined area has its own ventilation system. See Section 3.8.4.8 for additional information on building construction.

Except for the roof, the control room structure is surrounded by other portions of the 105-N and 109-N Buildings. Thus, the walls of the control room are not subjected directly to wind forces unless the service area of the 105-N building is blown away. In this unlikely event, the tornado pressure (70.5 lb/ft^2 actin on the windward wall of the control room) would not overstress the reinforced concrete structure nor otherwise damage the equipment located in the control room. Suction forces would be minimal because the control room, which is only 15 feet high, is shielded and protected by the towering walls of the confinemer structure, located to the south and west. The basement area and the control room are connected through the ventilation system and are adequately vented to atmosphere. The control room ventilation system was upgraded in 1988 to upgrac control room accident habitability (Ref. 57). The new ventilation system provides dual emergency air intakes with in-line high efficiency filtration anc sufficient capacity to pressurize the control room. The control room and its HVAC system are independent from all adjacent areas. Radiation and toxic gas monitoring and protection from volcanic ash is provided on the emergency and normal air intakes. Automatic isolation of the normal control room air intake and realignment of the ventilation system by means of a new damper arrangement is provided.

The new emergency ventilation system is designed to the seismic requirements detailed in SDC 4.1 (Ref. 28), and is protected from tornado missiles. In addition, the existing duct work, used by the emergency system in the control room, is supported seismically to the same requirements. This system will, therefore, be operational after all accidents postulated by the Functional Design Criteria (Ref. 58).

The control room structure was modified as necessary to provide tornado missile protection and radiation shielding for the postulated accident situations. 21

3.3.2.2.14 Irradiated Metal Segregation and Storage Basins

This reinforced concrete tank is located at the southeast corner of the 105-N Building. It is 135 ft long, 49.5 ft wide and 24.5 ft deep and contains two 101-ft long compartments for storing irradiated fuel elements. One compartment is 19 ft wide and the other 21.5 ft wide. Each is 24.5 ft deep and is separated from the other by a concrete divider wall. The metal-segregation basin is a compartment, 26 ft by 29.5 ft, which is also 24.5 ft deep and is located at the east end of the storage basin. Wall thicknesses are typically 3 ft thick, except at the vertical light chaises, which are 1 ft wide and 1 ft deep and spaced 8 ft apart in a staggered fashion to assure a minimum wall thickness of 2 ft. The tank bottom, which is nominally 3 ft thick, increases to a thickness of 5 ft, 9 in. in a bank around the edges in the east half of the tank. The bottom slab rests on the ground and acts as a solid mat foundation for the structure.

The storage and segregation basins are covered with a steel-frame structure that supports the roof, insulated-wall panels and the overhead crane systems. The steel framing of the superstructure extends into the metal-transfer area, that has a roof height of 49 ft above the ground compared with a height of 22 ft for the roof over the irradiated metal storage and segregation areas. The latter are nestled between the confinement walls of the 105-N outlet pipe space on the east, the confinement walls of the 109-N pipe gallery on the south, the metal transfer area on the west and partially by the concrete walls of the decontamination-service area of the reactor building on the northeast. As a result of its shielded location, the storage and segregation areas would not be subject to maximum tornado effects unless the tornado was directly from the northwest, which is not in the prevailing tornado wind direction. Even then, the shielding and containment walls of the 105-N and 109-N Buildings, which are much higher than the roof over the basins, would protect the latter from maximum wind effects in the event of such a tornado. For this reason, it is doubtful that much of the roof or the north wall would be damaged, if any. On the other hand, it is expected that the metal-transfer area would lose some of its metal siding and roofing during the postulated tornado, as well as one or both of the

large roll-up doors located at the north side of the metal-transfer building. Since the roll-up doors would probably be torn from their guides at wind velocities of around 125 mph, the building compartment would then be adequately vented to minimize structural damage to the rest of the structure due to the 0.75 psi ambient pressure change.

For the reasons stated above, the metal segregation and storage basins would not be exposed to peak wind forces, even if the tornado passed directly over the area. Thus, it would not be able to blow enough water out of the basin to be of concern. According to Reference 16, there is no substantial evidence that a tornado can move a significant amount of water through any type of sucki mechanism, although there is evidence indicating that water has been blown from ponds and swimming pools. In order to move water through lowering of the pressure in the tornado core, the outside diameter of the tornado would have tc be smaller than the least dimension of the body of water being affected, or less than about 50 ft when considering the metal storage basin. A tornado generating maximum wind speeds of 175 mph would have an outside diameter of around 300 feet (Ref. 17). Thus, it may be concluded that the tornado will not impair the integrity of the basins for safe storage of irradiated fuel elements contained therein.

3.3.3 Tornado-Generated Missiles

The various types of tornado-generated missiles are described in Section 3.3.3 The effects caused by these missiles striking critical facilities are described in Section 3.3.3.2.

3.3.3.1 Description of Tornado-Generated Missiles

The following tornado-generated missiles criteria are extracted from SDC 4.1, Rev. 10 (Ref. 28) and have not changed since January 1985.

The effects of the following tornado-generated missiles on Category I structures, systems and components shall be analyzed. Protection from penetration and secondary effects of scabbing shall be provided.

Missile	Weight (lb)	End Area (ft2)	Horiz. Veloc. (ft/sec)		Max. Height (ft)
			Reactor	Nonreactor	
Wood Plank 4"x12"x12'	139	0.29	114	88	200
3" Std Steel Pipe, 10' Long	76	0.067*	80	59	100

*Value given is gross cross-sectional area for use in penetration calculations.

NOTE: Vertical missile velocity is 2/3 missile horizontal velocity. The horizontal and vertical velocities should not be vectorially combined.

These DBT criteria values were established by DOE document UCRL - 53526, "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites," prepared by Lawrence Livermore National Laboratory, February 1984. The document contains a tabulation of detailed tornado design parameters. 21

These requirements are applicable in all cases except where adjacent structures or high ground create a potential for more damaging missiles or missile impacts at higher elevations.

These missiles contain a significant degree of conservatism, specifically in the assumptions that the maximum wind speed (165 mph) acts over the full height of a building and that they appear at some arbitrary location in the tornado windfield and are simply allowed to fly.

Other tornado-generated missiles that have been considered in prior tornado evaluations for the N Reactor site correspond to Spectrum II missiles defined in the NRC Standard Review Plan 3.5.1.4, and include the following:

No.	Description	Weight (lb)	Velocity (ft/sec) Horizontal	Vertical**
(A)	Wood plank 4" x 12" x 12′	115	110	105
(B)	Steel rod 1" x 3′	9	24	24
*(C)	Steel pipe 6" sch 40 x 15′	287	55	30
*(D)	Utility pole 13.5" x 35′	1125	55	30
*(E)	Steel pipe 12" sch 40 x 15′	750	35	30
*(F)	Automobile 20 ft^2 area	4000	50	20

*These velocities apply only at heights under 20 ft above grade. Velocity above this height is 0.

**Do not combine with horizontal velocity. Use only for evaluating roof structures.

NRC Spectrum II missiles are intended to be used in the design of new facilities. Provision for these tornado-generated missiles, in this manner, provides a conservative level of protection for any type of missile that might be airborne during the specified design tornado. During the analysis of new structures, it is customary to assume that the particular missile is traveling at the theoretical speed and that it hits the barrier, head-on, at the most vulnerable point. For new facilities, missile protection can be provided by making the barrier walls of concrete and/or steel of sufficient thickness to stop the specified missiles.

When considering the potential effect of tornado-generated missiles on older structures, not built to current tornado standards, additional factors of a practical nature should be considered in addition to the theoretical properties of the above missiles. For example, a review of the technical literature and observations of actual tornado effects indicate that the Spectrum II missiles are not likely to (1) become airborne during a tornado having a maximum wind speed of 175 mph, (2) be traveling head-on with the wind, if airborne, and (3) hit the target head-on at its most vulnerable point. Thus, the probability of a particular missile being generated and hitting a critical piece of equipment at N Reactor is substantially less than 10^{-7} assigned to the occurrence of the Hanford Site design tornado.

The most recent study of tornado missiles at the N-Reactor site was performed by McDonald in 1988 (Ref. 55). The tornado missile parameters given in the tabulation of UCRL 53526 are based on computer studies performed by Dr. McDonald. The study was motivated by concern for potential perforation by tornado-generated missiles of a 42-in. diameter vent pipe located on the roof of Building 105-N. Elevation of the roof is 70 ft above ground level. Centerline of the pipe is at an elevation 73 ft above ground where there is a horizontal run of exposed pipe of about 25 ft. The primary objective of the study was to determine the potential for impact of a spectrum of missiles and to recommend appropriate design criteria. The study, while still conservative, utilized a more realistic estimating approach based on computer tornado simulations, which is entirely appropriate for such low probability (10^{-7} per year) events.

The missiles considered in the study were: 21

1. 2x4-in. timber plank weighing 15 lb
2. 4x12-in. timber plank weighing 139 lb
3. 3-in. diameter standard steel pipe weighing 76 lb
4. Automobile weighing 4000 lb.

Trajectory calculations were made using a computer code (TMTC) which has successfully simulated observed missile trajectories, damage paths and impact points from several actual tornados. From the results of the trajectory calculations, maximum missile speeds and the associated heights were established, and from these, design recommendations were then formulated. A secondary objective was to provide the maximum tornado wind speed and atmospheric pressure change which could occur at the 70 ft elevation of the roof of the 105-N Building.

For purposes of the study, missiles 1 and 4 were added to the two missiles listed in the Hanford Site DBT criteria.

2x4 inch Timber Plank

The 2x4-in. timber plank is typical of the most common missiles found in tornad damage paths. They originate from the destruction of houses, barns, sheds, mobile homes, one- and two-story apartment or motel buildings, or other timber or timber-roofed structures. In the 100-N Area aerial photos suggest the sourc of the 2x4 timber missile could be from the destruction of office trailers, storage sheds, single-story timber-roofed buildings, or materials from construction sites.

4x12 inch Timber Plank

Sources of 4x12-in. timber planks include boards used for scaffolding, loose timbers used for skids, pieces from heavy equipment packing crates, and heavy timbers from roofs of shed and storage buildings.

3 inch Diameter Steel Pipe

Missiles typical of 3-in. diameter standard steel pipes are pieces of scaffolding, appurtenances on storage tanks, vent pipes, street light poles, electrical conduit, etc. They could be loose, loosely attached, or securely connected to a structure.

Automobile

The 4000-lb automobile represents various vehicles including pickup trucks, for lifts, and light duty cranes. The automobile is not readily modeled by the TMT computer code because it does not become airborne like the other three missiles Automobiles tend to slide, roll or tumble. Frictional forces that develop as t automobile comes in contact with the ground tend to retard movement. These frictional forces were not considered in the TMTC code, but some indication of the automobile movement was obtained.

Model for Hanford Site

The Standard Design Criteria (SDC 4.1) (Ref. 27) provides for a maximum horizontal wind speed of 165 mph for reactor facilities. The standard further requires that this wind act over the full height of a building. The 165-mph design wind speed in SDC 4.1 is interpreted as a peak gust (2-3 seconds) horizontal wind velocity component at 33 ft above ground level. The tornado model used in the study for the trajectory calculations used the 165-mph maximu

wind speed, but provided a more refined approach in that the wind speed was allowed to vary with height and approach zero at ground level. While the practice of assuming a uniform wind over the height of a building is practical and conservative, it fails to recognize the frictional drag that reduces wind speed near the ground. Reduction of wind speed near ground is realistic and consistent with physical observation.

The model is scaled for intensity and spatial considerations by specifying three parameters:

1. Maximum tangential wind speed V_m
2. Translational wind speed T
3. Radius of maximum tangential wind speed R_o.

Selection of the maximum tangential wind speed and the translational wind speed determines the maximum horizontal wind speed of the tornado being modeled. The translational speed is adjusted to be typical of the type of tornado being modeled. The radius of maximum tangential wind speed determines the edge of the tornado damage path, which is defined by the limit of 75-mph winds. The value is selected to be typical of the tornado being modeled. 21

The first consideration in modeling the tornado windfield for the Hanford Site was to achieve the 165-mph horizontal wind speed at 33 ft above ground. The translational speed and the radius of maximum tangential wind speed specified were typical of a tornado of this intensity, based on records of tornadoes from the National Severe Storms Forecasting Center in Kansas City, Missouri. The three parameters selected were maximum tangential wind speed of 130 mph at 33 ft above ground, translational speed of 43 mph, and radius of maximum tangential wind speed of 171 ft.

Aerodynamic Flight Parameters

During motion, the missile either maintains a constant attitude relative to the wind velocity vector or has a tumbling motion. For a body to maintain a constant attitude, the resultant dynamic force would have to be applied at the exact center of missile mass, which is highly unlikely, given the turbulence and velocity gradients in the tornado windfield. Since no stabilizing effect is

produced by the flow of air past the missile, the assumption was made that missiles tumble in a random fashion.

Missile Release Wind Speed

Potential missiles are either attached, loosely attached, or unattached. The wind speed at which a missile is released from its attachment is called the missile release wind speed (MRS). As a tornado translates along its path, the point where the missile is located prior to release experiences wind speeds of varying magnitudes and directions. When the tornado wind speed reaches the missile release wind speed, the missile is then free to be transported by the wind. The direction that the missile moves depends on the direction of the tornado wind vector at the instant of release. This case is much more realisti than simply placing the missile at some arbitrary location within the tornado windfield and allowing it to fly.

The missile release wind speed greatly affects the path taken by the missile and determines if the missile will move at all. If the tornado wind speed is less than the minimum missile release wind speed, the missile will remain stationary. If the missile releases at a small wind speed, the drag force is small; hence, the initial acceleration is small and the missile may not travel very far before falling to the ground. The wind speed at which a missile will release can be calculated, if its form of attachment is known.

The wind speed at which the 2x4 timber plank released was estimated to be 100 mph. Values between 100 and 150 mph were assumed for the 4x12 timber plank. The higher value might be appropriate for the destruction of a heavy roof timb The release wind speed for a steel pipe could range from 100 to 150 mph. It required a value above 100 mph to produce any significant movement of the missile. A value of 65 mph was estimated to cause an automobile sitting at rest with brakes locked to slide along the ground. This value was used as the release wind speed. The missile release wind speeds used in the study are summarized below:

Missile	Missile Release Wind Speed, mph
2x4 Timber Plank	100
4x12 Timber Plank	100 - 150
3 inch diameter Steel Pipe	100 - 150
Automobile	65

Equations of Motion

The tornado missiles were modeled dynamically as three-degree-of-freedom particles. The TMTC computer code numerically solved the equation of motion and calculated missile position, velocity, and acceleration as a function of time. Required input data to the code included the tornado windfield characteristics, missile flight parameters and initial conditions of the missile including location.

21

Tornado Missile Trajectory Calculations

The TMTC code was run a number of times for each of the four missiles considered in this study. The principal variables were initial missile location and missile release wind speed (MRS). The locations of the missiles across the width of the tornado path were varied along with the initial height of the missile above ground. In some cases, as discussed above, the missile release wind speed was varied over a reasonable range of values.

Five items of information were of principal interest:

1. Maximum horizontal missile speed
2. Height at which maximum horizontal missile speed occurred
3. Maximum height achieved
4. Missile speed at maximum height
5. Maximum vertical missile speed.

Three different release heights were tried in the study, 12, 33, and 50 ft. Missiles that released at 12 ft achieved a higher maximum horizontal speed than those that released at 50 ft above ground. This is because as the tornado approaches the missile and the missile releases from its restraint when the wind speed reaches 100 mph, the missile at 50 ft releases at a larger distance

from the tornado center than the one at 12 ft. Because the wind speeds increas with radial distance from the center up to the radius of maximum wind speed, the missile released at 12 ft height is affected by a larger wind speed gradier than the one at 50 ft; hence, it achieves a higher horizontal speed along its trajectory.

Results of the missile trajectory calculations suggest the following conclusions. The 2x4 timber plank could reach a horizontal speed of 135 mph between the 70 and 80 ft level where the vent pipe is exposed. The vertical speed could reach -21 mph (downward) at roof level. Thus, since the 42-in. diameter pipe is circular, it could be struck by the 2x4 timber plank at a resultant speed of approximately 140 mph.

4x12 Timber Plank

For the 4x12 timber plank missile the maximum horizontal speed achieved by the missile was 116 mph. This speed occurred at a height of 12 ft above ground as the missile was descending. The maximum height attained by the missile was 30 ft, but the horizontal speed at that height was only 91 mph.

Like the 2x4 timber planks, the 4x12 missiles reached greater heights when released from 12 ft than when released from 50 ft. However, none of the 4x12 timber missiles reached a height of 70 ft. Therefore, the 4x12 in. timber missile would not reach the height of the vent pipe and thus is not a missile of concern for design purposes.

3 inch Diameter Steel Pipe

The 3-in. diameter steel pipe missile reached a horizontal speed of 60 mph at impact with the ground. To achieve this speed, the missile was released at 70 ft with a missile release speed of 150 mph.

The missile gains almost no altitude after release. It simply falls to the ground under the influence of gravity. While it is falling to the ground, though, the horizontal speed continues to increase. The maximum horizontal missile speed is a function of initial height above ground at release. The pipe missile can reach a speed of 30 mph, if released at 70 ft above ground. The missile would not reach the 73 ft level required to strike the vent pipe.

The 3-in. diameter steel pipe is not a threat to the vent pipe on the roof of Building 105-N. Even if the pipe missile were released at an initial height of 70 ft (roof level), it would not strike the vent pipe.

Automobile

Results of the automobile missile trajectory calculations are summarized in Table 3.3-5. Only eight runs were required. The maximum horizontal speed achieved by the automobile was 12 mph. The auto was released at a height of 12 ft above ground, but this was only done to impose wind speeds at 12 ft on the auto, rather than those at the height of the center of gravity of the automobile. The automobile moves only a short distance under the influence of the winds. If exposed to the windfield for a longer period of time, the horizontal speed would increase. However, this would not be an accurate representation of the true behavior, because the effect of frictional forces between the ground and the auto are not taken into account. The auto is artificially held in the windfield for a few seconds and then allowed to fall to the ground under the influence of gravity. Vertical speed components have no meaning. Under the circumstances described, the automobile could reach a horizontal speed of 15 mph. It remains in contact with the ground and has no meaningful vertical speed. The automobile is of no concern relative to the vent pipe. 21

The results of the missile trajectory calculations are summarized in Table 3.3-6 and were extracted from Tables 4, 5, and 6 of the study.

Based on the above it was concluded that only the 2x4 timber plank is capable of reaching the height of the exposed 42-in. diameter vent pipe. The other three missiles considered (4x12 timber plank, 3-in. diameter steel pipe and automobile) would not be lifted to the roof level of Building No. 105-N by the DBT wind. Exposed pipe should be designed to resist the 2x4 timber plank traveling at approximately 140 mph. This value represents the resultant of the horizontal and vertical components of the missile (135 mph and -21 mph, respectively), which is the worst case, since the pipe is circular.

Maximum missile speeds at elevations below the roof level are summarized in Table 3.3-7.

The maximum atmospheric pressure change (APC) calculated for the Hanford Site DBT was 82 lb/ft^2 at 70 ft above ground. This value is about one and one-half times greater than the 54 lb/ft^2 specified in SDC 4.1. It was recommended that criteria for the DBT should include an ambient pressure decrease of 82 lb/ft^2 at a rate of 30 $lb/ft^2/sec$, hold for 2.7 seconds and return to ambient at the same rate. Appropriate combinations of aerodynamic wind pressure and APC shoul be considered in design of building walls and roofs. These combinations includ wind alone if the building is vented, wind plus one-half maximum APC or APC acting alone if the building is sealed.

As discussed in Section 6.2.5, these recommendations were used to confirm the ability of the mixing duct for the Hydrogen Mitigation Project (H-791) to withstand the DBT and maintain its structural integrity.

3.3.3.2 Effects Of Tornado-Generated Missiles On Critical Plant Facilities

Tornado-generated missiles may cause damage of the following types: (1) local penetration or perforation of walls and roofs, (2) instability effects on the whole structure caused by impact of massive objects, and (3) damage to equipmer located beyond the barrier wall. Sometimes, secondary missiles may be generate inside a protected area which may impact and damage critical equipment. Spalli and damage to inside equipment can be prevented by making the barrier walls sufficiently thick. In general, the first seven missiles described in Section 3.3.3.1 are of the penetration type and have masses which are relatively small with respect to the target mass. Missile type 8 was used in some analyses to evaluate instability effects on structural components of buildings and large pieces of equipment. Analyses for determining missile penetration of 100-N building components are preserved in Reference 1. A result of these analyses are presented below.

Concrete wall thicknesses required to prevent missile penetration and the production of secondary missiles are presented in Table 3.3-7. The thickness values are based on missile penetration tests performed on similar types of missiles and published in the technical literature. In general, initial missi velocities utilized in the tests were equal to, or greater than, those specifi in Section 3.3.3.1 for a 165-mph tornado. When test velocities were significantly greater, the thickness values reported in the literature were

scaled down by the square root of the respective velocities. The Rotz equations (Ref. 21) determine the threshold thickness required to prevent concrete spalling by solid steel and pipe missiles. These equations are based on data obtained from missile penetration tests described in Ref. 21 and 23. Design thickness to preclude spalling should be 10% to 20% greater than the threshold values predicted by the Rotz equations. | 2

It is noted from Table 3.3-7 that all of the missiles, except No. 9, are stopped by 7.5 in. or less of concrete. A 12-in. concrete slab is required to stop missile No. 9, consisting of a 4,000-lb automobile traveling initially at 50 ft/s (34 mph). Penetration tests of missile types 1, 3, and 4 indicate that the stopping power of a reinforced concrete block wall, 8 in. thick with interior spaces filled with mortar, is not quite as effective as a solid concrete wall, 6 in. thick (Refs. 24 - 26). | 21

As described in Section 3.8, the concrete walls and roofs of the 105-N and 109-N confinement structure are greater than 12 in. thick, as are the walls around the inner and outer-rod rooms and their roofs. The wall of the ECCS bypass structure and the walls and roof of the 105-N control room are 12 in. thick. Thus, none of the missiles identified in Section 3.3.3.1 would be able to penetrate into the above building areas, nor cause significant spalling of interior concrete surfaces.

As discussed in Section 3.3.2.2.1, the Spectrum II missiles could penetrate the walls of the 181-N superstructure as they are constructed of hollow concrete blocks, especially if the particular missile hit between rows of reinforcing steel. However, the energy of a missile would be mostly dissipated when penetrating the wall, and before it could penetrate the corrugated-metal fire-protection walls surrounding the emergency diesel engines. Penetration would be greatest for the 3-in., 76-lb pipe missile travelling at 80 ft/s when compared with the 6 in. diameter, 267-lb pipe at 25 ft/s or any of the others. One of the slender missiles might enter the 181-N river water pump house if it hit the vent grill end-on. This grill is located near the roof of the west wall of the superstructure. If aimed exactly in the right direction, the missile could damage the controls and governor of one of the diesel engines. Such an event has a low probability of occurrence because the missile would probably | 21

have to come from across the river, which is about 2,000 ft wide opposite from the N Reactor plant. In addition, the west (left) bank of the river is a deser and is not inhabited. It is also concluded that both diesel engines located in the 181-N river pump house would not be damaged by a single missile.

Considerable damage to the 181-N superstructure could be caused by missile No. 9, typified by the 4,000-lb automobile traveling at 50 ft/s (34 mph): If i hit the concrete-block wall between concrete columns, it would penetrate the wall and be stopped by impacting one of the diesel engines and/or its concrete pedestal. Since a rolling/tumbling mode over the ground is more likely than being completely airborne, the missile would be stopped by the concrete pedesta and cause only minor damage to the diesel engine supported on top. However, if airborne, the automobile could hit the engine head-on and cause damage to the engine which might prevent it from operating. If the missile should hit one of the concrete columns, it would be stopped thereby, although damaging the columr and possible causing it to "jack-knife." Loss of one column would not cause the roof or the superstructure of the 181-N building to collapse (Ref. 1). From the above discussion, it is evident that a Spectrum II missile could put one of the emergency cooling system diesel pumps out of service if it were to impact the 181-N building at a vulnerable location. However, as discussed in Section 3.3.2.2.1 the occurrence of two such missiles, causing both diesel engines to shutdown, is not considered credible.

As discussed in Section 3.3.2.2.9, the batteries, power panels and conduit located on the main floor of the 182-N Building were potential targets of missiles penetrating the concrete superstructure walls. It was also determine that probability of a missile on this equipment was less than the probability of the tornado occuring which is less than (10^{-7}) per year.

As discussed in Section 3.3.2.2.8, the ECCS supply line is exposed to potentia missiles at the west side of the 105-N Building and the most recent (Ref. 54) 1988 analysis of tornado missile impact on the exposed ECCS piping concluded the line would remain stable and intact during and after impact. Also, the only missile (the 2x4 missile) determined to attain a height above the roof elevation discussed in Section 3.3.3 (Ref. 55) would not damage the exposed ECCS piping. This conclusion supports the following analysis reported earlier

Should the exposed section of ECCS piping be hit by a tornado-generated missile, except possibly for missile No. 9, the missile would most likely be deflected, causing only minor damage to the pipe and its supports. In the unlikely event of a direct, head-on collision, the wood missiles would shatter, except that the utility pole would tend to rebound. | 21

Based on the Ballistic Research Laboratory (BRL) formula (Ref. 21), it can be shown that none of the steel missiles have the capability of penetrating the 0.5-in. wall of the 30-in. pipe. For example, considering the 6-in. pipe missile and using the BRL formula, $tp = E^2/3/672D = 0.127$ in., where t_p is the thickness of steel plate required to stop the 6-in. pipe missile, E is the kinetic energy (13,500) of the missile in ft-lb, and D is the diameter of the missile (6.625) in inches. The steel plate in the BRL formula is supposed to be rigidly backed up. In this example, the wall of the 30-in. pipe is not rigid but would deflect about 7-in. in order to absorb all of the kinetic energy of the impacting missile. However, the actual deflection would be less than 7 in. because the kinetic energy would be dissipated by local deformation of the missile, penetration of the pipe insulation, deflection of the pipe wall and bending along the pipe length, plus rebound energy. Thus, the missile would not penetrate the pipe nor cause sufficient local deformation to prevent it from functioning adequately during and after the incident. | 21 | 2

At the point where the ECCS pipe comes out of the ground at the west side of the 105-N, it is partially protected by the concrete foundation of the gas-storage shed and the basement-access well. Thus, missile No. 9 would have to be airborne to produce maximum impact on the exposed, vertical section of the ECCS pipe. The direct, head-on, impact of the 4000-lb automobile would cause local deformation of the pipe wall as well as bending deflection of the vertical section of pipe. If all of the kinetic energy of the impacting missile is dissipated in deforming the pipe wall, the depression would be about 17 inches deep over a 4-foot section of pipe. On the other hand, if the missile energy is dissipated by bending-deflection of the 20-foot vertical span length, the maximum pipe deflection would be approximately 3.5 inches. Actually, some of the kinetic energy would be utilized in crushing and deforming the front of the automobile, some in deforming the pipe and some in plastic-bending of the vertical pipe in beam action. The amount of energy absorbed in each failure mode of energy | 21

absorption is not known. However, a conservative estimate of the maximum depression of the pipe wall is about 12 inches, based upon the above values expressing upper-limits for the two pipe modes considered. Thus, it is conclu that none of the missiles described in Section 3.3.3.1 is capable of penetrati the ECCS pipe wall, nor damaging the ECCS supply piping to the extent that its capability for transporting water would be reduced significantly.

The effects of tornado wind forces on the spent fuel storage basin is discusse in Section 3.3.2.1, which included the possibility that some of the insulated-metal-siding would be blown from the northwest wall. This event would expose the storage basin to the full impact of the tornado-generated missiles describ in Section 3.3.3. However, spent fuel elements are stored inside the two compartments, under at least 14 feet of water. The walls, which are 2 to 3 fe thick, would not be penetrated by any of the missiles, especially when it woul most likely strike the walls at an angle and rebound. Major impact would be caused by missile No. 9 dropping at an angle and hitting the top edge of one o the outer basin walls head-on. Other than causing minor cracking of the concrete, no significant structural damage is likely to occur. The spent-fuel elements, contained in canisters and stored under water in the two compartment are not likely to be damaged. In case one or more fuel elements are damaged o crushed by a particular missile, any loose radioactive materials would still b contained within the storage basin. In the unlikely event that missile No. 9 hit one of the structural steel columns which support the roof, it would plastically bend the column and possibly shear the anchor bolts. However, failure of one column would not cause the rest of the structure to fail or the storage basin roof to collapse. Thus, damage to the metal storage basin and its superstructure, imparted by tornado-generated missiles, would not prevent the storage basin from adequately performing its design function during the incident.

3.3.4 Design Basis Volcanic Ashfall Loading

A volcanic ashfall is considered to be a potentially occurring natural phenomenon in the category of tornado and earthquake. All new Category I structures, systems and components are required to meet the design basis ashf criteria (Ref. 28) as follows:

Thickness of ashfall:

6 inches uncompacted
3 inches compacted

Duration and rate of ashfall:

Maximum duration of ashfall	24 hours
Maximum ashfall rate	0.45 inches per hour for 6 hours
Average ashfall rate	0.25 inches per hour

Design ashfall loading: 13

16 psf (pounds per square foot) dry, compacted
20 psf wet, compacted

Density of ash:

Dry, loose	31.2 pcf (pounds per cubic foot)
Dry, compacted	62.4 pcf
Wet, compacted	80.5 pcf

Ashfall loading is to be considered a live load and is to be combined with other normal live loads. Dry ashfall in combination with snow loading will normally produce the highest loading for design. Ashfall loading is not to be combined with earthquake or wind loads.

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TABLE 3.3-1

DBT WIND PRESSURES

Surface	Cp	q(pst)'f
Windward Wall	0.8	56
Side Walls	-0.7	-49
Leeward Walls		
North & South Walls	-0.5	-35
East & West Walls	-0.42	-29
Roof	-0.7	-49

q_z = 70 lb/ft^2

Positive pressure acts inward on the structure

21

TABLE 3.3-2

MASONRY FACTORS OF SAFETY AGAINST THE DBT

Member	Masonry Compression Stress (psi)	F.S.[1]	Reinforcement Tension Stress (psi)	F.S.[2]	Masonry Shear Stress (psi)	F.S.
North/South Wall	309	3.6	14,500	2.5, 2.0	23	2.3[3]
Bond Beam, North/South Wall	224	4.9	9,200	3.9, 3.1	11	4.7[3]
West Wall	274	4.1	9,400	3.8, 3.1	--	1.7[4]
Bond Beam, West Wall	143	7.8	12,300	2.9, 2.3	--	--

Notes:

1. Allowable masonry flexural compression stress = 110 psi.
2. Reported factors of safety are: (vs. 36,000 psi allowable, vs. 28,800 psi allowable)
3. Allowable shear stress = 52 psi.
4. Based on in plane-out of plan interaction.

TABLE 3.3-3

DRAG COEFFICIENTS AND TORNADO PRESSURE LOADS

	Load	CLOSED BUILDING (h:b:L=1:1:1) CD	CLOSED BUILDING (h:b:L=1:1:1) p (lb/ft^2)	CLOSED BUILDING (h:b:L=1:4:4) CD	CLOSED BUILDING (h:b:L=1:4:4) p (lb/ft^2)
Walls					
Windward	a	0.9	70.5	0.9	70.5
Leeward	b	-0.5	-39.2	-0.3	-23.5
Sides	c	-0.6	-47.0	-0.4	-31.3
Roof Uplift					
Maximum	d	-0.7	-54.8	-0.8	-62.6
Ambient WP*					
Full WP	e		-108.0		-108.0
W P/2	f		-54.0		-54.0
Combined Effects					
Total Structure Load		1.4	109.7	1.2	94.0
Windward Wall	a		70.5		70.5
Windward Wall	e		-108.0		-108.0
Leeward Wall	b & f		-93.2		-77.5
Side Walls	c & f		-101.0		-85.3
Roof**	d & f		-108.8		-116.6

21

*Bursting effect due to external pressure drop of 0.75 psi, which is equal to (-) 108.0 lb/ft^2. WP/2 occurs simultaneous with maximum tornado wind velocity.

**Not including dead load. Local wind pressures at ridges and eaves will be greater than average values listed herein.

TABLE 3.3-4

MAXIMUM HORIZONTAL AND VERTICAL SPEEDS ATTAINED BY AUTOMOBILE

Initial Missile[1] Location, ft			Missile[3] Release Speed	Maximum Horizontal Speed		Maximum Height[4]		Maximum Vertical Speed	
x	y	z[2]	(mph)	Speed (mph)	Height (ft)	Height (ft)	Speed (mph)	(mph)	Comment
300	200	0	65	10	0	0	0	--	
300	150	0	65	8	0	0	0	--	
300	100	0	65	7	0	0	0	--	
300	50	0	65	10	0	0	0	--	
300	0	0	65	12	0	0	0	--	
300	-50	0	65	11	0	0	0	--	
300	-100	0	65	11	0	0	0	--	
300	-150	0	65	11	0	0	0	--	
300	-200	0	65	12	0	0	0	--	

[1]Value of x is arbitrarily selected at some distance ahead of the tornado.

[2]Automobile is initially located at ground level, but is assumed to be initially affected by wind at 12 ft level.

[3]Wind speed of sufficient strength to overcome friction and move automobile on the ground.

[4]Automobile, in effect, remains in contact with ground.

TABLE 3.3-5

TORNADO MISSILE TRAJECTORY RESULTS

Missile	Initial Height(ft)	Max. Horiz. Speed Speed(mph)	Max. Horiz. Speed Height(ft)	Max. Height Height(ft)	Max. Height Speed(mph)
2" x 4" Wood Plank	12	143	108	112	140
	33	129	126	126	129
	50	107	98	131	54
4" x 12" Wood Plank	12	116	12	30	91
	50	88	5	67	58
3" Diameter(1) Steel Pipe	12	22	0(2)	12	0
	50	53	0(2)	51	0
	70	60	0(2)	70	0

21

(1) Missile gains essentially no altitude after release.
(2) Maximum horizontal speed occurs at impact with ground.

TABLE 3.3-6

MAXIMUM TORNADO MISSILE SPEEDS VERSUS HEIGHT

Missile	Maximum Horizontal Missile Speed, mph*	Height, feet
2x4 Timber Plank	60	130 - 140
	130	100 - 130
	145	80 - 100
	135	70 - 80
	120	50 - 70
	110	0 - 50
4x12 Timber Plank	75	60 - 70
	80	50 - 60
	90	30 - 50
	120	0 - 30
3-inch Diameter Steel Pipe	30	70 - 71
	45	60 - 70
	50	50 - 60
	55	20 - 50
	60	0 - 20
Automobile	15	0 - 5

*Missile speeds are rounded up to nearest 5 mph.

REQUIRED CONCRETE THICKNESS TO STOP MISSILES

	Horizontal Velocity (fps)	Kinetic Energy (Ft-lb)	Required Concrete (t = in.)	Method of Qualification	Reference Number
1. Wood Plank (2"x12")	147	18,000	6	Test at 125 fps	22, 24
2. Plywood Sheet	220	42,100	6	Test at 190 fps	22, 24
3. Steel Siding	220	75,200	6	Test at 221 fps	22, 24
4. Wood Plank (4"x12")	110	21,600	6	Test at 125 fps	22, 23
5. Steel Rod	24	80	2.0	**Rotz Equation	22
*6. Steel Pipe (6")	55	13,500	7.4	**Rotz Equation	22
*7. Utility Pole	55	52,800	6.2	Test at 204 fps and t $V_{1/2}$	22
*8. Steel Pipe (12")	35	14,300	7.0	**Rotz Equation	22
*9. Automobile	50	155,300	10 12	Time History Analysis Calculations and Test at 76 fps and t V_2	14, 16 23

*For heights less than 20 ft above ground.

**Rotz Equations:

$$T_{ss} = \frac{15.5\ W^{0.4}\ V_s^{0.5}}{f'_c\ D^{0.2}}$$

$$T_{sp} = \frac{5.42\ W^{0.4}\ V_s^{0.65}}{f'_c\ D^{0.2}}$$

T_{ss} = Required concrete thickness (in.) for solid steel missile

T_{sp} = Required concrete thickness (in.) for steel pipe missile

W = Weight of missile (lb)

D = Missile diameter (in.)

V_s = Missile striking velocity (fps)

f'_c = Concrete compressive strength (psi)

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3.4 WATER LEVEL (FLOOD) DESIGN

This section describes provisions made for flood protection in the design of Seismic Category I facilities. Related information is presented in Section 2.4, including material on the following subjects: hydrological site information, river elevations under flood conditions and requirements for flood protection. 2

3.4.1 Flood Protection

N Reactor plant is located on the right (south) bank of the Columbia River, which has an average ground elevation of 453 ft mean sea level (MSL). The river bed is clearly marked in the terrain and flows between a high right bank and a low left bank, which slopes gently upward to the west at about 1% grade. The 7-mi stretch of river, upstream and downstream from N Reactor, is essentially straight. Thus, the main flood flow is expected to be parallel to the river channel as it passes by the site. 2

N Reactor plant was designed originally to accommodate the maximum flood that could be expected within 100 yr and which would produce a maximum river elevation at the site of 415 ft MSL. Seismic Category I structures, systems, and components are identified in Section 3.2.1. Except for the 181-N river pump house (intake structure), which is located at the river's edge, all plant facilities are located well above the 100-yr flood level. The main, or operating, floors of the various buildings are located at elevation 454.5 ft MSL. Flood waters would have to exceed this level in order to enter the buildings and their respective basements, except possibly for backflow through a few basement floor drains from clean (nonradioactive) building zones. However, backflow is not possible unless flood levels exceed about elevation 430 ft MSL, corresponding to the elevation of the lowest basement floor. The approximate size and elevations of the foundations for Seismic Category I structures are presented in Table 3.4-1. From these data it is evident that maximum groundwater levels associated with flood conditions will have little or no effect on these foundations.

The 181-N river pump house was designed conservatively to accommodate a maximum river elevation of 415 ft MSL, corresponding to the 100-year flood level. Stability of the structure was established in the original design calculations by assuming that the 100-yr flood occurred at the same time as the design earthquake and when one pump well was dewatered, resulting in maximum buoyancy effects. The operating floor of the 181-N building is at elevation 421 ft MSL. Design features are described in Section 3.8.4.1, including riprap provided to protect adjacent river banks against flood erosion. Emergency procedures for shutting down the nuclear reactor in the event of flood conditions are described in Sections 2.4 and 13.3.

3.4.2 Probable Maximum Flood Conditions

As discussed above, the N Reactor plant is constructed on the high, right bank of the Columbia River. This location is well above the 100-yr flood level, which was used in the original design of the plant, including the design of the 181-N pump house. In 1972, a new probable maximum flood (PMF) was established for the site by the Corps of Engineers having a peak flow of 1,440,000 ft^3/s. See Section 2.4.3. This postulated flood would produce a river level of elevation 423 ft MSL at the site, which is 8 ft higher than that produced by the 100-yr flood. However, this added design flood level is well below the foundations of the Category I structures located on the high, right bank of the river. See Table 3.4-1. By definition, the PMF and the maximum design earthquake (SSE) do not occur at the same time.

The Corps of Engineers in establishing the PMF did not consider any potential earthquake-induced water release. A preliminary effort to quantify the risk of earthquake and random dam failures was performed by Lawrence Livermore National Laboratory and is described in Section 2.4.4. This preliminary assessment indicates the potential for exceeding the PMF water level at the plant site, including the potential for exceeding site grade elevation of 454.4 ft MSL. The assessment also indicates that approximately 20 hr advance warning would occur which would permit time for reactor shutdown.

During the PMF the foundation of the 1304-N emergency dump tank (EDT) would likely be inundated by groundwater but not the tank itself. This tank is located near the top of the incline which forms the right bank of the river. The bottom of the circular foundation mat is at elevation 418.75 ft MSL; whereas, the bottom of the tank shell is at elevation 423 ft MSL. The tank is protected to elevation 431 ft MSL (minimum) by a reinforced concrete ring wall and 30 ft, or more, of compacted gravel backfill. Thus, the emergency dump tank would survive the PMF without damage.

The 181-N pump house, which is located at the edge of the Columbia River, is partially underwater during normal periods of river flow. The operating floor, which is also the top deck of the 181-N substructure, is at elevation 421 ft MSL. Therefore, the substructure would be completely underwater during the PMF. This condition is discussed below.

The river bed, west of the 181-N pump house, has an average elevation of about 360 ft MSL. Allowing for water on the west flood plain, the river would have an average cross-section of about 140,000 ft^2 during the PMF, corresponding to a maximum flood elevation of 423 ft MSL. Thus, the average velocity of water near the site, at flood stage, would be about 10.3 ft/sec. Using Bernoulli's equation, the corresponding hydrodynamic pressure acting on the south face of the 181-N building would be: $p = w(V_0)^2/2g = 62.4(10.3)^2/64.4 = 103$ lb/ft^2. This is a conservative value because the velocity of the current near the shore will be less than the velocity near the middle of the stream. In addition, the pump intake structure, located upstream at the WPPSS plant, will tend to deflect stream-flow away from the bank and towards the middle of the river. Due to its proximity downstream from the WPPSS pump structure, the 181-N building will be in a protected eddy created by the WPPSS pump structure. This condition will also minimize the scouring action that would otherwise occur along the river bank around the 181-N pump house and downstream in the vicinity of the emergency dump tank.

The hydrodynamic pressure (103 lb/ft^2) would produce a resultant force of about 331,000 lb on the south wall of the 181-N substructure. This magnitude

of lateral force is less than 1/20 of the horizontal earthquake reaction associated with the 0.25 g earthquake used in seismic calculations presented in Reference 1. These calculations demonstrate that the 181-N pump house would withstand an earthquake producing a maximum horizontal ground acceleration of 0.25 g and a maximum vertical acceleration of 0.188 g acting simultaneously with an assumed flood elevation of 415 ft MSL. When the design flood elevation is increased to 423 ft MSL, buoyancy effects on the 181-N building increase by about 969,000 lb, which is about 15% of the vertical seismic reaction provided for in Reference 1. Thus, it is concluded that the 181-N river pump house will remain in place and not be damaged during the PMF.

The 181-N superstructure is constructed of reinforced-concrete components and block-filler walls. It is located on top of the operating floor of the substructure and houses two emergency core cooling system diesel engines, low lift pumps and auxiliary equipment, a third ECCS pump, and diesel drive located in the adjacent 181-N Annex. It stands on the 181-N river pump house operating floor and is attached to the southwest corner of the 181-N superstructure. The exhaust system, intake air filter, and a ventilator for the diesel engine are located on the annex roof which is 17 ft above the operating floor. The superstructure will eventually contain 2 ft of water during the PMF, via leakage through the door seals. However, there is nothing inside which will be permanently damaged or prevent operation of the emergency diesels and low lift pumps if the operating floor is covered with 2 ft of water. The superstructure will not be subjected to the full dynamic effects of the postulated flood; as it is protected by the presence of the security fence, circulating-water pumps, piping, supports, and auxiliary equipment located on the operating floor to the south and west of the superstructure building. The air intakes to the emergency diesels are located outside the superstructure building, the lowest portion of the intake piping being at elevation 424.75 ft MSL. These air intakes, it is judged, would not be affected operationally by wave action (2-ft maximum), since in their protected location they would not be exposed directly to maximum wave and flood conditions. The concrete block walls, doors, and the reinforced concrete framework of the superstructure will not be damaged structurally by the PMF

and will withstand dynamic forces caused by 2 ft of flooding, including floating debris.

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3.5 Missile Protection

This section postulates and defines potential missiles and the sources of energy which create them. The events which generate these postulated missiles are considered singular, causing production of only a single missile. The postulated missile that causes failure of a safety system is considered to be a single failure. No other failures of that safety system or other safety systems are assumed to simultaneously occur except when two systems share a common component that may be damaged.

Some of the postulated missiles are not credible. These noncredible missiles are identified and the justification for not being a credible source of damage is given.

Missiles produced from pipe breaks are discussed in Section 3.6.

The original plant design criteria did not specifically call for protection against missiles. However, the designers did apply basic principles regarding location of equipment to avoid potential interaction upon failure of a piece of equipment. The design provides for redundant systems, separation of components, and location and orientation of rotating equipment to reduce potential interaction. The result is that the reactor is inherently safe from onsite generated missiles from the standpoint that plant missile damage from a single event will not prevent a safe shutdown. 21

3.5.1 Missile Selection and Protection

3.5.1.1 Internally Generated Missiles (Outside Confinement Zone I) 21

The plant systems and components which need to be protected from onsite missiles are those that are necessary for achieving and maintaining a safe shutdown of the reactor. Safe shutdown of the reactor for the purpose of this discussion is defined as shutdown of the reactor to a subcritical

condition with normal recirculation or emergency core cooling. Key structures necessary for safe shutdown (as shown in Figures 3.5-1 and 3.5-2) are:

Safe shutdown recirculation cooling	Safe shutdown emergency core cooling
105-N Reactor Building	105-N Reactor Building
109-N Heat Exchanger Building	181-N River Pump House
153-N Switchgear Building	182-N High-Lift Pump House
181-N River Pump House	
182-N High-Lift Pump House	
184-N Plant Service Boiler House	

Structures containing rotating equipment postulated to be a potential source of missiles are the 185-N, 182-N, 184-N and 109-N Buildings. Systems requiring protection from missiles are the reactor coolant systems [reactor cooling system (RCS) and emergency core cooling system (ECCS)] and the reactivity control systems [horizontal control rod (HCR) system, graphite and shield cooling system (GSCS), and ball safety system (BSS)]. The seismic classification of the structures is given in Section 3.2 and the safety system components are listed in Tables 3.2-1 and 3.2-2.

3.5.1.1.1 Potential Rotating Equipment Missile Sources

Rotating equipment considered as possible sources of missiles are those subject to loss of control or load. These are the Hanford Generating Plant Turbine Generators, the primary pumps/drive turbines, the ECCS low-lift diesels, the ECCS high-lift diesels, the fog Spray diesels, the onsite turbine-generator and the emergency seal water turbine.

The Washington Public Power Supply System (WPPSS) Hanford Generating Project electrical generators are located in the 185-N Building about 650 ft from the reactor, and are oriented so the plane of rotation is parallel to the reactor and its support facilities. Postulated missiles from the drive turbine are discussed in Section 3.5.1.3.

Six RCS recirculation pumps are located in the 109-N Building and oriented with their planes of rotation parallel to the nearest reactor face. In addition, each pump is located within steam-generator cells with 5-ft thick reinforced concrete walls. The turbine drives are adjacent to the south end wall of the cell. Thus, the wall is between the turbines and the reactor coolant pumps and piping. An additional wall separates the 109-N and 105-N Buildings. Postulated missiles from the drive turbines are discussed in Section 3.5.1.3.

Three ECCS low-lift pumps and their diesel engine drives are located in the 181-N Building about 1,000 ft from the 109-N Building, and three high-lift ECCS pumps as well as two fog spray pumps and their diesel engine drives are located in the 182-N Building about 250 ft from the 105/109 Building complex. Diesel engine drives for engineered safety feature systems are provided with independent governors. A no-load engine runaway would require shear of the coupling and be accompanied by failure of the two overspeed controls. The pumps and diesels are located in the basements of their respective buildings under re-enforced concrete ceilings. Postulated missiles are discussed in Section 3.5.1.3. 21

The onsite ac plant service generator set is located in the 184-N Building, and is also oriented such that its plane of rotation does not intersect the reactor or any of its ECCS components. The probability of the generator overspeeding is less than that of the driving turbine (see Section 3.5.1.3) because of coupling failure. The generator is judged to be a low probable missile source for damaging the reactor. 21

The emergency seal water turbine is located in the basement of the 109-N Building. Postulated missiles from this source are discussed in Section 3.5.1.3.1.

There are three potential sources of other onsite missiles; other rotating equipment, pressurized equipment and systems, and stored kinetic energy. Only those facilities which might produce a missile that could affect reactor coolant circulation, the ECCS, the BSS or the HCR system will be identified. 21

Rotating equipment powered by electrical motors does not have a potential for overspeed missile production because electric motor speeds are fixed by design in multiples of power supply frequency.

A small (187 kVA) diesel-driven generator located on the first floor of the 182-N Building is used for auxiliary power to motorized valves. It is on the east side of the Building and separated from the ECCS equipment in the west side of the basement by a steel reinforced concrete floor. The rotational plane is in the north-south direction so that a postulated missile from this source would not strike the ECCS or RCS. In addition, this is considered a highly improbable missile source because of its infrequent operation.

Pressurized equipment and systems have been fabricated and installed in accordance with original construction specifications or applicable ASME Pressure Vessel Codes. Such equipment is periodically tested and inspected to assure its integrity. Thus, there is a low probability for missiles from this source. Years of missile-free operation attest to this fact. Protection is provided primarily by redundance of cooling and reactivity control systems.

Potential missiles from stored kinetic energy sources consisting of nuts, bolts, and studs are a very low probability. These items have been installed under procedural control to assure safe stress levels as prescribed by the ASME Boiler and Pressure Vessel Code. This, along with periodic maintenance and inspections and the small mass of these fasteners, make damage of critical systems from this missile source inconsequential.

3.5.1.1.2 Potential Missile Targets

The BSS (which is within confinement Zone I) and the HCR safety system (which is outside confinement Zone I) are two separate reactivity control systems, each capable of rapidly achieving and sustaining reactor subcriticality. Since they are physically separated by thick steel reinforced concrete walls with the BSS on top of the reactor and the HCRs on both sides of the reactor, a single missile, either from inside or outside confinement Zone I, would not cause both systems to fail. In addition, both systems are provided

missile protection by being enclosed within reinforced concrete on all sides, top, and bottom. Additional discussion is given in Section 3.5.1.2. There is no rotating equipment within these enclosures capable of generating missiles. 21

The two separate coolant systems (RCS and ECCS) share only the distribution piping on the front and rear of the reactor within confinement Zone I. They also share supporting components at the 181-N River Pump House and the 182-N High-Lift Pump House, but their components are physically separated. The ECCS is described in Section 6.3, and the RCS is discussed in Chapter 5. The RCS has redundant power sources for recirculation and redundant heat dissipation systems. The protection for the coolant system is fully described in Section 3.5.1.1.3. 21 2

The switchgear for the onsite and offsite electrical power supplies, which are described in Chapter 8, are in the 153-N Building. This building is partitioned with onsite ac switchgear in the west half and offsite ac switchgear in the east half. The 153-N Building is located adjacent to and perpendicular to the plane of rotation of the plant service TG; however, the 153-N Building is a thick-walled concrete structure minimizing the probability that a missile from the turbine would impact with onsite or offsite plant switchgear. Moreover, the equipment powered by this facility is not required by the ECCS or reactivity control systems for safe shutdown.

3.5.1.1.3 Coolant System Protection

The 181-N River Pumping Facility houses three ECCS low-lift diesel-driven pumps; two are in the reinforced concrete block building on the otherwise exposed deck of 181-N Facility and one is in the 181 Annex. Four electrically-driven circulating raw water (CRW) pumps are also in the 181-N Facility. These motors are mounted on the exposed deck, the pumps are each in separate below-deck compartments, as are each of the two ECCS pumps. Potential missile sources are the diesel engines and their starting air supply piping. The compressed air supply to the diesels is a part of the ECCS system; however, each engine has an air accumulator with sufficient storage capacity to assure starting. The small size piping of the compressed

air supply would generate only small missiles, which might damage one of the two diesels, but is unlikely to penetrate the building wall and damage a CRW pump. Either one CRW pump on offsite power, or two pumps using onsite power, or one ECCS low-lift pump is needed for achieving and sustaining safe reactor shutdown. Similarly, any single flying missile would damage only one system. Consequently, the 181-N Facility contribution to reactor cooling is safe from internally generated missiles by redundancy or existing barriers.

The two CRW pipe lines are underground from the 181-N to the 109-N Heat Exchanger Building, and are thereby protected from missiles. The ECCS line from each 181-N Low-Lift Pump is also underground to the 182-N Building except for the ECCS bypass. The ECCS bypass siphon tank and the demineralized water storage tank are exposed and setting at ground level. However, there are no pressurized systems or rotating equipment as potential missiles sources in the vicinity of the ECCS bypass or the demineralized water storage tank. Moreover, missile damage to the ECCS bypass or demineralized water storage tank would not affect the RCS.

Within the 182-N Building, the three diesel-driven ECCS high-lift pumps, two diesel-driven confinement system fog spray pumps (which also provide backup supply to the graphite and shield coolant system), and five electrically driven high pressure injection (HPI) pumps (four required for normal RCS operation) are located in the basement. The basement floor is 24 ft below grade. The ground-level floor is made of steel reinforced concrete. The above-grade structure is of steel reinforced concrete and, at some wall panels, reinforced concrete block. The ECCS and the HPI systems are thus protected from small airborne missiles. The diesel engines are all provided with a load governor and an overspeed trip to shut off fuel. Diesel engine failures are not known to produce high-energy missiles. For these reasons it is not credible that the HPI system in the 182-N Building could be damaged by a diesel overspeed generated missile. Also, of the five HPI pumps, only two are required for a safe, orderly reactor shutdown without ECCS cooling.

The ECCS pump diesel engines are located side-by-side with the fog spray pump diesel engines. As noted earlier, diesel engine failures are not known

to produce high-energy missiles, but because of their proximity, it is conceivable that one diesel drive could damage another diesel drive. However, the diesel engine governor system should prevent generation of overspeed missiles. Runaway, no-load overspeed potential is highly unlikely because of the hard coupling of diesel engines to the pumps. Damage to either or both the ECCS diesels or confinement system fog spray diesels would not impair normal RCS operation.

A 42-in. diameter line branches from the CRW pump lines, enters the 182-N basement and passes above and between the two fog spray diesel pumps. It then connects to the 30-in. diameter header, which runs the full length of the 182-N Building providing supply to the high pressure raw water pumps, the low pressure raw water pumps and the fog spray, diesel-driven pumps. The 30-in. diameter header is also supplied by the low-lift, diesel-driven pumps. Although the 30-in. diameter supply header passes above the fog spray diesel engines, for reasons noted earlier, it is not considered credible that an engine generated missile could damage the line to the extent that flow out of the break would prevent orderly, safe shutdown on either emergency core cooling from the low-lift pumps or on recirculation cooling with heat dissipation to the CRW system. The 182-N Building is provided with a large sump and a 42-in. diameter gravity drain to the river. This system is sized to prevent flooding from major pipe breaks. 21

The normal reactor cooling is from the 109-N Building with the HPI piping to the reactor recirculation system routed underground from the 182-N Building to the 109-N Building. Thus, the HPI piping between the 182-N and the 105-N Buildings is not subject to damage from missiles generated onsite from sources outside Zone I.

The RCS HPI system piping conforms to ASA B31.1, 1955 and HWS-6897 (Ref. 1). HWS-6897 greatly surpassed the minimum code in effect at the time of installation and compares to the current Section III, "Nuclear Power Plant Components," of the ASME Code. The valves on the high pressure piping are oriented with the stems in a plane parallel to the ECCS pumps and piping so that a potential valve stem missile would not directly strike any ECCS 21 21

components. Any valve stem missile hitting the ECCS piping would be a low angle glancing blow or a high angle hit resulting from an energy dissipating deflection of the stem.

A compressed air system, composed of two compressors and three air receivers, is located on the ground floor of the 182-N Building. This system supplies compressed air to the diesel engines starting air accumulators located in the basement. Except for the piping to startup air accumulators in the building's basement, the basement systems are protected from the supply system by the steel reinforced concrete floor. The air piping in the basement is a part of the ECCS except that each diesel engine has an air accumulator with sufficient air to assure startup without the main air system. The compressed air system is discussed in Section 9.3. A potential missile from the 1 1/2-in. air piping could impact one diesel engine or one HPI pump, or it could impinge on the discharge pipeline from either system. Since the potential missile would be small, it is unlikely but conceivable that it could penetrate the discharge piping of one but not both systems. Only two high-lift, diesel-driven pumps are essential for attaining shutdown on ECCS cooling, so the loss of one diesel-driven pump to a missile is not a critical loss. Further, two diesel-driven pumps would supply sufficient reactor coolant with a punctured line. It is not considered credible that a missile associated with the air system in 182-N Building would incapacitate either the ECCS or the HPI system required for normal RCS operation.

There are two separate ECCS pipelines from the high-lift pumps within the 182-N Building to the reactor in the 105-N Building. The active line is routed underground to the 105-N Reactor Building. Northwest of the reactor, the active line rises vertically from underground, then horizontally across the low roof part of the 105-N and connects to the reactor ECCS manifold from the north. The reserve line is serviceable and intact but valved off inside the 182-N Building. The reserve line provides a redundant emergency coolant route from the 182-N Building to the reactor building. Its route is underground to the 109-N Building and through the 109-N Building outside of Zone I, then passes on the east side of the reactor still outside Zone I and connects to the reactor ECCS header through a check valve.

In close proximity to the active ECCS line where it makes the vertical rise from below ground onto the roof of the 105-N Building, there are 20 cylinders, containing either nitrogen or compressed air, connected to manifolds with spares, all of which are chained in an upright position. Similarly there are four argon cylinders manifolded together plus some spares also chained in an upright position. Such gas cylinders are always stored in a vertical position. However, if the valve should be broken off of one of these cylinders, it might become a rocket type missile, provided enough room were available for the steel cylinder to accelerate to a damaging speed. In this location, the cylinders are too restrained and too close to the ECCS line to attain a damaging speed before striking the line.

From the point where the active ECCS line starts across the 105-N roof to the ECCS manifold, the system is protected by distance and shadowed by structures from potential onsite missiles, except possibly from extremely high trajectory missiles (HTM).

Just prior to entering the 105-N Building at an elevation of 22 ft, the active ECCS line is branched into two lines. Inside the 105-N Building the two active ECCS lines go down to the -15 ft elevation where they connect to the ECCS manifolding. Damage to either the active or reserve ECCS line from 182-N to 109-N from an onsite-generated missile is very improbable. No single missile could damage both lines. The two lines connect to a common manifold outside Zone I. Compressed air bottles, chained in place and additionally restrained following the III/I walkdown program, represent the only potential missile source that could damage the manifold. Manifold damage would not affect normal reactor recirculation cooling. The manifold connects through eight lines to the 16 RCS inlet headers. These headers and connectors attached to the individual pressure tubes are the only components that are common to both the RCS and the ECCS. The area is adequately protected from missiles generated outside Zone I by the thick Zone I confinement walls of steel reinforced concrete, and are not within the general trajectory pattern of any potential onsite generated missiles.

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The primary missile protection of the RCS and ECCS is provided by redundant systems and redundant components within systems. Considerable additional protection is provided by existing structures and having many components below grade.

3.5.1.2 Internally Generated Missiles (Inside Confinement Zone I)

Confinement Zone I contains the reactor, its recirculating coolant system, and the BSS. Each system has a redundant counterpart, the ECCS for cooling, and the rod safety system for reactivity control located outside Zone I. However, certain interconnecting piping and valves between the ECCS supply and the RCS are located within Zone I, and some of the ECCS piping is common to the RCS piping.

The only missile sources inside confinement Zone I are valve components, the circulation pumps and pressurized air/gas cylinders. Most of the large RCS piping is of welded construction, thus, greatly reducing the potential for missiles from kinetic energy sources. The pumps have a very low probability of overspeed failure because the drive turbine controls must fail coincident with loss of water to the pump intake. In addition, the drive turbine would fail before the pump under overspeed conditions. The air/gas cylinders located in the rear face piping areas of Zone I are used for valve operation. These cylinders are safety chained in an upright position and have had additional restraints added due to the III/I walk down program. They are accessible, to be disturbed only during shutdown periods when cooling flow demand is low. Due to the confined space, the cylinders are judged not able to attain damaging jet-propelled speed. Except for the common ECCS and RCS piping at the inlet to the reactor, any missile generated within confinement Zone I can only damage the RCS leaving the ECCS intact. Much of the ECCS-RCS common piping is protected by a thermal barrier during reactor operation, and all of the piping is in areas where the potential for damaging missiles is very low. The common inlet header piping has 16 parallel paths and common pressure tube inlet connector piping has 1,003 parallel paths; failure of one of these inlet headers or a number of inlet connectors would not jeopardize reactor cooling because of reverse flow capabilities from other pressure tubes.

Within the confined area of the BSS there are no potential sources of internally-generated missiles. 2

Even if a missile is postulated, since the BSS is composed of 107 individual ball hopper channel units that are fail-safe, it is not credible that a single missile could incapacitate a sufficient number of hopper mechanisms to render the system inoperative. Furthermore, missiles in the compartment where the BSS is located could not penetrate to the compartments where the HCR system is located (outside of confinement Zone I). Thus, there are no known sources of missiles that could damage the BSS--even if a missile did eject, it would not prevent the BSS from providing the safe shutdown function; and if this occurred, the totally redundant HCR system would shut the reactor down and maintain it in a subcritical condition.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

The 100-N Area complex contains nine turbines. Two are located in the 185-N Building, one in the 184-N Building, and six in the 109-N Building. The axial orientation of these turbines is such that the plane of rotation does not intersect the reactor. Figures 3.5-3, 3.5-4, and 3.5-5 show their proximity to the reactor and other critical facilities. These figures also show the ejection cone angle for postulated missiles from the turbines.

There is one additional turbine located in the basement of the 109-N Building. It is the emergency seal water pump drive turbine which is operated infrequently. This turbine rotates in a horizontal plane 11 1/2 ft below grade. It is a small turbine, 600 hp, as compared with the reactor coolant pump drive turbine of 9,100 hp. No missiles generated by this turbine could damage the ECCS, reactor control room or reactor reactivity control system. Because of its location, small size, rotational plane, and infrequent operation, this turbine does not have the potential for generating missiles which could impact safe shutdown of the reactor. Therefore, there will be no further discussion on this turbine in this section.

3.5.1.3.2 Turbine Characteristics (Station)

The station service drive turbine is an impulse, single-element, single-flow, horizontal-shaft, multistage (5), nonextracting, condensing-type of turbine. It is rated to produce 15,625 kW at a speed of 3,600 r/min using dry or saturated steam at 55 lb/in^2 (gage) to 303°F, 115 lb/in^2 (gage) to 347°F, or 400 lb/in^2 (gage) to 448°F.

The turbine is equipped with two steam control valves, two stop valves, an operating governor, overspeed governor, low vacuum trip, and lubricating oil system. The shutoff valves are designed for quick action on trips. The high speed stop is presently set for 3,845 + 25 r/min and the overspeed trip at 3,935 + 25 r/min.

The pump drive turbines are of the horizontal-shaft, axial-flow, impulse type. These turbines are more completely described in Section 5.4.1. These turbines are designed for operating speeds of 900 to 3,600 r/min with an overspeed trip presently set at 4,200 r/min with steam conditions the same as that for the plant service drive turbine. The turbine drives are directly connected to the RCS pumps, and this arrangement reduces the potential for load rejection and actuation of the overspeed protection equipment. The WPPSS Hanford Generating Project turbine drives are discussed in Chapter 10.

3.5.1.3.3 Turbine Overspeed Protection

All of the turbines are equipped with overspeed controls which can automatically shut off the steam supply to the turbines. The shutoff valves are spring-loaded and are held open by the turbine pressurized oil system. The overspeed trips interrupt this hydraulic supply (pressure) allowing the spring to close the shutoff valve. The drive turbines are hard coupled to the system pumps and are of insufficient power to reach trip speed with the pump connected. In addition, the 184-N plant service turbine is also equipped with an electrical load loss trip.

The present turbine testing program includes the following:

- The pump drive turbines are tested during prestartup runs by manually tripping the overspeed device. The plant service turbine is tested during prestartup by actually overspeeding the turbine with no load on the generator and with the normal governor high speed stop bypassed.

- A functional test of the pump drive turbines overspeed trip is made on an annual basis. The test is performed with the pump disconnected and by manually controlling the turbine speed to trip.

3.5.1.3.4 Turbine Valve Testing

The pump drive turbines are equipped with two throttle and trip (T&T) valves which are given the following tests currently deemed necessary for missile protection:

1. Prestartup Test

   a. With the turbine operating between 400 and 500 r/min, the T&T valve hand control is varied to verify speed control (response).

   b. With the turbines operating at a low speed, the emergency trip buttons are separately activated in both the 105-N control room and on the local control panel to verify trip action.

   c. With the turbine operating between 1,000 and 1,200 r/min, the vacuum breaker valve is slowly opened until the low vacuum trip switch closes the T&T valves.

2. Weekly Tests

   a. Each T&T valve is tested weekly, and the steam shutoff valves are given major inspections every 5 yr. This inspection includes disassembly sufficient for measurement of wearing surfaces and for valve seat roughness, cutting and clearance.

The station service generator turbine drive is equipped with two steam control and two stop or shutoff valves. These valves are given the following tests:

1. Prestartup Test

   a. The turbine, without a load on the generator, is checked for steady operation at 3,600 r/min. Separately, the "emergency trip" button on the control valve casing and the trip switch in the control room are activated to show that each will cause the control and stop valves to close.

   b. A test for low condenser vacuum pressure is conducted in the same manner as for the pump drive turbines.

2. Weekly Tests

   a. The operation of the stop valves is checked by operating the "test-for-stop-valve" until the stop valve starts to close.

The turbine control valve, along with the turbine and other turbine components, is given a major disassembly inspection on a 5-yr frequency.

The environmental steam conditions for all of the turbine valve tests are the same as the operating conditions which are 145 lb/in^2 (absolute) and 303°F saturated for both the pump drive turbines and the plant service generator drive turbine.

3.5.1.3.5 Turbine Missile Probability Analysis

A very significant source of rotational energy in the balance of plant is associated with the turbines. The following probabilities are considered in the determination of the likelihood of a turbine-missile accident leading to

damage of structures, systems, and components required for safe plant shutdown:

P_1 = The probability of the generation and ejection of high energy missiles from a turbine-generator.

P_2 = The probability that a missile strikes a specified target given its generation and ejection.

P_3 = The probability that the missile strike damages its target in a manner leading to unacceptable consequences.

The total probability of a turbine-missile damaging safety-related plant components P_4 is given by the product of the individual probabilities defined above:

$$P_4 = P_1 \times P_2 \times P_3 \qquad (3.5\text{-}1)$$

The following sections provide information on turbine-missile dynamic characteristics, turbine missile trajectories and analytical results.

3.5.1.3.5.1 Probability of Missile Formation (P_1)

Any of the three types of turbines on the reactor site can act as sources of missiles. These are: (1) the two WPPSS Hanford Generating Project 400-MWe 43-in. last-stage, General Electric Turbine Generators (TG); (2) the 15.6-MWe General Electric TG; and (3) the six 10,000-hp Elliott RCS Pump Drive Turbines. Their characteristics have been defined in the following.

3.5.1.3.5.1.1 Washington Public Power Supply System--Turbine Missile Size and Formation Probability (P_1)

The turbine stages for these 1,800 r/min units are of similar size and design as those presently being installed in nuclear service by General Electric.

The probability of missile genesis (P_1) has been determined by GE for turbine rotor failures at or near running speed (a low-speed burst due to brittle fracture), and at high overspeed or runaway conditions (a high speed burst due to ductile yielding). The following annual failure probabilities have been calculated from information contained in a GE Memo Report (Ref. 1):

Annual low-speed burst (below 120% speed)	8.7×10^{-9}
Annual high-speed burst (from 120 to 180% speed)	5.0×10^{-9}
Average annual wheel failure	1.4×10^{-8}

In this analysis, adjacent turbine wheels (or stages) are categorized by GE into three groups (designated stage groups I, II, III) rather than considering individual stages for each turbine flow (Ref. 2). This approach reduces the complexity of calculations and is justified on the basis that hypothetical missiles of significant similarity are produced by adjacent stages. The approximate expected number of missiles generated after a hypothetical nuclear wheel burst is postulated by GE as 16 fragments of four size classes. Each size class contains a specified number of fragments with characteristic shape, size, mass and velocity range.

For the low-speed burst condition, GE considers the last stage to be two orders of magnitude more likely to fail than the next more likely stage. Therefore, stage groups I and II do not make a statistically significant contribution to missiles occurring from low-speed failure and are eliminated from further consideration. For the high-speed burst case, all stages are designed to about the same general stress level, and therefore, are about equally likely to fail by general ductile yielding in the event of runaway.

Appropriate information regarding missile nomenclature, size, shape, weight, energy and velocity is presented in Table 3.5-1 and also in Table 1 and Figure 1 of another GE Memo Report (Ref. 2). The postulated missile characteristics for wheel groups and missile types for WPPSS 43-in. last stage units are shown in Table 3.5-1.

3.5.1.3.5.2 General Electric Station Turbine and Elliott Drive Turbines--Missile Size and Formation Probability (P_1)

The probability of failure of a turbine rotor for all causes was taken as 10^{-4}/yr based on the assessment of Bush (Ref. 3). The probabilities were as follows:

Annual low-speed burst (below 120% speed)	0.8×10^{-4}
Annual high-speed burst (from 120 to 180% speed) Where turbine load is a generator	0.2×10^{-4}
Where turbine load is a direct-coupled pump subjected to 10 h annual test in the uncoupled mode	2.3×10^{-8}

The drive turbines are incapable of attaining 120% speed when coupled to an intact RCS pump. Overspeed due to a governor failure is possible only if the pump impeller was dry or uncoupled from the turbine. The impeller is uncoupled from the turbine annually for test of the overspeed trip. This test, while carried out carefully, appears to provide the significant period of overspeed risk. The period of uncoupled test is taken as 10 h/yr. In recognition of this, the annual probability of overspeed failure of a pump drive vs. a TG drive is reduced by the fraction 10/8766.

Fracture of the turbine discs into 90°, 120°, and 180° segments was considered. The 120° segment had the greatest kinetic energy and was used in the computations. The total energy of a disk fragment is made up of translational and rotational energy. Tests by Westinghouse (Ref. 4) and analyses of General Electric (Ref. 5) indicate that the translational energy is of much greater significance in the ability of the fragment to penetrate the turbine casing, and that the rotational energy could be considered to be dissipated in blade and vane crushing.

The penetration of the turbine shell by a disk fragment was computed using the BRL steel plate equation referenced in a Bechtel report (Ref. 6) where the impact energy was taken as the translational kinetic energy of the

fragment. Penetration of the shell is considered to provide a fragment plug from the shell whose velocity is the same as the residual velocity of the escaped disk fragment. The characteristics of the turbine rotor stages are listed in Tables 3.5-2 and 3.5-3.

3.5.1.3.5.3 Probability of Damage to Missile Targets (P_3)

The damage probability (P_3) is defined as the probability that a missile strike might damage safety-related components or equipment in a manner leading to unacceptable consequences, i.e., the loss of the ability of the component to perform its intended safety function.

The damage of components housed within protective concrete structures and buildings can be related to missile penetration of concrete by a consideration of the potential for spalling (from the side opposite the missile contact surface) and perforation. The missile strike velocity necessary to just start spalling (V_{ss}) is used here as a conservative limit for initiating damage, where the onset of spalling is presumed to threaten the functioning of safety-related equipment. This approach is adequate for plant concrete structures that contain components and equipment required for safe shutdown.

The thickness of concrete that will just be perforated by a missile is given by the Ballistic Research Laboratory formula (Ref. 6 & 7):

$$T = \frac{427}{(f'_c)^{1/2}} \frac{W}{D^{1.8}} \frac{V_s}{1{,}000} 1.33 \qquad (3.5\text{-}2)$$

Where:

T = Thickness of concrete element to be just perforated (in)

W = Weight of missile (lb)

D = Diameter of missile (in). [For irregularly shaped missiles, an equivalent diameter (D_{eq}) is used.] The equivalent diameter is taken as the diameter of a circle with an area equal to the projected frontal area of the noncylindrical missile.

V_S = Missile strike velocity (fps) (assumed equal to the missile ejection velocity)

f_c = Compressive strength of concrete (lb/in^2).

Upon rearranging Equation 3.5-2, the minimum missile strike velocity that will just start perforation (V_{sp}) is given by:

$$V_{sp} = 1{,}000 \frac{(f'_c)^{1/2} D^{1.8} T}{427W}^{3/4} \qquad (3.5\text{-}3)$$

An estimate of the thickness that will just start spalling is given by:

$$T_s = 2T \qquad (3.5\text{-}4)$$

where:

T_s = Concrete element thickness that will just start spalling (in)

The missile strike velocity that will just start spalling (V_{ss}) is then given by:

$$V_{ss} = 1{,}000 \frac{(f'_c)^{1/2} D^{1.8} T}{854W}^{3/4} \qquad (3.5\text{-}5)$$

For conservatism in evaluating damage effects, air resistance is neglected (the missile ejection velocity is assumed equal to the strike velocity) and all missile strikes are considered to be normal to the target surface. In addition, the damage probability of equipment and components located outside protective structures are assumed to be unity over the range of missile velocities capable of causing a strike. This conservatism is introduced for lack of a suitable analytical base for estimating critical damage to these components.

In recognition of the fact that the missile probability of strike, P_2, is computed only for missile velocity values sufficient to cause spalling of protective structure, P_3 is unity over the appropriate velocity range and zero otherwise:

$$P_3 = 1, \text{ for } V_{ss} \geq V \geq V_u$$

$$P_3 = 0, \text{ for } V < V_{ss}$$

where:

V_u = Maximum missile ejection velocity

V = Missile velocity

For a given missile impacting a specific concrete barrier, the minimum damage velocity (V_d) is defined as the normal strike velocity component necessary to just cause spalling (V_{ss}), as determined from Equation 3.5-5.

V_d may assume the larger of the following values:

V_{ss} = Strike velocity to just start spalling

V_r = Minimum velocity necessary to reach the target (for components and equipment with little or no protection).

3.5.1.3.5.4 Missile Trajectories

Missile trajectories are discussed in the following section.

3.5.1.3.5.4.1 Probability of Missile Strike (P_2)

The strike probability (P_2) is the probability that a missile strikes a critical plant region, given its generation. The product of the probability of missile genesis (P_1) and the probability of missile strike (P_2) yields the total probability that a turbine missile will impact a specified target.

Strike probability inherently depends on the angular distribution of ejected missiles relative to the turbine wheel plane. Missile trajectories are specified by two angles: the elevation angle θ_1 and the angular variation about the wheel plane θ_2 (Ref. 8). See Figure 3.5-6.

Missile strike probabilities are categorized on the basis of elevation angle and surface hit. Missile trajectories are classified as HTM ($45° \geq \theta_1 \geq 135°$) or low trajectory missiles [(LTM), $0 \geq \theta_1 \geq 45°$, $135° \geq \theta_1 \geq 180°$], and surface hits as horizontal (top) or vertical (side). The following scheme applies:

1. HTMs, horizontal surface strike
2. HTMs, vertical surface strike
3. LTMs, horizontal surface strike
4. LTMs, vertical surface strike.

Categories (1) and (4) generally result in the highest calculated strike probabilities. The remaining two cases are insignificant due to lower strike probabilities and the low damage probability compared to cases (1) and (4). The negligible damage probabilities are a combination of small missile strike areas and the high obliquity of missile impact angles. Therefore, cases (2) and (3) are omitted from further consideration. Due to orientation of the N Reactor turbines, target interactions, in the large majority of cases, are possible only with HTMs. Turbine failure missiles, ejected between θ_1 angles of 180° to 360°, are also eliminated from consideration because they are assumed to be effectively contained by the turbine pedestal and turbine building floors.

Missile strike probabilities are calculated by a method similar to that used by GE (Ref. 8) and Westinghouse (Ref. 9). The analysis is based upon the fundamental assumptions of Semanderes (Ref. 9), the assumption of a uniform distribution of missile ejection energies and the recognition that target-turbine elevation difference is small. It is described in the following section.

3.5.1.3.5.4.2 Turbine Missile Impact Probabilities on an Equal Elevation Plane

The missile impact probability is the probability that a missile strikes a critical plant location zone, given its generation and ejection. The strike probability depends on the angular distribution of the ejected missiles relative to the turbine wheel plane. Missile trajectories have been specified by Semanderes (Ref. 9) as a function of two angles: 1) the rotation angle, θ_1, about the shaft axis, and 2) the cone angle, θ_2, relative to the wheel plane. The probability of the rotation angle, θ_1, is distributed uniformly about the full circle of the wheel. The cone angle, θ_2, of the fragments has been approximated by engineering representatives of the two major U.S. turbine manufacturers (Ref. 2, 9) as uniform probability distributions over -5° to +5° angle for the internal wheels and 0° to 25° for the end wheels of a hood. The uniform distributions are used because of the resulting simplification of calculations with reportedly minor adverse effect on accuracy (Ref. 7, 8 and 9).

System Definition and Analysis

The arrangement used in this analysis is shown in Figure 3.5-6. The turbine is located at the center of a coordinate system and is the source of missiles landing on the target zone located at a distance r from the turbine center and a horizontal plane angle θ_3 measured from the plane of the burst wheel.

The assumptions used are:

1. The turbine wheel breaks apart into n equal fragments.

2. The probability of angle θ_2 is independent and uniform between $\theta_{2\ell} < \theta_{2\ell} < \theta_{2u}$.

3. The probability of angle θ_1 is independent and uniform over the sector angle of the fragment.

4. The target zone is small relative to the exposed area, and is defined by the dimensions Δr and $r\Delta\theta_3$.

5. Atmospheric friction is negligible.

6. The vertical displacement between the horizontal planes at wheel axis and the target zone is negligible compared with r.

The impact probability analysis of Semanderes (Ref. 9) is based upon the following geometrical relationships for trajectories generated with the above assumptions:

$$\tan \theta_2 = \tan \theta_3 \cos \theta_1 \qquad (3.5\text{-}6)$$

$$\sin \phi = \sin \theta_1 \cos \theta_2 \qquad (3.5\text{-}7)$$

$$x = r \sin \theta_3 \qquad (3.5\text{-}8)$$

$$y = r \cos \theta_3 \tag{3.5-9}$$

$$r = \frac{V^2}{g} \sin 2\phi \tag{3.5-10}$$

The impact probability analysis utilized in this evaluation was developed using the foregoing assumptions with minor approximations introduced for the purpose of simplifying the numerical application (Ref. 10). The strike probability for a specific initial velocity V is estimated by the function:

$$P\ (\text{strike/V}) = \frac{n \cos \phi}{\Delta\theta_2\ 2\pi} \left| \frac{\tan 2\phi}{2} \right| \frac{\Delta A_{th}}{r^2} \tag{3.5-10}$$

where:

$$\sin 2\phi = \frac{gr}{V^2}, \quad |\theta_2| \geq \theta 2_{2max} \tag{3.5-11}$$

where:

$$P\ (\text{strike/V}) = 0, \quad |\theta_2| > \theta_{2max} \tag{3.5-12}$$

where:

$$\sin \theta_2 = \sin \theta_3 \cos \phi \qquad (3.5\text{-}13)$$

The simplifications utilized in this development are restricted to small values of cone angle, θ_2. The error in estimate introduced by their use is:

Internal wheel, $| \theta_2 | \geq 5°$, .4% low

External wheel, $| \theta_2 | \geq 25°$, 15% low

The foregoing equation was compared with the random-number-generated computer tabulations of Semanderes (Ref. 9) by examination of several constant velocity cases. The results were in good agreement.

The strike probability for the condition that the velocity or energy level is not fixed but is distributed between $V_u > V > V_i$ was derived from conditional probability. In the following:

$$P(\text{strike}) \equiv P(\text{strike}/V)\ P(V^2) \equiv P(\text{strike}/V_u)\ P(V_u, V_\ell, V_i) \qquad (3.5\text{-}13)$$

Where the probability distribution of the fragment energy level, V^2, is uniform. For the case of a horizontal area strike, the second term becomes:

$$P(V_2, V_\ell, V) = \frac{\cos 2\phi_u}{2 \cos \phi_u \frac{1 - V_i^2}{V_u^2}} \ln \left[\frac{\tan \frac{\phi_\ell}{2}}{\tan \frac{\phi_u}{2}} \right] \qquad (3.5\text{-}14)$$

The subscripted elevation angles are determined from equation 3.5-10 for the appropriate velocity value. V_ℓ is the minimum velocity required to reach the target or cause damage.

It can similarly be demonstrated that the vertical target equivalents of equations 3.5-11 and 3.5-14 are:

$$P(\text{strike}/V) = \frac{n \cos \phi \mid \tan 2\phi \mid}{\Delta\theta_2 \, 2\pi \quad 2\tan \phi} \frac{\Delta A_{tv}}{r^2} \qquad (3.5\text{-}15)$$

$$\text{and } P(V_2, V, V_i) = \frac{\cos^2 \phi_u}{2 \cos^2 \phi_u \left[1 - \left(\frac{V_i}{V_u} \right)^2 \right]} \left[1 - \frac{\sin \phi_u}{\sin \phi_\ell} \right] \qquad (3.5\text{-}16)$$

Nomenclature:

V = Speed of fragment leaving turbine housing

θ_1 = Rotation angle of initial velocity measured about turbine axis of rotation

θ_2 = Cone angle of initial velocity measured from plane of turbine wheel

$\Delta\theta_2$ = Total cone angle in which fragments are emitted

ϕ = Elevation of initial velocity measured from the horizontal plane

θ_3 = Angle of center of impact measured relative to plane of turbine wheel in the horizontal plane

r = Radial distance from turbine to center of impact measured in the horizontal plane

A_{th} = Area of target in the horizontal plane

A_{tv} = Area of target in the vertical plane

n = Number of pieces (uniform) into which the failed wheel separates

P( ) = Strike probability

g = Gravitational acceleration

Subscripts:

u = Upper level (maximum) missile ejection velocity

i = Lower level (minimum) missile ejection velocity

1 = Lower limit of effective velocity for target-missile interaction

3.5.1.3.6 Analytical Results

Damage potential for a fragment strike was based upon spalling velocity calculations for 3,000 lb/in^2 reinforced concrete. (See Table 3.5-4.) The spalling velocity was then utilized as a lower limit value in the computation of missile impact probability.

Table 3.5-4 gives the combined units, HTM, LTM and total probabilities (P_4) of turbine missile damage to various emergency-related plant components. Table 3.5-5 gives the HTM, LTM, and total P_4's corresponding to each turbine safety-related target on the site. A further breakdown into the combined (P_2) (P_3) strike damage probabilities is provided in Table 3.5-6.

The cumulative probability of a missile from any turbine damaging a critical system is estimated as $P_4 = 6 \times 10^{-10}/yr$. Even should this highly improbable event occur, the ability to achieve safe shutdown would not be impaired. The potential damage to a safety-related system is considered sufficiently improbable that specific protective barriers are not required. As may be observed, this result has been strongly influenced by the original area layout in which the turbine placements were carefully arranged so as to minimize LTM interactions.

3.5.1.4 Missiles Generated by Natural Phenomena

N Reactor is located very remote from other facilities, and the land in its vicinity is relatively free of materials which could become tornado-borne missiles. Analyses of key facilities to withstand tornadoes and tornado potential are discussed in Section 3.3.2.

The 181-N facility is slightly recessed into the riverbank and is downstream from the WPPSS Hanford Generating Project pumping plant which protrudes further into the river. Therefore, it is not in the path of fast-flowing flood waters. Any flood-borne missile would strike the 181-N facilities in a slow moving eddy pattern. Only the Probable Maximum Flood (PMF), as described in Section 2.4, would be above the 181-N deck and even then would cover the deck with only 2 ft of water. The CRW pump motors are protected from waterborne missiles by a chain link fence. Neither water nor waterborne

missiles would damage the ECCS diesel drives as the building, fencing and other equipment would intercept missiles. In addition, the ECCS diesel-driven pumps are mounted 5 ft above the deck.

There are no other conceivable natural phenomena which could produce missiles at the N Reactor site.

3.5.1.5 Missiles Generated by Events Near the Site

N Reactor is at the center of a 5 1/2-mi radius exclusion area within the Hanford Reservation. There are no other operating industrial facilities within this area. Consequently, there are no potential missiles from this source.

The potential for missiles from a train explosion is very remote, as the only railroad within the exclusion area is owned, operated and controlled by DOE. Further, the major traffic on this railroad is in support of N Reactor. Shipments to N Reactor consist of reactor fuel elements, water treatment chemicals, helium, and large electrical gear. Most of these materials are incapable of generating missiles. The trains operate at very low speeds in the vicinity of N Reactor. The only nearby road or highway crossing is just outside the entrance gate to the WPPSS Hanford Generating Project where the vehicles are slowed for the entrance-exit gate check. Thus, an accident involving a train and truck or car is quite unlikely.

Commercial traffic occurs on state highways within the exclusion area but about 4 1/2 miles from the reactor at the nearest point. This distance is too great for missile consideration from this source. Truck traffic closest to and within 100-N Area is almost exclusively in support of N Reactor operation. Number 6 fuel oil for the station service boiler is the predominant cargo. This fuel has low volatility and a high flash point, making it an unlikely material to explode. Other materials delivered are gasoline and compressed gases. Trucks carrying these materials are stopped at the entrance gate to 100-N Area. Movement within the area is at a slow speed to assure no damaging accidents.

The Columbia River past N Reactor has essentially no barge traffic because there is no navigation channel, and the dams upstream do not have navigation lockage facilities. Consequently, the potential for missiles from a barge explosion is nil.

There is no possible missile source from pipeline explosions as the nearest transmission line is approximately 35 mi away.

The nearest military facility is the U.S. Government Yakima Firing Range whose nearest boundary is approximately 20 mi away. The range is used mostly for artillery and tank firing. Aircraft traffic is very light, and the impact area for the ordnance expended from the aircraft is 30 mi west of the site. The delivery patterns are well within the boundaries of the range area. The firing range offers no credible missile source. See Section 2.2.

3.5.1.6 Aircraft Hazard

There are no federal airways or airport approaches within 2 mi of N Reactor. Air routes may now pass close to the reactor since the CAA has removed the original flight restrictions over the Hanford Reservation. Security patrol helicopter overflights of sensitive areas are administratively prohibited.

There are no airports located within either 5 or 10 mi of N Reactor. The closest airport is at Richland, 25 mi away with approximately 46,000 movements per year currently. A second airport at Pasco, 35 mi away, has current movemen of approximately 92,000 annually.

The Yakima Firing Range's closest boundary is approximately 20 mi.

Because of the distances to airways, airports, and military installations, and because of the low level of airport operations, analysis for aircraft hazard is not necessary.

3.5.2 Structures, Systems and Components To Be Protected From Externally-Generated Missiles

The structures and systems to be protected from external missiles are the same as were identified in Section 5.3.1. The degree of protection from tornados is discussed in Section 3.3, and protection provided for other externally-generated missiles has been discussed above in Sections 3.5.1.5 and 3.5.1.6. The following discussion primarily addresses protection for system components.

Protection from missiles is primarily provided by redundant systems and by redundant components of subsystems within a system. As discussed previously, a single missile caused component failure within either the normal RCS or the ECCS leaves the other system intact for achieving and sustaining safe reactor shutdown. Assuming a second component failure coincident with a missile caused failure, protection is still provided by redundancy within systems as stated below.

In the 181-N Building, only one of the two CRW pumps using offsite power, the two using onsite power, or one of the two low-lift pumps are required. Any combination of two component failures will leave either one system intact or sufficient capacity in the other.

In the 182-N Building, two of the three high-lift diesel-driven ECCS pumps or two of the five HPI pumps are required. Therefore, two simultaneous component failures will not incapacitate both systems. In addition, the HPI pumps, normally supplied from the 1,000,000 gal demineralized water tank, can also be fed from the other three water storage tanks. Further the ECCS high-lift pumps can be fed by the 181-N CRW pumps.

From the 182-N Building to the reactor are two separate and independent pipe lines for the ECCS supply. The normal RCS, with alternate powered circulation and alternate heat dissipation systems, basically relies on the single HPI line from the 182-N Building into the 109-N Building. The routing of this line is underground and not located near either of the ECCS lines.

Two simultaneous line failures would not prevent adequate reactor shutdown cooling.

The 181-N CRW pumps and the HPI pumps rely on redundant electrical service. Offsite electrical power is supplied to the Hanford 230 kV Loop from two sources. 100-N is connected to the Hanford 230 kV Loop by two independent ties. In addition, approximately half of the operating equipment is connected to offsite power, fed by the dual Hanford 230 kV Loop ties with onsite power connected to the other half operating load. Onsite power is supplied by the station service TG set in the 184-N station service power plant. With two 181-N CRW pumps on each power system and two of the high-lift injection pumps also on each power system, simultaneous failures, should they occur in the electrical system, would leave the ECCS intact. Consequently, no combinations of two component failures affecting the two power systems will prevent safe reactor shutdown and continued cooling.

The 109-N Building houses the heat exchanger, the coolant circulating pumps, and piping. Missile protection is provided for the heat exchangers and the coolant piping by massive reinforced concrete shielding and confinement enclosures. However, the primary protection against loss of cooling is provided by the independent ECCS. The six pump drive turbines that are located outside of the concrete cells provide redundant cooling capacity since only three pumps are necessary to achieve and maintain safe reactor shutdown.

The 153-N Building houses switchgear for the two power systems. The building is partitioned separating the two systems. Either system can provide adequate power for attaining and maintaining safe reactor shutdown. Again, backup protection is provided by the ECCS for reactor coolant and by their independent dc system for reactivity control.

The reactivity control of N Reactor relies on HCRs with the vertical BSS as a backup. There are 84 control rods with independent fail-safe activating circuits. The rods are individually, hydraulically operated, and each has an accumulator to provide power for actuation on loss of either dc power source or any automatic or manual trip (scram). There are 107 ball channels each

with a loaded ball hopper enclosed in the reactor top shield. The ball hoppers have a fail-safe mechanism designed to drop the balls automatically upon interruption of both independent dc safety circuits by automatic and manual trip conditions. The ball system is effective even if five of the channels should fail. The BSS is located in an enclosure separate from the HCR system. 2

Because of the fail-safe ball drop and rod scram features, neither system will fail as a system, rather only an individual component (rod or ball channel) is a credible failure. The ball system is effective with 102 of the 107 channels, and the rod system is effective with 80 to 82 of the 84 rods, depending on location of the out-of-service rods. Therefore, no two components failures will incapacitate either reactivity control system.

Because of redundancy and physical separation neither the RCSs nor the reactivity control systems require additional missile protection.

3.5.3 Barrier Design Procedures

When N Reactor was designed, no specific criteria for missile protection existed. However, orientation of turbines was considered, which along with other design considerations makes the reactor resistant to missiles in the following respects:

1. The WPPSS Hanford Generating Project TGs for commercial power production are housed in a separate building (185-N) located approximately 650 ft from the 105/109-N Building and are oriented such that the turbine blades rotate in a plane parallel to all the near 100-N building faces rather than in a plane intersecting the buildings.

2. The local TG set is housed in 184-N and is oriented such that its turbine blades rotate in a plane parallel to the 105/109-N Building.

3. The pump drive turbines in the 109-N Building rotate in a plane parallel to the reactor and the 182-N Building. The 181-N Building is within the trajectory cone for a postulated 109-N turbine missile, the nearest turbine is 365 ft from the diesel-driven pumps.

4. The confinement system structure consists mainly of massive reinforced concrete walls, floors, and roofs, and is not vulnerable to damage by missiles of the type that could be generated by failure or pressurized or rotating equipment.

5. The RCS pumps are located in individual cells with reinforced concrete walls between cells.The pumps are driven by through-wall shafts from turbine and pony motor sets located in the 109-N turbine bay outside of the confinement system.

6. The RCS pumps drive turbines and attached pony motors are mounted in a single row in the turbine bay with no missile barriers between them. Failure of a drive turbine could potentially cause failure of one or more other drive turbine/pony motor sets. If all drive turbines were damaged by missiles, the reactor would automatically be scrammed and placed on once-through emergency cooling. The drive turbines, pony motors, and RCS pumps are not used for the emergency cooling function. Failure of any one of the five normally operating drive turbines would cause a reactor scram but would not cause a need for emergency cooling. The drive turbine blades rotate in a plane parallel to the confinement structure rather than intersecting it.

7. The HCR system, BSS, and ECCS are located such that they are not potentially exposed to missiles from failure of high pressure piping nor failure of rotating machinery and are in reinforced concrete enclosures.

8. Multiple redundancy of cooling systems and power sources.

In conclusion, N Reactor is adequately protected from missiles.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED BREAK OF PIPING

The criteria and bases for structural design of the reactor coolant system (RCS) piping are discussed in other sections of Chapter 3. This section summarizes the results and methods of analysis that have been carried out to verify that the reactor, as designed, is protected against dynamic effects associated with the postulated break of piping. The criterion applied for evaluation of the analytical results is that the limiting pipe break shall not affect the ability to safely shut the reactor down.

Regulatory Guide 1.46 and the NRC Standard Review Plan Sections 3.6.1 and 3.6.2 requires systems or components important to plant safety or shutdown that are located proximate to high- or moderate-energy piping systems and that are susceptible to the consequences of failures of these piping systems be identified. The identification should be related to predetermined piping failure locations in accordance with Section 3.6.2.

High-Energy Fluid Systems are fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met: 21

1. Maximum operating temperature exceeds 200°F

2. Maximum operating pressure exceeds 275 psig.

Moderate-Energy Fluid Systems are fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

1. Maximum operating temperature is 200°F or less

2. Maximum operating pressure is 275 psig or less.

Protection from pipe whip need not be provided if any of the following conditions will exist:

1. Both of the following piping system conditions are met:

   a. The service temperature is less than 200°F

   b. The design pressure is 275 psig or less

2. The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from shipping by plant design features, such as concrete encasement

3. Following a single break, the unrestrained pipe movement of either end ofd the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety

4. The internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

Protection from pipe whip need not be provided if any of the following conditions will exist:

1. The service temperature is less than 200°F

2. The design pressure is 275 psig or less.

The following types of breaks are to be considered:

1. Longitudinal breaks in piping runs and branch runs with nominal pipe sizes of 4 in. and larger.

2. Circumferential breaks in piping runs and branch runs exceeding a nominal pipe size of 1 in.

3. Through-wall leakage cracks in piping and branch runs exceeding a nominal pipe size of 1 in., where the crack opening is assumed as 1/2 the pipe diameter in length and 1/2 the pipe wall thickness in width.

The high and moderate energy lines which may require additional protection should be identified; and an analysis of the potential effects of the postulated pipe breaks and safety evaluations must be performed.

3.6.1 Postulated Piping Failures in Fluid Systems Outside of Confinement

3.6.1.1 Design Bases

At N Reactor, the ECCS system is the only safety-related system whose components are in close proximity to high and moderate energy pipe lines. 21

3.6.1.2 Description of ECCS

A pipe break analysis, as discussed in 3.6.2, was carried out in 1968 as a result of concerns that had developed regarding potential consequences of pipe whip. The 105-N portion of the RCS piping was considered to be most vulnerable in relation to effects upon the functional capabilities of the ECCS and of the confinement system; consequently, the analysis was limited to that area.

In the most recent safety review of the N Reactor performed by the Westinghouse Independent Safety Analysis Task Group (Ref. 3), issue B.2.3, the following concern was stated:

> The potentials for propagating damage to ECCS piping through pipe whip from an initial event must be assessed and adequate protection must be shown to exist.

The resolution for this concern was provided in the Plan to Reprogram Safety Upgrades (Ref. 4) with the following two recommendations which are required to be resolved before restart:

1. Either re-route the V-4 actuation control cables (in the rear pipe spaces) outside of the pipe whip zone or determine by analysis that the potential for pipe rupture is acceptably low (B.2.3).

2. Conduct a walkthrough in 182-N and 105-N inlet pipe spaces to evaluate the potential for damage to ECCS components from a whipping pipe (B.2.3).

In response to the second recommendation, a walkdown of the ECCS was performed. Boundaries of the walkdown consisted of piping and components identified as essential ECCS items by the Seismic Category Three Over One Walkdown Program (described in Section 3.7). Excerpts from the III/I reports, describing the boundaries and system descriptions, including the ECCS actuation circuitry are provided below.

The N Reactor areas walked down for ECCS pipe whip hazard identification were:

1. <u>105 Building</u>

Room 2	Room 24	Room 33	Room 174	Corridor 4
Room 3	Room 25	Room 37	Room 180	Corridor 5
Room 5	Room 26	Room 47	Room 502B	Corridor 7
Room 6	Room 27	Room 108	Room 507	Corridor 11B
Room 7	Room 30	Room 136	Corridor 1	Corridor 22
Room 8	Room 31	Room 172	Corridor 2	
Room 9	Room 32	Room 173	Corridor 3	

2. <u>109 Building</u>
   Turbine Drive Bay
   Corridor 19

3. 181 Building
   All areas

4. 182 Building
   All areas

The ECCS provides emergency cooling by establishing once-through cooling to all supply risers in the 105-N inlet pipe space which distributes coolant flow to the individual pressure tubes through individual tube inlet connector pipes.

The ECCS initially draws upon stored demineralized water, and then upon treated Columbia River water to cool the fuel in the reactor. The system pumps the water through the reactor pressure tubes into the dump tank, then to the Liquid Effluent Retention Facility. A more complete description is provided in Section 6.3. 21

The scope of the ECCS walkdown is reported in Reference 6, and consists of five tasks.

1. 105N ECCS Inlet Manifold to V-3 valves and V-3, V-4 valve operating air supplies

2. Distribution System from V-3 valves to reactor inlet risers

3. Reactor inlet system from reactor inlet risers to pressure tubes

4. Reactor outlet system from pressure tubes to V-4 valves, including RV-2 relief valves and CV-3 check valves

5. Outlet System from V-4 valves to emergency dump tank overflow pipe.

Task 1

This walkdown commenced where the 30-in. active ECCS pipe emerges from the ground outside of the 109-N Building, up the side of 109-N to the roof, and across the roof to 105-N. The 30-in. pipe is then tee-reduced to two 20-in. pipes which descend into Corridor 5 in 105-N. The 20-in. pipes then reduce to 16-in. pipe prior to two check valves (CV-60) in Corridor 5. These two lines provide water to the 16-in. ECCS manifold in Corridor 5.

The ECCS manifold routes water to the eight V-3 valves. Walkdown boundaries are all in Corridor 5 and include CV-61-1 and V-62-1 valves to the east; and V-19, V-15, and RV-1 valves to the west.

The walkdown included the V-3 valves and operators. The local operator air supply, air piping, and solenoid valves were also covered as were similar air supplies for the V-4 valves. The Category I electrical conduit required to activate the V-3 valves was walked from the operator solenoid valves to where the conduit penetrated the 105-N Building walls.

Task 2

The eight 8-in. supply pipes from the V-3 valves to the 105-N confinement penetrations to the east, and west 8-in. inlet headers including the CV-1 check valves, the V-28 valves and the V-18 flush line valves.

The sixteen 4-in. supply pipes from the 8-in. inlet headers to the reactor inlet risers including V-1, V-9, CV-2 and CV-2R.

Task 3

The sixteen 18-in. inlet risers including the CV-5 check valve and V-38 vent valve.

The 1,003 inlet connectors including the V-11 valves, venturi flow meters and the pressure tube inlet nozzles.

Task 4

The pressure tube outlet nozzles to the 1,003 outlet connector pipes to the sixteen 18-in. outlet downcomers.

The 18-in. outlet downcomers including the CV-3 check valves, the V-2 valves, the V-8 valves, and the V-21 and V-33 vent valves.

The sixteen 6-in. pipes from the outlet downcomers to the 10-in. FL(HP) header then to the eight V-4 valves including the V-43 valves and the RV-2 relief valves.

The V-23 valves were also included in Task 4. Note that these valves are locked open and do not require actuation for the ECCS operation.

Task 5

The 10-in. pipe from the eight V-4 valves to the 16-in. FL(HP) headers. These are located in the east and west outlet pipe spaces in 105-N.

The continuation of the 16-in. FL(HP) east and west headers to the V-23 valves, then to the 24-in. drain line.

The continuation of the 24-in. drain line through a penetration into the 109-N pipe then 32-in. pipe to a pipe penetration exiting the 109-N pipe gallery.

The drain pipe continues underground to the vicinity of the emergency dump tank. This pipe was seismically qualified by the ECCS Resolutions Task Force.

The pipe from the 105-N lift station to the emergency dump tank overflow pipe was also included in the Task 5 scope. This pipe is underground within confinement and is not subject to III/I damage.

The Seismic Category I Emergency Power System raceway was also walked down (Ref. 7). The Emergency Power System includes portions of the safety-related 125 Vdc, 32 Vdc, and 24 Vdc systems.

Voltage	Battery	Location
125 Vdc	A & B	105-N, Room 9
125 Vdc	A	181-N
125 Vdc	A	182-N
32 Vdc	A	105-N, Room 174
24 Vdc	D & E	105-N, Room 6

The 125 Vdc A & B systems in Building 105-N supply power to the ECCS logic relays, solenoid valves for V-3 and V-4 operation, and the No. 1 and No. 2 low- and high-lift diesel speed controls and air start solenoids. Low-Lift Diesel No. 3 start system and speed control are powered by 125 Vdc batteries in the 181-N Building, and High-Lift Diesel No. 3 Start System and speed control are powered by 125 Vdc batteries in the 182-N Building. The 32 Vdc batteries power the start-up range and intermediate range flux monitoring instrumentation.

The 24 Vdc batteries D and E power the display instrumentation for CVV 23 and 24 open and close status, V-3 and V-4 open and close status, and the seismoscope trip indicating lights.

The entire length of the raceway was walked, photographs were taken, and the raceway qualified either by analysis or by Design Analysis Guidelines Acceptance Criteria (Ref. 8).

Also included in the walkdown program as potential pipe whip hazards was the monitoring and actuation circuits in Room 6 (Ref. 9). These include:

- The low level and subcritical flux monitor system circuits that originate in the control panels in the control room (145), pass through conduits in Room 6, and are routed to the instruments on top of the reactor.

- The IRFM/SRFM conduits that are in Room 6.

- The emergency power system circuits for the low level and subcritical flux monitor system that initiate in Battery Room 174 pass through Rooms 8 and 9 and into conduits in Room 6.

- Room 9 is the main 125 Vdc emergency power supply room and contains the "A" and "B" bus dc batteries which are the ones normally in service. The 125 Vdc circuits are routed through Room 8, across Corridor 2, and into Room 6.

In Building 182-N, the major ECCS components are the three high-lift diesel-driven pumps, instrumentation, and independent stored energy systems consisting of dc storage batteries, compressed air accumulators, and diesel fuel oil systems. 21

These subsystems and associated components summarized below were walked down for potential hazards from pipe break and pipe whip.

- Three high-lift diesel pumps
- Gearboxes
- Three diesel engines
  - Fuel oil system
  - Air start system
  - Cooling system
  - Lube oil system
  - Air intake and exhaust
- Control system
  - Instrumentation-tachometer
  - DC power supply
  - Starting-air solenoids
- ECCS Category I Piping

Lastly, the major ECCS components located in Building 181-N, the three low-lift diesel-driven pumps, instrumentation, and independent stored energy

systems consisting of dc storage batteries, compressed air accumulators, and diesel fuel oil systems were also included in the walkdown program for potential pipe whip hazards.

These subsystems and associated components summarized below were also included in the walkdown program for potential pipe whip hazards.

- o Three low-lift pumps
- o Gearboxes
- o Three diesel engines
  - Fuel oil system
  - Air start system
  - Cooling system
  - Lube oil system
  - Air intake and exhaust
- o Control system
  - Instrumentation-tachometer
  - DC power supply
  - Starting-air solenoids
- o ECCS Category I Piping

3.6.1.3 Description of High- and Moderate-Energy Lines Creating Potential Hazards

Pipe whip hazards were identified based on their proximity to essential ECCS equipment and the operating parameters of the system. A total of 18 potential hazards were initially identified. These initial hazards were reviewed against NRC Standard Review Plan general and Regulatory Guide 1.46 criteria. Those hazards falling into the low-energy piping category were dismissed. The hazards falling into the high-energy category were identified as requiring further assessment. Seven of the potential hazards were determined to be in the low-energy piping category.

It was concluded that the remaining 11 potential pipe whip hazards identified would be further assessed, and resolved as necessary, prior to restart of

the reactor. None of these potential hazardsinvolved piping targets of 18 in. or larger. Therefore, it has been concluded that the original pipe whip analysis performed in 1968 still represents the bounding case. 21

3.6.2 Determination of Break Locations and Dynamic Effects Associated With the Postulated Break of Piping 21

3.6.2.1 Criteria Used to Define Break Location

21

The break locations selected for the 1968 dynamic analyses were determined according to the following criteria:

1. Locations were chosen for postulated pipe breaks that would have the greatest potential for propagation of additional breaks.

2. The limiting case, as determined in criterion 1, above, was defined in terms of: (a) the potential for exceeding the design capabilities of the emergency core cooling system and (b) the maximum energy release to the confiner and the potential for exceeding the design capabilities of the confinement system.

3.6.2.2 Analytical Methods to Define Forcing Functions

The 1968 analysis to define forcing functions was confined to the determination of jet reaction forces only. The calculational model consisted of a single flowpath representation of the RCS and the reactor core, as described in detail in Reference 1. 21

3.6.2.3 Dynamic Effects Associated with the Postulated Break of Piping Break Locations

The locations chosen for analysis (Ref. 1) were within the 105-N portion of Zone I of the Confinement System, which contains almost the total lengths of the 16, 18-in. inlet and outlet risers (and downcomers). An elevation view of a longitudinal section of this area is shown in Figure 3.6-1. In this figure are shown numbers that correspond to postulated break locations, all

of which are directly adjacent to an elbow, a tee, or a change in pipe diameter. At each of these locations (numbered 1 through 8) a break was assumed upstream and downstream of the fitting, with the exception of Location 4 in the inlet risers, at which the reactor coolant is normally stationary. Breaks within the elbows were analyzed but were found to have less severe consequences than those discussed herein.

Also shown in Figure 3.6-1 is a coordinate system used for reference in describing the directions of reaction forces that would be generated at the various designated locations.

Analytical Results

The results of the dynamic analysis are shown in Table 3.6-1. In each case analyzed, a sudden, complete double-ended circumferential break was assumed to occur, and reaction forces were calculated over the period of time required to identify the maximum values that would be reached; the forces given in Table 3.6-1 are the minimum and maximum values calculated over a 10-second period. Conclusions relevant to the effects of the forces are listed for each break analyzed.*

Table 3.6-1 shows that a break of an inlet riser immediately downstream of Elbow No. 1 could have the most severe consequences. The downward reaction force upon the elbow would have an initial magnitude of 60,000 lb and would reach a value of 250,000 lb approximately 200 milliseconds later.

The potential effect of the force can be judged by envisioning the upstream section of the inlet riser as a cantilever beam restrained at the juncture with the inlet (cold leg) manifold. Distances from the centerline of the elbow and possible locations of restraint are as follows.

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* Some of the conclusions given in Table 3.6-1 are more conservative than those given in the source document (Ref. 1), primarily because of differences in judgement with respect to the possible actions of pipe hangers.

Restraint	Distance to C_L Inlet Riser, ft
105 - 109-N wall (5-ft thick concrete)	11.5
Centerline of 24-in. Inlet Manifold	19.5

The 105/109-N wall penetration is provided with a flexible, cooled sleeve having an outside diameter of 34 in.; thus, a downward deflection of approximately 8 in. would occur before the pipe contacted a rigid support (the concrete wall). For this reason, it was assumed that the pipe would be cantilevered initially about the welded connection with the 24-in. manifold. The resulting bending moment would produce a stress of approximately 250,000 psi due to flexure alone. This may be compared with a yield strength of 28,000 psi for the carbon steel of which the pipe is fabricated. 21

In view of the above considerations, it would be expected that a plastic hinge would form first at the inlet manifold juncture and (very probably) second, at the plane of the wall. The ultimate conclusion was that in spite of the energy dissipating in deflection of the pipe to the bottom of the wall penetration and forming a plastic hinge at that point, the pipe could strike the CV-3 check valve directly underneath with sufficient force to initiate a break of the outlet riser. Thus, the limiting case would be almost simultaneous failure of inlet and outlet risers connected to the same group of reactor pressure tubes. A break immediately upstream of the elbow (Case 2 in Table 3.6-1) could have the same result. Note that the postulated break in the 18-in. inlet riser did not impact any piping of smaller diameter, only the 18-in. outlet check valve CV-3. 21

The consequences of the dual-riser break described above have been analyzed by the RELAP4 code and the CONTEMPT code. Of principal concern was the maximum pressure that would develop in the confinement system as the result of the simultaneous break of the two 18-in. risers. Results of the pressurization analysis were that; the peak confinement system pressure calculated for the postulated accident was 3.55 psig, occurring at 5.2 seconds after the break. This is well below the design pressure of 5.0 psig. 21

A dual riser break would have a depressurization transient similar to one that would result from a cold-leg crossover break but more rapid because of the relative sizes and locations of the pipe breaks. The maximum fuel temperature would be well below 1900°F, the temperature at which fuel failure would occur (Appendix 15.6A of Chapter 15).

The foregoing considerations, which deal with the LOCA-related aspects of the postulated accident, provide assurance that the ECCS and the confinement system would function within their design capabilities. Further assurance for the favorable outcome of the break just discussed was provided in the resolution of the Westinghouse Independent Safety Analysis, Concern B.2, Item 3 (Ref. 3). On assessing the potential damage to ECCS piping and components due to pipe whip, it was noted that:

> with regards to pipe whip, Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," specifies that a whipping pipe may impact and rupture adjacent piping only if the impacted piping is of smaller diameter or lighter wall thickness. Discussions with other contractors have confirmed that normal design practice accepted by the NRC does not require evaluation of adjacent pipe failure of equal dimensions. In addition, piping thickness and weld integrity measurements have been completed at selected locations. These measurements substantiate that aging has not significantly reduced the pipings' resistance to impact loads.

Thus, the above postulated effect related to the CV-3 valve is not a credible accident, and it is not required for analysis of pipe breaks. Further, the ability to place the reactor in a safe shutdown condition would depend entirely upon the proper operation of the reactor safety shutdown systems (the horizontal control rod system and the ball safety system) and the ECCS. See Sections 3.9.4, 3.9.7, 4.5.1, 4.5.2, and 4.6. It has been concluded that

the operation of the safe shutdown on ECCS would not be impaired by the postulated accident for the following reasons: 21

- o Neither the rod or ball systems penetrate the RCS boundary.
- o The horizontal control rod system is physically isolated from the RCS piping.
- o Although some of the components of the ball safety system could be exposed to water issuing from a crossover break, it is extremely unlikely that such conditions would prevent the system from operating as designed.
- o As discussed in Chapter 15, Section 6.5, either of the two systems would be capable of rapidly shutting down the reactor in the event of a LOCA. 2

However, as discussed in Section 3.6.1.2, the walkdown of the ECCS system for potential pipe whip hazards resulted in eleven identified potential hazards. Further analyses and resolution of these potential hazards is a required condition for restart. Therefore, it has been concluded that at the time of restart, there will be no unanalyzed pipe breaks or pipe whip concerns which could adversely affect the capability of safely shutting down the N Reactor and/or initiating required engineering safeguard features. 21

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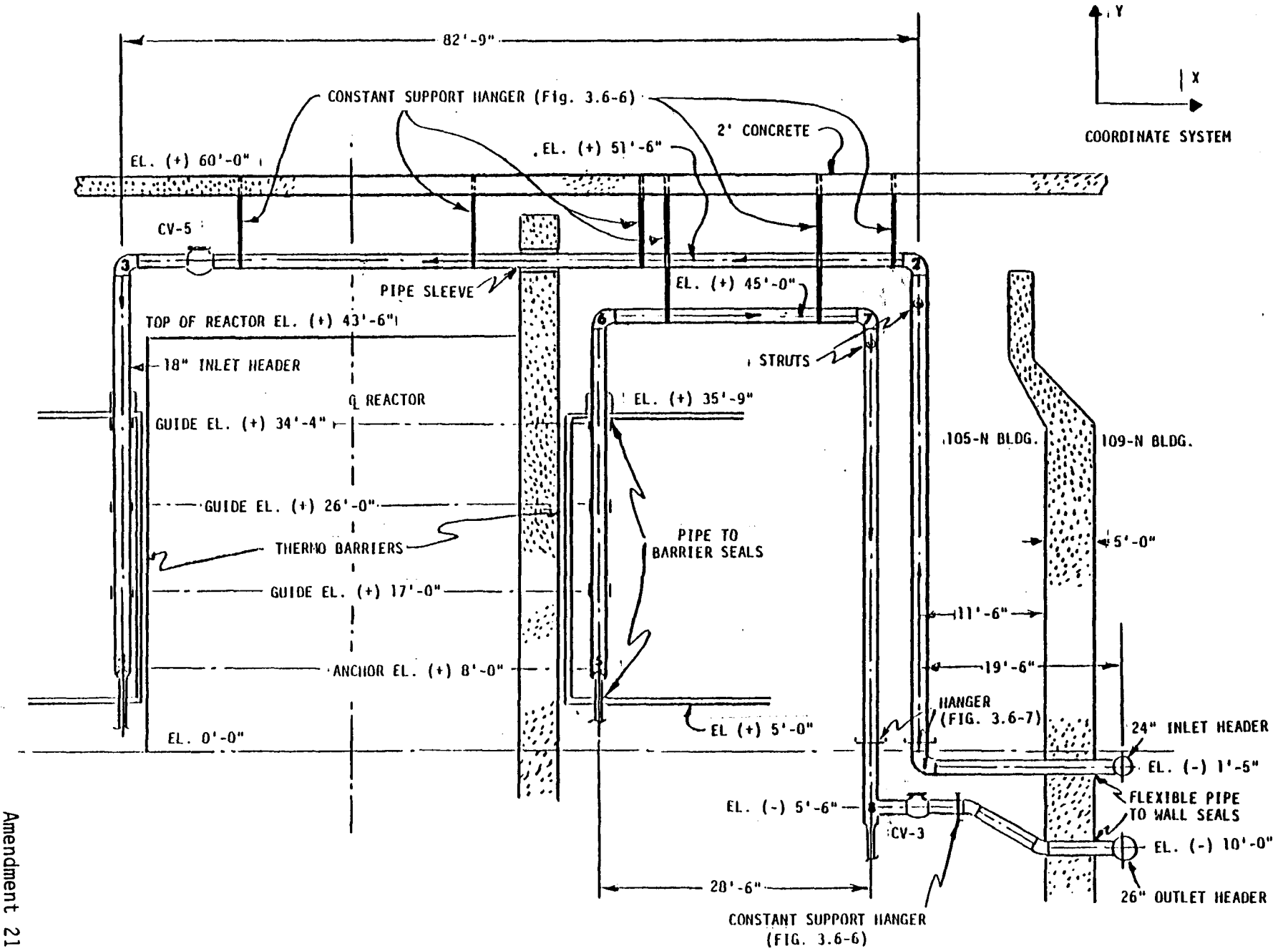
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ELEVATION SECTION
THROUGH ZONE I - 105-N

Figure 3.6-1

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3.7 Seismic Design

This section describes seismic criteria and procedures used to assure that the N Reactor will be adequately resistant to earthquakes that may affect the Hanford Reservation. The conservatism of these criteria and design procedures is compared with the seismology of the region and to the Safe Shutdown Earthquake (SSE) for the site. The methodology for validating the design of N Reactor Seismic Category I systems to the SSE is discussed, as well as actions taken to assure that these critical systems will function properly during the SSE. For background information, the original design criteria and subsequent seismic design chronology is presented in the following summary. 21

Background and Summary

N Reactor is located on the Hanford Reservation in Eastern Washington and within seismic Zone 2, as established by the Uniform Building Code (UBC) (1955-1958). The maximum horizontal ground acceleration for UBC seismic Zone 2 is usually considered to be 0.16 g, which corresponds to approximately intensity VII on the Modified Mercalli (MM) scale. See Figure 3.7-1 for the location of the Hanford Reservation on the seismic zone map of the United States. 21

The Hanford Reservation has been the subject of geologic and seismological studies to support the siting, construction and operation of nuclear facilities since 1944. Local and regional structural geology mapping and the installation and operation of a regional and site seismic network were included in these studies. The main geologic structures which contribute to the seismic exposure on the Hanford Site are the Umtanum Ridge-Gable Mountain (U/G) trend, the Rattlesnake Wallula alignment (RAW), Yakima Ridge, and the Saddle Mountains. The Hanford Reservation is an area of low magnitude seismicity, and during historical times the largest earthquakes occurred outside the Hanford Reservation producing an estimated maximum acceleration of about 0.03 g on the reservation. See Figure 2.5-30, and Section 2.5 for details. When N Reactor was designed in 1960, an earthquake having an intensity VI (MM) (0.06 g) was considered as a sound working value for 21

engineering design, and even though improbable, an intensity VII (MM) was considered an upper limit.

N Reactor is the ninth of the production reactors, and like its predecessors, has operated since 1963 employing a seismic trip setting of 0.007 g for horizontal ground accelerations, which corresponds to the minimum ground acceleration associated with an intensity of IV (MM). None of the Hanford reactors have been shut down from seismic trips.

N Reactor was designed in accordance with the earthquake requirements presented in the 1955 and 1958 editions of the UBC. The non-safety plant buildings were designed for 0.07 g ground acceleration, the reactor operation systems for 0.10 g ground acceleration, and the safe shutdown systems for 0.20 g ground acceleration.

	UBC	Range Allowed
Safe shutdown systems, Zone 3	0.20 g*	0.15 g and up
Systems for Plant Operation, Zone 2 (for instance, secondary system) (see Table 3.7-4)	0.10 g	0.07 g to 0.16 g
Normal plant structure, Zone 2 (see Table 3.7-4)		0.07 g to 0.11 g
Zone 1		0.15 g

From the beginning, seismic design considerations were given special emphasis. During the course of the original design, Holmes & Narver, Inc., with G. W. Housner as consultant, was engaged to evaluate the structural and seismic adequacy of the 105-N Building and critical components located therein. Three studies were completed. The first study (Reference 10) provided added seismic criteria for the 105-N and 109-N Buildings, process unit, metal storage and fuel storage basins, reactor coolant system piping,

*0.20 g is above the UBC minimum for Zone 3 and generally was equivalent to 0.25 g because of the conservative design practices used for safety system designs, i.e., the UBC permitted a 1/3 increase in stress limits for seismic design, which, as a general guideline was not used in the N-Reactor design.

reinforced concrete chimneys, analytical methods for seismic evaluation, and the 181-N pump house. These seismic criteria were incorporated into the final design.

In the second and third studies (References 13, 14, 15), the designs for the following were reviewed and investigated for structural and seismic adequacy:

1. Secondary shielding walls, floors and roof structure enclosing confinement Zones I and II and attached portions of confinement Zone III
2. Common wall and structural ties between the 105-N and the 109-N Buildings
3. Seismic joints between the portions of the 105-N Building defined in Item 1. above and the adjoining portions of the building
4. Metal segregation and storage basins
5. Ventilation exhaust tunnels
6. Reactor control room
7. Exhaust filter building (117-N)
8. Connector support structure and thermal barrier located in the inlet pipe space of the 105-N Building.

A number of specific recommendations were made, including one for strengthening the structural frames of the thermal barrier, all of which were incorporated into the final design.

In addition, although only portions of the RCS are safety related, the entire system was designed as if it were Seismic Category I. Several engineering organizations were involved. Initially General Electric Company analyzed earthquake forces on the RCS located in the 105-N Building. This analysis used the statically equivalent force method. Burns and Roe and their contractor, Bergen Pipe Support Co., reviewed these and performed additional calculations to determine reactions on the RCS and ECCS outside the 105-N Building.

Holmes & Narver, Inc. evaluated the seismic adequacy of the reactor coolant system in 1960 and the inlet piping support structure in 1962. The Electric

Boat Division of General Dynamics Corp. performed computer and dynamic analyses of the RCS resulting in improved pipe supports to minimize lateral pipe deflections in event of an SSE.

Over the years of N Reactor operation, seismic requirements for nuclear plants were strengthened, and the SSE, OBE, and Seismic Category I classifications discussed in Section 3.2 were defined by the NRC.

In March 1965, the Advisory Committee for Reactor Safeguards recommended that AEC production reactors should be operated under the same basic safety criteria that applied to licensed reactors. An extensive study of the seismology of the region and a re-evaluation of the adequacy of the seismic design of N Reactor systems critical to nuclear safety followed, and in December 1967, the U.S. Coast and Geodetic Survey (USGS) established, at the request of the Division of Reactor Licensing (DRL), a maximum design basis earthquake for the region, by concluding that:

> "an acceleration of 0.12 g on Glacio-Fluvial deposits at the Hanford Site would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. In addition, an acceleration of 0.25 g on the same site geology will represent the ground motion from the maximum earthquake likely to affect this site."

The value of 0.12 g was used for the OBE until 1985 when the value was changed to the current criteria of 0.06 g.

The value of 0.25 g corresponds to today's Safe Shutdown Earthquake (SSE). This postulated peak ground acceleration is about four times greater than the ground acceleration (0.06 g) produced at the site by the maximum historical earthquake.

In the 1970's, a 0.25 g SSE was also established for the Fast Flux Test Facility (FFTF) and the Washington Public Power Supply System (WPPSS) WNP-2. Both the FFTF and WNP-2 were reviewed by the NRC and the ACRS. The NRC

issued favorable Safety Evaluation Reports (SER) (References 34, 35). Additional investigations in support of Hanford reactor siting (Refs. 37, 37) and the Basalt Waste Isolation Project in the 1980's have also provided extensive and comprehensive geologic and seismological characterization of Hanford. In general, seismic design based upon a 0.25 g peak horizontal ground acceleration has been demonstrated to be adequately conservative during NRC reviews of nuclear reactors at the Hanford sites. A probabilistic assessment completed in the early 1980's by LLNL for DOE-ONS suggested a 30,000 year return for 0.25 g for the entire Hanford Reservation (Reference 38).

During studies for some new facilities at the 200 Area in 1987-88, the suitability of the 0.25 g SSE was again examined. It was postulated that if the Umtanum Ridge-Gable Mountain trend (U/G) was a continuous fault, it would be capable of an earthquake as large as RAW and a Richter magnitude of 6.7 was suggested. This hypothesis was reviewed, and it was determined that the possibility of U/G being a single earthquake source with large earthquake potential had been addressed during the licensing of WNP-2. In addition, field studies done by the Basalt Waste Isolation Project (BWIP), WPPSS, and Puget Sound Power and Light (PSPL) (Reference 36) all supported the interpretation that U/G was a segmented source. Accordingly, it was concluded there was no basis for changing the design basis 0.25 g SSE, and that the NRC staff position for licensing WNP-2, that Gable Mountain was segmented, was still valid. 21

A continuous program has been underway to upgrade and validate N Reactor critical systems to this SSE. The critical systems that have been validated are those structures, systems, and components necessary to maintain the plant in a safe shutdown condition on emergency cooling. This class of structures, systems, and components is identified as Seismic Category I structures and systems. Refer to Section 3.2.1 for a list of Seismic Category I systems.

During the period 1963-1971, the N Reactor staff conducted a number of independent reviews of the design criteria and seismic adequacy of the

structures, systems, and components (References 5, 9, 11, 17, 18, 19, 20, 21, 22, 23).

Then, in 1967, following the establishment by the USGS of the 0.25 g SSE, the Division of Reactor Licensing (DRL) initiated a program to determine the adequacy of the N Reactor Plant seismic design.

The firm of Nathan M. Newmark - Consulting Engineering Services, was retained to assist in the seismic evaluation. The review and evaluation included the following:

1. Low-lift and high-lift diesel driven ECCS pumps
2. a. Emergency ECCS pipelines
   b. ECCS tank bypass, siphon, and cross-tie piping
   c. Modified (new) ECCS piping between 182-N and 105-N Buildings
   d. Existing ECCS line from 182-N to 105-N Buildings
3. ECCS tank
4. Process piping and primary coolant system - ECCS connection
5. Pipe support details
6. Moderator graphite stack
7. Controls and batteries.

The results of the study are summarized below:

The seismic adequacy of the graphite stack and shields was confirmed.

Potential stress areas of the diesel fuel oil supply piping were replaced and a fire protection system was incorporated. The pump casings for the low and high-lift pumps were replaced to meet NDT requirements.

A new 30 in. ECCS line was installed between 182-N and 105-N under Project 411. The 30 in. pipelines from 181-N to the ERW tank were reanalyzed for brittle fracture and found satisfactory. Lateral support for the ECCS bypass structure was provided in all directions, independent of the ERW tank. Freeze protection was also provided. The ECCS bypass design was

modified such that failure of the ERW tank would not impair the ECCS system. The annular space between the vertical ECCS pipe and the two floor slab penetrations in 105-N was filled with semi-rigid material capable of transferring lateral loads to the concrete slabs without damaging the pipe. The practice of welding pipe to pipe supports was discontinued. Tests and analysis of the valves required for ECCS operation confirmed they would remain operative at accelerations of 0.5 to 0.9 g. The supporting racks for essential batteries were reinforced to withstand an equivalent horizontal force factor of 0.5 g.

Some safety system components could not be satisfactorily analyzed and were therefore tested for dynamic response by shake tests. Accelerations attained during the shaking tests were greater than 0.8 g and the natural periods of vibration were less than 0.3 s. The following components (Reference 16) operated successfully under these extreme conditions, assuring satisfactory operation during a SSE. 21

1. Horizontal control rod system including the drive mechanism and the solenoid valve rack
2. The V-3 and V-4 solenoid control valves located in the ECCS
3. The ball safety system including the drop gate, closing and holding solenoid and the ball drain valve.

The Newmark study for the DRL also included a review of the 105-N and 181-N Buildings, 116-N ventilation stack, emergency dump tank, and 181-N gantry crane (References 3, 6). The objective was to determine the degree of conservatism used in the original design as compared to the 1969-70 licensing requirements. As a result of DRL review questions, the 116-N stack was subsequently analyzed by Dr. C. V. Chelapati using time-history inputs based on the 0.25 g SSE criteria and computer code DYNRESP (Ref 23). It was shown that the margin of safety against collapse during the SSE is approximately 1.3 (see Section 3.8.4.6).

The AEC (ERDA) stopped its licensability studies in February 1971, however, the seismic review performed by and for the DRL had confirmed that the

original seismic designs for critical structures and systems at N Reactor were conservative, and that the critical items studied would continue to function during and after the 0.25 g SSE.

In 1984-85, Impell Corporation was retained to evaluate the structural integrity of the N Reactor core and the functional operability of the critical safety system components within the core when subjected to an SSE (Reference 32). Impell was asked to assess the integrity of the core involving several postulated conditions, including the loss of 100% of the graphite keys. Impell conducted these analyses in two phases which resulted in the conclusion that the reactor core would maintain its structural integrity, and safety systems would remain functionally operable in the event of an SSE (Reference 25). The results are presented in Section 3.8.3.8. Refer to Section 3.7.3.14 and 3.7.3.15 for additional information regarding modeling methods and damping criteria used.

During 1987-88, an extensive seismic upgrade program was undertaken as part of the return to service (RTS) program. This program featured a review of over 1000 drawings to prepare safety equipment lists for environmental qualification purposes (Reference 39) and III/I walkdowns conducted by specially trained and qualified teams to identify and correct any potential hazards to safety class structures and systems resulting from earthquake induced failures of non-safety (Category II) structures and systems (Reference 40). In 105-N, 109-N, and 117-N Buildings, which contain most of the safety related components, a total of 828 potential problems were identified and corrected. Of these, 560 were resolved by analysis and 268 required some form of fix.

Seismic reanalyses were performed for the following Seismic Category I structures:

Structures

- o 166-N
- o 181-N
- o 182-N

- o 105-N
- o 109-N
- o Confinement structure
- o Control room - Room 6 (cable spreading)
- o ECCS silo
- o Reactor block.

Analyses and walkdowns were performed of the following systems:

Systems

- o Emergency Core Cooling System and trip functions (once through path only, 181-N through to 1316-N standpipe)
- o Ball Safety Systems (hoppers to exit drain valves)
- o Horizontal Control Rod System (scram hardware only)
- o Confinement Vent and Vacuum Valves
- o Reactor Trip System (seismoscopes)
- o Auxiliary Systems - Lube Oil, Diesel Starting Air, Fuel Oil, Jacket Cooling Water, etc.
- o Graphite and Shield Cooling System (portions necessary for once-through cooling)
- o Fog Spray System supplying GSCS
- o Instrumentation and electrical supply for the above.

The above systems and structures would prevent a seismically induced LOCA in the important once-through ECCS piping; protect the confinement from over-pressurization in the unlikely event of a steam line break; and provide the capability to safely shut down the plant after an SSE and maintain it in a safe shutdown condition.

In summary, the designs of Seismic Category I structures, systems and components have been validated to 0.25 g SSE by one of the following methods:

- o Calculations that accounted for the dynamic behavior of the particular structure or component through the use of statically equivalent earthquake forces

- o Dynamic analyses using computer programs for evaluating seismic behavior
- o Dynamic testing to determine seismic response of certain mechanical and electrical components that could not be satisfactorily evaluated by one of the above methods
- o III/I walkdowns of all Seismic Category I structures, systems and components to identify and resolve potential hazards due to failure of Non-Seismic Category I structures, systems and components.

3.7.1 Seismic Input

This section describes seismic input used in the reanalysis of the 105-N, 109-N, 181-N, and 182-N structures and components and includes acceleration spectra curves, critical damping values, support media, and relative conservatism with respect to the seismology of the region.

3.7.1.1 Design Response Spectra

Acceleration spectra curves used in the original design of critical N Reactor components were based upon response data for the 1940 El Centro earthquake (Reference 1). This earthquake occurred near El Centro, California on May 18, 1940, and was the first strong-motion earthquake in the United States for which fairly complete seismic records were obtained. (See paragraph 3.7.1.5 for a comparison of soil conditions at El Centro with those beneath the N Reactor site.) The design curves, shown in Figure 3.7-2, are shown for comparison purposes and are based upon the spectra curves prepared in 1959 for the El Centro earthquake by Holmes & Narver, Inc. with the assistance of Prof. G. W. Housner from the California Institute of Technology, Pasadena. The design curves were normalized to a maximum ground acceleration of 0.2 g, consistent with the lower seismicity of the Hanford site (Reference 2).

The design response spectra used for the seismic reanalyses and upgrade effort of the 105-N, 109-N, 181-N, and 182-N Buildings during 1987-88 are the safe shutdown earthquake (SSE) spectra given in SDC 4.1 (Reference 41) and shown on Figure 7.3-3. The spectra have a 0.25 g maximum acceleration and are the same as those used in the Impell seismic evaluations in 1985 which included a soil-structure interaction analysis, a nonlinear analysis of the

reactor core, and an evaluation of the safety systems. In addition, sensitivity studies were performed to verify the analytical models. The SSE design response spectra for horizontal motion are presented in Figure 3.7-4 (Reference 32). The spectra were normalized to 0.25 g zero-period acceleration (ZPA) for motion in the horizontal direction. For the vertical direction, the same spectra were normalized to 2/3 of the horizontal spectra with a 0.167 g ZPA.

21

3.7.1.2 Design Time History

Design acceleration time histories were synthesized from the design response spectra. The spectra resulting from the 105-N/109-N and 181-N histories enveloped the design spectra, while the 182-N history made a close fit. Specifically, the 7% response curve was used for synthesis in the 181-N work, and 5% was used in the rest. Three independent time histories, two horizontal and one vertical, were generated and applied concurrently in the soil-structure analyses of all but the 182-N Building. The symmetry of the 182-N Building produces negligible coupling effects; therefore, the three directions can be analyzed separately with one history scaled to the proper maximum accelerations.

Impell used design response spectra to generate three acceleration time histories, one for each orthogonal direction: two horizontal and one vertical. These acceleration time histories constitute the free-field input ground motions that were used for the SSI analysis of the major N Reactor facilities housing the Reactor Core. The methods used for generating the free-field artificial time histories as part of the Phase I work are described in Reference 29. Figures of the three orthogonal time history motions are also contained in this reference. The east-west (E-W) time history motion is shown in Reference 25, Vol. II.

3.7.1.3 Critical Damping Values

Engineering judgment weighs heavily in the selection of values of critical damping to be used in design. After evaluating the piping and support systems proposed for the ECCS and RCS, critical damping for these systems was estimated to be around 1.7%. In 1967, Dr. N. M. Newmark and Associates

concurred in a report to the AEC Regulatory Staff (Reference 4) that this damping value was reasonable. Values of critical damping used in the original design of other important components of the 100-N facility are given below. These design values are equal to or less than SSE values of critical damping presented in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," issued October 1973 (Reference 26).

	Percent of Critical Damping	
	100-N Design	Reg. Guide 1.61
Reinforced concrete shear walls	5 to 7	7
Welded steel frame structures	1 to 2	4
Welded steel pipes with rigid supports	1 to 2	2 to 3

Damping values presented in Table 3.7-1 were used in an internal design review performed in 1968 (Refs. 5 and 6). The Impell evaluation was based on Regulatory Guide 1.61 Damping Values and are contained in Reference 32.

In 1987-88, seismic evaluations of 105-N, 109-N, 181-N, and 182-N Buildings were made. The buildings are mostly of reinforced concrete construction. The 182-N Building has a bolted steel superstructure, and the superstructure covering the diesels on the 181-N deck has reinforced masonry walls. Structure damping values used were those of Regulatory Guide 1.61 or less. A value of 10% was used in the reinforced masonry walls of the 181-N Building superstructure. This is the same as that of a draft EPRI seismic margins review guideline, Reference 42. Draft DOE guidelines, Reference 43, recommend 12% for walls with strains beyond yield. For block walls, Standard Review Plan, Section 3.8.4, Appendix A (Reference 8) specifies the same 7% damping value as reinforced concrete. Since the walls were at or above yield, the 10% value used was considered conservative in light of the above comparisons. The 105-N and 109-N Building analyses conservatively used 5% for the reinforced concrete compared to 7% in Regulatory Guide 1.61. Soil damping values are discussed in Section 3.7.1.4.

3.7.1.4 Supporting Media for Seismic Category I Structures

Major structures at N Reactor are supported on undisturbed soil consisting mainly of compacted gravel. An upper layer, consisting of about 50 ft of medium-dense gravel, rests on about 50 ft of very dense gravel. These two gravel beds consist of well-graded sand and gravel mixtures with some silt and occasional cobbles and boulders. In some local areas, the gravel is devoid of fine sand, and in other localities it may be cemented. The gravel layers are underlain by approximately 400 ft of blue-clay mixed with variable amounts of silt and fine sand. The clay layers are highly compacted and possess the physical properties of soft rock. The above clay and gravel media rest on a series of basalt layers having a total depth of over 10,000 ft. Soil properties for the above media are described in Table 3.7-2. The dimensions and depth of the excavations for the major buildings are summarized in Table 3.7-3.

The dynamic soil properties used in the Impell evaluation are presented in Table 3.7-5 and Figure 3.7-6. In addition, the engineering properties used by Impell and based on the above profile are presented in Table 2.5-5. 17

Soil properties given in Tables 3.7-2 and 3.7-5 are appropriate to all analyses if the zero depth is referenced to elevation 454 ft, the ground level of the 105-N, 109-N, and 182-N Buildings. The soil stiffness is specified to a depth of 250 ft, below which a stiff bedrock condition is used. See Section 3.8 for general descriptions of the buildings and soil embedments. The 105-N, 109-N, and 182-N Buildings are embedded 24 ft to 26 ft down from the 454-ft elevation. The 181-N Building has soil up to the 420-ft elevation on the east side, while the opposite side is dredged down to the top of the basemat at elevation 360 ft. 21

3.7.2 Seismic Analysis

3.7.2.1 Seismic Analysis Methods

The methods of analysis are basically tied to the soil-structure analysis for each building described below. Thus, time-history analysis is the basis for generating the seismic loads throughout the buildings. Stick models

were used in all but the 182-N analysis, so that torsional effects are accounted for by offset mass and stiffness centers. The 182-N Building has the same structural and mass properties in the two horizontal directions, therefore, eccentricity and coupling effects are not significant.

EQE Inc. (Reference 44) developed in-structure response spectra for the 105-N and 109-N Building complex and described the method as follows.

The layout of the N Reactor facility can be divided into three major structures for the purpose of calculating SSI response. These structures are: the process area (including the reactor itself) which is supported by a basemat about 50 ft by 90 ft; Building 105-N, the central portion of which surrounds the reactor and is about 140 ft by 210 ft; and Building 109-N which lies to the south of Building 105-N and has general dimensions of about 200 ft by 430 ft for the major portion of its structure. Both buildings are primarily constructed of shear wall systems supported by strip footings and both contain structures peripheral to their major portions. These peripheral areas would have a secondary effect on the reactor response, and therefore were not included in the SSI analysis.

SSI response of major structures is calculated by the CLASSI system of computer programs using the substructure approach to SSI. The substructure approach divides the SSI problem into a series of simpler problems (typically three), solves each independently, and superposes the results. The three steps are: determination of the foundation input motion; determination of the foundation impedances; and analysis of the coupled soil-structure system. The methodology to perform SSI analysis of structures whose foundations are assumed to behave flexibly is an extension of the substructure approach.

Free-field ground motion. Specification of the free-field ground motion entails specifying the control point, the frequency characteristics of the control motion (typically time histories or response spectra), and the spatial variation of the motion. In general, the control point is specified on the free surface of soil or rock, the control motion is acceleration time histories, and, in most cases, vertically incident plane waves are assumed,

which defines the spatial variation of motion once the soil properties are identified.

Foundation input motion. The foundation input motion differs from the free-field ground motion in all cases, except for surface foundations subjected to vertically incident waves. The motions differ for primarily two reasons. First, the free-field motion varies with soil depth. Second, the soil-foundation interface scatters waves because points on the foundation are constrained to move according to its geometry and stiffness. For vertically propagating seismic waves impinging on rigid surface foundations, the foundations input motion is the same as the free-field motion.

Foundation Impedances. Foundation impedances describe the force-displacement characteristics of the soil. They depend on the soil configuration and material behavior, the frequency of the excitation, and the geometry of the foundation. In general, for a linear elastic or viscoelastic material and a uniform or horizontally stratified soil deposit, each element of the impedance matrix is complex-valued and frequency dependent. For each rigid foundation, the impedance matrix is 6 x 6, which relates a resultant set of forces and moments to the six rigid-body degrees-of-freedom. 21

Structure model. The dynamic characteristics of the structures to be analyzed are described by their fixed-base eigensystem and modal damping factors. The structures' dynamic characteristics are then projected to a point on the foundation at which the total motion of the foundation, including SSI effects, is determined. For structures supported on multiple foundations, their "pseudo static" characteristics, which describe their behavior to their support displacements, must also be obtained. The structural characteristics are then projected to a point on each foundation.

SSI analysis. The final step in the substructure approach is the actual SSI analysis. The results of the previous steps--foundation input motion, foundation impedances, and structure model--are combined to solve the equations of motion for the coupled soil-structure system. For a single rigid foundation, the SSI response computation requires solution of, at

most, six simultaneous equations--the response of the foundation. For a structure supported on N foundations, the solution of 6N simultaneous equations is required.

The SSI analyses of the N Reactor facility were carried out using a multiple foundation version of the CLASSI system of computer programs. These programs are used for SSI response calculations for structures that are supported either on multiple foundations or flexible foundations.

3.7.2.2 Natural Frequencies and Response Loads

Dominant building frequencies determined in the 1987-88 reanalysis, involving deformation of the massive substructures with a fixed base, are as follows.

Building	Direction	Frequency, Hz
105/109	NS	12
105/109	EW	14
105/109	Vertical	28
181	NS	7
181	ES	18
181	Vertical	32
182	NS/EW	~13
182	Vertical	~15

3.7.2.3 Procedure Used for Modelling

The procedure used by EQE for modelling is described fully in Reference 44 and briefly summarized below. Three key elements are involved: free field ground motion, foundation impedances, and the structural models.

Three synthetic acceleration time histories were generated for use as the three-field input motions for the SSI analyses. Two time-histories were scaled to a maximum acceleration of 0.25 g for use as the horizontal components and the third had a 0.167 g maximum acceleration as the vertical component.

The frequency content of the synthetic time histories was such that their response spectra enveloped the N Reactor SSE design response spectra at 5% damping, satisfying the US NRC criteria set forth in the Standard Review Plan, Sec. 3.7.1. It was also verified that the time histories met the NRC criteria for statistical independence between components as set forth in Regulatory Guide 1.92.

For the purpose of calculating impedances, the N Reactor structures were considered to be supported on two separate rigid foundations; Building 105-N and the reactor core being supported by one, and Building 109-N by the other.

A study of the available structural drawings showed that the foundations of the three structures are connected to each other, either directly or through the structures, to a varying extent. The reactor basemat is connected to Building 105-N through two heavy shear walls which run east and west along either side of the basemat. This connection causes both foundations to move as a single unit. Hence, both were considered to be supported on the same foundation. 21

The orientation of the shear wall running east and west would tend to cause the foundations of both buildings to move as a single unit in the E-W direction. However, the connection between the buildings in the north-south (N-S) direction is not completely rigid; therefore, impedances were calculated for two separate foundations, coupled only through the soil, and the restraints caused by the connections between the buildings and foundations were included in the structural model.

The procedure used to compute the impedances involved first using the CLASSI program GLAY to calculate Green's functions, which are steady state dynamic force and displacement relationships between a point source and observation points as various specified distances on the surface of the layered half-space. This was done for a number of specified frequencies. The soil properties used for the GLAY analysis were those used in Reference 31 and shown in Table 3.7-6. The Green's functions were then used, along with the above to generate the foundation impedance functions at the specified

frequencies. This was accomplished using CLASSI program CLAN. The impedance functions were calculated by applying the constraints of rigid body motion between all subregions of each foundation to obtain a 12 x 12 complex-valued impedance matrix at each specified frequency. The impedance matrices give the total steady state dynamic force and displacement relationship, including phase information, of the rigid massless foundations on the soil deposit.

Lastly, a structural model of the coupled Building 105-N, Building 109-N, and the Process area was developed for use in the SSI analysis. Initially, a separate model was defined for each of these three major structures. The final coupled model was formed by joining the structures and the Process Area as necessary. From the eigenvalue extraction of the model, twenty-three modes were obtained with frequencies up to 35 Hz.

The flexibility of floor slabs was not included in the model. A separate study showed that, except for the roofs of both buildings, the floors of the west rupture monitor room (El. -16 ft), and electrical room corridor (El. 16 ft), all floor slabs have fundamental frequencies above 25 Hz, which is above the frequency range of amplified response to earthquakes. The dynamic behavior of the slabs noted above was included in models used for the vertical analysis of components or equipment that is supported away from the walls.

A description of the 105-N and 109-N Building model and method utilized by Impell in the 1985 evaluations of the reactor core is provided in Reference 29 and discussed in 3.7.3.3.

3.7.2.4 Soil-Structure Interaction

All buildings were analyzed for soil-structure interaction using time history methods. The CLASSI system of computer programs was used for the 181-N, 105-N and 109-N Buildings, while FLUSH was used for the 182-N analysis. The CLASSI analyses use typical finite element models for the structure, and account for soil effects with impedances (frequency dependent springs and dampers) determined from iterated free-field soil properties and the kinematics of a rigid basemat. These analyses are fully three dimensional

in nature, and they account for torsional effects. The FLUSH analysis is based on a two-dimensional model of the building and soil, while the three dimensional effects, specifically radiation damping, is approximated in the program by viscous dampers. The foundation is not assumed rigid. Both types of analyses employ vertically incident plane waves with either horizontal or vertical particle motion.

In effect, all the seismic control motions were specified at the 454-ft elevation. Iterated soil properties in the free field, away from the buildings, were determined with SHAKE analyses which produce the shear modulus and damping vs depth. Such properties were used in all analyses.

Embedment effects are accounted for in the FLUSH program itself for the 182-N Building analysis. In the CLASSI analysis of the 181-N Building, the one sided embedment was accounted for by adding one half the impedance change associated with the embedment of an equivalent circular cylinder. This approach requires that the motion at a massless foundation be determined for input to the CLASSI program. This was done using a FLUSH model of a E-W slice of the building, and the resulting motion was used for NS excitation as well as the EW excitation. 21

3.7.2.5 Development of Floor Response Spectra

Floor response spectra at 5% damping were obtained from the time history solutions at appropriate directions and floor locations. Frequency dependent factors, as discussed below in the 105 and 109-N Buildings, were applied to account for embedment. ASME Code Case N-411, which calls for damping graduated from 5% to 2% in passing from 10 Hz to 20 Hz frequencies, was followed. Spectral accelerations at damping values lower than 5% were obtained by scaling with the square root of the ratio of 5% and the desired damping value. This is an appropriate trend found among seismic records. Finally, all spectra were broadened plus and minus 15% to account for uncertainties in the frequencies and other parameters.

3.7.2.6 Three Components of Earthquake Motion

Dynamic structural models were used to calculate the structural responses to the two horizontal components and the vertical component of ground motion.

Section 3.7.8 was used as a source for the material properties that were used to calculate the stiffness of the structures. In general, the stiffness of reinforced concrete structural members were calculated using gross cross sections.

Mass calculations included the deadweight of the structure and major pieces of equipment. The mass of nonstructural elements (e.g., partitions), small pieces of equipment (e.g., electrical cabinets), and piping was taken as a uniform load distributed across the whole floor.

3.7.2.7 Combination of Modal Responses

Discrete models of the graphite and reactor core structure components were coupled to the dynamic structural model of the 105-N/109-N Buildings.

Structures with load-resisting systems were connected to each other and analyzed using coupled dynamic models, except where it could be shown that coupling did not significantly influence relevant structural responses.

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

The interfaces between Seismic Category I and Non-Seismic Category I structures and plant equipment have been reanalyzed for the dynamic loads and displacements of both Seismic Category I and Non-Seismic Category I structures and plant equipment. In addition, all Non-Seismic Category I structures met one of the following requirements:

1. The collapse of any Non-Seismic Category I structure will not cause the Non-Seismic Category I structure to strike a Seismic Category I structure or component.
2. The collapse of any Non-Seismic Category I structure will not impair the integrity of Seismic Category I structures or components.

3. The Non-Seismic Category I structures are analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.
4. The collapse of Non-Seismic Category I structures will not prevent the functioning of Seismic Category I structures or components.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Variations in structural properties, damping, soil properties, and soil/structure interaction parameters could result in the shifting of calculated periods to resonant peaks in the floor response spectra. To account for the effects of such variations on the shape of calculated floor response spectra, smooth design envelopes (which incorporate a minimum shift of the periods corresponding to calculated resonant peaks) were developed and used in seismic design of systems, equipment and components.

21

Embedment effects for the 105-N/109-N Buildings were accounted for by adjustment to the floor spectra. Independent reviews of the seismic response spectra for the 105-N and 109-N Buildings generated by EQE noted several areas of conservatism in the analysis. The original analysis by EQE was conducted with the soil between the ground surface and foundation level completely removed. The control point at which the design time history is applied was then at the foundation level in the free field. In addition, the envelope of the response of this time history over the design response spectrum has overshoots that would persist to floor spectra generation.

Consequently, the EQE floor spectra for locations in the 105-N/109-N Buildings were modified with factors accounting for the following two effects:

1. Excessive loading represented by overshoots of the calculated spectrum over the design spectrum was compensated for by frequency dependent ratios determined by these two spectra.
2. The effect of soil removal down to the foundation level was determined with FLUSH analyses conducted by Westinghouse Hanford. Models of the two buildings with and without this soil produced responses at various

locations. Comparison of the two model results had the effect of moving the control point from the foundation level in the EQE analysis to the 454-ft elevation and of adjusting the soil properties accordingly. Thus, frequency-dependent ratios were determined from the corresponding spectra for application to floor spectra to account for the embedment effect. These ratios were dependent on elevation in the buildings and excitation direction, and are also applicable to horizontal motion only. The minimum factor was about 0.7 in the 3 to 5 Hz range. The Westinghouse Hanford work was reviewed by EQE and it was requested that the CLASSI and FLUSH approaches be shown to give the same results. This was done, and EQE concurred with the use of the reduction factors.

Further details of justification and procedures for developing reduced response spectra for 105-N and 109-N Buildings are provided in Reference 45.

3.7.2.10 Use of Constant Vertical Static Factors

Seismic Category I structures, systems, and components are subjected to a vertical dynamic analysis with the vertical response spectra defining the input, with the exception that a static analysis can be performed in lieu of dynamic analysis as described in 3.7.2.1.9.

3.7.2.11 Method Used to Account for Torsional Effects

Significant eccentricity between the center of mass and the center of stiffness in a structure will induce torsional responses when the structure is subjected to horizontal components of ground motion. Such eccentricity was taken into account in the modeling of the structures. In addition, for nonsymmetrical structures, a stiffness analysis is performed to determine torsional effects on vertical structural elements resisting lateral loads. Inertial forces, determined from dynamic analysis of structures, are applied at the center of mass for each floor. Thus, torsion effects are introduced in each story by applying a twisting moment about the center of rigidity of the story under consideration. This moment is calculated as the sum of the products of the inertial force applied at the center of mass of each floor above and a lever arm equal to the distance from the center of mass of the floor to the center of rigidity of the story. The lever arm is not less

than the minimum eccentricity required by the Uniform Building Code. The torsional moment and story shear are distributed to the resisting elements in accordance with the provisions of the Uniform Building Code.

Symmetrical structures are analyzed in a similar manner for torsional effects using minimum eccentricities between the center of mass and the center of rigidity as defined by the Uniform Building Code.

3.7.2.12 Comparison of Responses

A discussion of the response spectrum and time-history method of dynamic analysis used in the seismic reanalysis of the N Reactor facility Category I structures is provided in Sections 3.7.1.1 and 3.7.1.2.

3.7.2.13 Methods for Seismic Analysis of Dams

Dams are not provided at the N Reactor Site. 21

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning and sliding effects of horizontal seismic loadings have been considered in combination with the effects of vertical seismic loadings. The seismic loads used consist of two horizontal orthogonal and vertical components of earth-quake motions. Each horizontal component is taken separately and is applied concurrently with the vertical component. For Seismic Category I structures, the results of the dynamic analysis have been converted to equivalent static loads at the mass points.

To calculate the capability of safety-related structures to resist overturning, the following load combination is generally considered

$$D + E' + Q^* + \text{Uplift}$$

Where:

- D = The dead load, also includes the weight of dry and saturated backfill, as applicable.
- E′ = Loads generated by the SSE.
- Q* = Lateral pressure, including seismic effect; on subgrade walls due to the dry and saturated soil as applicable.

Uplift = The uplift force, is taken as the weight of the water displaced by the structure, acting vertically upward and applied to the bottom surface of the basemat.

The overturning moments and the stabilizing moments are calculated about the lower edge (toe) of the basemat. The safety factor against overturning is calculated by dividing the total stabilizing moment by the total overturning moment.

To calculate the capability of safety-related structures to resist sliding, the load combinations considered are the same as those listed above for calculating the capability of safety-related structures to resist overturning. The safety factor against sliding is calculated by dividing the frictional force resisting sliding between the basemat and the soil by the summation of horizontal forces causing sliding.

The factors of safety against overturning and sliding for safety-related structures are discussed in 3.8.

3.7.2.15 Analysis Procedure for Damping

For structures and components, damping coefficients are selected as discussed in 3.7.1.3.

EQE (Reference 44) used both a rigid foundation approach and a flexible foundation approach when modeling foundation responses. The treatment of damping is different for the two approaches.

The flexible foundation formulation requires explicit treatment of coupling damping terms which obviously do not exist for the single rigid foundation case. These damping terms do not lend themselves to a modal representation. Hence, when applying the flexible foundation formulation in a reduced form to a single rigid foundation, a slight difference in the formulations remains. A series of sensitivity studies were performed to investigate the treatment of damping. The conclusion was to treat damping in two ways: for foundation response, damping is assumed to be of a hysteretic form; for in-structure

response, it is assumed to be viscous modal damping. The differences are small, however, in benchmarking the technique this treatment was necessary. 21

3.7.3 Seismic Subsystem Analysis

This section describes seismic reanalyses for the following subsystems: reactor coolant system, supports, ECCS, auxiliaries, wall penetration, and buried piping.

3.7.3.1 Seismic Analysis Methods

The general approach to the seismic subsystem analysis is identical to those procedures described in 3.7.2 for seismic system analysis, except for the soil/structure interaction effects. As discussed in Section 3.7.2.9, the effects of 22 ft of soil between the 105-N/109-N Building foundation level and the surface had not been considered in the original response spectra developed by EQE (Reference 44). These and other conservatisms in the analysis (damping) particularly affected the reanalysis of piping and equipment. With the concurrence of EQE, reduction factors were generated that could be applied to the EQE spectrum. 21

With the exception of the GSCS piping which had been reanalyzed using EQE spectra without reduction factors, and the Impell analysis of the graphite moderator, core, and shield structure, all piping and equipment seismic reanalysis was performed with the EQE spectra modified by the reduction factors discussed in Reference 45.

The reduction factors generated for the 22-ft soil replacement are repeated here as Figures 3.7-7 through 3.7-10. These factors vary with elevation and frequency and are applied to the peaks of the raw EQE spectrum. The resulting reduced spectrum is then broadened ±15% from each frequency peak to meet NRC requirements. An envelope is then used to connect the peaks and the resulting envelope spectrum used for seismic qualification. An example is shown in Figure 3.7-11. Note that linear interpolation may be used to generate reduction factors and spectra at elevations other than shown.

A comparison of the time history response spectra used in the EQE analysis with the required design response spectra are shown in Figures 3.7-12, 13, 14. Additional reduction factors can be obtained by dividing the design spectra acceleration by the time history acceleration at each frequency of interest. These factors can then be applied in addition to the soil replacement factors to the raw EQE spectra. The resulting response is again broadened ±15% to give minimum required spectra for seismic qualification. Note that this reduction factor is constant with elevation. An example is shown in Figure 3.7-15.

3.7.3.2 Determination of Number of Earthquake Cycles

The procedure used to develop an acceleration time history based on actual earthquake models is described in 3.7.1.2. Development of reduced response spectra, responsive to the 0.25 g criteria, and applicable to piping and equipment is discussed above in 3.7.3.1.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals occur below 50% of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake was determined in the following manner:

1. The fundamental frequency and peak seismic loads were determined by a standard seismic analysis, i.e., from eigenvalue extractions and a forced response analysis.
2. The number of cycles that the component experiences were determined according to the frequency range within which the fundamental frequency lies.
3. For fatigue evaluation, 0.5% (0.005) of these cycles were conservatively assumed to be at the peak load and 4.5% (.045) at or above the three-quarter peak. The remainder of the cycles had negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of an SSE is so small that it was not necessary to postulate the possibility of more than one SSE occurring during the 40-year life of a plant. Fatigue evaluation due to the SSE was not necessary because it is considered a faulted condition and thus not required by the ASME B&PV Code, Section III.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Modeling Techniques - Reactor Coolant Loop and ECCS Piping

The piping system components and component supports were represented by an ordered set of data which numerically described the physical system.

The spatial geometric description of the model was based on the as-built isometric piping drawings and equipment drawings. Node point coordinates and incremental lengths of the members were determined from these drawings. Node point coordinates were input on data sheets. Incremental members lengths were input on the data sheets. The geometrical properties along with the modulus of elasticity, the coefficient of thermal expansion, the average temperature changes from the ambient temperature, and the weight per unit length were specified for each element. The supports were represented by stiffness matrices, which define restraint characteristics of the supports. 21

Network models were made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section were used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section were obtained. These loads were incorporated into the overall load vector.

After all the sections had been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points was determined. The flexibility matrix was multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer

relationship, the deflections and internal forces were then determined at all node points in the system. The support loads were also computed by multiplying the stiffness matrix by the displacement vector at the support point.

The models used in the static analyses were modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment. The lumping of the distributed mass of the piping system was accomplished by locating the total mass at points in the system which approximately represent the response of the distributed system. Effects of equipment in the overall system were modeled when required. The supports were again represented by stiffness matrices in the system model for the dynamic analysis. From the mathematical description of the system, the overall stiffness matrix was developed from the individual element stiffness matrices using the transfer matrix associated with mass degrees-of-freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes were determined.

The effect of eccentric masses, such as valves and extended structures, were considered in the seismic piping analyses. These eccentric masses were modeled in the system analysis, and the torsional effects caused by them are evaluated and included in the total system response. The total response had to meet the limits of the criteria given.

The lumped mass model of a typical loop of the four-loop coupled model is shown in Figure 3.7-11. The total model was assembled from variations of this model. The coupled model was used to come up with the loads and stresses in the system.

3.7.3.4 Basis for Selection of Frequencies

All frequencies in the range of 0.25 to 33 hertz were considered in the analysis and testing of systems, components, and equipment. These are the frequencies excited under the seismic excitation.

If the fundamental frequency of a component was greater than or equal to 33 hertz, it was treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 hertz were not considered as they represent very flexible structures not encountered in this plant.

The frequency range between 0.25 hertz and 33 hertz covers the range of the broad-band response spectrum used in the reanalysis.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load method was used when the natural frequency of the equipment was not determined. If the equipment could be adequately represented by a single degree-of-freedom system, then the applied inertia load was equal to the weight of the equipment times the peak value of the response spectrum, curve. Seismic acceleration coefficients for multiple degree-of-freedom systems, which were in the resonance region of the amplified response spectra curves, were increased by 50% to account conservatively for the increased modal participation. 21

3.7.3.6 Three Components of Earthquake Motion

The simultaneous use of three components of earthquake motion was not a design basis requirement for this plant. However, the selected components and equipment were in the reanalysis to the requirements of Regulatory Guide 1.92.

3.7.3.6.1 Response Spectrum Method

The response spectra generated were developed considering three components of earthquake motion. The individual floor response spectra in each orthogonal direction were obtained by the SRSS combination of the collinear contribution due to the three directions of earthquake motion. These were used to predict the total response at each frequency.

3.7.3.6.2 Time-History Method

When the time-history method of analysis was used, the time-history responses from each of three components of the earthquake motion were combined

algebraically at each time step. The maximum response was obtained from this combined time solution.

For equipment, cable trays, supports for cable trays, and HVAC ducts, the three spatial components of the earthquake were considered in the same manner as structures described in Section 3.7.2.6.

The criteria used for combining the results of horizontal and vertical seismic responses of piping systems are described in Section 3.7.3.1.1.

3.7.3.7 Combination of Modal Responses

All piping and equipment that was reanalyzed was evaluated to the requirements of Regulatory Guide 1.92.

When the response spectrum method modal analysis was used, all modes except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10%) were combined by the square root of the sum of the squares (SRSS), as described in Section 3.7.3.3.1. When closely spaced modes were present, the SRSS method was modified to account for these. 21

3.7.3.8 Analytical Procedure for Piping

3.7.3.8.1 Analytical Procedure for Safety-Related Piping

The analytical procedures for piping analyses are described in Section 3.7.3.1.1. Methods to include differential piping support movements at different support points are also described.

3.7.3.8.2 Analytical Procedure for Safety-Related Piping

The design criteria and the analytical procedures applicable to components and equipment are as described in Section 3.7.3.1.2.

3.7.3.9 Multiple-Supported Equipment Components with Distinct Inputs

3.7.3.9.1 General

The procedure and criteria for the analysis of multiple-supported equipment components with distinct inputs are described in Section 3.7.3.1.2.

3.7.3.9.2 Independent Support Motion

Where piping is located in more than one building or supported at more than one building elevation, the existing criteria specify the use of an envelope of applicable floor spectra. The proposed alternate approach is to utilize Independent Support Motion (ISM), wherein each pipe support is driven by its applicable floor spectra rather than by the envelope spectra. The ISM response spectra technique has been in the ASME Code, Appendix N since 1978 and has been used extensively on BWR plants. Additionally, NUREG-1061 (Volume 4, Section 2.4) recommends that the ISM response spectrum method be allowed as an option in calculating the response of multiple-support piping with independent inputs. 21

3.7.3.10 Use of Constant Vertical Static Factors

Constant vertical static factors were not used in the seismic reanalysis of equipment and piping systems.

3.7.3.11 Torsional Effects of Eccentric Masses

When the torsional effect of an eccentric mass is likely to have a significant effect on the result of an analysis, the eccentric mass is included in the analytical mode. If the pipe stresses due to an eccentric mass are expected to be insignificant, the offset moment due to the eccentric mass is usually neglected.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

No buried Seismic Category I tunnels are utilized in this facility. Pipe penetration through walls and floors is by use of pipe sleeves.

Underground nuclear safety related piping is designed to safely resist operating loads and loads due to accident conditions which include seismic

waves passing through the soil media supporting these elements and relative seismic displacements between building and surrounding soil. Analysis of these underground pipes subjected to ground motion is based on their configuration and boundary conditions and the elastic properties of the soil and piping.

The method of analysis as suggested by N. M. Newmark in Reference 46 is used in the analysis of the bending stresses in buried pipes. The method of analysis developed in Reference 46 is used for predicting the stresses of buried pipes in the free field.

The method addresses primarily the axial stresses induced in pipe runs which are parallel to the direction of soil strain. Since the response behavior of a buried pipeline is influenced by its physical parameters and by the governing geotechnical and seismological parameters, these parameters are manipulated to determine the forces, moments, and stresses on the pipe element. The method neglects strains induced by ground curvature and the negligible dynamic effects on buried piping response.

This analysis method involved four (4) distinct phases.

1. Physical, geotechnical, and seismological data were assembled. The physical and seismological data were readily available. Geotechnical data were developed using engineering judgments, data available from a geologically similar site and combined with tabulated parameter values for the soil covering the pipes to form the geotechnical database.
2. Intermediate parameters such as the soil pipe interaction constant and maximum length, L′, over which friction effects are important, were calculated.
3. Pipes were then classified as long or short based on the magnitude of L′ relative to the length at which there was negligible additional influence on the forces and moments.
4. Finally, deflections, forces, moments, and code stresses were calculated and compared to allowables.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

When Non-Seismic Category I piping is attached to Seismic Category I piping, that portion of the other piping up to the nearest piping anchor or terminal point is also analyzed as Seismic Category I piping. The Non-Seismic Category I piping is designed to withstand the SSE without failing in a manner that would cause the Seismic Category I piping to fail.

A comprehensive walkdown of all Seismic Category I piping was also conducted which included documentation of all potential Non-Seismic Category I piping hazards by photographs followed by subsequent analysis of the potential hazard and hardware modification of fixes as required. The following guidelines for analyses, to be used only if screening techniques, such as zone-of-influence and impact energy, were not successful in problem resolution were developed (Ref 47):

21

1. Assumptions/Ground Rules

   Functional failure of Non-Category I components may be allowed. Support integrity must be maintained. Consequences of functional failure must not jeopardize Category I equipment (e.g., failure of water pipe can short IE electrical equipment). Sufficient clearance must be available to assure that deformation does not cause interaction with adjacent structures and jeopardize Category I equipment.

   Category I components can be protected by preventing failure/gross distortion of Non-Category I components (no missiles) or by providing shielding (missile protection).

2. Applied Loads

   Loading to be considered in the evaluations should include where applicable:

   - o Seismic inertia
   - o Gravity
   - o Pressure
   - o Thermal
   - o Interface reactions (i.e., nozzle loads).

Thermal is considered only if operating temperatures exceed 150° F.

3. Seismic Inertia Loading

Components and component supports will be evaluated using simplified models and conservative hand calculations. The first approach is to use the peak spectra acceleration of the applicable response spectra multiplied by 1.5 to obtain the maximum equivalent static seismic load.

As required, a mode/frequency analysis can be used to determine seismic loads. If the natural frequency of the component is in the rigid range ($\geq$30 Hz); an equivalent static analysis with ZPA accelerations can be used. Alternatively, the accelerant can be determined based on the fundamental frequency using 1.5 times the highest 'g' level at or above this frequency.

4. Damping

For III/I evaluations use damping as recommended by Regulatory Guide 1.61. Increased damping may be used when justified by test data/experience. For example, piping and conduit may use 12% damping. In this case apply the 5% spectra 'g' levels without the 1.5 multiplier discussed above. In any case a 'g' level less than the ZPA value is not allowed.

5. Combining Earthquake Directions

The three orthogonal components of seismic motion (loading) should be considered. Combination by square root sum of squares or other combinations allowed by Reg. Guide 1.92 is acceptable. Typically in piping/conduit analyses only the worse direction loading need be considered.

6. Seismic Load Determination for Piping/Conduit Supports (Contributory Mass)

The amount of contributory mass to be applied at each support should be one-half the weight of the uniformly distributed weight between two adjacent restraints acting in the same restraining direction, plus a

proportional amount of any concentrated weight between those two restraints. This contributory mass is then multiplied by appropriate spectra acceleration to obtain seismic support loads. In support analysis, all three orthogonal directions of seismic motion must be considered.

Uniform loads include but are not limited to weight of pipe, insulation, and fluid. Concentrated weights include valves, flanges, and pipe appurtenances.

7. Concrete Anchor Allowables

Site specific concrete anchor bolt allowables based on test data at N Reactor were not available. In lieu of this data, reduced concrete anchor allowable loads are provided for screening purposes. Anchor status should be determined to show existing anchors (1) adequate (2) require test or (3) need replacement. Table 3.7-6 represents the lower bound concrete anchor allowables which can be used to determine if existing anchor loads are acceptable, or if a verification test or replacement is required. Table 3.7-7 provides concrete anchor allowables for new designs. This table will also be used to screen existing designs. If calculated applied loading is larger than the allowable used for new design, rework of the support is required. 21

All concrete anchors which fall between the loads provided in Tables 3.7-6 and 3.7-7 shall be identified using the baseplate information form provided in Attachment 1 to provide data for the further testing.

Combined loading shall be calculated using the interaction equation for pure shear and pure tension as follows:

$$\left(\frac{S\ (\text{applied})}{S\ (\text{allowable})}\right)^{4/3} + \left(\frac{T\ (\text{applied})}{T\ (\text{allowable})}\right)^{4/3} \leq 1.0$$

Anchor or spacing and edge distance for 100% effective anchor performance shall be as follows:

Minimum anchor spacing = 10 bolt diameters
Minimum edge distance = 5 bolt diameters

Where required by application, anchor efficiency may be reduced on a straight line basis down to 50% at 5 bolt diameters center-to-center anchor spacing.

8. Component Allowable Stresses/Loads
   Guidelines on allowable stresses for III/I components are provided in Table 3.7-8.

9. Component Support Allowable Stresses/Loads
   Guidelines on allowables for III/I component supports are provided in Table 3.7-9.

3.7.3.14 Seismic Analyses for Reactor and Graphite Structure

3.7.3.14.1 Structural Analysis

Nonlinear seismic analyses of the reactor core were performed to access the functional operability of the reactor safety system and to assess the structural integrity of the core, the primary shield, and the reactor pedestal. Results of the evaluation are given in Section 3.8.3.

In Phase I (Reference 29), three analysis cases were evaluated:

1. As-built--All graphite keys active
2. Extended-Operation--Graphite keys inactive in the fueled zone but active in the reflector
3. Extended-Operation--Graphite keys partially active in the fueled zone and active in the reflector.

The Phase I effort showed that Case 2 was the critical case and that it was the most conservative representation of long term reactor operation; therefore, it was the only case evaluated in Phase II.

Under Phase II, model sensitivity studies were performed to verify that the model discretization used in Phase I was satisfactory for the Phase II work. Then, Case 2 was reanalyzed accounting for modeling refinements and for refined material properties of the reactor core. In addition, horizontal control rod (HCR) T-block analyses were performed as part of Phase II. The T-blocks were not evaluated in Phase I.

3.7.3.14.2 Model Sensitivity Study | 21

For the model sensitivity study, a more detailed reactor core model was developed and analyzed. The response of the detailed model was then compared to that obtained using the Phase I model. All evaluations were performed for the extended-operation case with no keys effective in the fueled zone (Case 2).

The lumped-mass and beam model developed under Phase I. It consists of four lumped-mass, vertical sticks which are coincident. Two sticks represent the biological shield (one is the primary shield and the other is the inlet/outlet shield). The lumped-masses for these sticks are connected by linear-elastic beam elements. The other two sticks represent the graphite reactor core. Due to the highly nonlinear response of the graphite core, nonlinear elements are used for these sticks. The reflector region of the core is modeled as five lumped-masses and is connected by nonlinear friction and gap elements. The fueled zone is modeled as three lumped-masses which are also connected by nonlinear friction and cap elements. Because five lumped-masses are used to model the reactor core reflector zone, the model is referred to as the "five-layer model." To verify the five-layer model, a more detailed ten-layer model was developed. The same modeling concepts used for the five-layer model were used.

A comparison of response time histories for the two models showed that correlation was very good. In the N-S direction, the maximum variance in

the peak response in the core was three percent, and in the E-W direction, the maximum variance was twenty percent. The phasing of the response time-histories was also very good. Although a twenty percent variance is not excessive for nonlinear analysis, additional analyses were performed to assess the parameters which might affect the response.

Stiffness and mass distribution for the two models were assessed by performing frequency/mode shape analyses and by performing nonlinear static load-deflection analyses for both models. Also, variances due to small differences in structural damping and due to changes in convergence tolerance were assessed by performing additional nonlinear time-history analyses.

Variances in reactor core response due to the parametric changes were small. Details of the parametric studies and plots of comparison time-histories are given in Reference 25, Vol II. Based on these comparisons, the five-layer model with the Phase I convergence tolerance was considered to be adequate for the Phase II work.

3.7.3.14.3 Phase II Model

The structural model for the Phase II analyses of the reactor core represents the extended-operation case with no keys effective in the fueled zone. The Phase II model is identical to the corresponding Phase I model with these exceptions:

Nonlinear Gap Elements Between the Primary Shield and the Inlet/Outlet Shield. There is a 3/4 in. gap between the primary shield and the inlet/outlet shields which provides for thermal expansion of the inlet/outlet shields. In the Phase I analysis, the dynamic response of the concrete shields resulted in the displacements which close this gap; however, a gap element was not included to account for the impact. In Phase II, a gap element was added to account for the possible impact between the primary shield and the inlet/outlet shields.

Revised Properties of the Graphite Keys. Based upon new design criteria provided in Reference 32, stiffness properties of the (graphite) filler

block keys were revised. The properties of the pressure tube block keys were not changed by Reference 32; therefore, the original Phase I properties were used.

Revised Friction Forces. Friction between layers of graphite blocks is modeled using classical friction theory (i.e., the maximum lateral friction force is proportional to the normal force acting on the surfaces of the blocks). Due to the vertical earthquake, the normal force varies with time. To account for the variation in the normal force, the Phase I friction forces were conservatively reduced by the peak acceleration value of the vertical earthquake.

In Phase II, a more realistic approach was used to reduce the normal force. A friction force reduction factor was based on the root-mean-squared (RMS) value of the vertical acceleration time history. This value is considered to be more appropriate since it considers the time varying nature of the vertical earthquake acceleration and since it is directly related to the energy content of the motion. More details of this approach are covered in Reference 25, Vol. II. The Phase II approach resulted in larger friction forces and was therefore less conservative than the Phase I approach.

Revised Input Time Histories. The time histories used in the Phase II nonlinear analysis of the reactor core were based on the Phase II soil-structure interaction analysis.

Overall, the spectra are very similar. However, the Phase II spectra are higher in the 3 to 7 Hertz range. This is due to stiffer soil properties used in the Phase II soil-structure interaction analysis. Reference 25, Vol. II provides further details of the Phase II model.

3.7.3.14.4 Dynamic Response

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The nonlinear time history analysis of the reactor core was performed for the first 8.4 seconds of the twenty second input time history. The strong motion portion of the twenty second motion begins at about 1.5 seconds and tapers off at about 11.5 seconds. Thus, the duration of strong motion used

in the analysis is approximately 7.0 seconds. This duration of strong motion is longer than that of typical earthquakes. Therefore, the response predicted by the Phase II analysis is considered to be a conservative estimate of the response that would occur during a "real" earthquake. The response of components of the reactor core are described below.

Graphite Reactor Core. In general, the Phase II displacements were of lower magnitude than the Phase I displacements. This decrease in response was attributed to the increase in friction forces used in Phase II.

In the N-S direction, the reactor core slides essentially as a rigid body at its base with little relative displacement between other nodal points. This behavior is due to the core being unkeyed at the base for motion in the N-S direction. All nonlinear response is concentrated at this location. The displacements predicted by the Phase I and Phase II analyses in the N-S direction diverge at about 2.5 seconds. This is a result of sliding in the Phase I model which does not occur in the Phase II model due to increased friction forces.

In the E-W direction, the response of the reflector and the fueled zone is more like a shear building; that is, the response increases with height and the peak response occurs at the top of each structure.

Concrete Primary Shield. In general, the Phase II displacements are higher in magnitude than the Phase I displacements. This can be attributed to the increase in the Phase II input spectral accelerations at the fundamental frequencies of the primary shield.

Concrete Inlet/Outlet Shields. The displacement time history of the inlet/outlet shield in the E-W direction was compared for Phase I and Phase II and also the force time history for the gap elements between the inlet/outlet shield and the primary shield. The gap closes twice, producing two short duration load pulses in the gap elements.

Since the inlet/outlet shield is uncoupled from the primary shield in the E-W direction, the inertial load of the graphite core is transmitted through the pressure tubes to the inlet/outlet shields. The response of the core has decreased in Phase II due to increased friction forces; therefore, the inlet/outlet shield response is also decreased in Phase II.

3.7.3.14.5 HCR T-Block Analysis | 21

Graphite blocks, having the cross-sectional shape of a "T" line the horizontal control rod (HCR) channel. The T-blocks are 48 in. long and are "keyed" into the pressure tube blocks at the top and bottom of the T-block. Analyses were performed to determine whether or not a T-block could become dislodged during a seismic event and block the HCR channel.

To evaluate this, a block of graphite was subjected to a worst-case seismic excitation, and its displacement response was monitored. The worst-case seismic excitation was obtained from the nonlinear dynamic analysis of the reactor core. Frictional resistance of the block was due to the surface contact between two blocks and was a function of the coefficient of friction and the block's weight. Displacement time histories of the input and response were obtained and indicated the maximum seismic displacement of the block was 0.09 in. A translation of the T-block of this magnitude will not restrict the HCR from entering the channel.

An additional concern was that breakage of the graphite keys or local arching of the core could result in the T-block overturning and falling into the HCR channel. Three potential cases for this to occur were evaluated. For the first case, the pressure-tube-block keys have cracked and dislodged and provide no restraint to the T-block. For the second case, the T-block has "walked" over the pressure-tube-block key. For the third case, the T-block keys have cracked.

Overturning due to a seismic event for the first and second cases is unlikely. The location of the T-block's centroid would result in a free standing T-block Ooverturning and falling into the HCR channel if it were not supported at the top key. If the core was arched such that the T-block was unsupported

at the top key the T-block would overturn and fall into the channel regardless of the occurrence of a seismic event. Therefore, if the T-block has not already overturned into the HCR channel due to local arching of the core, it is unlikely that it will overturn due to a seismic event.

Overturning due to a seismic event for the third case is also unlikely. A rigid body dynamic analysis shows that the block will not overturn provided that the block is not displaced more than 0.8 in. into the channel prior to the seismic event.

Based on these evaluations, and to provide an adequate margin of safety, the T-blocks will not restrict the HCR provided that the T-blocks do not extend more than 1/2 in. into the HCR channel prior to the seismic event.

3.7.3.15 Analysis Procedure for Damping

Damping values used for Seismic Category I subsystems are similar to those discussed in 3.7.1.3.

Damping Values. The model Impell used for the SSI analyses consists of mathematical representations of the reactor building (Building 105-N), heat exchanger building (Building 109-N), and a simplified representation of the graphite core and shield wall system. The buildings are primarily heavily reinforced concrete structures. The damping values and stiffness parameters used for each of the above structures are contained in Reference 32. A damping value of 5% was chosen for all reinforced concrete elements. For the SSE excitation, typically a 7% value is recommended (Reference 26). Thus, the 5% value used is conservative.

The soil damping values used in the SSI analyses correspond to soil strains obtained from a free-field analysis of the site's soil profile. The variation of soil damping and shear modulus with soil strain that was assumed for this analysis is based on the normalized data provided in Reference 32.

Soil Media for Category I Structures. The reactor building (Building 105-N), the heat exchanger building (Building 109-N), and the reactor core pedestal

are supported on undisturbed soil consisting mainly of compacted gravel. The soil profile consists of an upper layer of about 50 ft of medium-dense gravel, resting on a 50-ft layer of very dense gravel. The gravel layers are underlaid by approximately 400 ft of blue clay mixed with variable amounts of silt and fine sand. The clay layers are highly compacted and possess the physical properties of soft rock.

The soil profile used for this study is shown in Figure 3.7A-2. This soil profile consists of a 250-ft deep column with well defined layer properties. Below a depth of 250 ft, the soil was considered to extend to infinity with uniform, rock-like properties.

The low-strain soil properties corresponding to this soil profile are shown in Table 3.7-2. A one-dimensional wave-propagation analysis was performed with this model using the program SHAKE in order to obtain strain-iterated soil properties.

The reduction in soil shear moduli (G) with increased soil strain was considered in the free-field analysis along with an increase in soil damping. The degradation curves for various types of soil used for this analysis can be found in Reference 25.

3.7.3.15.1 Independent Review of Impell's Evaluation

Lawrence Livermore National Laboratory (LLNL) conducted an independent review, assessing the validity of Impell's evaluation (Reference 31). The LLNL analysis was accomplished in two major stages: linear soil-structure interaction (SSI) analysis of the overall N-Reactor structure complex and nonlinear dynamic analysis of the reactor core.

In the SSI analysis, CLASSI computer codes were used by LLNL to calculate the SSI response of the structures and to generate the input motions for the nonlinear reactor core analysis. The results were compared with results from Impell evaluation in which SASSI computer code was used. The comparison was made on in-structure response spectra obtained at several locations in the core, the primary shield, and the inlet/outlet shield. Moreover, the

impact of foundation modeling techniques and the effect of soil stiffness variation on SSI response were also investigated.

In the core analysis, a nonlinear dynamic analysis model was developed. The stiffness representation of the model was calculated through a finite element analysis of several local core geometries. Finite element analyses were also used to study the block to block interaction characteristics. Using this nonlinear dynamic model along with the basemat time histories generated from CLASSI and SASSI, several dynamic analyses of the core were performed. A series of sensitivity studies was performed to investigate the discretization of the core, the effect of vertical acceleration, the effect of basement rocking, and modeling assumptions.

The independent analysis and review of core response validated the order of magnitude of the displacement calculated by Impell. A number of secondary recommendations were made to refine the Impell analytical model and these were incorporated in following studies.

3.7.4 Seismic Instrumentation

The N Reactor seismic instrumentation consists of two separate and distinct systems as follows:

1. Three seismoscopes are installed at an appropriate location in the basement of the reactor building. Two or all of the seismoscopes must react in order to automatically actuate the ball safety system and horizontal control rod system in the event of ground acceleration of 0.007 g. This ground acceleration is considered a minimum value corresponding to an intensity IV (MM) earthquake.
2. Upon occurrence of ground motion (earthquake), a triaxial strong motion accelerograph provides recordings of the resultant longitudinal, vertical and transverse motions of the foundation to facilitate an ensuing safety analysis.

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3.7.4.1 Seismoscope Trip System

The seismoscope system detects an earthquake intensity IV (MM), which is well below the seismic design basis for the plant and actuates a shutdown of the reactor by tripping the ball safety system and the horizontal control rod system.

The seismoscope system consists of three independent pendulum switches and relays mounted on a common base plate and pad at EL 454-ft MSL and is located in the flux monitor room, Room 178. Each pendulum switch and relay make a complete detection circuit. The pendulum switches respond to any motion of the earth in a lateral direction and are adjusted to close contact when the horizontal component of earthquake reaches intensity IV (MM).

A momentary contact of one of the switches "locks in" the corresponding relay which, in turn, operates contacts to actuate an annunciator drop in the control room. When any two of the three seismoscope relays "lock in," the ball safety system and horizontal control rod system are also activated. Since the seismoscope relays "lock in" on tripping, they must be manually reset in order to return the system to normal.

The basic element of the seismoscope is a pendulum suspended by a short length of music wire and critically damped in an oil-filled dashpot. Sufficient amplitude of motion causes a cone shaped male contact affixed to the pendulum to touch a ring-shaped stationary contact. See Section 7.2.1.2

2

The seismoscope equipment is designed for reliability and is located in a Zone II ventilation area provided with adequate heating and ventilation.

2'

3.7.4.2 Strong Motion Accelerograph

A triaxial strong motion accelerograph* is attached to the top of the concrete foundation which supports the reactor core and primary shields. It is located at the west end of Corridor No. 3, at EL 438.5-ft MSL, in the basement of

*Model SMA-1 Strong Motion Accelerograph equipped with 0.25 g accelerometers and with both a model HS-1 horizontal starter and a model VS-1 vertical starter, manufactured by Kinemetrics, Inc.

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the reactor building. The accelerograph is housed in a light-tight, water-tight, cast-aluminum base and cover and records photo-optically on 70-mm film. The instrument remains in a standby condition until an earthquake causes a starter to actuate a camera light source and film drive motor.

The starter senses the initial earthquake ground motion and actuates the accelerograph to full operation in less than 0.1 second if the horizontal ground acceleration exceeds 0.01 g. The unit operates for as long as the starter detects the earthquake plus 18 seconds. It can record a single earthquake or a sequence of earthquakes and aftershocks lasting as long as 25 minutes, depending upon the amount of unexposed film in the magazine. After an earthquake is recorded, the film take-up magazine can be replaced by a spare and the film processed.

An event indicator shows when an earthquake has been recorded. Before an event, it is black; after an event, it is white. A key-locked switch is used for calibration checks and for resetting the event indicator.

The instrument is powered by self-contained batteries which are replaced routinely on a planned maintenance schedule.

3.7.4.3 Control Room Operator Notification

Each of the three seismoscopes connected into the reactor shutdown system when tripped will actuate an annunciator in the reactor control room. A trip of two or more seismoscopes will shut down the reactor via the BSS, and, in this event, the seismic activity will be confirmed by comparison with the response of the triaxial strong motion accelerograph.

3.7.4.4 Comparison of Measured and Predicted Responses

Before resuming operation following a shutdown associated with an earthquake, the integrity of all critical reactor systems and equipment would be determined by inspection and testing. Particular emphasis would be placed on inspection of piping and pipe supports.

The information provided by the triaxial strong motion accelerograph could be used to determine the maximum acceleration of the building foundation and, from the traces, acceleration spectral curves could be developed and compared with those used for the plant design and subsequent evaluations.

The Hanford seismoscope network would provide information on the magnitude of seismic events in the N Reactor area, and this information could be compared with the information from the triaxial strong motion accelerograph to obtain information on building response to seismic events.

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TABLE 3.7-4

N REACTOR SEISMIC CRITERIA (1960-1962)

Importance of Item	Seismic Category	Horizontal Force Factor = C	
		Rigid Items*	Flexible Items
General	Non-Seismic	UBC Zone 2 (See below)	UBC Zone 2 (See below)
Important for Plant Operation	Non-Seismic	0.10	0.20
Critical with respect to Nuclear Safety	I	UBC Zone 3 (or) 0.20	UBC Zone 3 (or) 0.50

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UNIFORM BUILDING CODE (UBC)

The 1955 edition of the Uniform Building Code specifies the following values of C for seismic Zone 2:

Part or Portion	Value of C	Direction of Force
Floors, roofs, columns and bracing in any story of a Building or the structure as a whole**	$\frac{0.30}{N + 4\ 1/2}$	Any direction horizontally
Bearing walls, non-bearing walls, partitions, free standing masonry walls over 6 ft in height	0.10	Normal to surface of wall
Cantilever parapet and other cantilever walls, except retaining walls	0.50	Normal to surface of wall
Exterior and interior ornamentations and appendages	0.50	Any direction horizontally
When connected to or a part of a building: towers, tanks, towers and tanks plus contents, chimneys, smokestacks and penthouses.	0.10	Any direction horizontally
Elevated water tanks and other tower supported by a building	0.06	Any direction horizontally

*A rigid item is not one that responds essentially as the ground, including structures having a natural period of less than about 0.1 second.

**N is number of stories above the story under consideration, provided that the designing floors or horizontal bracing, N is only number of stories contributing loads.

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Table 3.7-6

Minimum Safe Working Loads - All Existing Concrete Anchors

Bolt diameter (in)	Tension (lbs)	Shear (lbs)
1/4	150	400
3/8	250	800
1/2	650	1350
5/8	750	1900
3/4	1200	2700
1	1900	4500
1-1/4	2700	6000

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Notes:

1. Allowables in this table are based on test data for substandard installations, and will yield a minimum safety factor of 2 on ultimate capacity.

2. These allowables are valid only for anchors verified in good condition by visual inspection, e.g., no excessive spalling, not loose, and base plates not lifted off.

3. These are minimum safe working loads for non-Category 1 installations. Higher working loads may be allowed based on an individual proof test.

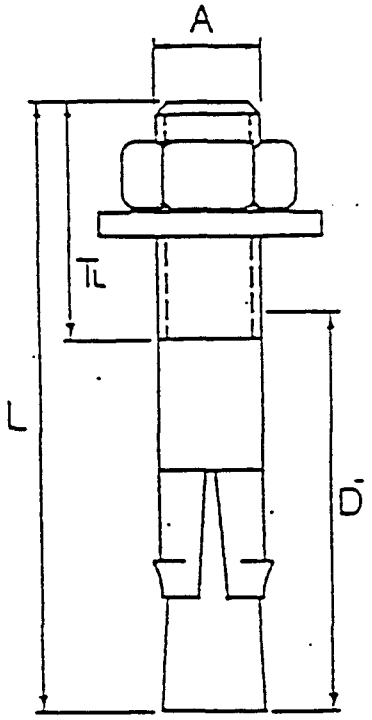


Table 3.7-7

Safe Working Loads - New Designs
Hilti Kwik Bolts

INSTALLATION: Drill hole the same diameter as the bolt. Drive bolt into the hole so that at least six threads are below the top surface of the fixture. Tighten nut. Resistance will increase after third or fourth turn.

NOTE: Loads listed are based on a safety factor of four when installed in accordance with manufacturer's recommendations.

It is permissible, under the following conditions, to use a safety factor of 2. Thus, design safe working loads may be taken as twice the table values for a support installation where the two adjacent support installations have safety factors of 4 or greater.

A (Bolt Diam.)	L	TL	3500.psi Concrete Safe Working Load		
			D(Embd.)	Tension	Shear
1/4	1 5/8	3/4	1 1/8	330	590
	2 1/4	3/4	1 3/4	645	590
	3	3/4	2	740	590
3/8	2 1/8	7/8	1 5/8	580	1190
	2 3/4	1 1/8	2	735	1190
	3 1/2	1 1/8	3	1010	1250
	5	1 1/8	4	1110	1250
1/2	2 3/4	1 1/4	2 1/4	1315	2020
	3 3/4	1 1/4	3 1/2	2205	2020
	5 1/2	1 1/4	4 1/2	2555	2470
	7	1 1/4	5 1/2	2775	2470
5/8	3 1/2	1 1/2	2 3/4	1575	2865
	4 1/2	1 1/2	3 1/2	2095	2865
	6	1 1/2	4 1/2	2685	2865
	8 1/2	1 1/2	6 1/2	3500	3730
3/4	4 1/4	1 1/2	3 1/4	2410	4040
	5 1/2	1 1/2	4	3115	4040
	7	1 1/2	6	4235	4410
	8 1/2	1 1/2	7	4925	4410
	10	1 1/2	8	5310	4410
1	6	2 1/4	4 1/2	3875	6750
	9	2 1/4	7	5530	6750
	12	2 1/4	9	5530	8175
1 1/4	9	3 1/4	5 1/2	5500	8985
	12	3 1/4	8 1/2	8055	9180

Table 3.7-8

Component Stress Criteria(4)

Component Type	Stress Condition	Criteria
Pipe	Tension, Bending*	Lessor of 1.875 Sy (1) 0.75 Su
Electrical Conduit	Tension, Bending	33 KSI (2) (3)
HVAC Duct	Bending Moments-Buckling	See Tables 3.3.3 & 3.3.4
Components Supports	All-Use Table 3.4.1 Criteria	
Cast Iron or other non-ductile materials not including concrete	All	Lesser of 0.9 Sy 0.7 Su

Note 1
a) Su = Ultimate strength at operating temperature
Sy = Yield strength at operating temperature
Recommended Su and Sy values are provided in Table 3.4.2

b) For straight pipe without elbows or fittings. Failure in elbows and fittings is acceptable if supports are provided on both sides and contents are not a problem. Otherwise ASME Section III Class 3, 1980 Stress Intensification Factor must be applied.

c) Do not apply net section reductions to threaded joints for Schedule 80 pipe less than or equal to 2-1/2 inches diameter. For larger diameters treat on a case bases using net section for stress calculations.

d) Based upon ASME Section III, Class 3 allowables for level D loads. Where stress allowable - 3 Sh and Sh is the lessor of 1/4 Su or 5/8 Sy.

Note 2
Based on Su = 45 KSI and Sy - 30 KSI. Section properties of steel electrical conduit are listed in Table 3.3.2 for reference. Weights per foot assume the conduit is 40% full of cable.

Note 3
Use 30% of allowable stress for threaded couplings and 50% of allowable stress for elbows.

Note 4
Criteria is to be used for evaluating both existing and new designs for non-Category 1 components.

*For bending, a bending modulus, k, such that $\sigma = \sigma_{b/k}$, is permitted. K is defined by Table A-9221 (a)-1 of ASME Sec III, Appendix A-9000. For example, K = 1.27 to 1.7 for thin and thick pipe respectively.

Table 3.7-9

Component Support Stress Criteria

Component Type	Stress Condition	Criteria
Structural Steel	Tension, Bending*	1.2 Sy (2) Lessor of 0.7 Su
	Shear	2/3 (Tension, values)
	Compressive Load	2/3 buckling load
	Welds	Base material shear
Catalog Items	All	Manufacturer's Load Capacity for Level D or F.S. = 2
Concrete Anchor Attachments	All	See Section 3.2
Embedded Anchors (3)	All	F.S. = 2 if concrete controls. Structural steel criteria if steel controls.

Notes:

(1) Su = Ultimate strength at design temperature
Sy = Yield strength at design temperature
F.S. = Factor of safety

Recommended Su and Sy values are provided in Table 3.7-10

(2) Based on ASME Section III Component Supports Level D loads. Where stress allowable equals lessor of 0.7 Su or 1.2 Sy.

(3) Use minimum concrete allowable compressive strength of 3500 PSI. Higher strength may be used when justified by test results.

*For bending, a bending modulus, K, such that $\sigma = \sigma_{b/K}$, is permitted.
K = 1.5 on rectangular sections
= 1.1 on I-beam sections
= 1.3 on thin tube, pipe sections
K is defined by ASME III, Appendix A-9000

Table 3.7-10

Recommended Material Properties

For New Designs utilizing A36 materials

Su = 58 KSI Room Temperature
Sy = 36 KSI

For Existing Designs with ductile materials (A7)

Su = 55 KSI Room Temperature
Sy = 33 KSI

NO

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3.8 Design of Seismic Category I Structures

Seismic Category I structures are defined and listed in Section 3.2.1. Design information, structural acceptance criteria and physical descriptions for these Category I structures are presented below, except for the irradiated fuel storage facility which is covered in Section 9.1.2. Although not classified as Category I, the 105-N control room is also described below. See Figure 3.5-2 for an overall layout of facilities containing safety-related shutdown systems.

3.8.1 Concrete Confinement Structure

This section provides the following information on the confinement structure:

1. The physical description
2. The applicable design codes, standards and specifications
3. The loading criteria, including loads and load combinations
4. The design and analysis procedures.

Structural acceptance criteria, materials, quality control programs and construction techniques are described in the specifications referenced herein. Testing and inservice inspection programs for the confinement structure are described in Chapter 6.

The confinement structure consists of the central portion of the 105-N building, which houses the nuclear reactor, and that part of the 109-N building which contains components of the reactor coolant system (RCS). The walls and slabs of this massive concrete structure provide nuclear shielding, a 5 psig confinement barrier between the reactor and its coolant system and surrounding building areas and protection against external missiles. General features of the 105-N reactor building are shown on Figures 3.8-1 to 3.8-9. The various building areas and rooms are identified by numbers described in Table 3.8-1. The 5-psig confinement boundary in each building is shown with a heavy, dark line located on the pressure side of the various walls and slabs of the confinement structure. Corresponding

details for the 109-N heat exchanger building are shown on Figures 3.8-10 and 3.8-11.

3.8.1.1 Physical Description of the Confinement Structure

The confinement structure is constructed of reinforced concrete, except for eight steel roof trusses and two steel box girders located near the top of the 105-N building. The principal structural walls are from 3 ft 0 in. to 5 ft 9 in. thick and provide nuclear shielding as well as confinement capability. The vertical walls and roof slabs which surround the reactor, the pipe gallery and heat-exchangers are called "secondary" shielding walls to distinguish them from the "primary" biological shields which enclose the reactor core.

The 105-N confinement structure is approximately 213 ft long (east-west direction), 142 ft wide (north-south direction) and 82 ft high. The reinforced concrete foundations and intervening grade slabs are located approximately 22 ft below ground level so that the top of the 105-N confinement structure is about 60 ft above ground level. The roof of the 105-N building, at this location, is 10 ft above the top slab over the confinement structure. Thus the 105-N portion of the confinement structure is enclosed on three sides and the top by other parts of the 105-N building. It is contiguous with the 109-N confinement structure on the south.

The central portion of the north wall of the 105-N confinement structure is similar in construction to the inlet face of the primary reactor shield, except that it is only 3 ft thick. Permanent steel forms, provided during construction, locate penetration sleeves and contain high-density concrete. These sleeves, or access holes, correspond to the pressure tube lattice spacing of the reactor. See Section 3.8.3.2. Shielding is provided by high-density concrete plugs, which are removable from the outside. Pressure integrity is provided with expandable neoprene plugs, which must be removed from the inside in order to refuel the reactor. Access holes are also provided in the south barrier wall for servicing the outlet face of the reactor for non-routine, shielded access to the reactor, but not on a one-to-one basis with reactor pressure tubes, as provided through the inlet

barrier wall. Each outlet face, access hole is provided with a shielding plug and an expandable plug similar to those described above for the front face.

The reinforced-concrete roof slab, which forms the upper pressure boundary of the 105-N confinement structure, is supported by a structural steel framework consisting of metal decking, floor beams, trusses and girders. This roof slab varies from 1.5 to 3 ft thick, depending upon location and shielding requirements. The metal decking, concrete roof slab and encased roof beams, the latter spaced about 5.5 ft apart, are supported by the bottom flanges of the roof trusses. There are six, three-span roof trusses (spaced 11 ft apart) and two, single-span trusses. Each truss is 134 ft long and 9 ft high, except that the single-span trusses are only 45 ft long. The bottom of the trusses and the roof slab are located 59 ft 4 in. above ground level, which is at El. 454.5 ft above MSL. The conventional roof, at the 70-ft level of the 105-N building, is supported on the top flanges of the roof trusses.

The outer ends of the roof trusses rest on the east and west walls of the confinement structure; whereas, the intermediate supports rest upon two steel box-girders, spaced 44 ft 4 in. apart. The ends of the box-girders bear upon the inlet and outlet barrier walls. Two intermediate supports are also provided for the box-girders so that the centerline distance between girder supports is 28 ft 0 in., 38 ft 9 in. and 32 ft 6 in., respectively, measured north to south. The ends of the girders rest on bearing plates and are bolted to the top of the concrete walls; whereas, sliding-rocker units are provided at the intermediate supports. The rocker units are also bolted to their supports to prevent uplift during pressurization of the confinement structure. In addition to resisting vertical loads, the box-girders are designed to resist horizontal reactions due to pressure and earthquake. To accommodate these horizontal reactions, the ends of the two girders are tied structurally to the top of the confiner walls by a series of high-strength bolts. These penetrate the walls horizontally and are anchored to weldments which bear upon the outer wall surfaces. 21

Also included in the 105-N building, but outside the confinement structure, are the following major rooms or equipment areas: the reactor control room; shielding rooms for housing the reactor control rods; the ball safety system auxiliary equipment and gas driers; a water storage basin for spent fuel; auxiliary equipment rooms for instrumentation, electrical switchgear, motors, fans, compressors, pumps, valving and piping and the maintenance shops and offices. Personnel access to the 105-N confinement structure from adjacent building areas is through shielding doors located at each corner of the confinement structure and at each floor level. Inflatable seals, good for 5 psi, are provided on these doors.

The 105-N building is divided into five ventilation, or confinement, zones. The central space, within the 5 psi confinement structure, is categorized as Zone I. Rooms adjacent to the reactor block not in Zone I, such as the left and right-rod rooms, are subjected to radiation sources but do not house components of the RCS. These areas are in confinement Zone II. Areas immediately beyond Zone I and II are in Zone III and include corridors adjacent to shielding doors leading to Zones I and II. The space between the confiner roof at the 60-ft level and the conventional roof at the 70-ft level is in Zone III, except for the two machinery rooms and the Zone I exhaust chamber, which are in confinement Zone I. Additional information on the various confinement zones is presented in Table 3.8-2 and 3.8-3. In general, portions of the 105-N building outside of Zones I and II are of conventional steel construction with concrete floors, corrugated metal siding and insulated roof decking.

The 105-N reactor building is contiguous with the 109-N heat exchanger building, with a common wall between the two buildings. This common wall is of reinforced concrete, 5 ft 0 in. thick and about 390 ft long. It extends vertically 76 ft above its foundation, which is 22 ft below ground level. The two buildings are tied together by a normal complement of dowels plus six additional clusters of #18S reinforcing bars. The latter are designed to tie the two buildings together in the event of an earthquake producing 0.25 g equivalent seismic reactions at the interface. The seismic steel clusters extend across the roof of the 109-N building and into intersecting walls and

diaphragms of the 105-N building. Structural elements are very rigid and provide a definite load path between the two structures. Dowels provide 1,725 $in.^2$ of steel reinforcement above the 38-ft level. Six clusters of seismic reinforcement provide an additional 848 in^2.

If the dowels take all of the horizontal reaction at the roof line of the 109-N building, due to the 5 psi pressure load, and the six clusters take all of the north-south seismic load, then the dowels see an average tensile stress of 3,750 psi due to confinement pressure, and the clusters see about 10,900 psi due to the 0.25 g earthquake reaction. On the other hand, if it is assumed that the combined pressure and seismic reaction at the roof line is taken evenly by the dowel and cluster steel, the average stress would be around 4,175 psi. These stresses are well below the maximum allowable stress (20,000 psi). In addition to the steel reinforcement described above, keys are provided in the mating concrete surfaces to resist shearing forces, should the postulated earthquake occur in the east-west direction.

To assure leak tightness, elastomeric waterstop, 9 in. wide, is provided along mating surfaces between the 105-N and 109-N confinement structures. The 105-N confinement structure is separated on all sides from the rest of the 105-N building, by a continuous expansion joint, 2 to 3 in. wide. This isolation joint extends vertically for the full height of the 105-N building and begins at the top of the foundations. Ventilation ducts, located just below the roof of the 109-N building, penetrate the common wall in order to allow pressure equalization inside Zone I during a postulated reactor incident.

The 109-N confinement structure houses most of the RCS piping, the pressurizer, steam generators and RCS pumps. The pressurizer is located in a penthouse, just south of the 105-N/109-N common wall. The steam generators and RCS pumps are installed in six isolated rooms, or cells, and are serviced from a large pipe gallery located on the north side of the cells. The pipe gallery, the steam-generator cells and a seventh cell, containing auxiliary equipment, are surrounded by thick walls and slabs of reinforced concrete.

Similar to the 105-N building, this concrete structure provides nuclear shielding, a Zone I confinement barrier and missile protection.

The 109-N confinement structure is approximately 390 ft long, 56 ft high and 145 ft wide. Shielding walls are 5 ft 0 in. thick, except for the upper section of the south wall which is 5 ft 9 in. thick. The reinforced concrete roof is located 40 ft above ground level. It is 3 ft 6 in. thick, except at the pressurizer cell where it is 2 ft 8 in. thick. The pressurizer cell, or penthouse, is 35 ft long, 26 ft wide and 43 ft 3 in. high, inside dimensions. It spans across the pipe gallery in a north-south direction, along the extended-centerline of the reactor unit. The pressurizer cell protrudes above the roof of the pipe gallery a distance of about 48 ft. Three vent structures also extend above the roof 12.5 ft and span across the pipe gallery in a north-south direction. These vent structures have side walls 1-ft thick and are integral with the gallery roof.

The pipe gallery, which is 35 ft wide inside, runs the full length of the 109-N confinement structure, and is located between the steam-generator cells and the 105-N building. Each of the six steam-generator cells and the auxiliary cell is 50 ft wide and 94 ft 3 in. long inside and is separated from adjacent cells and from the pipe gallery by shielding walls 5 ft 0 in. thick. The north shielding wall of each steam-generator cell contains three large vent openings near the roof and three sump openings at the bottom for passing RCS piping into each cell from the gallery. Any steam-generator cell can be isolated from the pipe gallery by filling the respective sumps with water and by rolling shielding doors in front of the upper vent holes.

A shielded, access corridor runs along the basement side of the cell structure on the south. A shielding door is provided at each of the passageways to the seven cells from the access corridor. An air-lock is located at each end of the shielded corridor for personnel access. In addition, two large shielding doors are located at the east end on the pipe gallery, one at the basement level and the other at ground level. These two doors, as well as the ones on the two air-lock chambers, are provided with inflatable seals good for 5 psig. The access corridor doors to the seven cells provide

radiation shielding; however, they do not have inflatable seals and are not air-tight.

The penthouse (which houses the pressurizer), the auxiliary cell and the pipe gallery are designed for an internal pressure of 5 psig; whereas, the six steam-generator cells are designed for an internal pressure of 12 psig. Additional information on the confinement zones for the 109-N building is presented in Table 3.8-3. The south portion of the 109-N building, beyond the confinement structure, the Turbine Bay, is of conventional steel-frame construction with corrugated metal siding and an insulated concrete roof-deck. Primary and backup drive units for the six RCS pumps are located in this area, as well as the dump-condensers, various pumps, piping and auxiliary plant equipment.

It is noted that the secondary shielding walls, which form the vertical sides of the confinement structure, are supported on a series of continuous spread footings. The bottom of these reinforced concrete footings is located between 22 and 29.75 ft below normal grade level and rest upon undisturbed soil, consisting of compacted gravel alluvium. See Section 3.7.1.5. Footings are constructed 12- to 19-ft wide in order to equalize soil pressures beneath the confinement structure. Soil pressures are generally less than 8,000 psf, including earthquake effects. Most of the major footings are 30 in. deep, although some of the more massive walls have footings 42 in. deep. The steam generators in the 109-N building and the reactor located in the 105-N building are supported on isolated matt foundations. Intervening basement floor slabs are supported on compacted backfill. Components of the foundation structure are tied together to resist forces due to pressure, earthquake and thermal loads. Likewise, steel and concrete components of the roof structure are also tied together structurally to resist the above forces.

Initially, the 109-N building contained only five steam-generator cells. However, structural provisions were made during construction of the west wall of Cell No. 1 to add one or two additional cells at a future date. The sixth steam-generator cell and pipe-gallery extension were constructed

in 1965-1966. These items are integral parts of the confinement structure. It is unlikely that a seventh cell will ever be added to 100-N plant, although provisions were made in the construction of the west wall of Cell No. 6 to add another steam-generator cell.

Design criteria are referenced in Section 3.8.1.2, below, for the two buildings. Additional information on the ventilation and reactor confinement systems are presented in Section 6.2.

3.8.1.2 Structural Design Criteria

The design of the 105-N and 109-N buildings is based upon the following criteria, codes and specifications. The numbers used to identify the various items correspond to references listed under Section 3.8.

1. Holmes & Narver, Inc., "Earthquake Resistance of the Hanford New Production Reactor," HN-148, June 1960.
2. "Hanford Works Architectural-Civil Standards," HWS-1000, 1960.
3. "Basic Reactor Building Requirements, 105-N Design Criteria," HW-57031, June 1959.
4. "Technical Criteria and Bases, Section 0.2.2-Earthquake Resistance," May 1961.
5. "Uniform Building Code," 1955 and 1958 Editions, Pacific Coast Building Officials Conference.
6. "Specification for the Design, Fabrication & Erection of Structural Steel for Buildings," Fifth Edition, American Institute of Steel Construction (AISC).
7. "Building Code Requirements for Reinforced Concrete," 1956 Edition, American Concrete Institute (ACI).
8. "Manual of Steel Construction," Fifth Edition, American Institute of Steel Construction (AISC).
9. "Wind Forces on Structures," ASCE Paper No. 3269, American Society of Civil Engineers (ASCE) 1940.

10. Davis, H. S., "Thermal Considerations in the Design of Concrete Shields," ASCE Paper No. 1755 ST 5, American Society of Civil Engineers (ASCE), Sept. 1958.
11. "National Fire Codes," National Fire Protection Association International, Boston, Mass.

3.8.1.3 Loads and Load Combinations

The following loads were considered in the design of the confinement structure: dead, live (floor, snow, crane reactions, etc.), thermal, internal pressure, partial vacuum, hydraulic forces due to basement flooding, earthquake and wind. However, all of the above loads were not applied simultaneously. Moments, shears and stresses were computed separately for each load or for simple combinations of similar loads. The results were then combined in realistic combinations to obtain maximum effects on structural components, such as walls and floor slabs, and on integral building frames. Load combinations covered normal and abnormal conditions of plant operation, including accident conditions (Table 3.8-4). In order to minimize crack size in the concrete walls and slabs, which make up the pressure boundary of the confinement structure, maximum stresses were kept below normal design limits. (Although the Uniform Building Code permits a one-third increase in the allowable stresses when considering lateral forces in combination with other loads, this permissible stress increase was not utilized for the reason stated above.) Design was based primarily on requirements of the UBC and the Building Code for Reinforced Concrete of the American Concrete Institute (ACI). Detail design was executed during 1959 and 1960, which predates the current design code for this type of nuclear structure by about 15 years.

1. Dead Load--Dead Load consists of the weight of the reinforced concrete used in construction plus the weight of permanent equipment such as the reactor block, steam generators, piping etc. By definition, the reactor block includes the graphite core, thermal shield, primary biological shield and related reactor equipment, piping and foundation. The bulk of the confinement structure is made with normal structural concrete weighing about 150 lb/ft^3. However, various types of high-density

concrete were used in the construction of the reactor shields, the inlet barrier wall, small sections of the 105-N confiner roof located above the discharge pit, around door frames and in shielding doors. Magnetite concrete, weighing about 230 lb/ft^3, was used in some of the doors: whereas, ferrophosphorous concrete weighing around 300 lb/ft^3 was used in others. Details on the various types of high-density concrete used in the construction of the confinement structure are presented in Section 4.5.5 (see also Ref. 13).

2. Live Load--Concrete floor areas inside Zone I and corridors leading to Zone I of the 105-N building are designed for a live load of 200 lb/ft^2, plus a moving concentrated load of 8000 lb. Gratings and restricted roof areas are designed for a live load of 125 lb/ft^2. The roof slab above the discharge pit contains several access holes filled with removable step plugs and is designed for a live load of 300 lb/ft^2. Live loads for the 109-N building include 300 lb/ft^2 for grade-supported slabs (located in the pipe gallery, cells and access corridors), 100 lb/ft^2 (hung load) on roof areas over the pipe gallery and the steam generator cells, plus a snow load of 20 lb/ft^2 during the winter months.

Traveling bridge cranes are provided in the following locations:

109-N Building	Capacity (tons)
Pipe gallery	25/5
Turbine drive bay	35/5
Steam generator cells	5
Auxiliary cell	10
South roof area (traveling gantry)	3.75
105-N Building	
Above reactor block	2 (See below)
Metal transfer area	60 (two 30 in. ton hooks)

The bridge unit over the reactor block in the 105-N building is a 12-ton unit used during reactor construction; however, it supports only a

2 ton motorized trolley at the present time. In addition, there are 13 monorail systems in the 105-N building, some inside the confinement structure, rated from 1/2 to 10 tons in capacity. There is also a 10-ton monorail located above the dump condensers in the 109-N building.

3. Thermal Loads--The original design criteria required that walls of the confinement structure and tunnels be cooled, or otherwise protected, so that the following temperatures are not exceeded during periods of normal reactor operation:

		Buildings	Tunnels
Maximum concrete temperature	°F	130	160
Maximum differential temperature across wall thickness	°F	50	72

After the reactor was started up, it proved impractical to keep maximum air temperatures at upper elevations in the 105-N pipe spaces below 130°F. Revised thermal calculations demonstrated that 160°F would be acceptable as the upper temperature limit with a maximum temperature differential of 50°F. In general, air temperatures within the 105-N confinement structure are less than 70°F at basement levels; whereas, they may approach 130°F at upper elevations in the inlet pipe space and 160°F near the roof of the outlet pipe space when the reactor is operating.

Structural walls and roof slabs of the 105-N confinement structure are designed for the 50°F temperature differential. Allowance was made for the variation in air temperatures which may exist vertically in the 105-N of the confinement structure when the reactor is operating. The temperature differential through the secondary shielding walls (which form the pressure boundary of the confinement structure) was assumed zero at the minus 16-ft level near the bottom of the walls, and to increase linearly to 50°F delta T at an elevation of plus 40 ft (Ref. 3).

Maximum air temperatures within the 109-N confinement structure are kept below 130°F during reactor operation. Structural members within

the 109-N confinement area are designed for the following maximum temperature differentials:

Zone I - Outside	Walls:	42°F	Roofs:	40°F
Zone I - Zone I	Walls:	50°F		
Zone I - Zone II or III	Walls:	40°F		

In the first case, exterior walls and roof slabs are insulated on the side exposed to the weather. Thus, the total temperature differential provided for in the design is greater than 50°F.

4. Pressure--During a postulated reactor incident, the pressure inside Zone I of the confinement structure may build up to a maximum in a short period of time and then decrease slowly. See Section 6.2. However, the duration of the transient pressure-loading is long enough that resulting stresses can be calculated on a static basis. To accommodate this pressure cycle, the exterior walls and roof slabs are designed for the peak internal pressures presented below:

105-N Building		
Inlet and outlet pipe spaces	+ 5.0 psig	- 2.0 psig
109-N Building		
Pipe gallery and auxiliary cell	+ 5.0 psig	- 2.0 psig
The six steam generator cells	+12.0 psig	- 2.0 psig

In the latter case, the pipe break is assumed to occur in an isolated steam-generator cell. The 12 psig pressure used in the design of the six steam-generator cells allows for a pressure differential of up to 7.0 psi for venting a cell to the pipe gallery, whereas, the pipe gallery is designed for a peak pressure of 5.0 psig.

Lower sections of walls of the 105-N confiner, bordering Zones II and III, may also be subjected to the following external pressures:

Zone II, except the spacer decontamination room	+ 0.5 psig	- 0.1 psig
Zone III, and the spacer decontamination room	+ 0.2 psig	- 0.1 psig

5. Hydraulic Forces--The floor and lower walls of the confinement structure may be subjected to internal hydraulic pressures in case the basement areas of Zone I are flooded with water during a reactor incident. For design purposes, stresses in the lower walls were computed, assuming that the water level in Zone I was at elevation 454.5 ft MSL, and were within normal design limits.

6. Earthquake--As described in Section 3.7, the seismic adequacy of the 105-N/109-N buildings and the parts and components of these facilities were reviewed at various times by Holmes & Narver Inc., H. S. Davis, R. L. Murray, G. Housher, Newmark and Hall, DOE, etc. In 1987 the Westinghouse Hanford Independent Safety Assessment Task 2-D group (Ref. 52) reviewed the seismic analysis of the 105-N/109-N complex performed by EQE Inc. in 1986-87 (Ref. 53).

   The EQE analysis used prevalent industry practices involving SSI analysis, 3-D dynamic analysis and utilized the Hanford DBE in SDC 4.1 (Ref. 50). The analysis included the representation of both buildings and implemented a model of 105-N/109-N structure and the reactor core. The free-field acceleration time histories used for this analysis met the USNRC criteria for frequency content defined in SRP 3.7.1 and assured the statistical independence between components as set forth in the Regulatory Guide 1.92. Structural damping used in the analysis was 5%, whereas the Regulatory Guide 1.61 recommends 7% for reinforced concrete structures in SSE analysis. The lower damping, as well as the free-field motion input at embedded foundation level, resulted in conservative floor response spectra. These response spectra were smoothed and

broadened to account for uncertainties, as discussed in 3.7. It was noted that although the analysis had been performed specifically to produce equipment response spectra, a modal analysis had been performed which provided insight into building response.

The modal analysis performed by EQE (Ref. 53) produced fundamental eigenvalues as follows:

	105-N	109-N
E. W.	12.7 CPS	10.2 CPS
N. S.	9.9 CPS	12.0 CPS
Vert.	28.4 CPS	--

Horizontal and vertical response spectra were produced by EQE at all levels of the two buildings. The horizontal amplitude at the foundation level at these frequencies is roughly .4 g and the zero period amplitude at the highest elevations are also roughly .4 g. Vertical zero period amplitudes are in all cases less than .3 g. Using these values, the relative values of design loads given in Table 3.8-5 and the ratio between allowables used in the design and todays extreme load "allowable" simplistic comparisons were made for vertical bending loads by the W Task group (Ref. 52). Based on calculations, it was determined that the increased vertical load ratio due to increased seismic forces is 1.14 whereas the increased moment capacity due to increased "allowables" is 1.86 providing a positive safety margin.

The ratio of present day shear allowables to 1960 shear allowables is 1.55 whereas the ratio of present day loads to 1960 loads is approximately 2. However, the shear walls are massive for shielding purposes and are not heavily loaded. From the EQE report it can be determined on a very conservative basis that the average lower story shears in building 105 are less than 60 psi which is well below present day allowables. It was concluded the risk of further operations is considered acceptably low, because of the conservatisms and comparisons discussed above. The risk is further reduced if it is considered that

N Reactor will operate only into the mid 90's further reducing overall exposure to these site-limiting events. Therefore, N Reactor can be safely operated with an acceptably low risk of major damage to the 105-N/109-N structures which might result from extreme natural phenomena such as earthquakes and tornados. It was suggested, however, that to dispel any doubt, a modern state-of-the-art dynamic analysis should be performed.

In 1988, to substantiate the above task group conclusion, which was based on generic and screening type calculations, additional analyses (Ref. 54) were performed. The scope of the study covered three general areas: 1) a generic roof slab load and moment comparison, 2) an analysis of all major shear walls in the buildings, and 3) an analysis of overturning for each building. The area included is inside the confinement boundary which is separated from the remainder of the building by Seismic Joints. The boundary of the analysis is depicted in Figure 3.8-33. A detailed analysis of all building elements for earthquake forces in combination with other loads was beyond the scope of the task and only earthquake load combinations were examined. Normal load combinations with appropriate multipliers in accordance with current ACI Code 349 requirements were not examined. The detailed information necessary to perform these analyses is best obtained from a finite element building model analysis which was not available. The EQE analysis utilized a stick model and assumed all floor and roof slabs to be rigid. 21

Working stress methods per ACI Codes in effect at the time of design were judged acceptable for normal loading conditions. The results of the analysis (Ref. 54) are summarized below.

Roof Slabs--Generic roof slab comparisons were made for three different thicknesses (18 in., 24 in., and 36 in.) and four different load cases found in NUSAR Table 3.8-5. The analysis is conservative in that it assumes earthquake loads correspond to the peak of the vertical 5% damped spectra. In reality the fundamental frequency could be either lower or higher than the peak depending on spans and boundary conditions and

damping would be at least 7%. The results show that current earthquake loads could be as much as 1.5 times higher than previously computed. However, section properties are at least 1.7 times higher than previously computed based on current ACI code methods, providing a positive safety margin.

Shear Walls--The shear walls were analyzed at each level using the results of the EQE analysis (Ref. 53). The analysis is very conservative because it assumes the maximum floor acceleration determined by EQE all occur at the same time. The results show that the maximum shear stress occurring in any wall of either building is 86 psi. The allowable shear in 3000 psi concrete is 109.5 psi which allows a positive safety margin. The 109.5 psi allowable is also very conservative for two reasons:

1) The concrete strength is undoubtedly well above 3,000 psi at this point in time (1988)
2) It does not include shear which would be carried by the reinforcing.

Overturning--Overturning was evaluated for both buildings. Major areas of conservatism in the analysis are:

- Maximum acceleration at each level were assumed to occur at the same time
- Embedment depth was not counted on to resist overturning
- The two buildings were assumed to act independently.

The results show that the minimum factor of safety against overturning is 3.93.

Steel Members--Steel members such as the steel roof trusses in building 105-N were not explicitly analyzed. However, an analogy can be drawn with the concrete roof slab analysis. The peak of the vertical response spectrum is 0.6 g whereas the previous vertical factor used was 0.1 g.

The maximum ratio between the two total loads is defined as follows: 1.6/1.1 = 1.45. The ratio between the two "allowables" is .09Sy/0.6Sy = 1.5 providing a positive safety margin. Additional information regarding conservatism is contained in DUN-5143 (Ref. 55).

Conclusion--In all cases evaluated there is a positive margin of safety against overstress or failure due to earthquake load combinations. Because of the low shear stresses and high margins of safety against overturning found in the analysis, there is a high confidence level that the areas not included, if analyzed would also be found acceptable. Therefore, the 100-N/109-N structural complex is judged acceptable for Return to Service. 21

7. Wind--The 105-N confinement structure is enclosed by exterior portions of the 109-N building, and, therefore, it is not exposed directly to wind forces. However, horizontal wind reactions from exterior structural members may be transferred to the massive concrete structure of the confiner. These reactions are small when compared with earthquake forces and consequently were not considered separately in the design. Also, design criteria specified that maximum wind conditions would not occur simultaneously with the design earthquake. The west wall of the 109-N confinement structure is exposed to the wind. However, in this case also, earthquake effects are more severe than wind. Tornado effects were not included in the original design of the 100-N plant. Updated information and analysis on the behavior of the plant during the postulated tornado is presented in Section 3.3. 21

8. Load Combinations--Summarized in Table 3.8-4 are various load combinations used in the structural design of the 105-N/109-N confiner. As noted above in the discussion of earthquake loads, these loads have also been used as input data in the 1987-88 seismic and tornado reanalysis of all Category I structures, systems and components. A particular live load may or may not be assumed acting, depending on whether or not its presence increases the stress in the structural member being considered. In the design of the 109-N confinement 21

structure, separate studies were made using the various load combinations assuming: (1) all of the cells are in operation (hot) and (2) all cells are hot except one. The isolated cell may be an exterior cell or a typical interior cell. The postulated pressure excursion occurs in only one of the operating cells, which may be an exterior cell or one of the interior cells. The most severe combination of loads for the various cases studied was used to determine the amount of reinforcement required for each wall and slab of the confinement structure. See Figure 3.8-12.

3.8.1.4 Design and Analysis Procedures

The 105-N building was designed by the General Electric Company; whereas, the 109-N building was designed by Burns & Roe, Inc. Similar design criteria were used by each design group. See Section 3.8.1.2. The common shielding wall and the structural ties between the two buildings were designed jointly. The design of both buildings was based upon classical theory, empirical equations and numerical methods of structural analysis used by the profession in the early 1960's. Concrete structures were designed in accordance with the working stress design method (WSD) presented in ACI 318-56 (Ref. 7). Structural steel components were designed in accordance with the 1946 and 1961 editions of the AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings (Ref. 8). These design procedures are based upon an assumed linear relationship between stress and strain and the conventionally-accepted, straight-line theory of flexure. To maintain these elastic design conditions, stresses were kept well below the material's yield strength during the loading combinations presented in Table 3.8-4. Plastic design methods were not used for structural steel members, nor was ultimate-strength design used for concrete structures.

Both buildings are massive, box-like structures constructed primarily of reinforced concrete. Neither structure is pre-stressed. In general, shielding requirements dictated wall and roof thicknesses. Sufficient reinforcement was then provided to take care of the structural loads described in Table 3.8-4, above. Stresses caused by thermal loads, associated with the 50°F temperature differential, were added directly to

those produced by other mechanical loads. For most structural components of the confiner, thermal stresses account for almost 50% of the allowable stress. The remainder provides for mechanical loads such as dead weight, pressure and earthquake. The data presented in Table 3.8-5 depicts the relative importance of the major loads when they act on the confinement structure during normal and abnormal plant operations.

During design analyses, the confinement structure was modeled by cutting it into a number of horizontal or vertical sections, 1-ft thick. This produced typical frames of reinforced concrete which could be analyzed by conventional methods of structural analysis. Fixed-end-moments were computed for each of the mechanical loads and applied to the joints of the analytical model. Actual end-moments for the frame models were then computed by moment-distribution. The ratio of the actual end-moment acting on a member to the corresponding fixed-end moment, is defined as ϕ and accounts for the relative rigidity of the member with respect to the rest of the concrete frame. Maximum moments for a particular frame were obtained by combining the effects of the various mechanical loads as specified in Table 3.8-4 and depicted in Figure 3.8-12. Maximum values of moment, based upon the most severe load cases, were then computed for both faces of the concrete members and at several locations, which included both ends of the member and at mid-span.

Moments were expressed in kip-feet so that the required amount of reinforcing steel was obtained from the expression: A_s = Moment/1.44 d, where the terms are described in Table 3.8-6. The factor 1.44 equals the allowable steel stress, 20,000 psi, times j divided by a conversion factor of 12,000 to change kip feet to lb-in. In the above expression, j was taken as 0.864 for preliminary design calculations. This value of j is conservative because the actual j for a thick wall, having reinforcing steel in each face, will be about 0.92 when the ratio of steel to concrete is around 0.005. When axial loads were present in a wall or slab, axial stresses were combined with bending effects, allowing for any eccentricity of the axial loads.

Thermal moments, assuming fixed-end conditions, were computed from the expression: $\alpha_c E_s t I/h$. Terms of this expression are also presented in

Table 3.8-6. The thermal moments were applied to the joints of the various frame models and distributed by moment distribution (Ref. 10). The uncracked concrete section was used to compute the moment of inertia (I_g) for the walls and slabs of the 105-N confiner and for the roof slab over the 109-N confinement structure. The effective concrete section, assuming that the concrete takes no tension, was used to compute the moment of inertia (I_c) for interior walls of the 109-N confinement structure, except for the common wall, which was considered part of the 105-N building. The use of the uncracked section for computing the moment of inertia is conservative as it allows for thermal moments greater than actually exist when the member is heated, by the ratio of I_g/I_c. The location of the centroidal axis of the effective concrete section is required for computing I_c. However, it cannot be determined until after the member has been designed and the amount of reinforcing steel is known. Thus, it was necessary to assume an initial value for I_c, in the latter case, and to modify the design calculations, accordingly, for the 109-N confiner walls.

Reinforcing steel was located near both faces of walls and slabs in a rectangular grid pattern. The amount of steel reinforcement provided in each direction and on each face of the member varied from around 0.04 to 0.08% of the area of the concrete cross-section. Most walls and horizontal slabs, which make up the confinement structure, were designed as two-way slabs in order to accommodate the transfer of loads to adjacent members and into the foundation. Placement of reinforcing steel around sleeves, embedments and door frames received special attention.

The walls and roof at the 60-ft level of the 105-N confinement structure were made as thin as possible in order to minimize dead weight at this level and to reduce pressure on the wall footings. The concrete roof deck is also reinforced by a rectangular grid of steel reinforcing bars located near the bottom of the slab and near the top. Individual bars are continuous across the roof, and when necessary, holes were cut through the webs of structural members to provide the desired continuity.

After the confinement structure was designed, the thermal stress (f_{st}) in the tensile reinforcement (located on the colder side of a concrete wall or roof slab) was checked by the following expression:

$$f_{st} = \phi\ \alpha_c\ t\ E_s\ (1-k)$$

The terms of this expression have been defined above and in Table 3.8-6. The factor ϕ is usually less than 1.0; however, it may be greater than 1.0 for a relatively thin member, such as a roof slab, attached integrally to a thick shielding wall. The factor k locates the neutral axis (or axis of zero strain). When axial compression or tension is present, the eccentricity of loading (e = M/P) must be considered in computing k. For a structural member, subjected primarily to bending, k will be around 0.35; whereas, when moments are small and axial compression loads are large, k may approach 1.0; or, if axial tension loads are present, k may be less than 0.35.

Thermal stresses were combined with corresponding stresses due to mechanical loads and the total stress kept below normal design values presented in the specifications (f_{sa} = 20,000 psi and f'_c = 1350 psi). No increase in these normal design values was permitted when considering seismic loads, acting on the confinement structure, in combination with other mechanical loads such as pressure and thermal loads. Shearing stress at lower wall elevations was found to be less than 50 psi during the 0.2 g design earthquake, and even smaller at higher building elevations. This compares with a maximum allowable shearing stress of 60 psi, permitted by the specifications for beams without web reinforcement.

As discussed above and in Sections 3.3 and 3.7 the tornado and earthquake designs for the Category I structures, systems, and components have been reanalyzed using current criteria (Ref. 51) and analysis procedures (see Sections 3.3 and 3.7). Results of the seismic reanalysis effort are presented in Sections 3.8.3, 3.8.4, and 3.9. 21

3.8.1.5 Structural Acceptance Criteria

The original criteria are discussed in Section 6, except for the tornado and earthquake acceptance criteria which are discussed in Sections 3.3 and 3.7.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

As described above, the confinement structure is constructed of reinforced concrete. Materials of construction and Quality Control procedures utilized are described in Section 6. The structure is designed for internal pressures which are very small when compared with pressures associated with conventional containments. To minimize out-leakage of air, interior surfaces of the N Reactor confinement structure are covered with special, elastomeric coatings, or paint. These are discussed below in 3.8.1.6.1 & 2.

3.8.1.6.1 Elastomeric Coatings

Interior surfaces of the confinement structure (Section 3.8.1) are covered with elastomeric coatings to improve leak-tightness. Included are concrete and steel surfaces located in the 105-N, 109-N and 117-N buildings, which are exposed to confinement Zone I air, plus Zone I and Zone II exhaust tunnels. In addition to providing an essentially leak-tight surface, the special coatings provide: (1) a surface which is more easily decontaminated, (2) better lighting levels in Zone I, (3) less surface cracking (by reducing the drying rate of concrete) and (4) longer filter life with less maintenance (by preventing dusting of concrete surfaces, which would otherwise be exposed directly to Zone I airflow). Only concrete surfaces forming the outer pressure boundary of Zone I in the 109-N confiner are coated. Divider walls between the steam generator cells are not covered, except for about 2.5 ft beyond their intersection with boundary walls, floors, and ceiling slabs. Confinement coatings include the following types of elastomeric paint and thicknesses:

System 25 (25 mil coating)

Prime Coat:	neoprene, thinned as needed.
Body Coats:	successive coats of neoprene to provide dry film thickness of 20 mils, minimum.
Top Coat:	Hypalon, white in color, to provide dry film thickness of 5 mils, minimum.

System 15 (15 mil coating)

Same as System 25 except that the body coat has a dry film thickness of 10 mils, minimum.

System 10 (10 mil coating)

Same as System 25 except that the body coat has a dry film thickness of 5 mils, minimum.

The neoprene for body coats was a thixotropic, one-part, air-curing solution of neoprene (polychloroprene) with suitable pigments and curing agents. Neoprene for prime coats was similar to above except that it did not have to be thixotropic and it was thinned slightly to improve penetration. Alternate coats were of different color to assure proper coverage. Film properties after curing 24 hours at 150°F exceeded the following values:

Ultimate tensile strength ---- 1,600 psi
Elongation at 75°F ---------- 450%,

when tested in accordance with requirements of ASTM D412 (Ref. 14).

The top coat was an air-curing, solution of Hypalon (chlorosulfonated polyethylene) with suitable pigments, extenders and curing agents. Film properties after curing at 24 hours at 150°F exceeded the following values:

Ultimate tensile strength ----- 1,300 psi
Elongation at 75°F ---------- 600%,

when tested in accordance with requirements of ASTM D412. Tests were also required to demonstrate that the finished coating could be readily decontaminated when washed for one minute with a solution of "Tide" detergent and water, using a soft-bristle brush and rinse water.

The air-cured coatings were shown to be resistant to heat and radiation by prior testing. After fully-cured specimens had been exposed to 230°F dry

heat for 20 hours and to total gamma radiation of 3×10^7 rads (2.63×10^9 ergs g^{-1} (C), physical properties did not change more than the following:

- o Hardness did not increase more than 90 Durometer, Shore A
- o Tensile strength was within 50 to 150% of original
- o Elongation was greater than 50% of original.

In addition, the material did not crack when bent 180 degrees over a mandrel 1/8 in. in diameter, nor show any surface cracking after being exposed to the heat and radiation levels described above. When subjected to flammability tests, the dry coating did not support combustion and was self-extinguishing when individual strips were removed from the flame.

Concrete and metal surfaces were specially prepared to receive the respective coatings. Metal surfaces included steel plate at door jams and at wall penetrations, as well as the under-side of the steel roof decking over Zone I in the 105-N building. Imperfections in concrete surfaces, such as holes and gouges, were filled smooth with a mixture of neoprene primer and fine sand prior to applying elastomeric coatings. Likewise, all caulking grooves around flanges of penetrations and at construction joints were first filled with elastomeric caulking compound and then primed with neoprene. Joints between sections of roof decking and others which could not be caulked satisfactorily, were covered with neoprene sheeting, 4 in. wide by 1/16 in. thick. The neoprene strips were bonded in place with special adhesive and then covered with a neoprene/hypalon coating.

System 25, consisting of 20 mil of neoprene and 5 mil of Hypalon top coating, was used on all interior concrete surfaces which form the pressure boundary of confinement Zone I. System 15, consisting of 10 mil of neoprene and 5 mil of Hypalon top coat, was used primarily as a "decontaminable" coating on interior walls, girders and in Zone I exhaust tunnels. System 10, consisting of 5 mil of neoprene and 5 mil of Hypalon top coat was used primarily in the 109-N portion of the confinement structure as a "decontaminable" coating on wall surfaces near the reactor coolant pumps and to extend the primary coating along the outer edges of the divider walls.

A special formulation was used on metal surfaces consisting of a rust inhibitive primer, plus 4 mils minimum of neoprene and 5 mils of Hypalon top coat. Abrupt changes in coating thicknesses were avoided by feathering. Elastomeric wall coatings in the 109-N building extend to within 4 in. of floor surfaces. The 109-N floors, within confinement Zone I, rest on the ground and are covered with a decontaminable, heavy-duty floor coating (Phenolene 300 system having a dry film thickness of 20 mil). This floor coating also extends along the cell walls about 4 in.

Elastomeric coatings applied to upper levels of the 105-N confiner walls and to ceilings, have been subjected to temperatures as high as 160°F during sustained periods of reactor operation. As a result, some of them have lost their elasticity or otherwise deteriorated. Coatings are inspected regularly and are removed when no longer considered suitable and replaced with new material.

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3.8.1.6.2 Elastomeric Confinement Seals

Interior surfaces of the confinement structure provide an essentially leak-tight envelope for an internal pressure range of minus 2 psig to a positive pressure of 5 psig, except for the 109-N steam-generator cells, which are designed for a positive 12 psig. In addition to the elastomeric coatings described above, leak-tightness is enhanced by the installation of confinement seals. These are of two main types, referred to as exterior seals and as interior seals. Exterior seals are located at major construction joints and around flanges at wall penetrations. Interior seals are those installed at wall penetrations and are designed to seal around electrical or instrumental wires and tubing runs, or they may seal interstices of stranded wires and cables.

Caulking grooves were provided at construction joints between concrete shielding walls and mating floor slabs and at other locations where structural components might move slightly with respect to each other, such as girder supports. Caulking grooves were also provided around steel frames and flanges of wall penetrations, including ventilation ducts, access plugs, pipe and conduit penetrations, canal liners, floor drains and door frames.

These caulking grooves in the concrete slabs were cleaned, filled with a elastomeric caulking compound and then tested individually, prior to placement of the elastomeric wall coatings. The wall coatings were run over the caulked joints and onto the respective steel flanges of the various pipe penetrations and door frames. The cured caulking compound used in exterior seals had the properties presented in the following table for Caulk No. 1 which conform to requirements of American Standards Association specification A 116.1-1960 for Class B materials (Ref. 15). Caulk No. 2, being more fluid than Caulk No. 1 prior to hardening, was used to saturate or impregnate webbing, batting or wiring bundles with sealant and for general potting applications. It met all the requirements for Class A materials as specified in American Standards Association specification A 116.1-1960.

	Caulk No. 1	Caulk No. 2
Tensile strength, minimum	350 psi	250 psi
Elongation, minimum	500%	500%
Hardness, Shore A Durometer	30-50	30-50

The above sealants are expected to function continuously at air temperatures up to 200°F and for at least 2 hours under a steam atmosphere at 250°F. For testing and approval purposes, samples of the cured caulking materials were first subjected to one of the following conditioning periods: (1) 30 days continuous at 200°F, minimum, (2) 15 days continuous at 210°F, minimum or 7.5 days continuous at 220°F, minimum. Conditioned samples were tested in accordance with ASTM D412-51T. In addition, radiation tests performed on cured specimens demonstrated that the caulking materials experienced less than 25% loss in tensile and elongation properties when subjected to 5×10^7 rads.

Material for Caulk No. 1 and No. 2 was provided as a two-part system consisting of a polysulfide-base compound and an accelerator and is commonly referred to as "Thiokol." No. 1 had a maximum viscosity of 15,000 poises, when placed; whereas, No. 2 had a viscosity of 500 to 1,000 poises.

A silicone adhesive, manufactured by Dow Corning, was used as a sealant-adhesive for wire and cable interstices and as an adhesive primer for silicone

rubber and silicone rubber foams. Several types of foaming silicone rubber were used for potting and sealing application where a room temperature curing, closed cell, rubber foam was required. Applications including filling of voids behind a plate or other obstructions, preparing a dam prior to installing another type of sealant, providing a resilient gas seal in components subject to expansion and/or contraction and sealing bundles of wire at wall penetrations.

All caulking and sealant materials were tested prior to acceptance and use on the job. Completion records were kept for each seal or joint made during construction of the confinement structure. Each joint was tested for leak-tightness prior to acceptance, which was witnessed and certified by independent inspectors. The over-all leak-tightness of the confinement structure was subsequently tested prior to final acceptance. Leak-tightness of the confinement structure is also determined on a scheduled basis. See Section 6.2.1.

3.8.1.7 Testing and Inservice Inspection Requirements

Described in Section 6.6.

3.8.2 Steel Containment

There are no conventional steel containers or Class MC components at N Reactor, as presently defined in ASME nuclear codes.

3.8.3 Internal Structures of Confinement System

3.8.3.1 Description of Internal Structures

The N Reactor does not have a reactor pressure vessel. The reactor core consists of:

- Graphite Moderator and Reflector
- Graphite Coolant System Tubing
- Horizontal Control Rod System
- Ball Safety System

- o Thermal Shield
- o Primary Reactor Shield (part of biological shield)
- o Zircaloy Pressure Tubes with metallic uranium fuel elements.

The graphite stack is 33 ft wide by 33 ft 4 in. high by 39 ft 5 in. long. It is composed of 89 alternating layers of graphite blocks. The moderator blocks are assembled in an interlocked, keyed open "lincoln log" lattice to permit venting of steam should a pressure tube rupture. The core graphite moderator which surrounds the fueled sections of the pressure tubes is about 22 ft wide by 25 ft long by 25 ft high. The core graphite is surrounded on all sides by the reflector graphite. The general arrangement of the core and graphite moderator is shown in Figure 3.8-13.

The reactor pressure boundary consists of 1003 pressure tube channels that run from front to rear (N-S), 640 graphite coolant tube channels that run from side to side (E-W), 107 ball safety system channels that run from top to bottom, and 87 horizontal control rod channels that enter from the sides. The core graphite serves to moderate fission neutrons to thermal neutrons, and locates and provides structural support for the pressure tubes, horizontal control rods, graphite coolant systems tubes, and ball safety system channels.

N Reactor operates with concentric tube-in-tube Zircaloy-clad, slightly enriched uranium. The fuel cladding is Zircaloy-2, metallurgically bonded to the uranium by a co-extrusion process. Individual fuel element lengths range from 15 to 26 in. This metal clad fuel has a characteristic slow thermal transient response relative to oxide fuels.

The normal graphite stack atmosphere is helium, circulated through the reactor core from front to rear. Gas plenums are located between the active zone and the reflector graphite at the inlet and outlet ends of the stack as well as at both sides. The helium cover gas system helps to keep the graphite moderator dry and minimizes in-leakage of air by maintaining a slight positive pressure. Gas sampling is used to detect water and air in-leakage.

The primary reactor shield is a concrete and steel structure which completely surrounds the reactor core, graphite reflector, and the thermal shield. It consists of the left and right shields, the top shield, and the inlet and outlet shields. The walls of this box-like structure are perforated by the numerous pressure tubes, graphite coolant tubes, instrument lines, and control equipment required to operate the reactor. The primary reactor shield rests upon concrete piers.

The concrete foundation also functions as a biological shield; however, it is considered part of the 105-N reactor building structure, as are the secondary shielding walls. The latter also function as walls of the reactor confinement structure. (The biological shield consists of the primary and secondary shields.) 21

As discussed in Section 3.7, the core, shield and support system structure was reanalyzed in 1985 using current dynamic analyses methods for seismic adequacy to the SSE. The results are discussed in 3.8.3.8.

In this section, the structural features of the reactor foundation, the thermal and primary reactor shields located in the 105-N portion are discussed, followed by discussion of support systems located in the 109-N portion of the confinement for the reactor coolant pumps, steam generators, and pressurizer. The fuel storage basin located in the 105-N building is also discussed. The chemical and physical properties of the shielding materials are described in Section 4.5.4 and Section 4.5.5. An isometric view of the reactor block, and its radiation shields, is depicted in Figure 3.8-13.

3.8.3.1.1 Reactor Foundation

The graphite-core structure, the thermal shields, and the primary reactor shield are supported on a reinforced concrete structure, or pedestal, which (in turn) is supported on a massive spread footing. See Figure 3.8-7 and 3.8-9. The pedestal is 47 ft 1 in. long, and 41 ft 6 in. wide. The top of the pedestal, beneath the bottom insulation layer, is at elevation 456.5 ft MSL; whereas, the peripheral region, beneath the four sides of the primary

reactor shield, is at elevation 455.5 ft MSL. The base plates beneath the primary shield are 1/2-in. thick and 48 in. wide, except that those beneath the inlet and outlet shields are only 40 in. wide. The base plates are keyed into the underlying concrete along all four sides and grouted into proper position. The left and right shields are rigidly welded to the base plates, whereas, they are free to expand outwardly (and inwardly) along the inlet and outlet plates relative to a fixed center point. The bottom and top sections of the inlet and outlet shields are keyed into the bottom base plates and into the top shield at their mid-points. The expansion gap around the inlet and outlet shields is covered with the expansion seal described above.

The gas membrane located beneath the bottom insulation layer is a continuous plate, 1/2-in. thick. It was prefabricated in rectangular sheets and drilled to match holes in the bottom of the reactor core. Plate assemblies were field welded to 4-in., wide-flange beams, previously embedded in the prior concrete pour. After the membrane was in place, the gap between the plate and the concrete between the embedded beams was filled with Portland cement grout. Cooling tubes were then placed on the top of the membrane and covered with molten pig-lead, which was subsequently tamped around the various cooling pipes to improve thermal conductivity. The top of the membrane is at elevation 456.5 ft MSL, as described above.

The reinforced-concrete pedestal beneath the gas membrane is solid for a depth of about 8 ft. It rests upon two exterior (east-west) walls and six center piers. The four corner piers are 7 ft 4 in. by 10 ft 8 in. in area and 10 ft deep. The two interior piers are triangular in shape, with a 6-ft base and an altitude of 6 ft 1 in. Placing of the piers provides a clear area of about 10 by 20 ft in size directly beneath the reactor for ball safety system collection equipment, three east-west passageways and two (north-south) cross tunnels. The top of the mat foundation, which is at elevation 438.5 ft MSL, also functions as the floor of the tunnels and ball-collection room. The mat foundation in this area is about 47 ft wide and 6 ft deep. It is highly reinforced and extends beyond the inner rod rooms on each side of the reactor core. Construction joints between the mat

foundation and adjoining floor slabs and at exterior footings are filled with elastomeric caulking compound. Components of the reactor support pedestal and the bottom foundation mat are of integral construction with continuous reinforcement between walls and slabs. The bottom of the reactor foundation mat is also tied structurally to the exterior footings of the 105-N reactor confinement structure in order to resist an internal confiner pressure of 5 psig.

The foundations of the reactor and shielding support structure have performed adequately since they were constructed in 1961. Measurements have indicated practically no vertical settlement and no differential movements of consequence.

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3.8.3.1.2 Thermal Shield and Bottom Insulation Area

The thermal shield encloses the graphite stack on all six sides, in order to protect the primary reactor shields from excessive heat. The six sides are referred to as the left, right, top, bottom, inlet and outlet thermal shields. Boron steel plates (1-1/16-in. thick) are used as the thermal shield material, except for the central sections of the inlet and outlet shields where cast iron blocks are used.

Details describing the thermal shield are shown in Figure 3.8-14, 3.8-15 and 3.8-16. Typical plate dimensions are presented in Table 3.8-7.

A typical boron-steel thermal plate is 1 1/16-in. thick, two lattice spaces wide and about 40 ft long, with two attached cooling pipes running lengthwise. The various plates are drilled to match penetrations located in the primary shield. The thermal plates, located on the left and right shields and those on the fringes of the inlet and outlet shield, are supported by brackets which are attached to the inside faces of the primary reactor shield. The boron-steel plates, which make up the top thermal shield, rest on channel-iron which provides a 4-in. gap between the thermal plates and the graphite moderator. The channels and the top thermal plates are, in turn, supported on top of the graphite-moderator structure. The bottom thermal shield plates

rest on angle-iron supports located between adjacent webs of the embedded I-beams, which support the graphite-core structure.

Except for the upper flanges, the I-beams are encased in a 16-in. layer of insulating concrete containing serpentine aggregate. The insulating concrete carries only the weight of the bottom thermal shield; whereas, the special I-beams transmit the weight of the reactor-core to the underlying structural foundation. Except for the top flanges of the I-beams, which are made of boron-steel plate, the webs and bottom flanges are carbon steel (A212 Grade B). The beams are 19 3/16-in. deep and have 1/2-in. webs. Top flanges are 2 1/2-in. wide and 1 1/16-in. thick; whereas, the bottom flanges are 4 by 1 in. in size. The insulating concrete and the special I-beams are separated from the structural concrete of the foundation by an impermeable gas-membrane. This membrane was fabricated from carbon steel plate (A201 Grade B), 1/2-in. thick, and was leak-tested prior to placing the insulating concrete. Structural shapes for brackets and miscellaneous supports are ASTM A7 carbon steel. Each of the boron-steel thermal plates have one or two cooling pipes welded continuously along its length. The corner plate assemblies on the inlet and outlet thermal shields, being approximately square in shape, have a W-pattern of cooling pipes, also welded continuously. However, pipes for cooling the special I-beams and the bottom gas membrane are embedded in chaises, welded to the members, in which pig-lead has been poured and tamped around the cooling pipes. The ends of the various pipes extend beyond the thermal plates and through the corresponding walls of the primary reactor shield, where they connect with supply and drain headers. Pipes attached to the thermal shield plates and assemblies are 3/4-in. (Sch. 80), type 304 stainless steel. Additional information on the shield cooling system is presented in Table 3.8-7 and in Section 5.5.

The thermal shield within the reactor pressure tube pattern is made up of cast iron blocks, two blocks per pressure tube, one on the inlet end and one on the outlet end. Typical interior blocks are 8 in. thick, 8 in. wide and 9 in. high and have chevronned edges around the periphery. Blocks located in the outside rows are similar to the interior blocks except that only three edges are chevronned. Each block is pierced by a hole,

3.5 in. in diameter, and is hung from a short piece of tubing, slightly smaller in outside diameter. This tubing bridges the gap between the interior face of the primary shield and the outer face of the graphite stack, on both the inlet and outlet faces of the reactor. In effect, the 2,008 thermal shield blocks are supported and cooled by the pressure tube assemblies.

The brackets and support system for the thermal shield were designed for a maximum horizontal acceleration of 0.2 g, produced by the postulated design earthquake. In addition, wide-flange beams, 4 in. deep, are installed along the left and right sides of the thermal shield. These structural members are located near the top and bottom of the graphite moderator structure and between the thermal shield plates and the primary shield. The 4-in. beams prevent shifting of the graphite moderator towards the primary shield and from damaging the thermal shield. Relative shifting in the direction of the pressure tube channels is prevented by the cast-iron thermal blocks, which are 8 in. thick and located within the pressure-tube pattern at the inlet and outlet faces of the reactor. The design of the various support brackets prevent excessive motion during periods of normal reactor operation and during the design earthquake but also provide for normal displacement due to thermal expansion and contraction.

In order to minimize the amount of heat transferred from the interior of the reactor core and from the thermal shield to the four walls of the primary shield, 18 Ga. aluminum sheets are attached to the inner steel faces of the primary shield. These reflector sheets are fastened to the primary shield with eight, 3/16-in. metal screws, 1/2 in. long, and are drilled to match penetrations in the primary shield. A typical reflector sheet, which is type 3003-H14 aluminum, is approximately 72 by 60 in. in size, although small pieces are used occasionally to provide the proper fit. Reflector sheets, 1/8-in. thick, rest on the boron-steel plate assemblies which make up the top thermal shield. These aluminum sheets are typically 144by 48 in. in size and are held in place with 5/16-in. studs, (which are welded to the thermal plates), 1-in. spacers, washers and hex-nuts. The design of the support system for the four sides and top of the aluminum reflector provides for differential expansion and contraction of the various shield components

as well as to prevent excessive movements that might otherwise occur during accident conditions or during an earthquake.

The thermal shields have performed satisfactorily since reactor startup in 1963. However, a potential problem area involves burn-out of boron contained in the thermal plates located on the side, top and bottom shields. This problem area was studied by Professors Murray and Elleman in 1975 for ERDA. They concluded that even after ten years of additional reactor operation, the thermal shielding effectiveness would still be significant, and that the worst that might happen would be some local increases in radiation levels due to activation (Ref. 29).

3.8.3.1.3 Primary Reactor Shield

The primary reactor shield is a concrete and steel structure which completely surrounds the reactor core, graphite reflector and the thermal shield. It consists of the left and right shields, the top shield and the inlet and outlet shields. The walls of this box-like structure are perforated by the numerous pressure tubes, cooling pipes, instrument lines and control equipment required to operate the reactor. The primary reactor shield rests upon concrete piers which surround the bottom insulation layer. The concrete foundation, located below the bottom gas membrane, also functions as a biological shield; however, it is considered part of the 105-N reactor building structure, as are the secondary shielding walls. The latter also function as walls of the reactor confinement structure. See Section 3.8.1.

The primary shield may be described as a massive structure of steel plate and bracing, filled with high-density concrete. The steel skin plates and structural members of the left side, the top shield and the right side are welded together to make a rigid frame. The concrete located inside the steel forms is also reinforced with conventional concrete re-bars. The inlet and outlet shields are not attached physically to the side and top shields, but are simply supported along their outer edges by means of corrugated keys. Thus, the inlet and outlet shields, which operate at temperatures corresponding to the inlet and outlet temperatures of the reactor coolant, respectively, are free to expand and contract independently of the side and

top shields, which operate at much lower temperatures. Gas seals cover the expansion gap between the inlet shield and the rigid frame (which makes up the top and side shields) and between the outlet shield and the corresponding faces of the frame. Gas seals also extend along the bottom of the inlet and outlet shields and attach to the bottom (fixed) section of the primary shielding structure. The gas seals are constructed of sheets of neoprene rubber, which are attached to the surface of the shield with adhesive, compression plates and cap screws. The gas seal system is designed to operate at about 0.5 psig during normal periods of reactor operation and for a surge pressure of 5.0 psig.

The top shield is supported by seven steel girders, specially fabricated for the application. The bottom flanges support a series of thermal plates and the ball safety system hopper assemblies. The four exterior vertical walls of the top shield consist of carbon-steel plates, which serve primarily as a form for containing concrete. Prior to erection at the reactor, shield components were prefabricated off-site. The basic component for the vertical walls was called a "crate," which consisted of a box-like structure made up of an inside and an outside steel plate, rigidly braced together but open on the top and bottom. Shield penetrations were provided by welding or expanding pipe-ends to the respective skin plates, which had been predrilled to a specified hole pattern. The length of the inlet and outlet crates are slightly greater than the width of the pressure tube pattern; whereas, the side shields are made up of a series of smaller units, including the corner crates. Crate and plate components of the left, top and right side shields are rigidly welded together, including the thermal plates and the seven structural girders located in the top shield.

The top primary shield is 46 ft 11 in. by 40 ft 4 in. in plan and 5 ft 5 in. deep. The left and right side shields are 43 in. thick; whereas, the inlet and outlet shields are 35 in. thick. In the pressure tube region, the primary shield is 47 ft 9 in. long and 41 ft 4 in. wide. The four shielding walls rest upon a steel base plate having a top elevation of 455.5 ft MSL. The upper surface of the top primary shield is at elevation 498 ft MSL.

Figure 3.8-13 is an isometric drawing showing the various shield components and their relationship to each other.

Crates used to erect the inlet and outlet shields are not welded rigidly together, although a light bead of weld metal was placed along horizontal joints to make the concrete forms (crates) water-tight when placing concrete. After concrete was placed and cured, an omega strip of light-gage steel was welded to the outside plates along horizontal joints, to provide a 5 psig gas seal. Successive tiers or crates, located in the inlet and outlet shields, are held together by vertical reinforcing bars embedded in the concrete.

High-density concrete was placed in the inlet and outlet shields by means of the preplaced aggregate method, one crate at a time. This method was used to assure that all intertices between and around embedded tube-sleeves would be filled with concrete of the specified composition and density. (The shielding concrete required for the inlet and outlet shields contained pieces of steel aggregate which tend to segregate out of the mix when conventional methods of concrete placement are used.) The conventional method was used to place different types of concrete in the left, top and right side shields. The various shielding concretes used to construct the primary reactor shield are described in Section 4.5.5, as well as their physical properties (Ref. 13). Reference is made to Figure 3.8-17, which shows the physical makeup of the shield, location of crate assemblies and the corresponding type of shielding concrete used in each section of the shield.

Prior to placing concrete in a particular section of the primary reactor shield, approximately 100 thermocouples were installed and leads run to common junction boxes. Cooling tubes were leak-tested prior to and after placing concrete and showed no leakage under a hydrostatic pressure of 300 psig when held for 10 minutes. After erection of the various shield components, the exterior faces received an enamel coating of air-curing epoxy-ester, except for the inlet and outlet shields within the gas-seal system. The outside surfaces of the latter shields were painted with a high-temperature inorganic coating rated at 600°F.

The primary reactor shield has performed satisfactorily since the 1963 start of reactor operation, except for two problem areas. The first problem to surface involved the gas-seal system located on the right-outlet face of the shield. Apparently the design of the gas-seal in this particular area could not accommodate the differential thermal expansion which occurred each time the reactor was started up and when shut down. This deficiency has required repairs periodically to the outlet gas-seal system, in order to minimize leakage of inert reactor gas. The second problem is related to the progressive growth that Zircaloy-2 pressure tubes experience under service conditions. See Section 5.3. The pressure tube assemblies are attached to the outer skin-plate of the outlet shields, and are designed to expand freely through expansion joints located on the inlet shield. However, the assemblies have been binding at the packing; consequently, they load the respective crates. As a result, both the inlet and outlet shields deflect outwardly more than originally expected. Shield movement is being monitored to assure that the shields will not be damaged by this loading.

3.8.3.1.4 Reactor Coolant Pump Support System

The reactor coolant system (RCS) is located in 105-N and 109-N. The portion of the RCS within 105-N consists of 16 parallel lines that conduct cooling water from an inlet water manifold in the heat exchanger building (109-N) to the reactor. Each of these 16 lines terminates in a vertical header to which is attached from 54 to 66 individual header to pressure-tube inlet nozzle connectors. Similar outlet risers and parallel lines conduct coolant from the pressure tube outlet nozzle-to-header connectors to an outlet water manifold. 21

The portion of the RCS located within 105-N needed for ECCS operation (from the CV-5 check valves on the core inlet side to the CV-3 check valves on the outlet side) was designed for the equivalent of the 0.25 g SSE. The rest of the RCS (to and from the pumps and steam generators) was designed for reactor coolant pressure boundary conditions, but has been upgraded to seismic category I during the 1987-88 seismic reanalysis program (see Section 3.9.3).

The RCS equipment in 109-N consists of six pumping and heat transfer loops located in six separate cells. Each loop (cell) contains two parallel steam generators, a circulating pump, and associated valves and instrumentation and measure 50 ft wide by 94 ft 3 in. long I.D. Piping and steam generators in each of the six cells can be isolated from the main header piping by means of isolation valves. There are five loops normally on-line with the sixth available as a spare or isolated for maintenance.

The six isolated rooms, or cells, are served from a large pipe gallery located on the north side of the cells. The pipe gallery, the steam generator cells and a seventh cell, containing auxiliary equipment, are surrounded by thick walls and slabs of reinforced concrete. The walls are 5 ft thick, the roof 3 ft 6 in. and the floor slab in the cell where the pumps are mounted at El. 0 ft 0 in. is 5 ft thick. Each cell is designed to withstand an internal pressure of 12 psig. As with 105-N, this concrete structure provides shielding and missile protection, in addition to serving as the confinement barrier. The entire confinement successfully passed on integrated leak rate test (ILRT) and high-pressure test in the second half of 1987, demonstrating its ability to withstand maximum calculated accident pressure transient (i.e., 3.9 psig), as well as its ability to satisfy 10 CFR 50 Appendix J requirements.

The reactor coolant pumps are stainless steel, single-stage centrifugal pumps that are mounted at El. 0 ft 0 in. on a 5 ft reinforced concrete slab that forms the roof of the integral access corridor at El. -16 ft leading to each cell. These pumps are driven by low-pressure steam turbines, which are installed outside of the steam generator cells in the 109-N building turbine bay.

An electric pony motor capable of driving the reactor coolant pump is mounted in each turbine shaft. These motors provide water circulation to prevent reactor damage during a scram transient and shutdown conditions that would follow complete loss of steam to the turbines. The pump-drive units

(turbines and pony motors) are located outside the shielded cells in the Turbine Bay area and are connected to the pump by an extension shaft through the cell wall.

The turbine drive shafts penetrate through the steam generator cell confinement walls to the reactor coolant pumps. These penetrations are pressure sealed and conform with confinement requirements. The pump bearings are provided with high-pressure injection seals through which makeup reactor coolant is supplied. There are no flywheels attached to these pumps.

3.8.3.1.5 Steam Generators Support System

The secondary coolant system removes heat from the reactor coolant system by boiling secondary water in the shell side of the steam generators. During dual-purpose reactor operation, the major fraction of the steam generated is routed to the WPPSS Hanford Generating Plant (HGP). The remaining steam is used to drive the reactor coolant pumps and the onsite turbine-generator unit and to keep the dump condensers warm so they are ready to take the full steam load in the event of a rapid load rejection. Condensate from the HGP or the dump condensers is pumped back to the steam generators for recycling. 21

The steam generators are installed at El. -16 ft in pairs within separate cells in the 109-N building. They are horizontal U-tubes with integral moisture separation equipment. Each steam generator construction incorporates a carbon steel shell with Inconel-600 U-tubes. Reactor coolant is delivered to a bottom-centered vertical manifold of each steam generator and flows up through a central channel of the annular manifold to the top half of the tube sheet, where it is directed through the horizontal U-tubes and leaves via the outside channel of the vertical manifold.

Each steam generator is provided with two support saddles that rest upon concrete reinforced pedestals. One support is firmly secured to its foundation pedestal with the other support designed to permit expansion and contraction on rollers. The concrete pedestals, in turn, are supported on an isolated matt foundation.

The secondary coolant system is not required for safe shutdown or accident mitigation.

3.8.3.1.6 Pressurizer Support System

The pressurizer is a vertical, cylindrical surge vessel with hemispherical top and bottom heads, fabricated from carbon steel clad with stainless steel. Pressure control is maintained by adjusting the temperature and steaming point of the pressurizer. Electric heaters in the liquid-containing section of the vessel are turned on and off as required to produce steam in the vapor portion of the vessel. Water sprays are provided to quench the steam and cool the liquid, should the pressure rise above the control point setting.

Reactor coolant inventory control is maintained by injecting coolant through the RC pump shaft seals and via locations near the pump suctions to replace coolant lost by coolant purification and sampling procedures. The pressurizer level controller provides automatic control of the injection water supply system.

The pressurizer cell, or penthouse, is centrally located over the pipe gallery along the extended line of the reactor unit. It is 35 ft long, 26 ft wide and 43 ft 3 in. high, inside dimensions. The pressurizer cell protrudes above the roof of the pipe gallery (El. 39 ft) a distance of about 48 ft to El. 87 ft 0 in. The pressurizer is supported at the bottom by steel girders spanning the 35 ft length of the penthouse and at the top and midpoint from the south wall of the penthouse by a supporting steel cantilever arrangement.

The 109-N building steam generator cells, pipe gallery and pressurizer penthouse cell are all an integral part of the 105-N/109-N confinement structure and have been qualified to the 0.25 g SSE (Ref. 51).

3.8.3.1.7 Fuel Storage Basin

The N Reactor fuel storage basin is used to store discharged fuel elements in a water environment for shielding and heat dissipation for a sufficient length of time to permit fission products to decay to acceptable levels for chemical processing.

The basin is located in the 105-N building in a separate facility adjacent to the reactor, and is divided into areas designated as North Basin and South Basin. Adjacent to these two areas are the fuel examination facility, the segregation area, and the cask loading pits. A water tunnel connects the basin area with the discharge pit. The North Basin area is covered entirely, and the South Basin floor is partially covered by a modular array of cubicles formed by boron and concrete posts and slabs. These cubicles house either aluminum or stainless steel canisters containing discharged fuel elements. The fuel storage basin facility is also part of the 105-N/109-N confinement structure and as such is qualified to seismic Category I (0.25 g SSE) (Ref. 51). 21

3.8.3.2 Codes, Standards and Specifications

The codes, standards, and specifications generally applicable to the confinement structures, including internal structures, is discussed in Section 3.8.1.2.

3.8.3.2.1 Thermal Shield - Criteria, Codes and Standards

The thermal shield was designed to the following criteria:

1. Limit total heat transferred to and generated in the primary reactor shield (coming from the reactor core) to less than an average of 125 Btu/ft^2-h.
2. Capable of removing a maximum local heat load of 5000 Btu/ft^2-h.
3. Minimize size of annular gaps at penetrations to prevent radiation streaming but to be large enough to accommodate relative movements due to thermal expansion and contraction.
4. Withstand an earthquake producing a maximum horizontal ground acceleration of 0.2 g. (This has been reanalyzed and qualified in the SSE, see 3.8.3.8.) 21
5. Withstand being displaced by pressure-buildup from a postulated rupture of a reactor pressure tube.
6. Permit replacement of pressure tubes.

The chemical and physical properties of the boron-steel plate and cast-iron blocks, used to construct the thermal shield, are described in Section 4.5.4. Other materials conformed to the following standard specifications:

Reference	ASTM Designations	
16	A7 -58T	Steel for Bridges and Buildings
17	A201-57T	Carbon-Silicon Steel Plates of Intermediate Tensile Ranges
18	A212-57T	High Tensile Strength Carbon-Silicon Steel Plates for Boilers and Other Pressure Vessels
19	A298-55T	Corrosion-Resisting Chromium and Chromium-Nickel Steel Covered Welding Electrodes.
20	A312-59T	Seamless and Welded Austenitic Stainless Steel Pipe
21	A371-53T	Corrosion-Resisting Chromium and Chromium-Nickel Steel Welding Rods and Bare Electrodes
	Federal Specifications	
22	QQ-A-359d	Aluminum Alloy Plate and Sheet (3003-H14)
23	MIL-E-19933	Electrodes, Welding, Bare (Austenitic Chrome-Nickel for Corrosive and High Temperature Services)

3.8.3.2.2 Primary Shield - Criteria, Codes, and Standards

Functional requirements of the primary reactor shield are as follows (Ref. 30).

1. Reduce neutron and gamma radiations originating in the reactor core to acceptable limits defined below.
2. Locate and support the thermal shield, pressure tube assemblies, coolant piping, auxiliary shielding blocks, control rods, safety systems, monitoring facilities, etc.

3. Support and protect the safety system ball-hoppers, embedded in the top shield.
4. Act as a barrier to the leakage of the inert gas from the inside of the reactor.
5. Contain reactor materials and gases in case of a reactor incident or earthquake.

At times personnel must have access within the confinement structure, adjacent to the outer faces of the primary reactor shield when the reactor is not operating and the reactor coolant piping system has cooled down. Thus, the primary reactor shield is required to reduce neutron fluxes to values which will not seriously activate surrounding reactor components and structural materials. The walls of the reactor confinement structure provide additional shielding so that radiation levels outside Zones I and II are below safe biological radiation limits at all times.

Steel tubing, plate and structural members used to fabricate the various shield components and crates, conformed to the respective ASTM Designations presented below:

Reference	ASTM Designation	
16	A-7-58T	Spec. for Steel for Bridges and Buildings.
31	A-15-58T	Spec. for Billet-Steel Bars for Concrete Reinforcement.
32	A-20-56	Spec. For General Requirements for Delivery of Rolled Steel Plates of Flange and Firebox Quality.
33	A-107-58T	Spec. for Hot-Rolled Carbon Steel Bars.
34	A-108-58T	Spec. for Cold-Finished Carbon Steel Bars and Shafting.
35	A-179-58T	Spec. for Seamless and Cold-Drawn Low-Carbon Steel Heat-Exchanger and Condenser Tubes.

Reference	ASTM Designation	
17	A-201-57T	Spec. for Carbon-Silicon Steel Plates of Intermediate Tensile Ranges for Fusion-Welded Boilers and Other Pressure Vessels.
36	A-210-58T	Spec. for Medium-Carbon Seamless Steel Boiler and Super-heater Tubes.
18	A-212-57T	Spec. for High Tensile Strength Carbon-Silicon Steel Plates for Boilers and Other Pressure Vessels.

3.8.3.2.3 Other - Codes, Standards, and Criteria

Criteria, codes, and standards applicable to the design of the reactor primary and secondary coolant systems are discussed in Section 5.4. Design criteria for the spent fuel storage basin are presented in Section 9.1.3.

3.8.3.3 Loads and Load Combinations

3.8.3.3.1 General

The load conditions, categories, and combinations that are discussed in 3.8.1.3 for the confinement structure design apply also, in general, to the concrete and structural steel internal support structures. Applicable loads and load combinations for which a structure is designed depends on the conditions to which the particular structure could be subjected. Load conditions are as follows:

a. Service Load Conditions:

   Under service load conditions, the loads are those encountered during construction, testing, normal operation, shutdown, and severe environmental such as the operating basis earthquake and the design basis wind.

b. Factored Load Conditions:

Under factored load conditions, the loads are abnormal loads due to postulated accidents such as loss-of-coolant accidents due to pipe rupture, shutdown earthquake, and tornado.

The load categories for loads under service and factored load conditions are as follows:

a. Service Load Conditions

(1) Construction--All events and loads during structural construction, excluding those during testing. 21

(2) Testing--All events and loads applied during structural integrity tests and pre-operational tests such as hydrostatic tests and pressure tests. Each testing event is considered to be mutually exclusive of other testing events.

(3) Normal Operation and Shutdown--All events and loads that could reasonably be expected during the operation, shutdown, and normal maintenance of the power plant. The magnitude of these events and loads are based on the probability of occurrence of at least once in the design life of the plant.

(4) Severe Environmental--All site-related environmental events and loads that could infrequently be encountered during the plant life such as the operating basis earthquake, design basis wind, and the flood of record.

b. Factored Load Conditions

(1) Extreme Environmental--All loads due to site-related environmental events which are credible but highly improbable. These events include the safe shutdown earthquake, design basis tornado, probable maximum flood.

(2) Abnormal--All loads due to postulated accident events. They include pressure, temperature, pipe rupture, shutdown earthquake and tornado.

These loads and load conditions are illustrated by the matrix shown in Table 3.8-4.

3.8.3.3.2 Primary Reactor Shield Loads

The primary reactor shield is designed for the following pressure, structural and thermal loads:

1. An internal pressure build-up inside the core of 5 psig, when the ambient pressure is atmospheric
2. An external (ambient) pressure of 5 psig, when the internal pressure is atmospheric
3. Weight of the neutron absorbing balls stored in hoppers, plus a uniform live load of 500 lb/ft^2 on the top shield
4. Weight of equipment attached to the vertical walls such as the thermal shields, cooling tubes, control rods, and pressure tube assemblies
5. Lateral forces caused by an earthquake producing a maximum horizontal ground acceleration of 0.2 g
6. Thermal effects caused by temperature differentials of up to 50°F through the top shield and up to 100°F through the vertical walls of the primary reactor shield.

Radiation shielding requirements dictated the thickness of the top and sides of the primary reactor shield, rather than the 5 psig pressure requirement.

As a result, shield failure is not likely to occur unless the internal pressure exceeds about 35 psig.

Maximum operating temperatures, used in design, are 160°F for the top shield, 200°F for the left, right and bottom shields (not including the bottom insulation layer) and 575°F for the inlet and outlet shields. During past years of operation, the above temperature limits have been maintained except for certain "hot-spots" on the top shield, which have run as high as 235°F at times. The "hot-spots" are mostly steel components of the ball safety system hoppers, located above the central portion of the reactor core, which are not protected by the thermal shield. These components scatter and attenuate radiation streaming upward from the reactor core through the open ball channels. Of the 19 thermocouples used to monitor temperatures in the top shield, 9 of them run below 160°F, 6 below 175°F, 3 below 225°F and 1 below 235°F, maximum. These temperatures have been reviewed and are judged acceptable with respect to design thermal loads.

The left and right shields and the top primary shields are cooled by circulating treated water through a series of type 304, stainless-steel pipe embedded in the concrete near the inside faces or embedded in tamped-lead chaises attached to the inside skin-plates of the respective shield components. The inlet and outlet shields are cooled by reactor coolant circulating in the pressure tube assemblies, which penetrate the inlet and outlet crate assemblies.

Design loads for other shield and reactor components are discussed in Section 4.5 including the physical and mechanical properties of the materials utilized. 21

Design loads for reactor coolant pumps, steam generators, and pressurizer are discussed in Section 5.4.

3.8.3.4 Design and Analysis Procedures

The design and analysis procedures used for the design of the confinement internal support structures described in Section 3.8.3.1 are discussed in Section 3.8.3.1.4.

3.8.3.5 Structural Acceptance Criteria

Structural acceptance criteria for confinement structures are discussed in Section 6.2.1 where the adequacy of the structural design is evaluated by determining the peak pressure, and the long-term pressures and temperatures of the confinement atmosphere during and following postulated accidents.

The acceptance criteria utilized for the seismic reanalysis of the structural integrity of the reactor and support structures discussed in 3.8.3.8 is provided in References 56 and 57. The basis included the American Concrete Institute code requirements for the seismic integrity evaluations of the primary shield and reactor pedestal.

During construction, special procedures were developed for the welding of the thermal shield components. These are discussed below.

3.8.3.5.1 Thermal Shield Welding

Welding of thermal shield components required special procedures because this type of shielding had not been used previously. Competent fabrication and welding procedures were developed at Hanford facilities prior to final design of the N Reactor thermal shields (Ref. 24). These procedures were incorporated in HWS-6587, "Specification for Fabrication of Thermal Shield, Insulation Layer and Aluminum Reflector Components," issued August 8, 1960 (Ref. 25). Separate welding procedures and welder performance qualifications were required for each of the following four combinations of base metals:

- Carbon steel to carbon steel.
- Carbon steel to boron steel.
- Boron steel to boron steel.
- Type 304 stainless steel pipe to boron steel plate.

Welding of carbon steel to carbon steel was in accordance with existing Hanford Plant Standards (Ref. 26.), except that welding procedure qualification and welder qualification was in accordance with Section IX of the ASME Boiler and Pressure Vessel Code (Ref. 27). Welding of carbon steel to boron steel and of boron steel to boron steel was performed in accordance with the above Hanford Plant Standard, except as follows. Approved welding processes included the shielded metal-arc method and the inert-gas-shielded metal-arc process. In the latter case, a welding grade argon atmosphere was used, mixed with a slight addition (up to 5%) of oxygen to reduce convexity of the weld face. Type 312 stainless steel was used as filler metal. Covered electrodes conformed to the requirements for type E312-16 electrodes, as described in ASTM Designation A298 (Ref. 19).

The stainless steel cooling pipes were welded to the boron-steel thermal plates by means of continuous seam welds laid down on each side of the pipe with machine-welding equipment. The welds were made using the inert-gas-shielded metal-arc process in which the feed rate of filler metal wire and the position of the welding head were carefully controlled mechanically. Filler metal was either type 312 or type 308L stainless steel. Electrodes of type 312 stainless steel conformed to the requirements of ASTM A 298, discussed above. Covered electrodes and bare wire of 308L conformed to ASTM Designations A298 and A371, respectively. Bare filler wire was used for continuous seam welding and occasional tack welding, whenever feasible.

Liquid dye penetrant techniques and spot radiographs were utilized to assure that competent welds were being executed during fabrication of thermal shield components. Spot examinations were made in accordance with requirements of Paragraph UW-51 of Section VIII of the ASME Boiler and Pressure Vessel Code (Ref. 28). In addition, production test plates were made for each 1000 ft of weld (500 ft of pipe) and subsequently used for cutting sections through samples of the welded pipe and for detailed examination. After the cooling pipes were welded to the various thermal shield plates, they were hydrostatically tested at 500 psig and accepted if they showed no leakage after a test period of ten minutes.

3.8.3.6 Material, Quality Control, and Special Construction Techniques

The materials of construction and Quality Control procedures utilized are discussed or referenced in the description of the internal structures, Section 3.8.3.1.

3.8.3.7 Testing and Inservice Inspection Requirements

Rigorous inspection was carried out during construction and in conjunction with the Quality Control assurance procedures for structural materials in effect at the time.

The structures specified in 3.8.3.1 are visually inspected when the reactor is shut down. Exposed structural steel items are visually inspected during refueling outages. The structural integrity of reinforced concrete members is evaluated by checking for spalling and excessive deformations. Representative samples of anchor bolts are tested for tightness.

A graphite surveillance program has focused on assessment of the current condition of the graphite moderator, prediction of future behavior, and mitigation of adverse behavior consequences in response to ASEP issues 136 and 137 (Ref. 61).

Additional discussion of the testing and Inservice Inspection requirements for the confinement structures are described in Section 6.

3.8.3.8 Seismic Reanalysis of Reactor and Support Structure

In 1985, an evaluation was completed of the structural integrity of the N Reactor core and the functional operability of the reactor's critical safety systems when subjected to a Safe Shutdown Earthquake (SSE). The SSE includes a ground acceleration of 0.25 g horizontal and 0.167 g vertical. The evaluation also included the graphite core and safety systems when subjected to a pressure tube rupture event, refer to Section 15.6.4.4. The two events were not considered to occur simultaneously. The seismic evaluation and results are summarized below, based on References 57, 58, 59, and 60.

3.8.3.8.1 Phase I

The evaluation was conducted in two phases. In Phase I analytical models were developed, and the response of the core was assessed when subjected to an SSE.

Failure modes of the graphite keys were identified and analytical models developed to calculate the reactor core response and to evaluate safety system components within the reactor.

Seismic evaluations included soil-structure interaction analyses to account for interaction effects between the soil media, the reactor core, and buildings 109-N and 105-N. These analyses provided input motions for a detailed analytical model of the reactor core. Analyses of the detailed core model provided response of the core's safety systems.

A worst-case scenario was assumed with frictional resistance and marginal graphite key resistance being neglected although it is known to exist. Also, where stiffness values were used for elements for which satisfactory test data were not available (i.e., ball channels), conservative estimates of stiffness were calculated. (Reference 59.)

The results of the Phase I evaluation indicated the reactor core would maintain structural integrity, and the safety systems would remain functionally operable during a postulated SSE.

3.8.3.8.2 Phase II

In the Phase II evaluation, a more restrictive case was considered in which no graphite keys were taken as active in the fueled zone. This case was taken as a conservative case of extended operation, since current surveillance indicates a loss of only approximately one-eighth of the total key population.

The seismic analysis methods used are developed and described below and summarized from Reference 58. 21

- Loads--Seismic loads were generated by performing a soil-structure interaction (SSI) analysis to account for the interaction effects between the soil media, the reactor core, and Buildings 105-N and 109-N. During Phase I, mathematical models were developed for the soil, the reactor core, and the buildings. The same models were used in Phase II.

- Structural Analyses--The reactor core model is highly nonlinear to account for the 89 layers of graphite blocks and the internal frictional forces of blocks sliding on one another. Other sources of nonlinear responses are in the gaps between the graphite keys, in which gap closure is a source of nonlinear response. The reactor core model was revised from the Phase I model due to changes in the material properties provided by UNC. The primary changes were the strength properties of the graphite. Also, the frictional force between the graphite increased and nonlinear gap elements were placed between the inlet/outlet shield and the primary shield.

  The detailed reactor core model was subjected to the earthquake time histories from the load generation effort, and the responses of the reactor's safety systems, components, and the biological shield were evaluated.

  In parallel with this effort, a sensitivity study was conducted to confirm that the Phase I detailed reactor core model was adequate to capture the local responses of the core's elements. A more detailed model was created by doubling the mass nodal points of the model. It was then subjected to the same seismic analysis. Both model responses were compared at several locations within the core and primary shield and the correlation between the responses was good.

- Safety Systems and Structure--The evaluation of the reactor's safety systems and structure were similar to the Phase I effort. The dynamic response of the core's elements was evaluated for functional operability and the reactor's shield and pedestal were evaluated for structural integrity. The evaluation consisted of comparing the system's capacity

to the seismic load demand. If the capacity exceeded the demand, the ratio of the two provides the margin of safety and is greater than unity. If the capacity is less than the demand, the converse is true. The safety systems and elements included:

- o Pressure tubes
- o Graphite coolant tubes
- o Vertical ball channels
- o Horizontal control rod (HCR) passages
- o HCR-T blocks
- o Graphite keys
- o Inlet/outlet shield gap closure
- o Biological shield and pedestal.

=1

The safety systems were evaluated for functional operability and, the primary shield, reactor pedestal, and graphite keys were evaluated for structural integrity. In addition, the reactor core and primary shield were evaluated for overturning. These evaluations are summarized below and in Tables 3.8-10 and 3.8-11.

3.8.3.8.3 Pressure Tubes

The maximum strain (e) in the pressure tubes due to the postulated seismic event is 0.17% strain. This strain represents a margin above the one percent strain criteria of 5.88. (See Table 3.8-10.) (Note that the one percent strain criteria has a margin of safety associated with it. Therefore, the factor of safety is above 5.88.) The pressure tubes meet the safety criteria.

3.8.3.8.4 Graphite Coolant Tubes

The maximum strain in the graphite coolant tubes is 0.12% strain and is below one percent strain. This represents a margin of 8.33 above one percent strain. No graphite coolant tubes exceed the one percent strain.

3.8.3.8.5 Vertical Ball Channels

In the graphite core, the maximum interlayer horizontal and vertical displacements of the vertical ball channels are 0.87 in. and 0.002 in. These displacements represent margins of 1.29 and 250 above the allowable displacements. At the interface of the ball channels and the ball hoppers at the top of the core, the maximum horizontal displacement is 1.2 in.

This displacement represents a margin of 1.04 above the allowable displacement. (Note that the allowable horizontal displacement depends on the location in the core. See Reference 25, Vol. I.) The vertical ball channels meet the safety criteria. 2

3.8.3.8.6 Horizontal Control Rods

At the interface between the primary shield and reflector, the maximum displacement is 0.63 in. At the interface between the reflector and the fueled zone, the maximum transverse displacement is 0.074 in. These displacements represent margins of 1.6 and 6.8 above the allowable displacements. (Again, note that the allowable displacement depends on the location in the core.)

3.8.3.8.7 Structural Integrity

Table 3.8-11 gives the maximum stresses in the primary shield, the inlet/outlet shields, and the reactor pedestal. The minimum factor of safety of these structures is 1.16. All stresses are less than ACI code allowable stresses.

The ACI capacity reduction factor for shear, $\underline{\phi}$, was not applied in these evaluations. This factor was not warranted in this evaluation, since the increase in concrete strength due to aging to date approximately offsets the capacity reduction factor. In addition, the strength evaluations were

performed using simplified conservative methodologies. This included neglecting reinforcing steel and using simplified member capacity equations. Concrete also has significant reserve strength due to strain rate effects. (Additional factors of safety are associated with the definition of the seismic event and with the soil-structure interaction analysis. These were not accounted for; therefore, the actual factor of safety is considerably greater than 1.16.)

Overturning is not considered to be credible for dynamic overturning moments that occur during a seismic event. Thus, overturning will not occur. Overturning of the core and the primary shield is also evaluated in Table 3.8-11. The minimum factor of safety of overturning is 1.12. This factor of safety is conservative since it assumes a statically applied overturning moment.

3.8.3.8.8 Graphite Keys

Based on the predicted relative displacements between nodal points in the graphite core model and the assumed probability distribution of the gap sizes, the graphite keys in the unfueled zone will not crack or fail.

3.8.3.8.9 HCR T-Blocks

Provided that the HCR T-blocks do not extend into the channel more than 0.5 in., the T-blocks will not slide or overturn into the HCR channel. If an HCR T-block extends into the channel more than 0.5 in., the T-block may prevent the insertion of the control rod. The maximum displacement calculated was 0.09 in.

It was concluded that the reactor core will maintain its structural integrity, and the safety systems will remain functionally operable under the postulated SSE.

3.8.4 Other Seismic Category I Structures

3.8.4.1 181-N Building

The river pump house (181-N building) is located on the right bank of the Columbia river just northwest of the high-lift pump building (182-N). It is constructed of reinforced concrete and provides support and suction wells for seven deep-well turbine pumps. Three of the pumps, two of which are located in the central bays, and one which is located in the 181-N Building Annex, supply river water to the Emergency Core Cooling System (ECCS). The other pumps (located in the outer bays, two on each side) supply river water to N Reactor during normal periods of operation. The three ECCS pumps are driven by diesel engines. The four circulating water pumps are electrically driven. Two of the ECCS pumps, diesel engines and auxiliary equipment are housed in a concrete superstructure located on the main floor of the river pump house, which is at 421 ft MSL. The third ECCS pump/diesel engine is located in a separate steel-framed metal building, which is adjacent to the masonry/concrete superstructure. The roof of the superstructure is at about 439 ft MSL. Architecture and structural details of the 181-N building are shown on Figure 3.8-18 and 3.8-19.

The cellular structure of the pump house rests on a matt foundation of reinforced concrete, 5 ft thick. Two trapezoidal-shaped keys extend downward another 6 ft into the ground along the west and east sides of the foundation slab. The slab is 124 ft long and 94 ft wide. The longer axis is parallel to the river bank, which is essentially north-south at this location. The top of the foundation slab is at 360 ft MSL. It rests on undisturbed soil consisting primarily of compacted gravel and sand. Two keys penetrate into the Ringold formation, a very thick layer of highly-compacted silty clay, containing some sand and gravel. The allowable bearing capacity of these foundation materials is 12,000 lb/ft^2; however, the design is based upon a maximum soil pressure of only 6000 lb/ft^2 for normal load conditions. Except for the space taken by four wing walls, the east side of the foundation slab, for a distance of 29.5 ft, is covered with gravel backfill to 421 ft MSL. The north and south sides of the foundation, beyond the walls of the pump house, are also covered with backfill, which varies from zero depth of

backfill at the west (river) side to 421 ft MSL at the other. Thus, the west side of the pump house is exposed to the water in the river above 360 ft MSL; whereas, the east side is contiguous with the bank of the river. The river bed and bank in the vicinity of the pump house are covered with 2 to 4 ft of riprap to 415 ft MSL. 2

The main floor of the river pump house varies from 1 to 3 ft thick. It is reinforced with steel bars, which are anchored to the walls of the six pump wells. The masonry/concrete superstructure, which encloses two of the ECCS pumps and diesel engines, is anchored to the top of the main floor slab with a series of rebar dowels. The superstructure is 37 by 51 ft in area and about 18 ft high. Twelve concrete columns support the roof slab as well as the four walls. The walls are constructed of hollow concrete blocks and are reinforced with vertical steel rods embedded in concrete fill and by horizontal layers of steel wire. The masonry walls are anchored to the concrete floor with dowels and to the horizontal bond beams near the roof, they are also keyed into the sides of the concrete columns. The roof slab is cast integrally with a series of beams and girders. Two openings in the roof, each 9 by 9 ft in size, are located above the ECCS pumps and are covered with metal hatch covers. 21

The 181-N main structure and ECCS low lift pump house have been classified as Seismic Category I. (See section 3.2, Table 3.2-1.) 21

The 181-N building was designed initially using conservative design parameters and a horizontal force factor of 0.1 g. The design was reviewed shortly thereafter and determined to meet the earthquake requirements specified in the Uniform Building Code for seismic Zone 3. Analyses performed in 1969 confirmed that the 181-N building would also withstand a more severe

earthquake in which the maximum horizontal ground acceleration was 0.25 g (Ref. 37). Although margins of safety against overturning and sliding in this case are less, they are adequately safe as indicated in the table below.

UBC Zone	Horizontal Force Factor	Margin of Safety Against Sliding		Margin of Safety Against Overturning	
		*375	415	375	415
	0	2.86	4.34	4.65	2.60
2	0.10	2.13	2.26	4.00	2.30
3	0.20	1.70	1.53	3.51	2.05
	0.25	1.54	1.31	3.31	1.95

* River Water Elevation ft MSL

Earthquake requirements, and not wind, controlled the design of the 181-N pump house, except for the guard tower. The effect of tornado loads on the superstructure is discussed in Section 3.3.2.

3.8.4.1.1 Seismic Reanalysis of 181-N

In 1987, in response to issues 12 and 72 of the ASEP (Ref. 61), an evaluation was made (Ref. 62) of the reinforced concrete frame and concrete block panels and superstructure of the 181-N building for the effect of the 0.25 g SSE defined in Reference 50. The evaluation was limited to the block wall superstructure and was based on analysis, performed by Kaiser Engineers Hanford (KEH) of the masonry walls enclosing the diesel pumps of the ECCS. The seismic loading was determined by the static equivalent method and the ultimate strength methods of ACI-318 were utilized to determine the capacity of the masonry walls. Z1

Based on the results of the evaluations it was concluded that the masonry walls enclosing the two diesel ECCS pumps have adequate capacity to withstand the SSE. The maximum stresses in reinforcing bars and concrete masonry for the most severe combination of seismic (or tornado) and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 75% of the ultimate strength of concrete masonry, and thus complies with the structural acceptance criteria contained in SDC 4.1 (Ref. 50). The design margin is 20%.

These results, however, were determined to be insufficient based on separate, independent reviews performed by the Westinghouse Hanford Independent Safety Assessment group (Ref. 52) and the ECCS III/I task group (Ref. 63). Accordingly, a dynamic finite element, seismic analysis was performed in 1988 of the entire 181-N complex, substructure, superstructure, Annex, gantry crane, and observation tower (Ref. 64).

The results of this more comprehensive analysis verified the initial conclusion that the 181-N building could safely withstand the SSE. Details of the evaluation are summarized below.

The Building 181-N structure was evaluated to verify conformance to Hanford Standard Architectural-Civil Design Criteria SDC 4.1, Rev. 9 (Ref. 50). Loads due to the SSE were combined with dead and live loads. Detailed evaluation was performed for those structural members that were the most highly loaded in comparison to their capacities. 21

Results are expressed in terms of factors of safety. The factor of safety for a structural member is defined as the ratio of its allowable stress or load capacity to the applied stress or load. A factor of safety greater than unity generally signifies that the member satisfies SDC 4.1 acceptance criteria, while a value less than unity implies that it does not.

The following structural and foundation components were evaluated:

1. Reinforced concrete members
2. Reinforced masonry
3. Structure stability
4. Soil-bearing pressure.

The applicable SDC 4.1 load combinations and acceptance criteria are:

1. Reinforced Concrete

   - Concrete compressive strength, f_c' = 3,000 psi, minimum specified value at 28 days
   - Reinforcement minimum yield strength, f_y = 40,000 psi.

   Applicable load combinations for reinforced concrete are the following:

   U = 1.1 (D + L) + 1.0 SSE
   U = 0.9 D + 1.0 SSE

   U = Dead Load
   L = Live Load
   SSE = Safe shutdown earthquake

2. Reinforced Masonry

   The superstructure block walls were constructed of hollow concrete units with vertical reinforcement bars embedded in concrete fill and layers of steel mesh for horizontal joint reinforcement. The original construction specifications included:

   - Masonry units for exterior walls were specified to conform to ASTM C90-52, Grade A. This provided a 2,000 psi minimum compressive strength.
   - Mortar was specified to comply with ASTM C270-54T, Type M. This provided 1,350 psi compressive strength.
   - Reinforcing bars were specified to conform to Federal Specification QQ-S-632, Type 11, Grade C, D, E, or G. This provided minimum yield strength of 40,000 psi.

Load combinations and allowable stresses for masonry walls were in accordance with Nuclear Regulatory Commission Standard Review Plan (SRP) Section 3.8.4.

3. Structure Stability and Soil Bearing Pressure

The following load combinations were used based on NRC foundation requirements for SRP 3.8.5.

$D + H + E'$
$D + H + W_t$

H = Lateral earth pressure

21

Minimum factors of safety of 1.1 are required against sliding and overturning. The allowable soil-bearing pressure for Building 181-N is 12 ksf per Section 3.8.4.1.

Seismic Loads

The total loads or stresses were compared with allowable values permitted by the acceptance criteria. Seismic responses with the control point at grade level Elevation 454 ft were generated.

Floor response spectra were generated from a single time history analysis. Ground motion input to this analysis consisted of uncorrelated time histories in the three global directions applied simultaneously.

This approach gives seismic responses equivalent to those attained by the square root of the sum of the squares (SRSS). Simultaneous application of SRSS loads on a structural member whose capacity is a function of more than one load component can be too conservative.

As an alternative, when evaluating structural members subjected to load interactions, 100% of the maximum response due to one ground motion component was combined with 40% of the maximum responses due to the

other two ground motion components. The 100-40-40 approach is simpler to apply and still conservative.

Results of the dynamic analysis are summarized below:

Reinforced Concrete Members

The following reinforced concrete members were evaluated:

- o Superstructure roof beam along Column Line I
- o Superstructure Column B-1
- o Superstructure roof diaphragm
- o East substructure wall.

The governing loads, ultimate strength capacities, and factors of safety are summarized in Table 5.2-1.

The roof beam calculated negative bending moment capacity and factor of safety were 97.9 k-ft and 1.5, respectively, by conservatively including only the bottom slab reinforcement within an effective width of four times the slab thickness.

The critical section for roof beam shear is at the interior column. A factor of safety of 2.5 was calculated for beam shear.

For the most highly loaded superstructure column a moment capacity and factor of safety of 112 k-ft and 1.15, respectively, were calculated. An ultimate shear strength of 41.9 kips and factor of safety of 1.14 were calculated.

The governing factor of safety was conservatively determined to be 1.9 for the superstructure roof slab.

The east wall of the substructure is the most heavily loaded concrete shear wall. In-plane shear on the east wall has a conservative factor of safety of 1.14.

The east wall is subjected to concurrent in-plane and out-of-plane bending moments. Conservatively including only one flange at the tension end of the east wall resulted in an in-plane moment capacity and factor of safety 517,000 k-ft and 1.8, respectively.

Under out-of-plane loading, the east substructure wall spans horizontally between the E-W guide walls. Out-of-plane loadings include the following:

- o Static earth pressure
- o Dynamic earth pressure
- o Wall lateral inertia
- o Hydrodynamic pressure from water inside the structure
- o Hydrodynamic pressure from water outside the structure.

21

These loads vary along the height of the wall. There is no net load due to hydrostatic pressure since it acts on both faces of the wall.

Out-of-plane bending of the east wall was checked at El. 405 ft -0 in. and El. 360 ft -0 in. The section at El. 405 ft -0 in. was found to govern, with a bending moment capacity at the transverse support and factor of safety of 8.1 k-ft/ft and 1.6, respectively.

Block Wall Panels

The reinforced concrete frames of the superstructure are infilled by block wall panels. These panels are constructed of 12-in.-thick hollow concrete masonry units reinforced with bars in the vertical direction and truss-type reinforcement in all horizontal joints. No positive mechanical attachment connects the panels to the superstructure columns at their vertical mortar joint interface. A horizontal bond beam is provided at the top of each panel, consisting of two courses of concrete-filled blocks reinforced by a total of four #5 bars. This reinforcement is lap spliced to dowels from the concrete columns.

The loading on a block wall panel consists of the following:

- o Vertical load due to dead weight and vertical seismic response of the panel
- o In-plane loads, due primarily to horizontal seismic loads from the superstructure roof and panels
- o Out-of-plane loads, due primarily to out-of-plane seismic response of the panel.

The block wall panels and bond beams at the south, west, and east sides of the superstructure were evaluated. The south panels have higher out-of-plane loads, while the west and east panels have higher in-plant loads. Resulting stresses and factors of safety are listed in Table 5.2-2 and, as shown, indicate overstress conditions for the west and east walls.

Further evaluation of the 181-N block wall masonry was performed using alternative acceptance criteria to that of SDC 4.1, Rev. 9 (Ref. 50), but still conservative and acceptable. Based on these criteria, which involved a change in the inelastic ductility demand factor (F_u) value from 1.24 to 1.4, it was concluded that the 181-N masonry walls are in conformance with the UBC, and are adequately reinforced, and attached to the main structure to provide excellent seismic resistance. An inelastic ductility to demand factor (F_u) value of 1.4 for the 181-N masonry walls in flexure is within the range of applicable recommended values for seismic acceptance criteria and can be obtained by applying the recommended 20% increase allowed for existing structures (F_u = 1.20 X 1.25 = 1.5 > 1.4). Further, it is concluded from the reference documentation that masonry walls of this quality would have reserve capacity with F_u values up to 2.0. An inelastic demand to capacity factor F_u = 1.4, therefore, is safe acceptance criteria for the SSE in-plane bending demands on the 181-N reinforced masonry wall panels.

The east-west wall panels can adequately withstand the SSE, and based on the results of the seismic analyses using this acceptance criteria will maintain a seismic factor of safety of 1.0 or greater under all conditions.

Structure Stability and Soil Bearing Pressure

Loadings applicable to evaluation of structure stability and soil bearing pressure include the following:

- o Structure weight
- o Buoyancy effects
- o Weight of additional soil supported on the mat foundation
- o Structure and supported soil seismic loads.
- o Static earth pressure
- o Dynamic earth pressure
- o Hydrodynamic pressure.

Resistance against sliding and overturning is due to the weight of the structure and soil that bears on the mat foundation, less buoyancy effects. In calculating the buoyancy effect, water was assumed to be at the normal water level, El. 390 ft -0 in.

Lateral loading towards the west (river) direction governs due to the earth pressures. Structure seismic loads were obtained directly from dynamic analysis. Inertial loads from the soil supported on the mat foundation were included by conservatively assuming the soil is rigidly attached to the substructure. Should the structure begin to slide or overturn, the resulting displacements will be sufficient to mobilize the active state in the backfill against the east substructure wall. Static earth pressures were based on the active pressure coefficient in SDC 4.1 The Monoobe-Okabe method was used to determine dynamic earth pressures. Hydrodynamic pressures from water outside the structure were determined as described for the east substructure wall.

A shear key along the east edge of the mat foundation is embedded in the Ringold formation. This key causes the sliding plane to form in the soil through the horizontal plane at the bottom of the key. An effective coefficient of friction of 0.7 was assumed. This corresponds to a soil angle of internal friction of 35 degrees, which is a typical value for most soils and should be conservative for the very stiff Ringold formation. A factor of safety of sliding of 1.5 was calculated. This is very conservative since lateral inertia of the soil on the mat foundation was included. The conservative factor of safety is still greater than the minimum allowable value of 1.1.

Overturning about the west edge of the mat foundation was investigated. A conservative factor of safety of 3.4 was determined. This is greater than the minimum allowable of 1.1. The net soil-bearing pressure was calculated to be 6.0 ksf. The factor of safety against the allowable pressure is 2.0.

3.8.4.2 181-N Gantry Crane - Original Design Description

The 181-N gantry crane is an outdoor facility, spanning the operating floor and superstructure of the river pump house. It is not directly required for safe operation of the plant; however, if it were to tip over during an earthquake, it might have adverse effects on continuity of operation during the emergency. For this reason, the gantry crane is classified as seismic Category I in Section 3.2. Design features, provided to prevent tipping, are described below.

The gantry crane is designed to raise, lower, hold or transport any hook load up to 125% of its rated capacity. The crane has two hoists: the main hoist is rated at 60 tons and the auxiliary hoist at 10 tons. The span between tracks is 104 ft and the distance between end-stops is about 120 ft.

Limit switches are installed which automatically interrupt power and apply brakes when the gantry bumpers approach within 20 ft of any one of the four end-stops. In addition to the brakes on the gantry drive and load

mechanisms, each side of the gantry crane has a rail clamp which automatically grips the rails when the crane is stopped. Additional information and crane dimensions are presented in Figure 3.8-20.

The crane was designed and constructed by the Pacific Coast Engineering Company of Alameda, California in accordance with standards issued by the Electric Overhead Crane Institute and the following design criteria:

1. A wind load of 20 lb/ft^2 on 1 1/2 times the projected area
2. Earthquake loads in accordance with Uniform Building Code for seismic Zone 2
3. Wind or earthquake forces acting either parallel or normal to direction of crane travel (However, wind and earthquake were not considered to act at the same time)
4. Lateral or longitudinal thrusts equal to 10% of the respective gantry crane reactions
5. Without rail clamps acting, a stability moment of at least 150% of the overturning moment due to earthquake, or wind and other forces.

The following loads and load combinations were utilized to compute maximum stresses:

1. Dead load plus weight of fully-loaded trolley, plus impact (20%), plus seismic or wind load
2. Dead load plus weight of fully-loaded trolley, plus lateral or longitudinal thrust (10%), plus seismic or wind load
3. Rated breakdown torque of motors
4. Dead load plus trolley, plus seismic or wind load, with crane not in operation
5. Collision with rail stops at maximum bridge speed.

For load conditions 1, 2, and 3, maximum stresses were limited to normal values specified by the Electric Overhead Crane Institute (Ref. 38) which are about 80% of design values permitted by the American Institute of Steel Construction for normal building design (Ref. 6). For loadings 4 and 5,

basic stresses could be increased one-third; however, for loading 4, stresses were kept below 90% of the material's elastic limit.

The following load data, corresponding to the most severe design condition, are of interest (Ref. 39):

Total dead weight of gantry crane	270,695 lb
Maximum reaction on one leg (two wheels)	164,750 lb

The latter item is made up of the following loads:

Dead load of crane	58,750 lb
Dead load of trolley and machinery	16,100
Live load	58,500
Live load impact	11,700
Longitudinal wind (20 lb/ft^2)	19,700
	164,750 lb

For the design of the gantry crane, wind effects (19,700 lb) are more severe than those corresponding to the earthquake (8,700 lb) associated with UBC Zone 2. On the other hand, earthquakes producing horizontal ground accelerations of 0.12 gravity and/or 0.25 gravity, would produce corresponding reactions of 24,400 lb and 50,800 lb, respectively. Substituting these values for the 19,700 lb wind reaction gives total loads of 169,450 lb and 195,850 lb, respectively, for the maximum reaction force on one leg of the gantry crane. These latter values represent increases of three and 19%, respectively, greater than the original design load of 164,750 lb. The 19% value is less than the 33 1/3% increase in design stress usually permitted for SSE conditions.

If the gantry crane responds essentially as a rigid structure, when subjected to earthquake movements in the east-west direction, it will have a safety factor of about 1.56 against overturning. If maximum vertical accelerations occur simultaneously with the maximum horizontal ground acceleration of 0.25 g, the safety factor becomes 1.3. These margins of safety would be

somewhat higher if the gantry crane is loaded, since any hook load would lower the center-of-gravity of the system.

Rail clamping forces were not included when computing the above safety factors against overturning. Each rail clamp will provide an additional resisting force of at least 25,000 lb. The clamps are located near the west side of the crane so that clamping reactions will increase the resisting moment 9% for tipping tendency to the west and 30% for tipping tendency to the east. Thus, if the crane should tend to overturn during a severe earthquake, it would tend to overturn towards the west and away from the 181-N superstructure. o 21

The effect of tornado loads on the gantry crane is discussed in Section 3.3.2.

3.8.4.2.1 Seismic Reanalysis of 181-N Building Gantry Crane

The 181-N building gantry crane was evaluated (Ref. 64) for the SDC 4.1, Rev. 9 (Ref. 50) 0.25 g SSE. Acceptance criteria are given in Reference 50. Current dynamic seismic analyses methods summarized in the preceding Section 3.8.4.1.2 were used and are described in Reference 64.

To obtain response of the gantry crane, equivalent single degree of freedom models were attached to the operating floor in the dynamic model. Properties of the equivalent models were selected to match the fundamental mode responses of the gantry crane in the horizontal directions. Fundamental frequencies of 2.6 Hz and 3.7 Hz in the N-S and E-W directions were determined from a more detailed model of the gantry crane. Properties of this model were derived from a previous analysis of the gantry crane (Ref. 65). The current gantry crane model contains refinement over the previous model to better represent the actual flexibility.

Gantry Crane Evaluation

The gantry crane is not a safety related component. Evaluation of the gantry crane was performed to verify if it could impact safety related components mounted on the operating floor. An approximate comparison based on scaling of design loads reported in Section 3.8.4.2 indicated that member stresses

should not exceed allowable values. Therefore, detailed stress analysis of the gantry crane was not performed.

Overturning of the gantry crane was evaluated. Acceptable performance was defined as a calculated factor of safety against overturning greater than 1.1 similar to overall structure stability.

Hold-down restraints for the gantry crane were at least partially installed. Walkdown determined that some of the anchors for the restraints are missing and that the restraints do not extend along the entire length of the crane rail. The hold-downs are therefore considered ineffective in preventing gantry crane uplift.

For the seismic evaluation, SSE loads were obtained from the dynamic analysis used for the 181-N structures.

The following factors of safety against gantry crane overturning were calculated:

Seismic - F.S. = 1.8

This factor of safety is greater than the required value of 1.1.

3.8.4.2.2 Seismic Analysis of 181-N Annex Building

The 181-N Annex is a structural steel frame. The siding and roofing are sheet metal panels. It stands on the 181-N river pump house operating floor attached to the southwest corner of the 181-N superstructure. The annex is about 9 ft wide, 40 ft long, and 17 ft high. There is an interior mezzanine 8.5 ft above the operating floor.

The annex houses a direct-drive diesel combustion engine used for an emergency cooling water pump. The mezzanine supports 2 compressed air tanks and a lubricating oil cooler. The exhaust system, intake air filter, and a ventilator for the diesel are mounted on the annex roof.

The 181-N Annex was analyzed and evaluated for the SSE motions using the 0.25 g SSE, and the structural acceptance criteria specified in SDC 4.1, Rev. 10 (Ref. 51). The objective was to determine if the structure could safely resist the SSE.

The design drawings specify that the 181-N Annex structure is designed for 0.25 g horizontal seismic loads. However, it was determined in Reference 64 that the SSE motions at the base of the annex would exceed 0.25 g due to the dynamic response of the 181-N substructure (Ref. 64). Therefore, seismic analyses of the annex were also necessary.

The 181-N river pump house seismic analyses included the development of SSE response spectra for the operating floor. These spectra, representing the potential N-S and E-W SSE input for the 181-N Annex, are included in Reference 64. Response spectra analyses for the annex using operating floor N-S and E-W response spectra were conducted. The analyses results were evaluated. 21

The results of the SSE analyses of the 181-N Annex Building (Ref. 66) show that the stresses in structural frame, and all connections, including attachments to reinforced concrete members and masonry panels in the 181-N superstructure are in conformance with SDC 4.1 criteria. The column base plate concrete anchor bolt loads are within the manufacturers recommended allowable. The 181-N Annex Building could safely resist the 0.25 g SSE.

3.8.4.2.3 Seismic Analysis of 181-N Observation Post

The 181-N raised observation post is a structural steel (mostly pipe sections) space frame tower 50 ft tall with a cab on top. The tower is 8 by 17 ft in plan. The observation post, which stands on the 181-N river pump house substructure, was previously analyzed for the 0.25 g free field SSE ground motions (Ref. 67). The subsequent SSE analyses for the 181-N river pump house (Ref. 64) shows the resulting operating floor motions at the base of the observation post tower significantly exceed the SSE free field motions, due to the response of the substructure. Therefore, it was necessary to reanalyze

the observation post for the SSE motions at the 181-N operating floor, as determined from the results of the Reference 64 analyses.

The seismic and tornado evaluation of the observation post documented in Reference 67 was reviewed. The mathematical model of the tower structure is adequate and was used for this SSE reanalysis. The identical model was constructed for ANSYS-PC/LINEAR (Ref. 68). Dead load and modal response spectrum analyses for the SSE were conducted for the structure.

Response spectra for the 181-N operating floor for the SSE based on soil structure interaction analysis control points at elevations 421 ft and 454 ft are documented in Reference 64. The SRSS combination of response spectra peak accelerations for the 421 ft and 454 ft elevation control points are 0.64 g and 0.62 g, respectively. The response spectrum for the 421 ft elevation control point was used for the modal analysis of the tower.

The dead load analysis results for the 181-N observation post (Ref. 69) show excellent agreement with the comparable analysis reported in Reference 67.

The 181-N tower structure member stresses due to the SRSS combination of SSE loads are within the allowables specified by SDC 4.1 (Ref. 51).

The tower SSE reaction loads on the reinforced concrete haunch supporting the tower on the operating floor are within acceptable limits. These SSE loads are lower than the 175 mph DBT reaction loads reported in Reference 67.

Once the 181-N raised observation post had been put into service, the occupants reported noticeable and uncomfortable tower sway caused by personnel climbing the stairs and walking in the cab. The vertical pipe legs were filled with grout to stiffen the tower and reduce the sway.

As a result, two additional structural questions developed. One, it was necessary to determine whether the added weight would overload the 181-N reinforced concrete substructure, i.e., the tower support haunches. Secondly, an evaluation for potential effects to the seismic response or resistance of

the tower was needed. It was determined that the grout in the tower legs increased the observation post total weight by 16% (maximum). It was found that the tower support haunches on the 181-N substructure remain within their structural load capacities.

The increased tower stiffness and weight potentially change its structural dynamic characteristics, and therefore, the seismic response or resistance. The fundamental natural frequency (N-S) for the tower with grouted legs is estimated as 3 H_Z. The calculated fundamental frequency for the original tower is 1.9 H_Z. From the N-S 181-N operating floor response spectra for control point elevation 454 ft, this change in the fundamental frequency will have minimal effect on the tower seismic (SSE) response or resistance. 21

It was concluded the 181-N raised observation post can safely resist 0.25 g SSE motions, with the structural stresses remaining below SDC 4.1 limits.

3.8.4.3 Demineralized Water Storage Tank and ECCS Bypass

During periods of normal reactor operation, the demineralized water storage tank is filled with 900,000 gal of demineralized water of which 452,000 gal are stored for ECCS use only. Each time the reactor is started up or shut down, the low-lift pumps at the 181-N river pump house begin pumping raw water to the standpipe of the ECCS bypass (silo), which is located just north of the demineralized water tank. If the level of water in the tank drops to within about 4.5 ft above the tank bottom, raw water will be delivered automatically to the high-lift pump suction header.

In general terms, the ECCS bypass is a liquid flow device which provides for the instant and automatic substitution of an emergency supply of cooling water for the normal supply when the normal supply is inadequate, while preventing mixing of the two liquids during normal operation (Ref. 40). The two supplies of liquid are connected by the air tank and an inverted U-tube, which extends above the highest level of water that can be attained in the ECCS and the demineralized water tank. An air pocket at the top of the inverted U-tube prevents mixing of raw water with treated water contained in the ECCS during normal operation. When the supply of treated water is almost

exhausted, the air pocket is vented to atmosphere, causing the emergency supply from the low-lift river pumps to flow into the ECCS system and to the high-lift pumps. In addition to the major piping components described herein, there are a number of vent pipes and a small drain line. These items automatically control water elevations in components of the ECCS bypass during normal periods of operation. The ECCS bypass is automatic and fail-safe in operation and requires no mechanical or electric devices in order to function. In other words, the ECCS bypass is a large valve with no moving parts, capable of automatically controlling the flow direction of emergency cooling water, depending only upon the water level in the ECCS. Construction details are shown on Figure 3.8-21.

The ECCS bypass at N Reactor includes the following major components: A concrete standpipe, two supply risers, outlet piping consisting of an air tank and an inverted U-tube, two large vent pipes and an overflow pipe leading back to the river. The standpipe is a reinforced concrete cylinder, 13 ft outside and 11 ft inside diameter. It rises 49 ft above ground level and is supported on a circular concrete foundation, 26 ft in diameter and 5 ft thick. The bottom of the slab is 14.5 ft below the ground surface, which in this case consists of a reinforced concrete slab, 6 in. thick. Although the major function of the standpipe, or "silo" as it is sometimes called, is to contain water, it also supports the air tank and system piping, most of which is above ground level. The top of the standpipe is covered with a steel plate roof, which is removable. Safety ladders are provided on the inside and outside walls to permit inspection of internals during EMS testing.

Earthquake requirements, rather than wind, controlled the lateral load design of the bypass structure. The original design was based upon an earthquake having a maximum horizontal ground acceleration of 0.2 g and a vertical acceleration of 0.15 g. Corresponding to this horizontal ground acceleration, internal damping of 2% of critical and a natural period of 0.14 seconds, the horizontal force factor was determined to be 0.375 g. This factor was used to compute shears and overturning moments at various levels along the structure. Fifteen percent of the total seismic load was applied at the top

of the structure and the remainder applied uniformly along its height. Stresses in the reinforcing steel were kept below 18,000 psi in order to assure a watertight structure. This is conservative because intermediate grade reinforcing bar was used in construction, which may otherwise be designed for 20,000 psi. The construction specifications called for 3,000 psi concrete (f'_c at 28 days); however, concrete placed in the foundation tested above 4,000 psi and that placed in the cylindrical walls tested greater than 5,000 psi in compression at 28 days old. Internal supports for the air tank and for the inlet and outlet piping were designed for a horizontal force factor of 0.5 g.

Although each of the water storage tanks in the vicinity of the standpipe may be called upon to supply reactor cooling water to the high-lift pumps during an emergency, only the demineralized water tank and the ERW tank are classified as Seismic Category I. (See Section 3.2.1.) Design parameters for the four tanks are presented in Table 3.8-8. The ERW tank was originally designed for a horizontal force factor of 0.5 g; the other three tanks were designed for 0.2 g. The demineralized water storage tank will survive the (0.25 g) SSE if the water level does not exceed 27.5 ft. The ERW tank currently functions as an equalizing reservoir for the fog spray pumps.

3.8.4.3.1 Seismic Reanalysis of ECCS Bypass (SILO)

Following fuel power operation, the seismic adequacy of the bypass structure was first evaluated in 1969 assuming that (1) the earthquake would produce maximum horizontal ground accelerations of 0.25 g and (2) design conditions would correspond to what are currently referred to as safe shutdown earthquake (SSE) conditions (Ref. 41). In this study, the horizontal force factor was determined to be 0.325 g and the vertical force factor 0.188 g since the damping would be about 7% of critical under SSE conditions. The investigation showed that the cylindrical structure, when filled with water, would respond essentially as a cantilever supported at its base, and that the higher modes of vibration did not contribute significantly to its dynamic behavior. Assuming that the concrete standpipe was fixed at ground level (452 ft MSL), the natural period would be 0.12 seconds. Whereas, if the structure were to vibrate about its foundation at 433.5 ft MSL, then its

natural period would be 0.174 seconds. In the frequency calculations, the modulus of elasticity of concrete was 4 million psi, corresponding to 5,000 psi concrete.

Calculation methods defined in ACI-505-54 (Ref. 42) were used to determine shears and moments along the vertical cylinder. As in the original design calculations, 15% of the total lateral load was applied at the top of the standpipe. However, the remaining load was distributed in a triangular manner starting with zero at the base. Although the triangular distribution proved more severe than a rectangular one by a factor of about 1.2, maximum stresses were less than about 1,000 psi (compression) in the concrete and 21,000 (tension) in the reinforcing steel. The latter value is but slightly greater than the maximum design stress (20,000 psi) permitted under normal conditions; whereas, maximum stresses under SSE conditions may approach the yield strength. Allowing for the passive soil resistance on vertical sides of the buried structure, the maximum soil pressure acting on the foundation is about 7,700 lb/ft^2, and the safety factor against overturning of the standpipe and its foundation is 1.89. These values are conservative when compared with the maximum allowable soil pressure (12,000 lb/ft^2) for this site and a desired safety factor against overturning greater than about 1.5 under normal loading conditions. Thus, the re-evaluation demonstrated that the original design calculations were conservative and that the bypass structure would be able to survive the 0.25 g earthquake with substantial strength to spare.

The ECCS bypass structure (SILO) was evaluated in 1987 for seismic response as a result of Issue 74 of the ASEP (Ref. 61) - "Review the SILO near the 182-N Building for strength and deformation of the pipe connection." The evaluation was performed by Kaiser Engineers Hanford (Ref. 70).

A three dimensional lumped mass computer model of the SILO was prepared. The analysis was performed using the general purpose finite element computer program - ANSYS developed and maintained by Swanson Analysis Systems, Inc. (Ref. 68). The elements used in the model consisted of spring elements to represent the rigid circular footing resting on elastic half-space, elastic

straight pipe elements to represent the reinforced concrete cylinder, and large diameter welded steel pipe and elastic curved pipe elements to represent elbows and bends in the welded steel pipe. The mass of the structure including the weight of the water was lumped at appropriate node points.

The results of the analysis yielded fundamental natural frequencies of the structure and significant participating modes. The response of the structure to seismic excitation was obtained by subjecting the model to the SDC 4.1 design ground response spectra for 7% damping. The modal response to the vertical and two horizontal excitations was combined by two methods; the Square Root of the Sum of the Squares (SRSS) method and the Complete Quadratic Combination (CQC) method. The CQC method is similar to the SRSS method in that a square root of the sum of the squares of the modal response terms is calculated. This produces more accurate results for systems with nonclassical modes of vibration or where two or more natural frequencies of vibration are approximately equal such as is the case with the SILO. 21

The model used in the SRSS analysis was based on empty overflow lines from the SILO to the river and from the SILO to the ERW tank. The model used in the CQC analysis was modified to include full overflow lines and actual connection details to the ERW tank.

A static analysis of the SILO was performed by subjecting the model to a vertical acceleration of 1.0 g to represent the dead load (DL) of the structure. The results of the DL analysis were combined algebraically with the dynamic values to obtain the worst case forces, displacements, and stresses at critical locations in the structure.

Based on the results of the analysis, it was concluded that the cylindrical walls are adequately reinforced to withstand the safe shutdown earthquake. The maximum stresses in the reinforcing bars and concrete for the most severe combination of seismic and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 85% of the ultimate strength of concrete, which is in accordance with the structural acceptance criteria contained in the SRP and SDC 4.1 (Ref. 50).

Overturning of the SILO under vertical and lateral seismic loading is below allowable limits and the SILO will not slide.

Stresses in the large diameter steel piping interior and exterior to the SILO are within allowable limits.

Soil pressure under the toe of the foundation during the SSE event is 15,150 psf. The ultimate soil-bearing pressure calculated on page 48C in Appendix A is 49,700 psf. Thus the soil pressure during the SSE exhibits a factor of safety of 3.28 (49700/15150) and is considered acceptable.

The analysis showed that four pipe supports at the SILO were overloaded. The support assemblies consist of anchor plates (anchors) which are held to the surface of the SILO by embedded rebar (#7 & #13) or Phillips Red Head Bolts (#19 & #20), and pipe or angle structures (supports) which are welded to the anchor plates and which provide the surface for support of the piping. The overloaded supports are listed below.

#7 SILO to 36 in. overflow
#13 SILO to 24 in. vent
#19 30 in. fog spray to ground
#20 30 in. fog spray to ground

A program to seismically upgrade these supports was initiated and completed in 1988 (Ref. 71). Therefore, the ECCS bypass SILO is seismically qualified in all respects.

3.8.4.3.2 Seismic Reanalysis of Demineralized Water Storage Tank

The Demineralized Water Tank was evaluated in 1987 for compliance to the SSE. The evaluation was based on seismic analyses prepared by Kaiser Engineers Hanford (KEH) and satisfies in part Issue 73 of the N Reactor Accelerated Safety Enhancement Program Plan (Ref. 61).

The Demineralized Water Tank is a 1,000,000 gallon capacity tank 75 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the Demineralized Water Tank is to:

- Provide a supply of warm demineralized water for ECCS use
- Provide a normal supply of demineralized water to the high and low pressure injection pumps
- Provide a reservoir for effluent water from the demineralizer plant.

In the event of an ECCS actuation caused by an RCS pipe break, the low-lift pumps and the high lift pumps would start. The Demineralized Water Tank is connected to the suction of the high-lift pumps and supplies the initial supply of warm water to the reactor. Once the supply of demineralized water is exhausted, the low-lift pumps supply river water to the high-lift pumps through raw water bypass silo. 21

The dynamic response caused by hydrodynamic pressure due to horizontal and vertical seismic loadings were determined.

In summary the KEH analysis showed the following (Ref. 73):

o Shell stress 1 ft from bottom plate	22.15 ksi
o Shell stress 8 ft from bottom plate	17.99 ksi
o Maximum bottom plate stress (secondary)	82.08 ksi
o Stress required for buckling	6.06 ksi
o Calculated buckling stress	1.96 ksi
o Sliding force	2,930 kips
o Force to resist sliding	2,730 kips
o Tank thickness to resist missile	0.236 in.
o Tank thickness	0.250 in.

The tank material is ASME SA-283 Grade C with a specified minimum yield stress of 30 ksi. The analysis shows that the stress in the tank shell is substantially below 90% of yield for the SSE.

The 82 ksi stress in the bottom plate is a secondary bending stress and is calculated on an elastic basis. In a ductile material yielding will occur and the stress self relieves. The few cycles of this resulting from the SSE should not cause a failure.

The KEH analysis shows that the tank will slide during the SSE. This is based on a coefficient of friction of 0.54. As discussed in the report, friction cannot be counted on to prevent sliding during a seismic event. Restraints should be installed to prevent sliding. These restraints will also reduce the secondary bending stress in the bottom plate.

As a result of this evaluation restraints were installed to prevent sliding (Ref. 74) of the Demineralized Water Tank in the event of the SSE. Reinforced concrete buttresses were installed around the tank. The buttresses were tied into the concrete foundation and into the tank. The buttresses were designed to restrain the tank from movement without any credit for friction between the bottom of the tank and the soil.

3.8.4.3.3 Seismic Reanalysis of Emergency Raw Water Tank

An evaluation of the ERW tank was performed in 1987 (Ref. 75) in response to ASEP Issue 73 (Ref. 61). Prior to this evaluation, the tank was analyzed in UNI-3133 Rev. 1, "N Reactor Balance of Plant Evaluation," Task 7 - Water Storage Tanks, M. R. Lindquist, dated January 1985. This analysis examined the Emergency Raw Water Tanks response to a 0.20 g earthquake. The analysis was part of a program related to plant life extension. The seismic analysis calculated tank shell stresses which are in essential agreement with the 1987 analysis and which show that the tank shell stress is below 90% of yield. The analysis did not evaluate sliding.

The Emergency Raw Water Tank is a 200,000 gal capacity tank 35 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the Emergency Raw Water Tank is to provide an auxiliary source of water for the for spray pumps suction and to store raw river water prior to conditioning for plant use.

In the event of a primary system pipe break, the fog spray pumps will start. The Emergency Raw Water Tank is connected to the suction of the fog spray pumps and supplies the initial supply of water until the 181-N Building low lift pumps are in operation.

The ERW tank is required for the safe shutdown of the N Reactor. 21

The dynamic response of the ERW tank caused by hydrodynamic pressure due to horizontal and vertical seismic loads produced by the design basis SSE (Ref. 50) were determined. The current operating level for the tank is 17 to 20 ft and a level of 20 ft was used in the analysis.

In summary the KEH analysis showed the following:

o Shell stress 10 in. from bottom plate	18.79 ksi
o Shell stress 7 ft from bottom plate	10.86 ksi
o Maximum bottom plate stress (secondary)	29.58 ksi
o Stress required for buckling	6.14 ksi
o Calculated buckling stress	2.77 ksi
o Sliding force	861 kips
o Force to resist sliding	440 kips
o Tank thickness to resist missile	0.236 in.
o Minimum tank thickness	0.250 in.

The tank material is ASME SA-283 Grade C with a specified minimum yield stress of 30 ksi. Based on the analysis, it was concluded that the stress in the tank shell is substantially below 90% for the SSE.

The 29.58 ksi calculated stress in the bottom of the tank is just below yield, is a secondary bending stress, and is calculated on an elastic basis. In a ductile material, yielding will occur and the stress self relieves. The few cycles of this resulting from the SSE should not cause a failure.

The analysis also showed the sliding force exceeds the force which resists sliding and it is concluded both from this analysis and from previous history of large unrestrained tanks during seismic events that the tank would move during the SSE.

Movement of the ERW tank could result in damage to the tank discharge line on the Emergency Core Cooling System (ECCS) side of the RWV-801-1R isolation valve. This could impact the availability of the ECCS. Tank movement also has the potential to cause the failure of the line from the tank to the suction of the fog spray pumps. Lack of suction to the fog spray pumps would negate the ability of the system to reduce Zone 1 pressure when required.

As a result of this evaluation, a program to seismically qualify the ERW tank for the SSE was implemented (Ref. 76). Reinforced concrete buttresses were installed around the tank. The buttresses were tied into the concrete foundation and into the tank. The buttresses were designed to restrain the tank from movement without any credit for friction between the bottom of the tank and the soil.

3.8.4.4 High-Lift Pump Building (182-N) - Original Design

Certain components of the reactor emergency core cooling system (ECCS) are located in the basement of the 182-N building, as well as components of the fog-spray and auxiliary pumping systems. Included are the diesel drive units, three high-lift pumps, two fog-spray pumps and associated piping. The ground floor of the building, which is also the ceiling over the basement, consists of a concrete slab supported by a series of integral beams and girders. Basement walls and the columns which support the main operating floor are constructed of reinforced concrete. In addition, there are a number of structural steel columns which provide independent support of

piping systems located in the basement. Architectural and structural details of the 182-N building are shown on Figure 3.8-22 and 3.8-23.

The walls of the superstructure, which are 12 in. thick and about 21 ft high, are made of hollow concrete blocks, anchored to a structural framework consisting of wide-flange columns, horizontal girts and diagonal bracing. A bond beam, constructed of reinforced concrete, ties the top surfaces of the four walls into the steel framework. In addition, a series of bolts and square washers connect the block walls directly to peripheral wall columns. Individual blocks are reinforced with vertical steel bars placed at about 32-in. centers and horizontal steel mesh located between each row of blocks. The built-up roofing is supported on a series of precast roof planks which span between the steel roof beams. The 182-N building, including the basement walls, the superstructure and the roof, was designed for a seismic factor of 0.2 g.

The respective diesel engines, drive units and pumps (for both the ECCS and the fog-spray systems) are supported on individual mat foundations which are isolated from the 182-N building structure. The foundations, equipment, diesel engines and piping for the ECCS and fog spray system are designed for a seismic factor of 0.5g. 2

3.8.4.4.1 Seismic Reanalysis - High Lift Pump Building - 182-N

In 1987 in response to ASEP Issues 12 and 13 (Ref. 61) an evaluation of the seismic adequacy of the masonry wall superstructure of the 182-N building was made. The evaluation was performed by Kaiser Engineers Hanford (KEH) (Ref. 78). 21

The features of the 182-N structure that appeared to the independent reviewers that established the ASEP issues to be the most vulnerable to damage from seismic loading are the exterior concrete masonry walls of the superstructure. The seismic loading in this part of the building is principally resisted by the structural steel frame supporting the roof. The structural steel frame of the superstructure, consisting of beams, girders and columns, is designed to resist lateral loads independently of the concrete-block filler walls.

These walls are non-load bearing and are constructed of hollow concrete blocks, 12-in. thick. Individual blocks are reinforced with vertical steel bars and concrete placed at about 32-in. centers and horizontal steel mesh located between each row of blocks.

The analysis performed by KEH utilized the methods and alternate calculations to reduce conservatisms used in the analysis of the superstructure of the 181-N building discussed in Section 3.8.4.1.1. Based on the results of the analysis, it was concluded that the exterior masonry walls in the superstructure of the High Lift pump house, including the anchorage of the masonry walls to the structural steel building frame have adequate capacity to withstand the SSE. The maximum stresses in reinforcing bars, concrete masonry and masonry anchors for the most severe combination of seismic and normal operating loads does not exceed 90% of yield strength for reinforcing bars and 75% of the ultimate strength of concrete masonry, and thus complies with the structural acceptance criteria contained in the SRP and SDC 4.1 (Ref. 50). 2

This analysis, while favorable, when reviewed by the Westinghouse Hanford Independent Safety Assessment Task group (Ref. 52) and the ECCS Seismic Task group (Ref. 63) was determined to be too limited in scope to provide assurance for the seismic capability of the ECCS components located within the building.

Accordingly a new, more complete seismic analysis of the entire structure was performed (Ref. 79). The purpose of this effort was to perform a structural analysis of the 182-N Building with respect to the requirements of Hanford Plant Standard SDC 4.1 (Ref. 50) to ensure its adequacy during a seismic event to preclude damage to the enclosed ECCS components. In addition to assessing structural integrity, in-structure response spectra were generated at selected locations for subsequent equipment and system evaluation. Since the engineering evaluation of the components of the ECCS was continuing on a parallel path, the analysis of certain localized structural components needed to be continually upgraded until the ECCS component configuration and support reactions were finalized.

The method of analysis utilized SSI analysis to predict foundation response to ground motion based on the SSE criteria given in Reference 50. Foundation response from the SSI analysis was then used as input data for the structural dynamic modal superposition analysis. Building responses (loads) and response spectra at various elevations were generated.

Structural members of the steel superstructure and reinforced concrete substructure were analyzed using loads generated from reactions to both seismic and gravity reactions. Evaluations of the structural steel framework, the masonry block walls, and the reinforced concrete substructure consisting of a concrete floor system, basement walls and footings were obtained.

The load combination used was 1.1 times the actual dead + live loads, plus 1.0 times the safe shutdown earthquake (1.1 (D+L) + 1.0 DBE) as required as a minimum in DOE 6430.1[8]. 21

Governing load combinations as reflected in the latest codes were selected by:

1. Using soil pressure and piping reaction forces where applied in the live loads in the SSE evaluation.
2. Using actual loads with years of demonstrated success. Since normal unfactored load combinations contained in ACI 349(11) are to provide for the uncertainties in estimating dead and live loads, these equations were not used because of the availability of actual loads.
3. The known dead and live loads do not appreciably contribute to the lateral forces. The adequacy of the structure to withstand the SSE lateral loads within code allowable stresses adequately ensures continued operability. The same allowable stresses were used for the SSE loads as would apply to the less severe factored OBE loads.

The factored load combination containing the operating basis earthquake (OBE) was not considered to be the controlling load combination and this was not specifically addressed. The 1.7 load factor times the 0.05 g earthquake (1.7 x 0.05 g = .085 g) produces less severe lateral loads than does the 1.0 load factor times the 0.25 g earthquake (1.0 x 0.25 g = 0.25 g).

The above analysis methodology is commensurate with that recommended in UCRL-15910, "Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards," Draft, for moderate to high hazard facilities.

The soil-structure interaction analysis was performed with the FLUSH computer program which performs a two-dimensional analysis of soil-structure configurations. An approximate three-dimensional capability is provided by including viscous boundaries on the sides of the model. Nonlinear stiffness and energy-absorbing characteristics of the soil are efficiently accounted for by iterative techniques.

Two soil-structure configuration models were constructed to determine the building responses. Both models used quadri-lateral elements for concrete, masonry and soil.

Low-strain soil properties and the strain dependency of the shear modulus and damping were taken from Section 3.7 of the NUSAR [3]. The approach used for iterating on these properties was to use the free field option. The converged free field soil properties were then re-entered into the model for the soil-structure analysis.

Soil layer spacing used criteria allowing transmission of 20 Hz signals through the soil. The KINT frequency interpolation multiples are about twice as fine as the minimum recommended. Building model and analysis details are provided in the report (Ref. 79).

Seismic excitation was defined in terms of design response spectra. An artificial time history was generated using MODQKE for a 5% spectrum. The control point for application of the excitation was at the top soil layer of each model.

Dynamic analysis of the building was accomplished with ANSYS modal analyses using separate models and separate excitation input for the horizontal and vertical directions. Frequencies, participation factors, and mode shapes

were then incorporated with an MDOF [4] modal superposition analysis to obtain separate horizontal and vertical responses at selected building locations. Independent horizontal and vertical response spectra at these selected locations were obtained using the SPECTR program [6].

Maximum accelerations for the existing building configuration were obtained as well as the response spectra at selected points in the building. The structural analysis consisted of evaluations of the structural steel superstructure and the reinforced concrete substructure. Seismic loads were calculated by using building responses generated from the dynamic analysis.

The analysis of the superstructure showed that the lateral stiffness of the walls is provided by the double angle cross bracing of the vertical exteriors wall frames. The exterior concrete block walls, attached to the vertical framing by means of metal clips with a single bolt embedded in the grouted block, cannot be depended upon to provide lateral load resistance. The concrete block walls are considered to be supported by the structural steel framing when the seismic forces are perpendicular (normal) to the wall. The seismic forces in the block wall which are parallel to the wall are considered to be resisted wholly by the block via shear, bearing, and friction. 21

The as-built wall cross bracings were found to be significantly overstressed when the building was subjected to horizontal seismic loads resulting from the Safe Shutdown Earthquake. The cross bracing members were considered to be effective in resisting both compression and tension loads. The capacity of the cross bracing system was found to be limited by the buckling strength of the compression brace. The capacity of the compression brace was found to be 36,800 lb compared to a demand of 81,800 lb in the E-W walls and a demand of 74,200 lb in the N-S walls. Therefore, if these compression members were not strengthened they would buckle inelastically and lose load resistance. The tension members would then be required to resist the entire lateral load. The tension members have a capacity of 107,800 lb at 90% yield compared to a demand of 163,670 lb.

Modifications to the cross bracing were designed to strengthen the lateral load resistance of the building. The method chosen to increase load capacity was to stiffen the existing cross bracings such that they could sustain the calculated additional compression loads without buckling. The affected columns and base plates were also evaluated for the increased loads and found to be satisfactory.

The ANSYS computer program was used in the analysis of the substructure. Computer models were made for both the floor system and basement walls. The reinforced concrete substructure consists of a floor slab integral with a beam/girder grid system supported by interior columns and by the box type basement walls around the perimeter. The basement walls having typical thickness of 2 ft -0 in. are supported on top of a continuous footing at an elevation of (-) 25 ft -6 in. The footing is 3 ft -0 in. thick, with a width of 9 ft -0 in. in E-W direction and 4 ft -0 in. in N-S direction.

The basic loads on each part of the structure were calculated and included; dead and live loads, superstructure reaction loads, dynamic soil pressure, and seismic and wind loads. The loads were combined as input data for the stress analysis in accordance with DOE Order 6430.1. The ANSYS computer program was used to determine the building member stress levels. The three orthogonal directions of seismic excitation were individually applied and the resulting loads were calculated by using the SRSS method. The resulting forces were tabulated and were used for verification of existing structural strength.

In the reinforced concrete members, the required reinforcing steel areas were calculated using the ultimate strength design method and using the resulting forces tabulated in the loads analysis. The evaluation was performed by using a combination of the largest load and least reinforced concrete section. All representative members were found to have adequate reinforcement to safely resist the applied loads in bending, shear, and torsion. In the seismic evaluation of 182-N, the 1963 code method was used to verify the adequacy of the torsional reinforcement. It was found that ample margins exist in the as-built structural members.

To summarize the above, the following conclusions were made.

Dynamic Analysis: As a result of the dynamic analysis, it was found that very little load amplification occurs in the load-bearing concrete or in the lateral masonry walls. For horizontal excitation, considerable load amplification occurs in the steel structure and the masonry that is supported by it. The load amplification is caused in part by the flexibility of the vertical steel bracing and in part by the bracing in the roof. Since girt line accelerations are not quite the average of the roof line and ground floor accelerations, there is also some load amplification associated with the out-of-plane deformation of the masonry and steel. For vertical excitation, amplification is associated with out-of-plane roof motion.

Structural Member Evaluation: Major structural members that are the primary load carrying elements were investigated. The stress analysis and code evaluation of the superstructure members were performed by hand calculations. The structural steel roof framing was found to be adequate. Both exterior and interior masonry block walls and anchors were evaluated and found to be acceptable. 21

The structural steel wall framing was found to be acceptable with the exception of the wall cross bracings. Modifications to the cross bracings were designed and installed (Ref. 80). As a result, all structural members will have adequate strength to resist the required SDC 4.1 (Ref. 51) seismic loads.

The reinforced concrete floor system, walls, and footing of the substructure were analyzed by using the ANSYS computer program. The forces calculated from the computer analysis were used to evaluate the concrete section design. Reinforcing steel provided in the concrete was found to exceed the required minimum quantities and therefore the concrete substructure is adequate.

3.8.4.4.2 Seismic Reanalysis of Filtered Water Tank

In response to ASEP Issue 73 (Ref. 61) an evaluation of the Filtered Water Tank was performed (Ref. 81) for the SSE specified in SDC 4.1 (Ref. 50).

The tank was analyzed to SSE equipment conditions to evaluate if the tank presented a hazard to equipment required for safe shutdown.

The Filtered Water Tank is an 850,000 gallon capacity tank 65 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the Filtered Water Tank is to:

- Provide a water supply to the demineralized water plant
- Provide a coolant supply to the spent fuel storage pool
- Provide normal suction supply to the filter water (FW-2) pumps.

The Filtered Water Tank is not a part of the ECCS and is not required to function after the SSE, but if it is functional, it may be used to supply water to the suction header of the 182-N Building high lift pumps by manually opening valves between the tank and the pump suction header.

The dynamic response caused by hydrodynamic pressure due to horizontal and vertical seismic loads on the tank were determined. The current operating level for the tank is 32.5 to 34.5 ft, an operating level of 34.5 ft was used in the analysis. The results of the evaluation are summarized below.

o Shell stress 15 in. from bottom plate	26.07 ksi
o Shell stress 8 ft from bottom plate	21.38 ksi
o Maximum bottom plate stress (secondary)	87.66 ksi
o Stress required for buckling	6.68 ksi
o Calculated buckling stress	4.07 ksi
o Sliding force	4,250 kips
o Force to resist sliding	2,750 kips
o Tank thickness to resist missile	0.236 in.
o Minimum tank thickness	0.250 in.

The tank material is ASME SA-283 Grade C with a specified minimum yield stress of 30 ksi. The analysis shows that the stress in the tank shell is substantially below 90% of yield for the SSE.

The 87.66 ksi calculated stress in the bottom of the tank is a secondary bending stress and should self relieve if the material deforms. The sliding force exceeds the force which resists sliding and it is concluded that the tank would move during the SSE. The result would be localized failure of the tank at pipe nozzles and limited leakage.

Since the Filtered Water Tank is not required to supply coolant to the Emergency Core Cooling System for the SSE and failure due to sliding would not endanger safe shutdown equipment, fixes for the tank are therefore not required for restart of N Reactor. 21

The Filtered Water Tank is required to be operable after the Operating Basis Earthquake (OBE) which has been defined (Ref. 51) for N Reactor as an earthquake with a horizontal ground acceleration of 0.05 g. Although as stated in the report, friction cannot be depended upon to resist sliding for the SSE with 0.25 g, it appears highly unlikely that sliding would be a factor in the event of an OBE. However, in accordance with the Prioritization Guidelines (Ref. 77) the need to determine the capability of the filtered water tank to resist sliding in the event of an OBE will be resolved within 24 months following restart.

3.8.4.4.3 Seismic Reanalysis of After Heat Removal Tank

The After Heat Removal Tank was added to the ASEP (Ref. 61) because of the close proximity and potential interface of the AHR tank with the ECCS. The tank was evaluated for the SSE in 1987 by Kaiser Engineers Hanford (KEH) (Ref. 82).

The After Heat Removal Tank is a 500,000 gallon capacity tank 50 ft in diameter, located south of the 182-N Building in the 100-N Area. The tank is supported on a reinforced concrete support ring and compacted oiled sand.

The function of the After Heat Removal Tank is to:

- o Store demineralized water
- o Supply demineralized water for secondary loop makeup during normal operation
- o Serve as a receiver for secondary loop and condensate returns.

The After Heat Removal Tank is not required to function after the SSE.

In summary, the KEH analysis showed the following:

o Shell stress 12 in. from bottom plate	4.88 ksi
o Shell stress 8 ft from bottom plate	1.26 ksi
o Maximum bottom plate stress (secondary)	12.79 ksi
o Stress required for buckling	6.60 ksi
o Calculated buckling stress	0.48 ksi
o Sliding force	233 kips
o Force to resist sliding	515 kips
o Tank thickness to resist missile	0.236 in.
o Minimum tank thickness	0.250 in.

The tank material is ASME SA-283 Grade C with a specified minimum yield stress of 30 ksi. Ninety percent of yield is 27 ksi. Based on the KEH analysis it is concluded that the stress in the tank shell and the tank bottom is substantially below 90% of yield for the SSE.

The calculated sliding force does not exceed the calculated force which resists sliding and the analysis concluded that sliding would not occur. However, previous experience with large unrestrained tanks during seismic events indicates that movement of the tank during the SSE would occur.

In the case of the After Heat Removal Tank, movement of the tank could result in failure of tank nozzles and leakage of water from the tank. However, the leak rate should be small and would not impact the operation of the ECCS.

Because the After Heat Removal Tank is not required to supply coolant to the Emergency Core Cooling System for the SSE and failure due to sliding would not endanger safe shutdown equipment, fixes for the tank to resist sliding during an SSE are not required. Within 24 months following restart of N Reactor the need to determine the capability of the tank to resist sliding in event of an OBE will be determined. 21

3.8.4.5 117-N Filter Building

The 117-N filter building is located 330 ft north of the E-W centerline of the reactor building and about 170 ft south of the 200-ft ventilation stack. It is constructed of reinforced concrete except for the roof, which is fabricated of steel plate and light structural members. Architectural and structural details for the 117-N building are shown on Figure 3.8-24 and 3.8-25. A filter cell is located in each of the four corners of the building, which is buried in the ground, except for the roof. A roof panel may be removed over each cell to provide access for servicing the filters. Each cell is approximately 19 ft wide, 27 ft long and 21 ft high, inside dimensions. The operating floor, which is at 0-0 elevation (454.5 ft MSL), consists of a concrete slab 12 in. thick and a series of access openings located over each filter cell. Each cell has two parallel banks of filters, each bank consisting of three stages. Thus, each cell has six access openings, each of which is covered with a removable shielding slab 12 in. thick. Each opening is about 2.75 ft by 9 ft in size and is surrounded on all sides by a concrete curb 12 in. high. Compression seals are provided at mating surfaces at the top of the curb and along the bottom of the respective shielding slabs. Tapered trunnions extend below the shielding slabs to guide them into place and to prevent them from moving laterally during an earthquake. Three buried exhaust tunnels (from the 105-N ventilation system) connect with the south side of the 117-N building (Figure 3.8-24). The tunnels, or ducts, from Zones I and II continue through the middle of the filter building and extend about 14 ft beyond the north wall, where they are terminated at a temporary barrier. (The space to the north is reserved for a second filter building in case additional filter capacity is needed at some future time.) The west face of the Zone II tunnel is also the north-south center line of the 117-N building. Separate ducts for handling

exhaust air from the filters are located on the west and east sides of the filter building and extend northward where they connect to the 116-N ventilation stack below ground level. The tunnel from Zone III of the 105-N building does not penetrate the 117-N filter building but connects to the south end of the east exhaust duct.

The four cells and the ventilation ducts within the 117-N filter building, are integrally supported on a rectangular foundation slab 18 in. thick, 100 ft wide and approximately 49 ft long. The top of the foundation slab is 22 ft below the operating floor and serves as the floor of the filter cells and the four exhaust tunnels. The space between the filter cells, above the two interior (Zone I and II) tunnels, is utilized as a shielded instrument room on the south and as a mechanical equipment room on the north. Floor elevations are minus 11 ft and minus 12 ft, respectively, for these two rooms. The two cells on the south are separated from those on the north by two access galleries, each 6 ft wide and having a sump at the far end extending to the minus 25 ft level. Concrete walls and floors are 12 in. thick, except for those for the instrument room, which are about 2 ft thick. See Section 6.5 for a description of mechanical equipment housed in the filter building.

The following load criteria were used in the design of the 117-N filter building:

Dead load	150 lb/ft^2	(slabs 12 in. thick)
Live floor load	250 lb/ft^2	
Earth pressure on exposed side walls:	30 lb/ft^2	times depth
Temperature differential for interior slabs:		50°F
Temperature differential across walls and floors in contact with soil:		20°F
Internal pressure: 10 in. equivalent water pressure (52 lb/ft^2)		

Except for the 250 lb/ft^2 live load, the above loads were also used for designing the underground exhaust tunnels. In addition, the tunnel structures were designed for the soil overburden (11 ft times 100 lb/ft^3) plus a 20-ton

truck (H-20) wheel loading (32,000 lb per rear axle). This axle loading produces an average pressure of 111 lb/ft^2 on the tunnel roof. Maximum stresses in the reinforcing steel, caused by various combinations of the above loads, were limited to the normal allowable design stress of 20,000 psi, not including earthquake stresses.

The seismic loads on the tunnels and the box-like structure of the 117-N building will be primarily from their weight and are not significant. For example, the lateral seismic load on a 12-in. wall during a 0.25 g earthquake would be approximately 36 lb/ft^2; whereas, the lateral force due to earth backfill varies from 360 lb/ft^2 at the top of a tunnel to about 660 lb/ft^2 near the foundation. In this case, seismic stresses would likely be less than 1,000 psi in the wall reinforcing steel. Thus, lateral earthquake reactions would have to be several times greater than 0.25 g before combined stresses would exceed design limits during the SSE.

The filter frames are vertical racks which support and seal in place the individual filter units that make up the filter banks in the four cells of the 117-N filter building. See Figure 3.8-26. Individual frames rest in flanged guide members, which are attached to the walls and center columns of the filter cells. Inflatable rubber seals, located on both faces of the frame, are pressurized (when in use) in order to form an airtight seal. Each frame holds 40 filter units - 9 ft wide, 21 ft 5 in. high, and 2 ft 8 in. deep. It weighs 1950 lbs empty, 8,750 lbs when loaded with activated charcoal filters and 3,950 lbs when loaded with HEPA filters. The frame, rubber seals and support guides are designed to withstand a face pressure of 12 in. water gage pressure or about 10,000 lbs. During an earthquake, frame movement is damped by friction at the base and by the perimeter seals, which provide a conservative damping coefficient of 2%. Using this damping value and the acceleration design spectrum curve for a 0.25 g earthquake, the corresponding seismic factor would be about 0.6 g. Six-tenths of the weight of the heaviest frame is only 5,250 lbs, which is substantially less than 10,000 lbs used in design. The shear stress generated on two dowel pins which would resist lateral motion of the frame in case the seals fail during the postulated earthquake, would be about 5,250 psi. This stress is

substantially less than the shearing strength (27,000 psi) of the dowel pins, which are made of 6061-T6 aluminum.

3.8.4.6 116-N Ventilation Stack

The 116-N ventilation stack is a reinforced concrete structure located about 500 ft north of the E-W centerline of the reactor building. It is used to transmit air exhausted from the 117-N filter building to an elevation of 201 ft above ground before releasing it to atmosphere. See Figure 3.8-27. It is supported on a concrete foundation, the top of which is located 22 ft below grade so that the total height of the cylindrical portion of the stack is 223 ft. The foundation is 4.5 ft thick beneath the stack and tapers to 2.5 ft thick at the outer perimeter, which is octagonal in shape. The concrete stack has an outside diameter of 15 ft at the top and tapers to 22.1 ft at the foundation. It has a wall thickness of 6 in. except for the lower 37.5 ft, where the wall thickness increases from 6 to 12 in. thick. Additional details are shown in Figure 3.8-28, including the amount of vertical reinforcement provided at various elevations along the stack height.

The stack was designed and constructed by the M.W. Kellogg Co. in 1961. Its design was based upon criteria presented in the following standards:

Reference	Standard	
42	ACI 505-54	Design and Construction of Reinforced Concrete Chimneys
5	UBC 1958 Ed.	Uniform Building Code, Section 2312 - Earthquake Requirements for Seismic Zone II

Wind pressures used in the design, conform to requirements presented in ACI 505 and a peak wind velocity of 100 mph at 30 ft above grade. They are as follows:

Vertical Location (feet above ground level)	Wind Pressure (lb/ft^2)
0 to 50	20
50 to 100	24
100 to 200	28

Lateral forces and bending moments acting on the ventilation stack, due to the above wind pressures, are more severe than those caused by earthquake forces associated with UBC seismic Zone II and, therefore, controlled the design of the stack. See Figure 3.8-29. The design also provided for an atmospheric temperature range of minus 20°F to 110°F and a maximum gas temperature of 150°F inside the chimney at the breeching. Maximum soil bearing pressures were kept below 6,000 lb/ft^2, which is a conservative value considering that bearing capacity for this particular location is in excess of 12,000 lb/ft^2.

Design calculations performed in 1969 demonstrated that the stack could withstand an earthquake producing a maximum horizontal ground acceleration of 0.25 g (Ref. 43). For this earthquake level, maximum stresses occurred in the reinforcing steel at about 140 to 170 ft down from the top of the stack and were found to be around 41,500 psi. This value of stress is somewhat less than the yield strength of A-15 intermediate grade reinforcing steel used in construction, which is expected to be around 50,000 psi, rather than the code minimum of 40,000 psi.* At the 0.25 g earthquake level, if any yielding were to occur in the stack reinforcement, it would be relatively small and localized in extent. Likewise, local yielding, if any, would tend to increase damping and to decrease spectral response. In addition, the modulus of elasticity would drop slightly causing the natural period of

*Yield strength of A-15 (intermediate grade) reinforcing steel used in the construction of the foundation for the emergency dump tank averaged 53,900 psi, based upon 21 tests. See Table 3.8-9.

vibration to increase in magnitude and the spectral response to decrease. Based upon the results of the study described above, it is concluded that the stack, which was designed originally based upon wind requirements, would be able to survive an earthquake having a maximum horizontal ground acceleration of 0.25 g. The 116-N ventilation stack was analyzed dynamically as a multi-mass system in 1970 by an independent consultant (Ref. 44). The results of these analyses confirmed the results obtained earlier and indicate the 116-N ventilation stack would perform satisfactorily during the specified earthquake, which had a maximum ground acceleration of 0.25 g. In the multi-mass analysis, the time-history input was based upon an earthquake similar to the N-S component of the 1940 El Centro earthquake but having a maximum horizontal ground acceleration normalized to 0.25 g. The proprietary computer code DYNRESP was used to perform the dynamic analyses, which included seven different cases of assumed stack behavior (Ref. 44).

The mathematical model of the ventilation stack used in the dynamic analyses is shown in Figure 3.8-30. Problem 100 described in Ref. 44 of the dynamic analyses was based upon an elastic solution utilizing "uncracked" section properties in which 8% damping was assumed for the first (primarily rocking) mode and 5% damping for higher modes of vibration for the stack and its foundation. The computer program generated time-histories of maximum stresses in the reinforcing steel at various stack levels and at times when stresses in the steel exceeded 40,000 psi. The review of these data indicate that maximum stresses exceeded 40,000 psi only for very short time intervals at any one point and then only one to about four times during the first 8 seconds of the simulated earthquake.

It is unlikely that the yield strength of the reinforcing steel is as low as the code minimum of 40,000 psi. If this should be the case, then Problem 100 indicates that maximum stresses in the reinforcing steel approached or slightly exceeded the yield strength a limited number of times. In this case, some tensile cracking of the concrete probably would likely occur, which would invalidate the solution based upon "uncracked" section properties. The stack computer problem was then rerun a number of times using section properties which allowed for tension cracking of the concrete. (To admit

tensile cracking is a normal design condition for reinforced concrete structures, in which the moment of inertia of the "cracked" section is taken as the average of the moment of inertia of the gross section and that of the transformed section.) Use of the "cracked" section properties in Problem 102 resulted in a natural period of 1.33 seconds for the first mode as compared with a value of 1.16 computed in Problem 100.

Damping in Problem 102 was assumed the same as for Problem 100; whereas, in Problem 101, 10% damping was assumed for the first (primarily rocking) mode and all higher modes of stack vibration. However, this change in assumed damping did not have a significant effect on the results. The shear wave velocity for the supporting soil was taken as 800 ft/s for Problems 100, 101 and 102. Conditions for Problems 103 and 104 were the same as for Problem 101, except the shear wave velocity of the soil was assumed to be 600 and 2,000 ft/s, respectively. The 800 ft/s value corresponds to the average shear wave velocity measured at the K reactor site, which is known to be similar to that at 100-N. The 600 and 2,000 ft/s values correspond to the lower and upper values of shear wave velocity, respectively, that had been measured in typical alluvium at the Hanford reactor sites prior to 1970. (These values of shear wave velocity were confirmed by field testing at the N site performed since 1970. See Table 3.7-2.) The dynamic analyses show that the assumed variation in soil properties on stack response are not significant and that maximum stresses in the vertical reinforcing steel are generally less than about 35,000 psi for conditions defined in Problems 101 through 104. In addition, computed values of stresses in the concrete were shown to be fairly low and that the reinforcement would have to yield before the concrete could fail in compression.

The effect which vertical ground accelerations have on steel stresses at various sections along the stack was also considered in the dynamic analyses. Vertical accelerations of 0.125 g and 0.25 g were assumed to act simultaneously with a maximum horizontal acceleration of 0.25 g, a conservative assumption. With 75% effective dead load acting, the maximum steel stress at a point 167 ft below the top of the stack was shown to be around 50,000 psi and that a small amount of yielding of the reinforcement

may occur. The concrete section at this point is "under-reinforced with respect to balanced-steel requirements"; thus, a small amount of yielding no way represents collapse. The ultimate bending resistance of the stack, based upon a yield strength in the steel reinforcement of 40,000 psi, is also plotted in Figure 3.8-29, as well as the seismic moments due to an earthquake having a maximum horizontal ground acceleration of 0.25 g obtained in Problem 102. It is shown that the safety factor against collapse of the 116-N stack is 1.30 or greater and that this minimum safety factor occurs at a point 167 ft below the top of the stack. These data confirm the conclusion obtained previously and demonstrate that the 116-N ventilation stack will safely survive an earthquake having a maximum horizontal ground acceleration of 0.25 g.

The Westinghouse Hanford Independent Safety Analysis Task group reviewed the seismic capability of the 116-N Stack in 1987 (Ref. 52) and concluded the N Reactor can be operated without undue risk from natural phenomena damage to the 116-N Stack and that the seismic design of the stack has undergone adequate evaluation.

3.8.4.7 Emergency Dump Tank (1304-N)

The Emergency Dump Tank is an integral component of the N Reactor's ECCS which is an Engineering Safety System. The ECCS has been classified (see 3.2, Table 3.2-1) as Seismic Category I. The function of the Emergency Dump Tank and associated piping is to:

- Receive the contents of the reactor coolant system during a reactor emergency
- Absorb thermal energy
- Vent excess steam and gas back to ventilation Zone 1 of the 105-N reactor building
- Transport excess reactor coolant system flow through the overflow line to the 1312-N LERF and through the vent line to the 109-N building.

The emergency dump tank is a large, cylindrical vessel with an essentially flat bottom and an elliptical head. The cylindrical shell is 61 ft in diameter and 48 ft high, not including the head, which has a rise of 15 ft 3 in. The vessel is supported on a reinforced concrete foundation and located between the railway tracks and the Columbia River, at the west side of the reactor building. Additional design details for the tank are shown in Figure 3.8-31. The purpose of the tank is to receive the contents of the reactor coolant system during a reactor emergency, to absorb thermal energy and to vent excess steam back to ventilation Zone I of the 109-N reactor building. During periods of normal reactor operation, it contains 685,000 gal of cold water; whereas, when completely full it has a capacity of 1,270,000 gal. 21

The main piping associated with the Emergency Dump Tank and shown on the drawings and specifications consists of the following:

- 24 in. Diameter Vent Line
- 30 in. Diameter Dump Line
- 30 in. Diameter Overflow Line
- 28 in. Diameter Sparger Line
- 18 in. Diameter Air Extraction Line
- 6 in. Diameter Drain Line.

The tank was designed and constructed by the American Pipe Construction Co. in accordance with requirements of API 620, "Large, Welded, Low-Pressure Storage Tanks," (Ref. 45) and the following job conditions:

1. An internal pressure of 5 psig when containing 1.15 million gallons of water
2. An internal negative pressure of 0.5 psig when almost empty
3. Hydraulic pressure when filled to highest point of the 24-in. vent-pipe with a concurrent pressure of 0.1 psig
4. Hydraulic pressure when filled to the top of the roof plus a proof-test pressure of 7.5 psig
5. A horizontal ground acceleration of 0.2 g due to earthquake

6. A wind pressure of 25 lb/ft^2 acting on 0.6 of the projected area of the tank exposed above ground
7. Thermal shock when the contents of the tank (615,000 gal of water) change in temperature from 35°F to 200°F in 10 minutes due to the addition of 450°F steam and water.

The tank is constructed of ASTM A-516, Grade 70 steel (Ref. 46).

Values of internal pressures, specified for the first three items listed above, were considered to act concurrently with earthquake and wind forces, in addition to the specified hydraulic and thermal design loads. No increase in allowable stresses was permitted when earthquake loads, or wind forces, were combined with the other loads. Earthquake, wind and the 7.5 psig test pressure were not considered to act at the same time. Pressures produced during the 7.5 psig proof test (load condition four above) are larger than those specified under the other load conditions. For this reason, they were used to compute shell thicknesses at various tank elevations. In these design calculations, a pressure head of 80.6 ft at the bottom of the tank was used with an allowable stress of 21,000 psi in the steel plate and a weld efficiency of 100%.

The concrete foundation and tie-down straps for the tank were designed by General Electric Company at Hanford. In order to allow inspection of shell surfaces during testing, the 7.5 psig proof test had to be performed before the concrete ring walls and the backfill were installed. This test condition produced maximum uplift on the foundation and controlled the amount of reinforcing steel in the foundation, as well as shell thickness. Now that the project is completed, the additional weight of the ring wall and the backfill counteracts any uplift forces and markedly reduces bending moments in the foundation matt. As a result, stresses in the reinforced concrete matt are much smaller, during normal and emergency load conditions specified above than they might have been otherwise.

The emergency dump tank is tied to its foundation with 58 anchor straps. During the proof test, these anchors developed a tensile stress of around

15,000 psi. This is about the same amount of direct tension that would be produced in the most-stressed anchor straps during the design earthquake plus 5 psig pressure acting on top of 53 ft of tank water. Anchorage of the straps to the foundation and to the shell of the tank is sufficient to cause the straps, which are ductile steel, to yield before tearing away from the tank shell or from the concrete foundation. Outward thermal expansion of the tank bottom during an emergency dump will produce bending stresses in the anchor straps in addition to direct tension. For a radial expansion of 3/8 in. the bending stress will be 6650 psi and the maximum combined stress about 25,000 psi, which is about 70% of the material's yield point.

Pipe penetrations of the ring wall are sleeved and provision made for thermal tank growth and relative movements which might occur during an earthquake, thereby avoiding localized stressing of the tank at places when pipes are attached. During erection, 100% of the vertical seams and roof welds were radiographed. Most all of the horizontal girth welds in the shell were also radiographed. Data describing the physical properties of the concrete and the steel used in erecting the structure are shown in Table 3.8-9. In all cases, physical properties determined during quality-control testing of the materials used in construction, substantially exceeded minimum values required by the specifications.

Neglecting the presence of the concrete ring wall and the backfill, calculations indicate that the emergency dump tank would respond at the following natural frequencies, depending upon depth of water contained in the tank:

Water Depth (ft)	Total Weight (10^3 lbs)	Natural Frequency* (Hertz)
0	385	50
28	5,385	13
50	9,223	10

*Essentially in a cantilever mode.

When the weight of the ring wall and the backfill is considered in the calculations, the increased mass would cause the tank to respond at somewhat higher frequencies, thus avoiding the frequencies of the acceleration spectrum for which dynamic response is a maximum.

Although the tank was designed in 1969 originally for a maximum horizontal ground acceleration of 0.2 g, it was re-analyzed later for 0.25 g and found adequate (Ref. 47). In these analyses, the sloshing behavior of water in the tank, shearing forces and overturning moments during the postulated earthquake, were determined by the procedure described in TID 7024, "Reactors and Earthquakes" (Ref. 48). The following design data and maximum stress values are of interest:

Stresses in the bottom shell plate (thickness of bottom shell plate 5/8-in.)

Hoop Stress

16,400 (psi) tension	(53 ft of water plus 5 psig)	
10,000 (psi) tension	(28 ft of water plus 5 psig)	

Longitudinal stress

Dead weight	234 (psi) compression	
5 psig uplift	1,470 (psi) tension	
0.25 g earthquake	2,110 (psi) compression	(neglecting compression in the anchor straps)

Stresses in anchor straps (58 straps, each 9 in. wide and 1/2-in. thick)

5 psi uplift	7,000 (psi) tension
3/8-in. radial expansion	6,650 (psi) bending
0.25 g earthquake	11,600 (psi) tension

Combined Stress 25,250 psi tension (which is less than yield strength of 36,000 psi)

Bearing on soil foundation (During 0.25 g earthquake plus full tank)

$p = 3,480 \pm 2,140$ lb/ft^2 F < 6,000 lb/ft^2 allowable bearing stress
12,000 lb/ft^2 bearing capacity of soil

Safety factor against overturning during 0.25 g earthquake: 5.72
(includes simultaneous effect of 0.125 g vertical acceleration)

Period of water sloshing in full tank: 4.5 seconds

Rather than construct a tank which could accommodate a negative internal pressure of 2.0 psig, the vendor chose to design for a vacuum of 0.5 psig and to provide four, 12-in. vacuum-relief valves. These valves will pass a total of 30,000 ft^3 of air per minute when the pressure drop is less than 0.5 psig. The high point of the roof, which is at 486.4 ft MSL, is connected to Zone I of the 109-N building with a 24 in. vent pipe.

Initially, the crest of the 30-in. overflow pipe was located at 460.75 ft MSL. However, a deep disposal basin was never funded so that the overflow from the emergency dump tank had to be discharged to the existing 1301-N facility. As a result of this design modification, it was necessary to raise the elevations of the overflow pipe 3.25 ft and the invert of the 24-in. vent pipe and the vacuum relief valves about 10 ft. This configuration change provided the hydraulic head required for transporting water to the 1301-N facility. In 1987, this was rerouted to the 1312-N LERF and tested. 21

The 24-in. vent pipe discharges into Zone I of the 109-N building at 496.75 ft MSL to minimize flow to Zone I in case three high-lift pumps are left on longer than necessary.

Under this new hydraulic configuration, the emergency dump tank will experience a maximum water pressure at the bottom of the shell equal to 63 ft of water each time the Equipment Maintenance Standards (EMS) flow test 21

is performed. During the EMS flow test the pressure at the high-point is atmospheric.

Should an emergency occur, the pressure in the vent pipe will be about the same as the pressure in Zone I of the 109-N building. During the worst accident, the pressure in Zone I is expected to rise to a maximum in a few seconds and then eventually decrease to zero. See Figure 6.2-8. The calculated maximum pressure during the incident is somewhat less than the 5 psig design pressure. The shortest time required to fill the dump tank during an emergency will be greater than 30 minutes, at which time the pressure in Zone I will have normalized back to an inch or less of water pressure. The original design criteria specified that maximum water elevation in the dump tank occurred simultaneously with the 5 psig maximum design pressure. This condition is not feasible because the pressure in the confiner decays faster than the dump tank is filled with spent reactor coolant, as described above. Consequently, when the dump tank is filled with water during an emergency, the tank will experience a maximum water pressure at the bottom of the shell equal to about 65 ft. This value is conservative when compared with 68.5 ft provided for in the original design (corresponding to a weld efficiency of 85%) and with 80.6 ft to accommodate the 7.5 psig proof test. As described previously, vertical joints between shell plates were 100% radiographed during construction, which (according to API 620) permits a weld efficiency of 100% and a safe design pressure head of 80.6 ft for rating the tank.

Another case of design conservatism is the assumption that the emergency dump tank is full of water during the postulated earthquake. Although, the duration of earth vibration is not specified in the design criteria, it is expected to be less than 30 seconds. Thus, the emergency dump tank would be only partially full during the first earthquake, as it takes at least 30 minutes to fill the tank during an emergency dump. For this reason, the design criteria provides for a severe aftershock, up to 0.25 g in magnitude, which must occur at least 30 minutes after the major shock which caused the emergency dump. Such a combination of events is improbable, but cannot be ruled out completely.

During an emergency dump of the reactor coolant system, the contents of the loop will be delivered to the emergency dump tank and distributed over the bottom surface by means of a single, straight section of sparger pipe. This pipe is 30 in. in diameter and is fabricated out of ASTM A516, Grade 70, carbon steel-plate (Ref. 46). It is about 59 ft long and contains 916 1-in. holes drilled in the upper surface. The sparger is supported from the floor of the emergency dump tank by means of a series of prefabricated steel trusses. The adequacy of this type of sparger was verified in 1963 during the "hot-dump" test performed prior to startup of N Reactor.

3.8.4.7.1 Seismic Reanalysis of Emergency Dump Tank

In 1987 the Westinghouse Hanford Independent Safety Analysis Task group reviewed the seismic capability of the EDT (Ref. 52). They found positive margins of safety in the review performed by Davis (Ref. 47) in 1969. It was noted that the flexibility of the tank walls was not accounted for in that analysis. Failure of the EDT on the piping downstream of the V-23 valves, exterior to the confinement structure could result in unacceptable consequences. It was recommended the EDT be qualified to the HWS SDC 4.1 criteria (Ref. 50). 21

The ECCS, seismic, wind and tornado task group, similarly, noted four apparent deficiencies in their 1987 review of the EDT seismic capability and recommended a seismic analysis be performed including internal piping and external connected accessories and pipe.

Because of the deviations noted above the Emergency Dump Tank, adjacent earthwork and associated piping were evaluated (Ref. 83) for structural integrity during a postulated Safe Shutdown Earthquake.

A seismic analysis of the Emergency Dump Tank was conducted utilizing the methods and technique outlined in Reference 83. Three percent damping was utilized for the steel tank and 1/2% damping for the liquid. The analysis

was conducted for four loading conditions and stresses were calculated as tabulated below.

Seismic Analysis Summary

Case	Loading Condition	Calculated Stress (ksi)	Allowable Stress (ksi)
A	Normal (maximum) (40.8 ft Liquid Head)	Primary Stress - 13.3 (Static Load) Primary + Secondary Stress (Dump - Non SSE) - 41.1	21 49
B	SSE (40.8 ft Liquid Head)	Primary Stress - 20.1	34.2
C	Severe Seismic Aftershock (52.5 ft Liquid Head)	Primary Stress - 27.7	34.2
D	Severe Seismic Aftershock (Tank Full - 63.5 ft Liquid Head)	Primary Stress - 36.3*	34.2

* The calculated stress exceeds the allowable stress by 6%. This is considered acceptable because of the inherent conservatism in the analysis and also as the actual yield strength of the tank material per DUN-8291 was 50.5 ksi, whereas the allowable stress of 34.2 ksi (used in the supporting calculations) was based on the minimum acceptable yield strength of 38 ksi for A-516-07, Grade 70 plate material.

Sliding Analysis - The results of the seismic analysis of the Emergency Dump Tank full of water (Loading Condition D, Table 1.0) indicates the applied lateral forces, exceed the resisting forces. As a result there could be relative motion between the structure (EDT and its foundation mat) and its attached piping. By analysis the amount of horizontal movement in any direction is estimated to be on the order of $\pm$ 0.8 in. Conservatively it is assumed that the vertical concurrent movement would be approximately 0.8 in.

(even though the vertical seismic response is only 2/3 the horizontal response).

The existing bellows assembly on the 30 in. diameter Dump Line (Purchase Order #R9F-555-12355) was designed for the following conditions:

a. Longitudinal Compression = 1.5 ft
b. Lateral Displacement = 0.204 in.

which was substantially smaller than the required 0.8 in. movement capability. Also, physical inspection of the bellows assemblies in the 30 in. diameter overflow line and the 24 in. diameter vent line revealed that these bellows were the "Uniroyal" rubber type and did not appear to provide the required ± 0.8 in. movement capability. In addition, the AWWA revised "Standard for Welded Steel Tank for Water Storage, ANSI/AWAA D100-84, March 18, 1985," requires all unrestrained flat-bottom tanks to provide a minimum of 2 in. of flexibility in tangential and vertical directions at all pipe connections. Z1

The existing expansion joints did not provide the required flexibility nor were they seismically qualified to resist cyclic loading generally experienced during a design basis seismic event. Hence a procurement specification HEDL-S-0166 dated 1/26/88 for the purchase of the replacement bellows, to be fabricated and tested in accordance with the requirements of ANSI B31.1 was prepared. Upon receipt, the new expansion bellows were installed (Ref. Work Authorization 154424, ECN 200683).

3.8.4.7.2 Soil Condition Analysis

The results of the slope stability analysis of the adjacent earthwork showed that the factor of safety slip circle analysis was greater than 1.1, indicating the slope of the ground at the Emergency Dump Tank is essentially stable. A little ravelling of the ground on the Westerly side however may occur, owing to the essentially unconfined state, but this will not jeopardize the integrity of the Emergency Dump Tank and its foundation (Ref. 83).

3.8.4.7.3 Associated Piping Analysis

The piping associated with the Emergency Dump Tank was analyzed (Ref. 83) for pressure, deadweight, thermal expansion, earthquake and tornado wind loading. Thermal expansion analysis was performed only when there was differential displacement between the piping and its attachment point.

The results of the piping and pipe supports analysis at the EDT are summarized in Reference 83 and included analyses and qualification of the following piping and associated supports:

- 24 in. Vent Line
- 18 in. Air Extraction Line
- 30 in. Sparger Line
- 30 in. Overflow Line
- 6 in. Drain Line.

3.8.4.7.4 Valve Access Platform and Walkway

The platform and walkway layout was redesigned to suit the revised configuration of the 24 in. diameter vent line and provide the required seismic capability. The layout arrangement, member sizes, and connection details are included in Appendix C of Reference 83.

Conclusions - It was concluded from the reanalysis that the Emergency Dump Tank, associated piping and fixtures were qualified to the SSE criteria of SDC 4.1 (Ref. 50). The following summarizes the safety modifications that were required. These were installed as indicated in Reference 84:

1. Replace existing expansion joint bellows on the 24 in. Diameter Vent Line, the 30 in. Diameter Dump Line and the 30 in. Diameter Overflow Line.
2. Reroute existing 24 in. Diameter Vent Line and provide new supports.
3. Rework supports outside the tank on the 18 in. Diameter Air Extraction Line.
4. Rework valve access platform and walkway to suit revised configuration of the 24 in. Diameter Vent Line.

3.8.4.8 105-N Control Room

The reactor control room is located on the southeast side of the 105-N building structure and just north of the common wall. The main floor is at ground level (454.5 ft MSL). The control room is the upper portion of a two-level, reinforced-concrete section of the 105-N building. The lower level, or basement beneath the control room, contains a major part of the reactor instrumentation and electrical switch gear. The reactor control room is about 98 ft long, 45 ft wide and 15 ft high. The concrete roof slab is 12 in. thick. It is supported at the periphery by the concrete side and end walls and at the middle on four concrete columns, each 16 in. square. The walls of the control room and the basement are typically 12 in. thick, except for the west wall of the basement which is only 8 in. thick. See Figures 3.8-1 and 3.8-2.

The main floor of the control room is 6 in. thick and is supported on a system of steel beams and girders consisting of 18 to 24-in. WF sections. The beams and girders are supported by the basement walls and by four interior steel columns. The columns are 12-in. WF sections and are spaced approximately 20 ft apart. They are located directly beneath the four concrete columns which support the roof slab of the control room. The entire control room and basement structure is supported on continuous footings located under the walls, and on individual concrete footings beneath the interior columns. This part of the 105-N building is separated from adjoining sections of the 105-N and 109-N buildings on the west and south, respectively, by a continuous expansion joint starting at the top of the footings. The basement area is about 146 ft long and extends 48 ft beyond the east wall of the control room. The concrete slab over this extended area is 8 in. thick and is also supported by an interior system of steel beams, girders and columns and the basement walls. The basement floor slab is 6 in. thick. It is supported on consolidated backfill and by the footings. The 105-N control room and basement will not be damaged structurally by the 0.25 g SSE.

In 1985 as part of the Plant Life Extension program a seismic analysis was performed by WHC system stress analysis (Ref. 85). As a part of this analysis a soil structure interaction analysis (SSI) was done which included the

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105-N/109-N building complex. The analysis utilized the HWS SDC 4.1, Rev. 9 (Ref. 50), Design Basis Earthquakes. Results of the SSI were then used as input to a 3-D ANSYS structural model of the control room. The purpose of the analysis was to check structural adequacy of the walls, floors, and roof slab and to produce response spectra for seismic qualification of control room equipment.

The results of the seismic analysis showed no overstress conditions except in the east wall of the stairwell down to room 6. This wall is slightly overstressed assuming the concrete has a maximum compressive strength (f'_c) of 3,000 psi, however, it is within allowable stress conditions if an f'_c of 4,000 psi is assumed. The 4,000 psi value is much more reasonable considering the age of the concrete. The results also showed that displacements were well within the 2 in. allowed for in the seismic joint.

It was concluded from the analyses performed that the Control Room Structure, both in its present configuration and with the addition of H-544 (control room habitability) equipment, meets the requirements of SDC 4.1 (Ref. 50), for Seismic Category I structures.

3.8.4.8.1 Seismic Reanalysis of Control Room/Room 6

In 1987 the Westinghouse Hanford Independent Safety Assessment Task group (Ref. 52) reviewed the seismic capability of the 105-N control room and concluded the following:

"Further seismic analysis of the 105-N reactor control room is not needed or recommended unless conditions such as additional floor loadings, etc., change sufficiently to warrant it. However, it is recommended that a III/I walkdown of the control room be implemented to identify and secure any potential missiles which may pose a hazard to safety class controls."

The III/I walkdowns were initiated in 1987. For the control room and cable spreading room the following items were inspected and are addressed in four reports (Refs. 86, 87, 88, 89).

- o Control room suspended ceiling panels
- o Control consoles, panels, boards, racks, annunciators, computers, graphic display panels, cabinets, cable trays and conduit in the control room
- o In room 6 battery racks, relay racks and cabinets, power supply and power distribution racks, and cable trays

More than 60 potential problems involving seismic restraint and tie-down were identified and resolved by analysis and hardware fixes.

3.8.4.9 Seismic Upgrade of the 109-N Turbine Drive Bay Building

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3.8.4.9.1 Description

The 109-N Turbine Drive Bay Building is located adjacent to the south wall of the 109-N confinement structure and contains the primary turbine drive units and backup pony motors for the six reactor coolant system (RCS) pumps, as well as the dump-condensers, various pumps, piping, and auxiliary plant equipment. The building consists of a reinforced concrete substructure and a steel-frame superstructure with an insulated concrete roof deck and sheet metal siding. Platforms are mounted on the roof structure which support the high-pressure steam and other piping systems, valves, and auxiliary equipment. The building is made up of 21 similar 18 ft -3 in. wide bays in the longitudinal (E-W) direction. The entire building is 410 ft long in the E-W direction and 60 ftde in the N-S direction. See Figures 3.8-10 and 3.8-11 for details.

The substructure includes all portions of the building at and below the operating floor at elevation 454 ft -6 in. It consists of a series of rigid joint two-span reinforced concrete frames in the N-S direction. The E-W direction is made of a continuous 18 in. thick wall at the exterior column line V and by rigid joint reinforced concrete frames along the two interior column lines R and T. The operating floor is a 5-in. thick reinforced

concrete slab. A 6-in. thick basement slab exists at elevation 438 ft -6 in. (16 ft below grade) on top of the foundation soil. This basement slab which is supported on the foundation soil, is not connected to the substructure wall or columns. The substructure columns and wall extend downward to elevation 430 ft -6 in. (24 ft below grade) and the footings to elevation 428 ft -6 in. (26 ft below grade). The turbine drives are supported on reinforced concrete pedestals.

The superstructure is of bolted steel frame construction and is internally braced in both directions. It is separated into two parts in the longitudinal direction by a seismic joint at column line 60. There are duplicate columns at each side of the seismic joint, and all beams are discontinuous at these columns. There are two bays in the N-S direction. The bay adjacent to the 109-N confinement structure is open and extends 36 ft from the operating floor at grade level to the roof. It contains a 35 ton gantry crane running the length of the building located 21 ft above the operating floor. The crane rails are attached to the columns at lines R and T. The outside bay has diagonal braces and contains piping systems, valves, and other equipment. It includes a structural steel mezzanine floor located 15 ft above the operating deck. This mezzanine supports sixteen dump condensers. The roof is a 4 in. thick reinforced concrete slab supported by steel beams. Structural platforms are supported by the building roof. These platforms consist of a pipe support structure at column line T, 46 ft above the operating floor and of two platforms 53 ft above the operating floor. Two gantry cranes (3 ton and 6 ton capacity) travel along the length of the building on these platforms. See Figure 2 for details.

3.8.4.9.2 Analysis for Seismic Upgrade

An evaluation (Ref. 91) for the effects of a Safe Shutdown Earthquake (SSE) event on the 100-N facility 109-N Building Turbine Bay, Decontamination, and Auxiliary areas was performed to support the N Reactor safety enhancement effort. The evaluation addressed the steel superstructure and the reinforced concrete substructure. The seismic behavior of this nonseismic Category I structure must not jeopardize seismic Category I safety-related electrical circuits which are routed through raceways that traverse the building areas.

All evaluations and supporting analyses were based upon seismic requirements contained in the Hanford Standard Design Criteria document, "Standard Arch-Civil Design Criteria, SDC 4.1, Rev. 10" (DOE, 1988). Analysis results were processed to assess the reactor facility non-Category I building structural response to the SSE event.

The qualification effort was limited to the 109-N Building Turbine Bay, Decontamination, and Auxiliary areas. The nonseismic Category I classification initially assigned to this structure was based on the safety and seismic classification associated with the enclosed steam system. Since the N Reactor steam system supports power production, its function is neither required to ensure safe shutdown and maintenance of a cold-shutdown condition nor required to provide continuous cooling by the Emergency Core Cooling System (ECCS). Consequently, the steam system is not critical to nuclear safety and the building structure enclosing it was not originally qualified to seismic Category I requirements. 21

Seismic Category I control and sensor circuits for the ECCS diesel engines and for the building Area 3 steam vent valves traverse the subject structure in electrical raceways. Recent system modifications and attention to the earthquake related potential for common mode failures has resulted in the requirement that these critical-to-safety electrical raceways not be compromised by the SSE event. Accordingly, the building structure must be qualified to the extent necessary to ensure that the nonseismic Category I building will not compromise the Category I raceways when subject to the effects of the SSE event.

A preliminary analyses had concluded (Ref. 90) it was very likely that this building and equipment could be qualified by means of a more rigorous analysis coupled with some minor structure upgrades. The initial analyses, therefore, justified the undertaking of a more precise, less conservative analysis that would qualify the 109-N Building Turbine Bay and east section of the building, and therefore the cable trays and circuits for an SSE event.

Based on this recommendation the 109-N Building Turbine Bay, Decontamination, and Auxiliary areas were re-evaluated for the effects of an SSE event. The response spectra synamic analysis technique, implemented by the finite-element method, was applied to obtain structural member stress resultants. The seismic event was characterized by the seven (7) percent damped site SSE response spectra (DOE, 1988). The analytical results were applied to support seismic qualification activities for these structures.

It was concluded that the 100-N Facility 109-N Building Turbine Bay, Decontamination, and Auxiliary areas are qualified for effects of the SSE event provided that the recommended conceptual modifications are implemented. The conceptual modifications includes steel frame lateral supports, strengthened cross braces and strengthened column-to-baseplate welds. The recommended modifications are conceptual, as prescribed by the project directive; sufficient information is provided to support modification design and cost assessment. A summary of the recommended modifications is provided in Table 3.8-12. Sufficient analyses and evaluations were performed to support the conceptual modification recommendations. Analysis and evaluation results are documented in related addenda. The location of results related to the affected areas are identified in Table 3.8-12.

Because this is a Return to Service issue, the commitment has been made that the required modifications will be made prior to restart of N Reactor.

3.8.5 Foundations

Major footings for Category I structures are supported on undisturbed soil consisting mainly of compacted gravel. The gravel beds which overlay the Pasco Basin and the underlying Ringold Formation are described in Sections 2.5.1.2 and 2.5.4. Data on subsoil distribution and soil properties are presented in Section 3.7.1.5 and in Table 3.7-2. General criteria and codes used to design the various footings are described in the following paragraphs.

3.8.5.1 Description of the Foundations

The reinforced concrete footings are described in Section 3.8.4 under the respective paragraphs which describe each Category I building. The foundation for the reactor core structure and the reactor shields are described in Section 3.8.3.3. The approximate size and depth of footing excavations are summarized in Table 3.7-3. The footings for each of the Category I structures are depicted on the sectional views of the respective building drawings. See attached drawings showing details of building construction and foundation.

3.8.5.2 Design Criteria, Codes and Specifications

Foundations were designed in accordance with requirements presented in the Pacific Coast "Uniform Building Code" and the general specifications listed in Section 3.8.1.2, including the American Concrete Institute, "Building Code Requirements for Reinforced Concrete" (Ref. 5 and 7, respectively). Specific subsurface conditions were determined by means of borings, test-pits and direct bearing tests. Allowable bearing pressures were kept below 8,000 lb/ft^2 for all structures, except that the maximum allowable pressure beneath portions of the 105-N and 109-N buildings, when considering earthquake effects, were specified as 10,700 lb/ft^2. Depending upon the building being designed, soil pressures were kept as uniform as possible for all of its footings in order to minimize differential settlement. Except for soil bearing pressures beneath the common wall (between the 105-N and the 109-N buildings), footings were designed for bearing pressure equal to about 2/3 or 1/2 the maximum of the 8,000 lb/ft^2 figure. Also, the size and shapes of the various spread footings were purposely designed to equalize bearing pressures in order to minimize differential settlement of the various building components.

The firm of Dames & Moore-Soil Mechanics Engineers performed a complete soils investigation of the N Reactor site. The results of this investigation are presented in Reference 49. Dames and Moore's recommendations for allowable bearing values are presented in Figure 3.8-32. They are considered to be conservative and to contain adequate safety factors to compensate for minor variations in the type and properties in the foundation soils.

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TABLE 3.8-2
PRIMARY CONFINEMENT STRUCTURE - ZONE I

105-N Building Zone I - Primary Confinement Zone (Restricted Access)

Locations	Remarks
a. Immediately above reactor	Confinement pressure limits of +5 psig to -2 psig. Operating pressure of 0.15" wg. Sealed doors and pipe penetrations. Complete filtration of exhaust air and release through stack only.
b. Inlet Face and "C" Elevator	
c. Discharge Face and "D" Elevator	
d. Pipe Spaces Containing Primary Coolant Piping.	
e. Zone I Ventilation Supply and Exhaust Ducts and Plenums	

109-N Building Zone I - Primary Confinement Zone (Restricted Access)

Locations	Remarks
a. Steam Generator Cells	Confinement pressure limits of +12 psig in cells to -2 psig. Operating pressure of -0.15" wg. Sealed doors and pipe penetrations. Normal ventilation exhausts to atmosphere. Ventilation isolation valves close for emergency confinement.
b. Auxiliary Equipment Cell	
c. Pipe Gallery	
d. Cell Access Corridor	
e. Pressurizer Penthouse	

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TABLE 3.8-8
DESCRIPTION OF WATER STORAGE TANKS

Water Storage Tank	Inside Diam. (ft)	Cylinder Height (ft)	Height of* Overflow (ft)	Tank Capacity (gal.)	Seismic Factor (g)	Roof Type
Filtered (FW)	65	40	34.5	850,000	0.1	Supported Cone t = 3/16 in.
Demineralized (DM)	75	32	27.5 (30.4)	1,000,000 Max. 350,000 Min.**	0.25	Umbrella t = 5/16 in. R = 60 ft
Emergency Raw Water (ERW)	35	30	16.5 (28.0)	119,000	0.25	Umbrella t = 3/16 in. R = 35 ft
Afterheat (AH)	50	36	34.1	501,000	0.1	Umbrella t = 1/4 in. R = 45 ft

NOTE: Tanks supported on concrete ring foundations filled with oiled sand.
Plate specification: ASTM A-283 Grade C.
Tank specification: AWWA No. D100-59

* The overflow heights in parentheses correspond to initial design conditions. The DM and ERW tanks are currently designed with overflows operated at the lower values of 900,000 gal and 119,000 gal, respectively, in order to satisfy 0.25 g (SSE) requirements.

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TABLE 3.8-10

EVALUATION OF SAFETY SYSTEM - SEISMIC ANALYSIS

SAFETY SYSTEM OR ELEMENT	COMPARISON OF ALLOWABLE AND PREDICTED PARAMETER	COMMENTS
1. Pressure Tube	$e_{allowable}/e_{predicted}$	
o Inlet/Outlet Shield and Reflector Zone	5.88	1. $e_{allowable}$ = 1% 2. No pressure tubes exceed the allowable strain
o Fueled Zone and Reflector Zone Interface	25.0	
2. Graphite Coolant Tube	$e_{allowable}/e_{predicted}$	
o Primary Shield and Reflector Zone Interface	8.33	1. $e_{allowable}$ = 1% 2. No coolant tubes exceed one percent strain.
o Fueled Zone Reflector Zone Interface	1000.0	
3. Ball Channels	$e_{allowable}/e_{predicted}$	
o Fueled Zone (Interlayer Interface)	1.29	1. Allowable horizontal displacement is 1.125 inch except for the interface between the primary shield and top of core where the allowable horizontal displacement is 1.25 inch. 2. No ball channels exceed the allowable displacement.
o Reflector Zone and Fueled Zone Interface	1.68	
o Concrete Pedestal and Core Interface	1.90	
o Primary Shield and Top of Core Interface	1.04	

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TABLE 3.8-10 (continued)

SAFETY SYSTEM OR ELEMENT	COMPARISON OF ALLOWABLE AND PREDICTED PARAMETER	COMMENTS
3. Ball Channels (continued)		3. Shield and core are "rigid" in vertical direction. Therefore, allowable vertical displacement of 0.5 inch is not exceeded. ($W_{allowable}/W_{predicted}$ = 250) 4. $W_{predicted}$ is obtained by a square-root-sum-of-the squares (in time) of the NS and EW responses.
4. HCR Channels	$W_{allowable}/W_{predicted}$	
o Primary Shield and Reflector Zone Interface	1.6	1. Allowable displacement in first 1.5 seconds of the earthquake is 1.0 inch for the Primary Shield and Reflector Zone Interface and 0.5 inch for the Reflector Zone and Fueled Zone Interface. 2. No HCR channels exceed allowable displacements.
o Reflector Zone and Fueled Zone Interface	6.8	
5. Graphite Keys	$P_{crack}/P_{predicted}$ $P_{Failure}P_{Predicted}$	
o Filler (or coolant tube) Block	9.4 11.6	1. Graphite Key Properties a. Coolant Tube Bloc P_{crack} = 4.67 kip P_{fail} = 5.76 kips b. Pressure Tube Blo P_{crack} = 4.5 kips P_{fail} = 5.67 kips 2. No graphite keys crac or fail
o Pressure Tube Block	3000.0 3800.0	

TABLE 3.8-10 (continued)

SAFETY SYSTEM OR ELEMENT	COMPARISON OF ALLOWABLE AND PREDICTED PARAMETER	COMMENTS
6. HCR T-Blocks	$W_{allowable}/W_{predicted}$	
	5.6	1. For translation, maximum predicted seismic displacement of a T-block is 0.09 inches, at t = 3.3 seconds.
		2. For overturning, the maximum translation before possibility of overturning is 0.8 inch. This was basis for conservatively defining allowable translational displacement as 0.5 inch.

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TABLE 3.8-11

STRUCTURAL INTEGRITY OF PRIMARY SHIELD - SEISMIC ANALYSIS

DISPLACEMENT OR STRESS	COMPARISON OF ALLOWABLE AND PREDICTED PARAMETER	COMMENTS
1. Relative Displacement Between Core and Shield	$W_{gap}/W_{predicted}$	
o NS Direction	2.4	1. $W_{gap}NS$ = 1.5 inches
o EW Direction	1.3	2. $W_{gap}EW$ = 1.5 inches
2. Maximum Seismic Stress in Primary Shield	$F_{predicted}/F_{allowable}$	
o Shear	1.19	1. Allowable stresses based on ACI 349-76.
o Bending	1.16	2. Allowables based on gross concrete sections and on minimum or no reinforcing steel.
3. Maximum Seismic Shear Stress in Reactor Core Pedestal	$F_{predicted}/F_{allowable}$	
	1.18	1. Allowable stresses based on ACI 349.76.
		2. Capacity does not include steel reinforcement; therefore, margin of safety is conservative.
4. Overturning of Primary Shield and Core	$F_{predicted}/F_{allowable}$	
o Primary Shield	1.2	
o Core	2.22	

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TABLE 3.8-12

SUMMARY OF RECOMMENDED CONCEPTUAL MODIFICATIONS

109N Bldg. Area	Recommended Modification Description	No. Required	Supporting Addendum Reference No.	Section No.	Page No.	Comment
3	Add lateral support	13	2	10.0	639	Attach steel frames to the confinement South wall at approximately elevation 36 feet
	Strengthen East-West braces; Column Line V	4	2	10.3	662, 660	Add structural angles to the back of existing double angles
	Strengthen Pipe Rack Steam Header Support Cross Brace (North-South, Area 3)	2	2	10.3	663, 659	Remove existing structural angles and replace with structural tube steel
	Strengthen Column Baseplate Weld at Column T, 59.9	1	2	10.4	669	Add additional weld material between column and baseplate
4	Add lateral support	10	3	8.1	526	Attach steel frames to the confinement South wall at approximately elevation 36 feet
	Strengthen East-West cross braces; Column Line V	16	3	8.3.2	649, 520	Add Structural angles to the back of existing double angles
	Strengthen cross brace at Column Line 71	2	3	8.3.2	652, 521	Add structural angles to the back of existing double angles
	Strengthen pipe rack steam header support cross brace E-W, Area 4	18	3	8.3.2	651, 521	Remove existing structural angles and replace with structural tube steel
	Strengthen Column-to-baseplate weld at Column T, 60.1	1	3	8.4.2	665	Add additional weld metal between column and baseplate

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TABLE 3.8-12

SUMMARY OF RECOMMENDED CONCEPTUAL MODIFICATIONS

109N Bldg. Area	Recommended Modification Description	No. Required	Supporting Addendum Reference No.	Section No.	Page No.	Comment
5	Add lateral support	8	5	10	448	Attach steel frames to the confinement East wall at approximately elevation 24 feet
	Strengthen Column-to-baseplate	1	5	11.6.2	582	Add additional weld metal between column and baseplate
6	Add lateral support	20	4	8.0	111	Replace existing cross braces
	Add column knee brace	8	4	8.0	111, 114	Add an additional knee brace, baseplate and anchor bolt assembly

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3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

During reactor operation, events such as startup, shutdown, scram and emergency cooling will create temperature, flow and pressure transients in the reactor coolant system (RCS). The following transients criteria and assumptions were utilized in the design and fatigue analysis of the various RCS components (Ref. 1). Subsequent review of operating conditions has demonstrated that the assumed design conditions were adequate.

1. Startup, Shutdown and Power Setback -- 2500 cycles during reactor life

   Temperature - Maximum rate of temperature change in any portion of the RCS: 5 degrees (F) per minute

   Pressure - Assumed constant during any transient: (System design pressure is 1825 psig, maximum. Pressures in this operating case were taken as 1600 psig in the rear outlet vertical headers and 1760 psig in the inlet vertical headers.)

2. Reactor Scram -- 1000 cycles during reactor life

   Temperature - Full-rod scram from equilibrium operation: Reactor cold-leg coolant temperature will remain essentially constant, while thermal transients occur in reactor hot-leg piping due to reduction of reactor delta-T. Coolant temperatures at the inlet, middle and outlet of an average length (52 ft) outlet connector during a scram are shown in Figure 3.9-1. Average metal temperatures at these connector locations are shown in Figure 3.9-2; whereas, average coolant and metal temperatures existing in an outlet, vertical riser are shown in Figure 3.9-3.

   Flow - Flow in a typical outlet connector during a scram is also indicated in Figure 3.9-1.

Pressure - Four injection pumps assumed available to limit pressurizer outsurge and pressure reduction: Operating pressure during a scram is shown in Figure 3.9-4 for the reactor outlet piping.

3. Emergency Cooling of Reactor Core - Six cycles (emergency dumps) during reactor life. System depressurization is due to a postulated, complete circumferential rupture of an inlet mixing header. Assumed operating conditions result in the greatest change in reactor inlet coolant temperature during transition to emergency cooling.

   Temperature - Average hot and cold-leg temperatures in the RCS are shown in Figures 3.9-5 and 3.9-6. Temperatures are based on maintaining saturated coolant temperatures corresponding to system pressure transients shown in Figure 3.9-8. Coolant temperatures at a typical inlet connector nozzle and at the inlet ECCS piping connection to the riser are also shown in Figure 3.9-6. Average metal temperature of an inlet vertical riser and at the riser connection to the ECCS are shown in Figure 3.9-7.

   Flow - ECCS flow during the emergency dump is shown on Figure 3.9-6. Flow in the hot and cold legs of the RCS is not specified due to the complexity of two-phase flow existing in the system following the postulated pipe rupture.

   Pressure - Pressure transients are shown in Figure 3.9-8 for two operating conditions. Emergency cooling starts when RCS pressure drops to about 325 psia, corresponding to the shutoff pressure developed by the high lift, emergency pumps.

4. Peak Temperature Transients Identified in Preliminary Design Calculations and by Tests - Each time the reactor is started up, operated for a period of time and then shut down, the reactor coolant changes temperature from cold to hot to cold. For design considerations, extreme temperature values for each cycle were defined ranging from 40°F for the cold condition to 600°F when hot. (Actual coolant temperatures are

usually greater than 80°F, when cold, and less than about 560°F, when hot.) For normal operating conditions, the maximum rate of temperature change during startup and shutdown was established at 300°F per hour for the RCS, except 200°F per hour at the RCS pumps. During reactor scrams the following design values were considered as being the maximum rates of temperature change of the reactor coolant: 80°F per minute and 2°F per sec. During an emergency dump the maximum rate of temperature change was established at 36°F per sec for 3 seconds.

Current operating limits for the N Reactor incorporated the normal temperature transients used in design, namely: 300°F per hour (200°F per hour for the RCS pumps) as the maximum rate of temperature change for the reactor coolant under normal operating conditions. These specifications and procedures do not provide limits for changes in reactor coolant temperature under scram or emergency conditions. However, the following maximum temperature changes of the reactor coolant have been observed during severe scram conditions: 50°F in 38 sec, 60°F in 1 minute and 100°F in 7 minutes without damage. During the hot dump test the bulk temperature of the reactor coolant dropped 100°F in 45 seconds and 225°F in 10 minutes. These observed temperature changes are significantly less than the maximum values established during design. Metal temperatures in the piping system lag behind observed changes in the temperature of the reactor coolant.

5. Operating Cycles - Design criteria recognized 2500 normal operating cycles during the life of the plant, plus 1000 scram cycles and six emergency dumps of the RCS.* A review of the operating records indicates that N Reactor was shutdown about 630 times from 1964 through 1984. Of these shutdowns, about one-third may be considered scrams and the rest normal shutdowns. When scrams occur from low power levels the thermal transients are less severe than under normal shutdown procedures. 15

*The original design criteria specified a design life of 25 years, minimum.

In any event, the total number of shutdowns (630) for 21 years of reactor operation is less than the 840 scram cycles (obtained by multiplying 1,000 times 21/25). The total number of shutdowns (630) is also significantly less than 2,940 cycles obtained by adding 2500 normal and 1000 scram cycles and multiplying the sum by 21/25. These data indicate that the original design bases for specifying operating cycles were conservative, and that the total cycles provided for in the original design are not likely to be exceeded during the remaining life of N Reactor.

The plant experienced one "planned" emergency cooling cycle during the hot dump test, which was conducted prior to power-mode operation of the N Reactor. Based upon approximately 20 years of past reactor operation, without having to resort to an emergency dump, it is unlikely that six emergency dumps will be experienced during the life of the plant.

3.9.1.2 Computer Programs Used in Transient and Fatigue Analyses

Transient thermal analysis was accomplished with the use of a computer program developed by Electric Boat Division of General Dynamics Corporation (Ref. 2). The program, written for an IBM 704 computer, calculates the transient temperature distribution, stresses and displacements in a long hollow cylinder due to thermal shock, in which the thermal shock temperature is an arbitrary function of time. The computer program was used to investigate the structural integrity of the connector nozzles, connectors and the inlet and outlet risers. Transient temperatures, free-body stresses and nozzles deflections were computed initially. Transient deflections were then used to obtain discontinuity stresses throughout the nozzle geometry. A fatigue evaluation was then performed in accordance with procedures presented in "Naval Reactor Tentative Structure Design Bases", (Ref. 3). The latter evaluates both the thermal shock stresses in a section of pipe undergoing step and ramp coolant temperature changes, and thermal stresses in tees and nozzles subjected to temperature gradients.

The transient thermal analyses indicated that the inlet connector nozzles were more highly stressed than the outlet connector nozzles. Final detailed

calculations were subsequently performed for a typical inlet nozzle, which provided for 2500 normal (startup and shutdown) cycles and six emergency cooling cycles (Ref. 2). The single nozzle analysis was considered to be at least representative of the most critically stressed nozzle on the inlet face of the reactor. The highest stress was found to occur on the outside surface at the fillet radius junction (between the nozzle and the inlet header) at the end of the sixth emergency cooling cycle. The combined usage factor for this condition and 2500 normal operating cycles was only 0.041. Since this value is significantly below the allowable usage factor of 0.800, the nozzle design was shown to be safe for the thermal conditions summarized in Section 3.9.1.1. The adequacy of the connector design and the calculational methods used in 1962 to evaluate thermal transients, have been verified by operating experience since 1964 in which the connectors have performed satisfactorily with no evidence of stress-imposed problems.

3.9.1.3 Experimental Stress Analysis of Connector Piping

Four representative connectors (No. 2, 5, 12 and 17) were load-tested in 1959 at the 189-D mechanical laboratory (Ref. 4). Each connector was cold-deflected a predetermined amount to simulate thermal expansion loads. Cold-deflection loadings caused piping bends, oval-shaped in cross-section, to become more so when subjected to load. This caused additional compressive stresses on the inside of the bend. However, maximum stresses measured from each of the connectors were within design limits and varied from 10,320 psi to 21,000 psi, under the superimposed deflection-loads. Care was taken to assure that each connector was in a stress-free condition when placed in the mockup jig. Stress coat (Magnaflux Corp. type ST-103A) and strain-indicating lacquers were applied first to evaluate stresses at all possible points of high stress on each connector and for determining final locations to install SR-4 strain gages. The SR-4 strain gages were then attached to the connector being tested and the deflection-loads applied. Initially, the connector was loaded and unloaded several times until the strain-gages had stabilized. Once stabilized, a series of five deflection loads were applied, strain gages read and then averaged to give the indicated strain.

Besides confirming that thermal expansion stresses would be within design limits, the tests indicated that additional supports were required for some of the longer connections in order to minimize sag. The design of the connector-support structure was subsequently modified to accommodate this feature.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

As discussed in Sections 3.2 and 3.7, during the development of the N Plant design (1958-1962), the 1955 and 1958 editions of the Uniform Building Code (UBC) Zone 3 requirements were used for the structures and systems needed for safe shutdown, emergency core cooling and confinement. In addition to the UBC Zone 3 requirements, additional conservatism was introduced by, in general, treating the faulted load conditions and stress requirements as "normal" design conditions. Thus the analytical methods used to evaluate the design did not depend upon use of elastic-plastic considerations.

However, in the seismic re-evaluation of these safety-related structures, systems and components conducted during 1985, 1987, and 1988 the analytical methods did utilize currently accepted limits and assumptions based on elastic-plastic stress and deformation limits. The general methodology is discussed in Section 3.7 and detail particulars are provided in the referenced reports prepared for each of the Seismic Category I structure, system and component that was reanalyzed for capability to withstand the SSE defined in SDC 4.1 Rev. 10 (Ref. 24).

Seismic reanalysis capability is discussed in Sections 3.9.2.2, 3.9.3.1 and 3.9.3.4 for mechanical components and 3.10.3 for instrumentation and electrical components.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The behavior of the reactor coolant piping located in the 105-N and 109-N buildings was evaluated by extensive preoperational testing which

utilized the following equipment and systems for determining piping stresses, movements and vibration characteristics:

1. Strain gages for evaluating piping stresses;
2. Dial indicators for measuring rotations and/or displacements of the mixing headers located in the 109-N pipe gallery and of the RCS pumps and piping located in the steam-generator cells;
3. Motion-indicator boards for making a permanent record of pipe displacements during operational and thermal cycling;
4. Motion-indicators provided on spring and constant-load pipe supports and hangers; and
5. Portable velocity pickups and/or accelerometers for measuring frequencies and oscillatory displacements.

In order to evaluate the entire piping system, a variety and a large number of instruments were required. Inlet and outlet piping loops No. 9 and 16, located in the 105-N building, were instrumented extensively, as were reactor coolant supply and return lines located in steam generator cells 2, 5 and 6. Instrumentation was also installed at several control points on the remaining six inlet and outlet piping loops, located on the right side of the reactor block, and on each of the hanger rods which support the loops at the lower trunnions. Likewise, instrumentation was installed at control points on reactor coolant piping located in steam generator cells 1, 3 and 4. The following numbers of instruments were utilized for the testing performed initially in 1964 and indicate the magnitude of the test program: 273 weldable strain-gages, 54 dial indicators, 113 motion-indicator boards and 308 motion-indicators on spring hangers and movable pipe supports. Motion-indicator boards, dial indicators and strain-gages were subsequently installed on: (1) the high pressure flush line for monitoring displacements and strains during the hot dump test, (2) the steam mixing headers located on the roof of the 109-N building and (3) the riser in the pipe gallery which connects to the pressurizer vessel.

Weldable-type strain gages were installed on pipe surfaces with a capacitive-discharge welder which did not damage the pipe surface. Lead-wires were

factory installed and hermetically sealed to each of the gages. This type of installation was required to meet service conditions which included moisture resistance at low temperatures and accuracy at temperatures up to 600°F. Instrument wires were collected, banded and run through several sealed penetrations through the confinement walls, to one of five read-out stations. Individual gages were calibrated and identified by switch number. Strain readings were taken manually using typical SR-4 strain-readout equipment manufactured by Baldwin-Lima-Hamilton (BLH).

Dial indicators were manufactured by Starret and were of four types, which provided maximum ranges of 0.20, 0.50, 1.00 and 3.00 in. respectively. The first gage listed above had 0.0001-in. graduations; whereas, the others had 0.001-in. graduations.

The motion-indicator boards were a unique installation and provided a permanent record of pipe displacement in a horizontal plane during a given operational period, which usually included several cold-hot-cold, thermal cycles. Each installation included a 8-1/2 by 11-in. drawing board, provided with hold-down clips for inserting paper. The board was attached to a metal frame, which, in turn, was welded to the fixed portion for a particular pipe support. A piece of tubing was located immediately above the board, in a vertical position, and attached to the moving part of the same pipe support, or to the pipe, by a mounting arm. An ordinary F-drawing pencil was then inserted in the tube so that the point of the pencil rested on the record paper, which had previously been clipped to the fixed drawing board. A weighted follower was then inserted on top of the pencil to maintain contact. The data and N-S axis were shown on each record paper for reference. Beside providing a trace indicating relative thermal movements between the pipe and its fixed support, the record provided information which showed whether or not the pipe was vibrating at the particular location and the magnitude. Upon completion of a particular test run or operational period, the record paper from each motion indicator board was removed and a new, dated sheet installed for the next run. Pipe displacements were monitored in this fashion during preoperational testing and for most of the first year of reactor operation.

While performing the acceptance test of the reactor coolant system, pipe displacements and strains were determined at the following temperatures (F) of the reactor coolant: 70, 300, 450, 525 and 70; 70, 490, 250 and 70; 70, 440, 70 and finally, 70, 500, and through the transient dump test.

The portable, vibration measuring equipment, which included a Raydata Vibration Analyzer, was used extensively during the first phases of preoperational testing. In these preliminary tests, cold water was pumped at increasingly high velocities through representative piping loops, which included connector piping and inlet and outlet loop No. 9, followed by high flows through several piping loops, simultaneously. See Section 3.9.2.4. Pipe motions and vibrations were monitored carefully during these tests to assure that the respective piping loops would not vibrate dynamically in resonant conditions, undetected by vibration calculations and computer analyses performed at an early date. This equipment was used later in the test program to evaluate vibration characteristics of isolated sections of reactor coolant pipe which, during previous test runs, had been observed to vibrate significantly or which were suspect. The vibration data in these cases were used to arrive at modifications required to eliminate the particular problem.

Strain-gage data were used to compute piping stresses due to combined pressure and thermal conditions. A digital computer program was utilized to translate strain readings into stresses. These stress values were compared with allowable design values and to stress values obtained previously during computer analyses performed by Electric Boat Division of General Dynamics Corporation. See Section 3.7.3. Stresses obtained from the strain data were in general agreement with those obtained during design, although somewhat lower in magnitude at a particular system temperature. The highest stresses, which approached design limits in magnitude, were observed in the RCS hot leg piping at two locations: in the T-section at the north end of the cell and in the section of piping which supplies hot coolant to the bottom of the steam generator most distant from the mixing-headers.

Horizontal pipe displacements, obtained with the motion-indicator boards, were also in general agreement with those predicted by the original design calculations and computer analyses. However, vertical displacements of several horizontal runs of piping, in the 109-N pipe gallery south of the mixing headers, were somewhat greater than originally provided for in the design. This condition was adequately accommodated for by replacing a number of trunnion or sliding-bearing supports with spring supports or constant load hangers. Data obtained from the motion indicators were used to "fine tune" the pipe supports for the entire reactor coolant system, and to assure that the various pipe supports and hangers had adequate range to accommodate thermal expansion and contraction movements experienced at the particular locations. Any deficiencies in the pipe supports were corrected prior to operating the reactor with nuclear fuel. Deficiencies were of a minor nature, except for those noted above.

Most recently, 1987-88, as part of the Return to Service (RTS) program, all Category I piping of ASME Class 1, 2 and 3 identified as required for RTS was reanalyzed for compliance to the defined SSE and the resulting combined loads. Results for Category I pressure piping systems are summarized in Section 3.9.3.1.5. Category I structures, systems and components as currently defined following qualification and upgrading are listed in Section 3.2 in Table 3.2-1.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Seismic qualification testing was performed on the following safety-related mechanical systems during 1968 and 1969:

1. Two components of the horizontal control rod system;
2. Critical components of the ball safety system and
3. The solenoid control valve for the V-3 and V-4 cone valves, used in the emergency core coolant system.

Vibration testing was performed on prototypes of the various mechanical systems which had been set up in the 1703-D development laboratory (Ref. 5). Two components of the horizontal control rod prototype were evaluated

dynamically. The drive mechanism was oscillated within a frequency range of 4 to 23 cps (hertz) and at accelerations which varied from 0.4 to 9.0 g. A rod scram during each vibration mode showed normal operation and compliance with design requirements. The rod was also scrammed after the drive chain had been intentionally loosened and jammed in an offset position. This simulated a possible chain position that might be attained while whipping back and forth during an earthquake. As the scram commenced, the chain tension became great enough to reposition the chain to its normal operating location. Scrams were completed without problem during all tests.

Critical components of the ball safety system were evaluated while being subjected to induced vibrations, including the ball-drop gate, the closing and holding solenoid, and the ball-drain valve. The first two components are critical to safe operation of a given ball hopper, while the latter holds the poison-containing balls in each reactor control channel after a ball drop. The ball-drop gate was subjected to a frequency range of 6 to 37 cps and forced accelerations which ranged from 0.5 to 3.2 g. The closing and holding solenoid was vibrated with a frequency and acceleration range which varied from 16 to 105 cps and 0.9 to 5 g, respectively. The two components were vibrated simultaneously with no inadvertent trips or other failures. In all cases, the hopper actuated on command and within design requirements.

A solenoid control valve, typical of those used to actuate the V-3 and V-4 valves, was vibrated with a maximum acceleration of 9 g at 43 cps and at a minimum of 0.1 g at 4 cps. The solenoid was actuated while in several different vibrating modes with no problems noted. It actuated on command and with no inadvertent trips or other failures.

During the seismic qualification testing described above, vibration frequencies and accelerations were more severe than would be anticipated for a postulated SSE with an intensity of 0.25 g earthquake. The critical components of the safety-related equipment operated adequately, and without failure, during the vibration tests.

3.9.2.2.1 Seismic Reanalysis Program

Most recently, 1986-88, seismic qualification testing of safety-related mechanical and electrical equipment to the requirements of SDC 4.1 Rev. 10 (Ref. 24) was addressed as part of the N Reactor Seismic and Environmental Equipment Qualification (EQ) program administered within the Accelerated Safety Enhancement Program (Ref. 25). Results of the Equipment Qualification Program are discussed in Section 3.11.1.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

N Reactor does not have a reactor pressure vessel. Discussion of the dynamic responses of the pressure retaining components of the reactor core is given in Sections 4.0 and 5.0. Also see Sections 3.9.1.4 and 3.9.3.1.5.

3.9.2.4 Preoperational Flow-Induced Vibration Testing

Early in the acceptance testing of the reactor coolant piping system, isolated portions of the piping system were subjected to maximum flow rates and velocities at room temperature. Vibration measurements were taken at the following locations when the reactor coolant pumps were operating at maximum operating rated speed (3600 rpm):

1. several inlet connectors;
2. corresponding outlet connectors;
3. at each of the spring hangers which support the horizontal runs of inlet headers No. 9 and No. 16, and outlet header No. 9;
4. vertical runs of inlet header No. 9 and outlet header No. 9, at the trunnion supports and at El 432.5 ft.

Vibration readings were taken at the pressure-tube nozzles and at points on the connectors near the headers. Vibration readings were initially taken at a number of control points as pump speeds were increased from 2000 to 3600 rpm. Measurements and observations of piping vibrations indicated no tendency for dynamic amplication at this range of pump speeds. At 3600 rpm, the following flow conditions existed in header No. 9 and connectors 3358 and 3359, respectively. It is noted that flow and velocity value reached during

the cold-flow test are substantially greater than corresponding values obtained during normal periods of reactor operation.

	During Cold-Flow Test	During Normal Reactor Operation
	Headers	
Flow (gpm)	19,500	12,000
Velocity (ft/sec)	32.8	20
	Connectors	
Flow (gpm)	309	200
Velocity (ft/sec)	28.2	18.5

Maximum vibratory displacements (0.001 to 0.0015 in.) were measured with the Raydata Analyzer for the inlet and outlet headers when the pump speed was approximately 3600 rpm. Smaller (peak) displacements occurred at slower pump speeds. The headers responded primarily at a frequency equal to about one-sixth of the pump speed; whereas, the connectors responded at a frequency equal to about one-third the pump speed to as high as one times the pump speed, depending on their length and configuration.

During various full flow and isolated flow tests of the RCS piping, using cold water, the behavior of piping components and supports was observed visually and also evaluated instrumentally. Amplitudes up to 8 mils were observed with water velocities of 30 fps in portions of loop No. 16, which was subjected to the most flow. During one test, in which the pump speed was around 3200 rpm, the bypass line in the northeast end of steam generator cell No. 5 vibrated severely, although the amplitude of vibration was small. This situation was subsequently corrected by readjusting the load taken by the three spring-hangers which support the bypass line. Various lengths of field-run pipe (less than 3 in. in diameter) were observed to vibrate more than desired, as well as the 8-in., low-pressure flush line, located in the outlet pipe space of the 105-N building. These problems were corrected by installing additional pipe supports on the respective piping runs. In general, the preoperational tests demonstrated that with a few exceptions, which were subsequently corrected, that the piping support system was adequate

and that it would be safe to proceed with later tests which required operation at higher temperatures and pressures.

Observations of piping system behavior made during the remaining startup tests and during the first year of nuclear operation, demonstrated excellent stability, except for problems that occurred during and following the hot dump test. Strain gages, attached to the high pressure dump line indicated a peak stress of 14,550 psi, which occurred at the beginning of the hot dump. One hundred minutes later, the stress had dropped to 3,640 psi and to only 1,400 psi 9 hours after the dump. All other monitored strain gages indicated maximum stresses were less than 3,000 psi. Likewise, vibrations of the dump line were innocuous during the dump. However, thermal expansion forces damaged a set of rollers on one support, which apparently bound up during the test. This problem was corrected by reinforcing the particular structural members which supports the expansion rollers.

The RCS piping at the west side of the pipe gallery and in the steam generator cell was subjected to steam water hammer phenomena when cold water was injected back into the system following the hot dump. This shocking of the system damaged some wall supported pipe supports by loosening the anchor bolts which tied the anchor plates to the building walls. The particular bolts had been installed with concrete inserts, which had to be replaced by larger units. Wall plates that had been attached with anchor bolts, which were originally embedded in the walls when the concrete had been poured, were not damaged. The forces experienced by the piping system and its supports following the hot dump test are believed to be more severe than those associated with a 0.25 g earthquake.

As an adjunct to the N Reactor fuel and component development program, a test facility which permits the simulation of in-reactor conditions of coolant flow, temperature, chemistry, pressure, etc., was constructed (reactor loop facility). The facility consists of a single reactor pressure tube, equipped with inlet and outlet nozzles; a piping loop which is fitted with the same valves and flow and pressure measuring devices as are used in-reactor; a pump which is capable of maintaining a total system flow in excess of normal

reactor demands and a coolant heating system. The test facility is well instrumented and is equipped with the controls necessary to permit operation for long periods of time on an automatic basis.

The test facility was and is currently used to test the performance of the various fuel element and spacer designs in full column loadings under simulated isothermal reactor operating conditions. Included in the tests was a determination of whether or not flow-induced vibration occurred along the fuel column length. Instrumentation was provided to detect any vibration or fluttering of the respective fuel column elements. After a particular test, visual examination of the various fuel pieces and spacers, which made up the test column, confirmed if vibration had occurred and determined the location within the column and its severity.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

Not directly applicable because N Reactor has no reactor pressure vessel. Pressure piping was reanalyzed for the SSE in 1987-88. The results are given in Section 3.9.3.1.5. 21

3.9.2.6 Correlations of Reactor Internal Vibration Tests With the Analytical Results

Not applicable, but see Section 3.9.3.1.1.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The original N Reactor primary coolant and associated special quality pressure systems components were procured, fabricated, and constructed in accordance with the requirements of the following codes and specifications. 21

- Specification HWS (Hanford Work Specification) 6897, 1960-61.
- Section I, ASME Boiler and Pressure Codes 1956-1959 Editions.

- Section VIII, ASME Boiler and Pressure Codes 1956-1959 Editions.
- ASA B31.1, American Standards Association Piping Code, 1955 Edition (Ref. 6) (now ANSI B31.1).

These codes provided the basis of design and allowable stresses under sustained loadings such as coolant pressure, deadweight and thermal expansion during heatup. The ASA code also establishes values of minimum wall thickness. It does not cover thermal stresses caused by rapid cooldown or heating of the respective piping systems, cyclic fatigue or earthquake. Transient thermal effects and fatigue are discussed in Section 3.9.1; whereas, earthquake effects on piping systems are included in Section 3.7 and 3.9.3.1.5.

As discussed in Sections 3.2 and 3.9.6, the ASME Section XI, 1977 ASME Edition - W79 Addendum was adopted, with DOE concurrence, as the basis code for inservice inspection, repair, and replacement of Class 1 systems of the N Reactor. Regulatory Guide 1.26 and the ASME 1983 code and addendum was used to later classify the N Reactor safety-related systems according to the ASME Class 1, 2, and 3 categories (Ref. 70). This section lists the major ASME systems/components identified as Class 1, 2, or 3 in Tables 3.9-1, 3.9-2, and 3.9-3. All other systems not covered by Classes 1, 2, and 3 are listed in Table 3.9-4.

The reactor coolant system (RCS) has been identified as a Quality Group A (Class 1 - Section III) system. That is, if it was being constructed at the present time Section III, Class 1 construction would be required. In addition, the requirements of Section XI for Class 1 systems would be invoked.

Since 1978 piping system modifications have conformed to American National Standards Institute (ANSI) Power Piping Code B31.1, 1977 (Ref. 21) and subsequent revisions (for example: Ref. 22) and criteria. Also, HWS 8732 "Specification for Carbon Steel Pipe Fittings, Castings, Forgings, 3/4 in. Studs and 3/4 in. Nuts for Nuclear Pressure Retaining Systems - N Reactor" is used in procurement. Moreover, the ASME "Boiler and Pressure Vessel Code," Section III may be used as an alternative to the original construction code.

3.9.3.1 Loading Combinations and Stress Limits

3.9.3.1.1 Pressure Vessel - Not Applicable to N Reactor
Startup, steady-state, power setback, other transients, and shutdown conditions are discussed in Section 3.9.1.1.

3.9.3.1.2 Reactor Coolant System (RCS)
ASME codes and code cases are discussed in Section 5.2.1, reactor coolant pressure boundary (RCPB) materials in 5.2.3, pressure tube design in 5.3, and major RCS components design loadings in 5.4. RCS design criteria are listed in Table 5.4-9. 21

3.9.3.1.3 Emergency Core Cooling System (ECCS)
The ECCS design codes and classifications and material specifications are discussed in Section 6.3.2.

3.9.3.1.4 Pressure Piping
The RCS piping design basis and materials specifications are given in Tables 5.4-7 and 5.4-8. The ECCS piping design parameters are listed in Table 6.3-1. The following paragraphs discuss the application of ASA B31.1 in the design of pressure piping at N Reactor.

Wall thicknesses for pressure piping were based upon the following equation, as required by ASA B31.1:

$$t_m = \frac{PD}{2S + 0.8\ P} + C$$

where t_m equals the minimum pipe wall thickness (in.), P equals the coolant design pressure (psig), D is the outside pipe diameter (in.), S is the allowable stress (psi) and C equals the minimum required corrosion allowance (in.). The design pressure for the reactor coolant system (RCS) was 1,825 psig. The allowable working stress is 17,500 psi for ASTM A155 KC70 pipe fabricated with A212 Grade B Plate, and 15,000 psi for ASTM Grade B pipe. Although 0.065 in. is the minimum corrosion allowance specified in

ASA B31.1, 0.090 in. was provided in the design of the connectors at N Reactor and 0.125 in. for the remainder of the larger sections of pressure piping. Wall thicknesses provided for all pressure piping was greater than the minimum specified by the above equation.

In addition to the hoop stress caused by coolant pressure, bending moments caused by dead weight (total pipe weight, including coolant, valves, system components and insulation) can cause significant longitudinal stresses in long (unsupported) sections of piping. Longitudinal stresses caused by dead weight were combined with longitudinal stresses caused by internal coolant pressure. The total longitudinal stress was also kept below the allowable stress given in the design code. However, this condition usually did not control the design as pipe supports and hangers were placed closer together, than otherwise required, in order to avoid potential resonant conditions sympathetic to earthquake ground motions. About 100 constant support hangers, in addition to regular pipe supports and hangers, were installed to support RCS piping runs in order to control and minimize dead-load stresses. Computer analyses demonstrated that dead weight and pressure-induced stresses do not exceed the allowable stress permitted by the code.

With respect to thermal expansion, ANSI B31.1.0 requires that the maximum calculated thermal expansion stress, S_E, shall not exceed the allowable stress range, S_A, defined by the following expression:

$$S_A = f(1.25\ S_c + 0.25\ S_h)$$

where S_c equals the allowable stress in the cold condition (psi), S_h is the allowable working stress in the hot condition (psi) and f is a stress range factor and is equal to 1.0 when the expected number of stress cycles is less than 7,000. The ASA code also provides the basis for calculating expansion stresses in piping components by defining values of flexibility and stress intensification coefficients.

It is noted that the allowable stress range (S_A) is defined as the expansion stress calculated for the hot condition minus that for the cold condition

and is larger than the allowable working stress. Thermal expansion stresses, if of sufficient initial magnitude, will be relaxed due to yielding or creep. Stress reduction will then take place in the hot condition but will reappear as a stress of reversed sign in the cold condition in an amount equal to that reduced by yielding in the hot condition. Thus, thermal expansion stress is not limited by stress occurring from static loads. For this reason, static and expansion stresses are calculated and evaluated under separate allowable stress criteria. The code also allows the difference between the allowable hot working stress (S_h) and the static load longitudinal stress (pressure plus dead load) to be added to the value of S_A defined above. Therefore, extra allowable expansion stress is available when the longitudinal stress due to static loads is below the allowable working stress. A similar approach was used to evaluate seismic stresses in combination with stresses due to thermal expansion, dead weight, and pressure. See Sections 3.7.3.2 and 3.9.3.1.1. 21

Pressure piping systems in the 105-N building were designed by General Electric Company; whereas, those in the 109-N building and other portions of the N Reactor plant were designed by Burns & Roe, Inc., using the same design criteria and codes. (However, the latter design firm chose to incorporate small amounts of cold-spring into the various piping loops of the RCS located in the 109-N building due to their relatively long, horizontal runs.) In the early stages of design, thermal expansion stresses were calculated by methods of piping flexibility analysis developed by commercial firms such as Tube Turns, Inc. and the M.W. Kellog Co. At that time, the Kellogg method was programmed for an IBM-650 computer and was somewhat limited in scope. Later, a more powerful program based on the matrix method of analysis (developed by J.E. Brock, Ref. 7) became available for the IBM-7090 computer. The latter program was used by General Electric Company and Burns & Roe, Inc. to evaluate stresses due to dead weight, pressure and thermal expansion in the final piping layout.

An independent and comprehensive review of the stresses in the RCS system, located in the 105-N building, was performed by Electric Boat Division of General Dynamics Corp. after the detailed design was fairly complete.

Electric Boat utilized the matrix method of piping flexibility analysis and an IBM-704 computer (Ref. 8). This program had sufficient capacity for detailed analysis of the complex piping configurations of the various RCS piping loops and connectors. The Electric Boat analyses, which were performed independently of the previous design calculations, found only three instances where stresses exceeded those allowed by the ASA Code. These overstressed conditions were corrected by revising the piping configuration to obtain additional flexibility in the affected areas.

3.9.3.1.5 Seismic Reanalysis of ECCS and Other Category I Piping

In 1987 and 1988, in response to recommendations by the Westinghouse Independent Safety Assessment Senior Review Committee (Ref. 30), a Seismic Task Force was organized to review all the safety class systems for seismic capability for the SSE defined in SDC 4.1 Rev. 10 (Ref. 24). The scope included previously identified ASEP seismic issues (Ref. 25) and all Seismic Category I structures, systems, and components and included III/I walkdowns to identify and correct any potential hazards to safety class systems (1) resulting from earthquake induced failures of non-safety class systems (3) (see Table 3.2-1 for listing of Category I items).

The results of the seismic reanalysis of structures are discussed in Section 3.8. The results of the reanalysis of the Category I piping systems are summarized herein. The piping systems evaluated included:

- Emergency Core Cooling System (ECCS) (includes RWS-2 portions)
- Graphite and Shield Cooling System (GSCS) - ECCS backup portion
- Fog Spray System (FSS)
- Reactor Cooling System (RCS) - Non-ECCS portion
- Confinement System (Steam Release Vents/Weatherheads/Vacuum Relief Valves).

The Seismic Task Force issued over 117 reports in total with about half of these devoted to the evaluation, analysis, and modifications to the above piping systems. Two summary reports (Ref. 26, 27) issued in the latter half of 1987 provide overview summaries and status of work accomplished at that

point. The reports also identified specific evaluations still required. The methods used in assessing and resolving the III/I problems comply with general industry practice. Field walkdowns were conducted to identify hazards (Category III) and targets (Category I). Screening techniques (Ref. 28), such as zone-of-influence and impact energy capability, were applied to resolve many potential problems. If screening rules failed, additional analysis was performed to assess the potential for failure. If the problem was not resolved, a fix was designed and analyzed. The analysis guidelines and procedures followed are given in Reference 29.

Each of the above piping systems was walked down by a qualified walkdown team. Potential problems were identified, documented, analyzed, and evaluated. If a modification or fix was required, the fix was designed and implemented. All fixes identified are required to be completed prior to restart of N Reactor. All of the safety-related portions and components of the above piping systems have been qualified to the SSE criteria of SDC 4.1 (Ref. 24). A brief listing of the scope of review and pertinent documentation addressing the analyses performed and fixes implemented follows. 21

A. ECCS - portions evaluated and upgraded are listed below.

- Three low-lift pumps, their diesels and auxiliaries (cooling water, jacket water cooling, lubricating oil, diesel oil, starting air, air intake and exhaust systems); inside 181-N.

  The analysis (Ref. 31) and six appendices totaling 30 volumes report the evaluation of 18 primary and auxiliary piping systems. A total of 58 fixes to the supports were implemented. The III/I walkdown in 181-N produced 229 problems, 88 that required fixes (Ref. 42).

- Three high-lift pumps, their diesels and auxiliaries (cooling water, jacket water cooling, lubricating oil, diesel oil, starting air, air intake, and exhaust systems); inside 182-N.

The analysis (Ref. 32) and 8 appendices totaling 51 volumes report the evaluation of 34 primary and auxiliary piping systems within Building 182-N. A total of 87 modifications to supports were designed, approved, and implemented. The III/I walkdown in 182 resulted in 287 ECCS problems and 127 fog spray problems, of which 166 required fixes (Ref. 43).

- o Buried diesel fuel piping from the bulk storage tank farm to the day tanks, including piping routed through the 166-N Building and the fuel oil unlocating trench, and from the day tanks to the 181-N and 182-N Buildings.

  The analyses (Ref. 33 and 34) reports on the evaluation of buried fuel oil piping from the bulk storage tank farm to the day tanks and through the 166-N Building and fuel oil unloading trench. Thirteen potential problems were resolved and 2 new supports were installed in the 166-N Building and 16 existing supports in the trench were modified. The lines from the day tank to 181-N and 182-N were evaluated in U-12010 (Ref. 35). The day tanks and supply and return lines were determined to be acceptable. One 1.5-in. line penetrating the 181-N Building required replacing hard grout with a soft material to provide some motion.

  Increased response spectra for the two buildings necessitated a re-evaluation of the diesel fuel oil lines in 166-N, 181-N, 182-N and the unloading trench. Analysis documents (Ref. 36) and (Ref. 37) reported the results.

It was concluded that the conclusions reached in the original analyses were still valid for the following components based on the revised response spectra:

1. Fog Spray Accumulator Tank in 182-N Building (Ref. 40)
2. Potable Water Tanks in 182-N Building (Ref. 38)

3. Air Receivers in 182-N Building (Ref. 39)
4. Diesel Fuel Oil Piping in the 166-N and 182-N Buildings and in the unloading trench (Ref. 33).

Fixes previously implemented for the Fog Spray Accumulator Tank (DC 87230), the air receivers (DC 87318), and the Diesel Fuel Oil Piping in the 166-N Building (DC 88031) and 182-N Building (DC 87266) are adequate to qualify these components for the Safe Shutdown Earthquake (SSE) with the revised response spectra. The Potable Water Tanks which are Seismic Category III do not pose a III/I hazard with the revised response spectra.

It was determined (Ref. 37) that the diesel fuel oil piping in the 181-N Building required the installation of twenty-six (26) new pipe supports and the modification of thirty-nine (39) existing supports. The new supports and the modification of existing support was accomplished by Design Change 87266 as follows: 21

- 105-N ECCS Inlet Manifold to V3 Valves and V3, V4 Valve Operating Air Supplies

  Forty one potential problems were identified and resolved in U-4506 Task 1 (Ref. 41) and 13 fixes made

- Distribution System from V3 Valves to Reactor Inlet Risers

  Six potential problems were identified and resolved in Task 2 (Ref. 41) with 2 fixes made

- Reactor Inlet System from Reactor Inlet Risers to Pressure Tubes

  Eight potential problems were identified and resolved in Task 3 (Ref. 41), one fixed was required

- Reactor Outlet System from Pressure Tubes to V4 Valves Including RV-2 Relief Valves and CV-3 Check Valves

  Seventeen potential problems were identified and resolved in Task 4 (Ref. 41), nine fixes were made

- Outlet System from V4 Valves to Emergency Dump Tank Overflow Pipe

  Seven problems were identified, all were resolved by analyses in Task 5 (Ref. 41).

The above five tasks including the ECCS supply lines entering the 105-N/109-N Buildings were also included in the RCS analysis discussed below in paragraph D, and reported in References 73 and 74. In that analysis, six new supports at the roof area were identified and installed plus a number of fixes.

- ECCS Buried Piping, Supply Piping to and Discharge Piping from the N Reactor

  SD-NR-DCA-88019 (Ref. 42) provides the structural evaluation of seismic loading on the large diameter buried piping that is part of the ECCS and FSS. Evaluations for the following lines are provided:

  1) RWS lines from Building 181-N to Building 182-N, including branch supply lines to ECCS tank bypass (silo) and ERWS tank
  2) RWS line from ERWS silo and DW tanks to Building 182-N, including branch connections to RWV-805 and FWV-817 isolation valves
  3) Primary ERWS line from Building 182-N to Building 105-N
  4) Overflow line from the Emergency Dump Tank 1304-N to 1301-N weir box and the branch line to the 105-N lift station penetrations
  5) Flush line from Building 109-N to Emergency Dump Tank 1304-N

6) RWS-2 supply line from Building 182-N to Building 105-N
7) Vent line from Emergency Dump Tank 1304-N to Building 109-N
8) RWS fog spray line in the valve pit between Building 182-N and circulating water pumphouse (check valve RWSV-551)
9) Soil stiffness calculations
10) RWS-2 supply line from 36 in. diameter line by 182-N Building to 109-N Building floor (Auxiliary Cell)
11) RWS-2/RW-1 fog spray line from 30 in. diameter line by 109-N Building (East face) to 109-N Building floor (Pipe Gallery).

Also included in this report is the seismic and tornado evaluation of the following demineralized water tank piping.

1) 12 in. DW-3 overflow piping including 6 in. branch connection.
2) 6 in. IW-3 supply line.
3) 4 in-MP, 4 in.-MPr and 3 in. nozzle connection.
4) 18 in DW-1 line including 14 in and 8 in. branches up to DWV-804-1 and DWV-805-1 respectively.
5) 12 in DW-1 and DWV-801-2 including 10 in. branch up to DWV-812-1.
6) 1-1/2 in. line nozzle connections.

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All lines evaluated in this report are within acceptable stress levels. The system is capable of withstanding a Safe Shutdown Earthquake (SSE) and will remain functional to deliver the intended water supply to the ECCS, FSS and 36 in. demineralized water supply line to the high-lift pumps. The effects of corrosion on the buried piping and tornado loading were assessed and concluded to not compromise the structural integrity of the piping.

A minor fix for two supports on the 12 in. DW-3 Demineralized Water Tank Piping was designed and implemented.

The 1301-N weir box was separately analyzed (Ref. 67) for seismic capability. Results of the analysis indicated that the structure, prior to the addition of the weir box cover, would be overstressed. However, with the cover the structure would have a considerable increase in strength and would be stressed at only 82% of code allowable stresses. Therefore, the weir box structure was judged to have adequate strength and to be in compliance with DOE Order 6430, ACI 349, and HPS SDC 4.1 (Ref. 24).

The reinforced concrete cover was installed on the 1301-N weir box as part of the N Reactor Project H-797. The purpose of the cover was to provide containment of the emergency coolant effluent and to provide structural support for the weir box walls in the event of a safe shutdown earthquake. The weir box cover also provides support for 3 pipes (6 in., 12 in. and 24 in.) that penetrate the cover. The three pipelines that enter the weir box by penetrations through the cover are not required for restart. Yielding of the pipeline is acceptable.

Analysis of the cover as installed (Ref. 68) indicated the 6 in. and 12 in. lines are acceptable, but that the 24 in. line either be disconnected by a removable spool piece or install an expansion bellow. The fix made was to install an expansion bellows.

Also as part of N Reactor Project H-797, a standpipe was installed in the N Reactor ECCS piping just downstream of the 1316-N valve box. The purpose of the standpipe was to provide a means for pressure relief in the event that downstream piping became constricted under faulted loading conditions such as the safe shutdown earthquake. The standpipe was analyzed (Ref. 69) where it was concluded that the subject standpipe will maintain its intended function of providing a pressure relief path under faulted load conditions.

B. GSCS

The GSCS piping necessary to provide once-through graphite cooling was qualified to Seismic Category I requirements. The portions of the system

external to 105-N/109-N Building located in Buildings 181-N and 182-N were included in the qualification of all Seismic Category I piping as discussed above for the ECCS reanalysis. The portion of the GSCS inside the 105-N/109-N Building was divided into four systems to facilitate the analysis.

The boundaries of the pipe systems to be qualified are defined in the analysis specification (Ref. 46). The systems include:

System 1 -- Graphite Cooling System (GCS) and return pipe exterior to the primary shield (Ref. 47)
System 2 -- Graphite Shield Cooling System (GSCS) piping from and to the Graphite Cooling System piping exterior to the primary shield (Ref. 48) 21
System 3 -- Graphite Shield Cooling System piping interior to the Primary Shield (Ref. 49)
System 4 -- Graphite Cooling Tubes interior to the Primary Shield (Ref. 50).

The results of each of the above analysis systems are summarized briefly below.

- System 1. Graphite Cooling System (GCS) and return pipe exterior to the primary shield (Ref. 47)

  GCS Supply Boundaries

  Location

  Auxiliary Cell - The 8 in. pipe from GCSV-303-1, -2, -3 check valves downstream from the pumps and 12 in. RWS backup supply valve RWSV-234-1, -2 valves from the RWS-2 underground line. These two line "T" in the auxiliary cell.

  Pipe Gallery - The 10 in. GCS line downstream from the tee to the 109/105 wall anchor.

Reactor Pipe Space Rear	-	From the 109/105 through a tee branch to the left right rod room ceiling anchors through reducer to 8-in. main supply header.
Left and Right Rod Rooms	-	From ceiling anchors to bottom header to the ten 2 1/2-in. distribution headers and 3/4-in. graphite cooling connectors at the reactor face plate which is the boundary for this system. Qualification of the internal graphite cooling tubes (system 4) is in the Impell Report (Ref. 63).
Reactor Pipe Space Rear	-	From the floor anchors in the left and right side rod rooms to the V32 valves.

GCS Return Boundary

Location		
Left and Right Rod Rooms	-	Boundary at reactor face plate 3/4-in. graphite cooling connectors to the 2 1/2-in. distribution connectors to the 10-in. collecting headers to the top header ceiling anchor.
Reactor Pipe Space Rear	-	From the left and right side rod room ceiling anchors, the 10-in. header branches "T" together to two 12-in. pipes across the top of the reactor and then branch to 105/109 anchor where one goes to the drain anchor at El. -16 ft.
Pipe Change to Valve Pit	-	From the above anchor through the pipe chase to the 6.5-in. orifice boundary in the valve pit.
Pipe Gallery Auxiliary Cell	-	The other 12-in. pipe returns from the 105/109 anchor across the pipe gallery to the boundary at the 301 isolation valve in the auxiliary cell.

Twenty two new supports were required to seismically qualify this system. The final results indicate positive margins of safety for all models of the GCS system. The usage factor for the connector due to 745 thermal cycles and ten cycles of seismic displacement was 0.0.

The III/I walkdown of this portion of the GCS resulted in sixty-six seismic III/I problems identified and analyzed which resulted in 17 fixes required.

The source of the water for GSCS once-through cooling is Raw Water Supply -2 (RWS-2). Seismic analysis of this pipe was conducted by Kaiser Engineers Hanford (KEH). A review of the KEH RWS-2 pipe seismic analysis and a walkdown of the 109 auxiliary cell and the 105 corridor 2 piping was conducted. The evaluation indicates that, for the auxiliary cell and No. 2 corridor, RWS-2 piping is seismically qualified by the static analysis performed and that dynamic analysis was not required.

- System 2. Graphite Shield Cooling System (GSCS) piping from and to the Graphite Cooling System piping exterior to the primary shield (Ref. 48).

  System boundaries are reported in Reference 46. Seismic qualification of the System 2 GSCS piping exterior to the primary shield is reported in DAR-SSA-8704 (Ref. 48). The evaluation and analysis resulted in eleven (11) additional supports required (8 identified by analysis and 3 by walkdowns).

- System 3. Graphite Shield Cooling System piping interior to the Primary Shield (Ref. 49).

  An evaluation for the effects of a Safe Shutdown Earthquake (SSE) event combined with the Design Load Condition on the N Reactor shields and Traveling Wire Flux Monitor (TWFM) interior cooling

systems was performed to support the Graphite Cooling System safety enhancement effort.

Boundary conditions for System 3 are defined in Reference 46.

The shield, TWFM, and graphite moderator are water cooled; the respective cooling systems share a common water supply. A compromised shield coolant piping system may affect coolant flow to the graphite cooling pipe which may preclude once-through cooling. To support the effort to provide a once-through cooling capacity, the subject evaluation addressed possible failure modes that may affect the interior shield and TWFM cooling pipe.

The evaluation, analyses and results are reported in Reference 49. Results of the evaluation to assess the effects of an SSE event combined with the Design Load Condition on the interior shield cooling system and on the TWFM show that neither the cooling pipe nor the thermal shield support structure are compromised. Results show that stress levels in the system components satisfy all relevant strength and fatigue related to B31.1 and AISC criteria.

- System 4. Graphite Cooling Tubes interior to the Primary Shield (Ref. 50).

  An analysis was performed to determine whether the reactor core will maintain its structural integrity and that the safety systems will remain functionally operable during the postulated earthquake and the pressure tube rupture.

  The evaluated reactor core's safety systems and elements included:

  - Pressure tubes
  - Graphite coolant tubes
  - Vertical ball channels
  - Horizontal control rods (HCR) passages

- o HCR-T blocks
- o Graphite keys
- o Inlet/outlet shield gap closure
- o Biological shield and pedestal.

The results of the evaluation are reported in Reference 50 and concluded the following:

- o Seismic Event
  Under the seismic loading the pressure tubes, graphite coolant tubes, horizontal control rod passages, vertical ball channels, and graphite keys capacities exceeded the earthquake demand. Thus, their margins of safety were greater than one. Likewise the structural integrity of the biological shield and pedestal was adequate when compared to design code allowables.

- o PTRE Event
  Under the localized PTRE loading, all of the safety systems meet the safety criteria. Thus, their margins of safety were greater than one.

  From the analyses conducted, it was concluded that no structural modifications are required to meet the N Reactor's extended operation condition when subjected to a Safe Shutdown Earthquake Event or a Pressure Tube Rupture Event.

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C. Fog Spray System

The Fog Spray System provides confinement pressure control and washes fission products and aerosols from the confinement atmosphere. On failure of the electrically driven high-pressure raw water supply to the fog spray stations in 105-N and 109-N, emergency fog spray water is supplied by two diesel-driven pumps located in 182-N Building. These pumps take suction directly from the low-lift pumps in 181-N. The diesel-driven fog spray pumps also provide a backup supply of the cooling

water for the Graphite and Shield Cooling System (GSCS) in the event of loss of AC power.

The controls for the Building 182-N No. 2 fog spray diesel as well as the No. 3 low-lift diesel are powered by the batteries and associated components in Building 181-N.

The following FSS components were walked down and analyzed:

- Branch pipe from CRW system check valve RWSV-551-1 to fog spray suction header
- Fog spray suction head to sectionalizing valve and isolation valve RWSV-552-1, between ECCS suction header and fog spray suction header
- Remainder of suction header, pipes, and valves to fog spray pumps.
- Fog spray pumps and diesel drives
- Diesel fuel oil system, lube oil system, starting air system, and jacket water supply from fog spray pump discharge pipes
- Fog spray pump discharge pipes, valves, manifold and supply pipe to 109-N.

The above piping lines and components external to Building 105-N/109-N were evaluated and determined to be seismically acceptable as part of the analysis reported in References 31, 32, 43 and 44.

Components and piping inside 105-N/109-N were also walked down and evaluated (Ref. 45) and included the following:

- Fog spray supply to graphite shield cooling system from 109-N entrance including distribution system
- Fog spray water supply piping from entry point in 105-N building to the fog spray stations
- Ten 105-N fog spray stations

- o 105-N fog spray networks
- o 109-N fog spray piping from entry point in 109-N pipe gallery to orifice plate FT 266.

The walkdown team identified 8 potential III/I problems. One of these was resolved by analysis. Seven required a fix to be completed prior to restart.

D. Reactor Coolant System

Within the 105-N/109-N complex, the primary coolant piping consists of the primary coolant supply (PCS) and primary coolant return (PCR) lines in the 109 building pipe gallery; the large diameter pipe lines in the steam generator cells; the 18-in. diameter vertical inlet and outlet headers in the 105 Building; the connector piping joining the vertical headers to the pressure tubes consisting of 1003 relatively similar lines connected to the inlet and outlet pressure tubes; the pressurizer piping consisting of the 10-in. diameter surge line and the 4-in. and 2-in. spray lines; and the lift station piping consisting of piping ranging in size from 12 in. to 36 in. in diameter. The lift station piping is located in the 105 Building, west side. 21

The above primary coolant piping was reanalyzed for seismic capability in 1988 (Ref. 73). As-built dynamic modal analyses were performed on twenty-five analytical piping models. These models represented the N Reactor seismic Category I primary coolant, pressurizer, and lift station piping systems. Response spectra based on SDC 4.1, Reference 24, was used as loading input. The criteria used to assess the resulting pipe stresses was B31.1. Initial analyses were performed with an allowable 1/2 S_h, with pipe stresses exceeding this value at a number of locations. Justification was developed to increase the seismic allowable stress from 1.2 S_h to 2.4 S_h in accordance with ASME Section III Code Case 1606. All PCS models passed the B31.1/ASME pipe stress criteria except for one model (PR-2) that addressed the pressurizer 2-in./4-in. spray line. Five supports were required for this model to assure that stresses satisfy appropriate criteria. The supports were

designed and DC 87337 was prepared (Ref. 74). In accordance with the RTS program implementation was deferred to the LTS program.

Fatigue analyses were performed on piping locations with high thermal expansion stresses, with the pressurizer tee the critical component. A conservatively calculated ASME Class 1 fatigue assessment indicates a present usage factor of 0.54 (allowable = 1.0) with a life expectancy under normal usage to the year 2007.

Support loads, nozzle loads, response spectra loading input, nozzle evaluations, and other contributing calculations are discussed in the Reference reports (Ref. 73, 74). In addition, the high-pressure raw water system piping elements in the 105-N/109-N Building were also analyzed and qualified to Seismic Category I (Ref. 75).

E. Confinement - Piping Systems

Category I systems within the confinement structures consist of the following:

- o Penetrations
- o Vent system
- o Moderator gas system
- o Emergency drain system
- o Filter system.

(1) Penetrations - are defined as connecting Zone 1 or Zone 2 areas (potentially radioactive) to Zone 3 non-radioactive. The penetrations and penetration testing results are described in detail in References 51 and 52.

All 1236 penetrations in 105-N/109-N were walked down and examined (Ref. 53).

During the seismic III/I walkdowns, it was determined that the non-seismic Category I steam pipes on the roof of the 109-N Building

were a possible hazard to the Category I confinement roof penetrations. Reference 54 describes the analysis conducted to verify that the steam pipe will not fail during an SSE seismic event.

There are six heat exchanger cells in the 109-N building. From each cell four 24-in. steam pipes rise upwards through roof penetrations and continue to the 46-in. main steam header. Steam is then routed from the header to the turbine generators, dump condensers, RCS pump drive turbines, surge tanks, and/or medium pressure system. In sum, there are twenty-four parallel 28-in. main steam pipes on top of the 109-N building and the 46-in. header.

Pipe stress, pipe supports and restraints, nozzle loads, and roof penetrations were evaluated for all piping. All penetrations were found acceptable and it was concluded that the main steam pipes in 109-N building and on the 109-N roof are acceptable for SSE seismic loads based on ANSI B31.1 piping requirements and other stress calculations. Three seismic snubbers were replaced; two on the south side with PSA-3 snubbers and one on the north with a PSA-10 snubber.

(2) Vent System (Ref. 55) - There are fifteen 6-ft diameter steam release vents on the 105-N/109-N roof which provide a total release area of approximately 390 ft^2. Two of the vents are mounted over the 105-N roof. The other thirteen vents are mounted on the 109-N pipe gallery roof.

Each vent assembly consists of a sealed weatherhead, a primary closure butterfly valve, and a backup closure butterfly valve. Each primary valve butterfly valve is normally held open hydraulically. The backup closure valves are operated by the compressed air system and have a secondary source of pressure supplied from locally mounted nitrogen cylinders.

During reactor operation, the weatherheads seal Zone 1 for ventilation purposes following minor incidents. If there is a major pipe break and Zone 1 pressure builds to 2.0 psig, holding pins will shear, weatherheads are blown free, and steam vents to atmosphere (a special weatherhead vents at 1.25 psig).

The 109-N ventilation system is equipped with ten supply and nine exhaust isolation valves. Each valve is equipped with an automatic backup valve.

Four roof-mounted relief valves (two on 109-N roof and two on 105-N roof) provide Zone 1 vacuum relief. These valves are passive and will open when Zone 1 confinement pressure is less than -0.25 psig.

The 105-N ventilation system is equipped with one supply isolation valve (Room 28) and three exhaust isolation valves (Room 505), all of which are hydraulically operated. Each valve is provided with a backup closure valve.

The 105-N/109-N Zone 1 Confinement Isolation System is further described in Section 6.2.

The following equipment was walked down:

- The two 6-ft diameter steam vent valves on the 105-N roof and the 13 steam vent valves on the 109-N pipe gallery roof
- The 15 backup closure valves for the above steam vent valves and their locally mounted nitrogen pressure supplies, piping and valves
- The four roof mounted vacuum relief valves (two on 109-N and two on 105-N)
- The 105-N ventilation system supply isolation valve in Room 28 and the three 105-N ventilation exhaust isolation valves in

Room 505 and the associated hydraulic actuation systems and backup valves.

NOTE: The 109-N ventilation valves (20 valves) were not included in the walkdown. These valves were included in the Penetrations, Task 1, walkdown.

The walkdown team identified twelve potential III/I problems in the Task 2 vent system. One of these was resolved by analysis. Eleven required some form of fix (Ref. 55) to be completed before restart.

(3) Reactor Moderator Gas System (Ref. 56) - The purpose of this system is to prevent deterioration of the graphite and Zircaloy-2 tubes in the reactor. It is a safety-related system only in that it provides a pathway for the relief of steam pressure in the event of a postulated pressure tube break. The system is comprised of the following: 21

- o Helium loading and storage facilities
- o Helium pressure reduction stations
- o Helium blowers and dryers
- o Helium piping into reactor
- o Control system.

The reactor moderator gas system is further described in Section 9.

Walkdown revealed that this system itself is not seismic SSE qualified. Although III/I fixes were identified, they will not be implemented since this alone will not qualify the system. It is not presently planned to upgrade this Moderator Gas System prior to reactor restart.

The walkdown team identified 10 potential III/I problems in the Task 3 Moderator Gas System. Two of these were resolved by analysis. Eight require a fix (Ref. 56).

(4) Emergency Drain System (Ref. 57) - The following equipment must be protected from damage by non-Category I failure:

- o Diesel drive unit
- o Pump, diesel fuel system
- o Controls, valves and piping
- o Zone 1, 36-in. diameter floor drains.

Discharge piping from the lift stations was not walked down as the piping runs to the crib are below ground. The 105-N Lift Station diesel pump does not use an air starting system. The two 36-in. diameter drains in Zone 1 were walked down as part of the GSCS (no III/I problems identified).

The walkdown team identified three potential III/I problems in the Task 4 Emergency Drain System. One problem was resolved by analysis. Two required a fix (Ref. 57).

(5) Filter System (Ref. 58) - The exhaust ventilation filter and diversion system is a seismic Category I system designed to prevent the release of contaminants to the atmosphere and supplement the confinement features in building 105. The filter system is housed in the 117-N building in four cells, A, B, C, D.

Each filter cell is designed to house three filter banks installed in series. Removable covers are provided above each filter bank to allow removal of filter bank and frame as a unit.

The first stage cell bank consists of aluminum moisture separators that prolong the effectiveness of the second stage bank of filters. Moisture separators are installed in cells, A, B, and D.

The second stage filter bank consists of high efficiency particulate air (HEPA) filters. The filters are designed for continuous

operation at a maximum temperature of 200°F. Minimum filter particulate efficiency is 99.9%.

The third stage filter bank consists of charcoal absorbers which are a minimum of 95% efficient for removal of elemental halogens.

A filter fog spray system is provided to accommodate a combined fission product decay heat generation of 72,000 BTU/hr. The discharge from the filter cells is routed to a single stack which exhausts to the atmosphere 200 ft above ground level.

The 117-N filter system is further described in Section 6.5. 21

The following equipment must be protected from damage by non-Category I failure:

- o Filter fog spray piping (in cells and RWS-2 supply
- o Filter frames
- o Thermocouples
- o Water entrainment separators
- o Filters: HEPA and charcoal absorbers.

The walkdown team identified 5 potential III/I problems in the Task 5 Filter System. Two problems were resolved by analysis. Three required a fix (Ref. 58) to be completed before restart.

3.9.3.2 Pump and Valve Operability Assurance

Testing of pumps is presented in Section 3.9.6.1. Testing of valves is presented in Section 3.9.6.2. 15

3.9.3.2.1 Seismic Re-Evaluation of 181-N Low Lift Pumps

As required by the Return to Service Program (Ref. 64), and included in the Seismic Task Force Program to seismically qualify all Category I components, a re-evaluation of the Low Lift Pumps in the 181-N Facility was performed (Ref. 66). 21

The purpose of the evaluation was to demonstrate that the three Seismic Category I, Low-Lift Emergency Cooling Pumps located in Building 181-N can withstand the revised Safe Shutdown Earthquake (SSE) loading as defined in Reference 24. Additionally, it was necessary to determine if the Non-Category I Circulating Raw Water Pump #4 (which shares a water well with the Low Lift Pump #3) presents a potential Seismic III/I hazard to that low lift pump.

The approach used to make the above assessments was to:

- Extract enveloping in-structure response spectra from Reference 24 for use in determining seismic loading
- Review for prior seismic analyses of the subject pumps and perform comparative analyses for revised loading
- Perform supplemental calculations as required to assess pump structural integrity
- Apply current approved Hanford structural acceptance criteria to determine margins of safety.

The three deep-well low-lift pumps located at the 181-N river pumphouse provide the emergency cooling water to the Emergency Core Cooling System (ECCS) and the Fog Spray System (FS). The combined discharge from two pumps is required to satisfy all of the emergency flow requirements. While some of the emergency conditions could be handled by one pump through a manual balancing of the available flow between the ECCS and the FS, the additional pumping capacity of the third pump (added by Project H-567) gives added safety against the need for manual balance procedures for a single pump source. In addition, the surplus water from two operating pumps eliminates the water level fluctuations in the ECCS Silo that could possibly occur with a single pump.

Based on the review of prior analyses and the supplemental calculations provided herein, it is concluded that the building 181-N Low Lift Pumps are structurally adequate to withstand the currently predicted SSE loading, and

the Circulating Raw Water Pump #4 will not compromise the integrity of the Low Lift Pump #3 (Ref. 66). 21

3.9.3.3 Design and Installation Details for Mounting of Pressure-Relief Devices

Design criteria and provisions for overpressurization protection of the reactor coolant system are described in Section 5.2.2. Testing of pressure relief valves are included in Section 3.9.6.2.

3.9.3.4 Component Supports

Supports for the various components and piping loops of the RCS were designed for the following loads, acting simultaneously: dead weight including insulation, weight of contents and seismic effects. Portions of the RCS located in the 105-N building, including pipe supports, were designed for the following effective earthquake forces: 0.5 g acting horizontally and 0.25 g vertically; whereas, the RCS system located in the 109-N building was designed for 0.2 g acting horizontally and 0.1 g vertically. See Section 3.7. Thermal expansion effects may produce reaction forces and pipe stresses when pipe expansion is restrained. Constant load hangers and spring supports were utilized in the pipe support system to minimize thermal expansion forces. Long sections of piping runs were supported laterally or guided, whenever feasible, in order to resist earthquake forces. o 21

As discussed above in Section 3.9.3.1.5 in paragraphs A and D, the piping supports for the RCS and ECCS piping systems within the 105-N/109-N Building were reanalyzed (Ref. 73, 74) as part of the RTS program. The loading included the effects of dead weight, thermal expansion, and Safe Shutdown Earthquake (SSE) effects. The supports analyzed include those involved in (1) the emergency core cooling and associated piping, (2) the pressurizer piping and (3) the lift station piping. Loads were obtained from Reference 73. 21

Material allowable stresses were based upon AISC handbook and related documents. When the analysis indicated that a given support would fail the

acceptance criteria, a fix was proposed. Additional analyses were then made to show the proposed fix was acceptable.

The majority of the pipe is restrained by single pipe supports, and these were analyzed individually. In some cases, the pipe is supported by three-dimensional structural frames that incorporate a number of individual pipe supports. These were analyzed as a unit to include all of the members of the frame structure. As a result of the piping and pipe supports analyses, a number of overstress conditions were identified at several locations. Additional new supports were designed and placed in locations such that stresses were reduced to acceptable levels. These changes were accomplished by DC 87307 and DC 87337. A summary of the support requirements and location of the supporting analysis is shown in Table 5-1 and Figures 5-1 through 5-4 of Reference 74. It should be noted that the modifications of DC 87337 (Pressurizer piping) have been deferred to the Long Term Service Program.

The pipe support structural analyses also resulted in fixes to existing supports being required at several locations. These are summarized in Table 5-2 for individual supports and in Table 5-3 for pipe support frame structures in Reference 74. The seismic piping analyses were performed without the presence of seismic snubbers and it was verified that RCS snubbers originally installed had been removed as reported in Unusual Occurrence Report 85-12.

The pipe support system for the reactor coolant system is described in Section 5.4.14.

As discussed in 3.9.3.1.5, the seismic reanalysis and upgrade of Category I piping supports involved, primarily, replacements and modifications of the supports and restraints for the systems.

3.9.4 Horizontal Control Rod Drive System (HCRDS)

The N Reactor horizontal control rod system consists of 84 control rods in a pattern 11 rods wide and 9 rods high with 3 rods omitted at each corner, and three spaces are not used for control. The rods are on a rectangular spacing 32-in. wide and 36-in. high. Forty-five rods enter from the right side of the reactor and 39 from the left side. The rod system functions in a dual capacity in that it provides normal reactor control and is also the primary reactor safety system.

The HCRDS consists of the mechanical-hydraulic drive mechanisms, control units, accumulators, hydraulic pumps and connecting pipes and valves and extends to the coupling interface with the reactivity control elements. See Section 4.5.1, Control Rod System Structural Materials, and Section 5.6, Rod Cooling System.

The rods are driven by a combination hydraulic cylinder and chain drive. Each rod is equipped with an individual drive. The two rod groups - one comprised of the 45 rods on the right side of the reactor and the other of the 39 rods on the left - are provided with a separate and totally independent power unit which supplies oil pressure through a header system to the individual rod drives.

3.9.4.1 Descriptive Information of HCRDS

The rods are driven by a combination hydraulic cylinder and a two-to-one ratio chain drive. Each drive has two independent control systems for controlling rod movement. One system is used to move the rods at either fast or slow speed for normal positioning of the rod for reactor control; the other is used for scram operation to quickly shut down the reactor in emergency situations. Power for scram operation of the rods is supplied by an individual hydraulic accumulator for each rod.

The design bases for the HCRDS and descriptions of the mechanical-hydraulic drive and normal and scram operation are provided below. The testing program employed to verify the design and operation is described in Section 3.9.4.4. 2

3.9.4.1.1 Design Bases - The design bases for the HCRDS are (Ref. 9):

1. capable of two speed for normal operation, namely:
   a. fast speed of approximately 30 fpm for coarse adjustment and for quick removal of rods during early phases of reactor startup;
   b. slow speed of approximately 4 fpm for fine adjustment.
2. capable of providing both normal and emergency reactivity control;
3. capable of scram function without dependency on auxiliary power source or action of secondary controls;
4. capable of 75% insertion of all rods within 1.25 sec after de-energization of scram hydraulic valve (EMS limit is 1.2. sec);
5. capable of normal or emergency rod insertion assuming a reactor core distorted by 25 years of service;
6. capable of a scram life of 1,000 cycles;
7. capable of rod position accuracy of $\pm$ 2 in.;
8. characterized by low maintenance requirements, ease of replacing parts and use of materials exhibiting neutron activation properties permitting reasonable personnel access for maintenance;
9. capable of scram operation during or following the design basis earthquake.

The HCRDS are located in the outer rod rooms (one on each side of the reactor) which are accessible during reactor operation. Therefore, the drive system functions in a normal atmosphere at ambient building temperature and in an area of low radiation exposure.

3.9.4.1.2 Mechanical-Hydraulic Drive

The mechanical-hydraulic drive is illustrated in Figure 3.9-9. Each rod is driven by chains affixed to the outer or "cold" or seal section of the rod. The chains pass around sprocket wheels which are mounted on the hydraulic cylinder and are attached to the center of the drive frame assembly. This arrangement - the cylinder moving over a fixed piston and piston rod - makes it possible to drive the rod at twice the speed of the cylinder. The cylinder rides on rollers on a track that is an integral part of the drive assembly

and is well supported within the total structure. The materials which are used in the fabrication of the HCRDS are described in Section 4.5.1.

The drive cylinder moves over a stationary piston which is mounted on a double-ended hollow piston rod anchored at the ends. This rod extends out of the ends of the cylinder through seals. The hydraulic fluid enters into and is drained from the cylinder through ports in the hollow piston rods. As the horizontal rod is driven into the reactor by the moving cylinder under scram conditions, the ports which are located near the piston are progressively covered by a choke or buffer tube on the outboard end of the cylinder. This tube fits closely over the piston rod thereby choking off the hydraulic fluid flow and causing the rod to decelerate to the point where mechanical damage to the rod and drive mechanism is avoided.

The hydraulic system on each horizontal rod includes dual pilot-operated check valves which "lock" the hydraulic fluid in place and prevent unintended movement of the rod in either direction. For a rod to be pushed out of or ejected from the reactor would require that pressures be built up within the reactor which are on the order of 50 times what the reactor vent system (see Section 9.3.6.1) will permit (3 $\pm$ 1 psig actuation pressure). The reactor vent piping is sized to limit the maximum internal reactor pressure to 15 psig (approximately 90 lb force on the rod tip). See Section 15.6.4.4. There are no documented cases of a check valve failure which would release the hydraulic fluid and permit the rod to move inadvertently.

3.9.4.1.3 Normal Drive System Operation

The normal drive system is shown in Figure 3.9-10. Separate hydraulic systems are used for normal control of the left and right side rods in order to maintain independence of action. To move a rod, hydraulic fluid is pumped from a reservoir to one side of the piston, while fluid is drained from the other side through one or two parallel constant flow orifices and back to the reservoir. For slow speed, fluid is returned through one small orifice; for fast speed, return fluid flows through both orifices.

Two pumps supply fluid to the normal drive system. The system is designed for 2,000 psig and operates at about 1,800 psig. This system is capable of operating the rod at either of two speeds; fast speed (approximately 30 ft/min) for coarse rod adjustment and slow speed (approximately 4 ft/min) for reactor control rod positioning. Only one of the two pumps can be operated at any one time to maintain pressure in the system. The two pumps are of different capacities. The smaller (normally-used) pump has a capacity of 3.25 gpm; the larger (spare) pump has a capacity of 5.25 gpm. (Under current operating procedures, the fact that the pumps are of different capacity is of no significance.)

The normal HCR drive return flow control valves govern the speed of rod withdrawal or insertion. These speeds are normally set at 30 ft per minute for fast speed and 4 ft per minute for slow speed. The construction of the flow control valves is such that component failure would tend to restrict flow or speed of rod travel rather than cause the speed to increase. Furthermore, each rod drive system has a separate set of speed control valves such that a malfunction of speed control would only impact one rod drive system. Speed flow control settings are routinely tested.

When the rods are stationary, the hydraulic pumps are needed only to make up for leakage. Since the pump operates continuously, the hydraulic fluid is bypassed through coolers back to the reservoir. A nitrogen-charged accumulator is provided on the pump discharge line to accommodate leakage for a reasonable length of time to prevent short-period cycling of the bypass system. A relief valve limits maximum system pressure to 1,900 psig. Two additional features of the normal rod drive system include: (1) solenoid valves to determine direction of rod drive and (2) check valves to prevent rod movement by forces other than controlled hydraulic pressure.

3.9.4.1.4 Scram Drive System Operation

The scram horizontal control rod drive system is markedly different from the normal control system in that individual accumulators, instead of pumps, serve as the primary source of hydraulic fluid for rod motion. Therefore, the energy required to insert the rods at high speed under emergency

circumstances is stored, and the ability to scram the rods is not dependent upon electrical power. Other significant differences include less complex controls, larger and fewer valves and the use of larger piping to conduct the hydraulic fluid to and from the drive cylinder. (The scram system is fabricated of 1-in. pipe and fittings; the normal control system is 3/8-in. tubing and fittings.)

The scram system is capable of accelerating the rods sufficiently to obtain a 75% rod insertion within 1.2 seconds of a system trip. Acceleration and deceleration rates are held within 5 g for protection of components from shock damage.

The scram drive is shown in Figure 3.9-10. During scram action, fluid flows out of the accumulator through a hydraulic-pressure-operated, fail-open scram valve and a check valve into the rod drive cylinder. The purpose of the check valve in the inlet line from the scram accumulator to the drive cylinder is to isolate the hydraulic fluid in the upstream scram lines from the normal drive hydraulic system. The accumulator vessel for each rod drive has a total capacity of about four gallons. Normally, the accumulator will contain approximately three gallons of hydraulic fluid and one gallon of compressed nitrogen . The two materials are separated by a free-floating piston, with the fluid in the chamber above the piston and the gas below. Charging the accumulator is accomplished by first pressuring it with nitrogen to 1,000 psi and then pumping in about three gallons of hydraulic fluid to give a terminal pressure of 2,700 psig. A magnetic-type level switch is provided for each accumulator to detect leakage of any of the hydraulic fluid into the gas chamber, and this is annunciated in the reactor control room. The hydraulic pressure in each accumulator is monitored and low pressure actuates an annunciator in the reactor control room.

Here, as with the normal drive system, the right and left side rods comprise two completely independent systems, each of which is equipped with two high pressure hydraulic pumps for charging the individual rod drive scram accumulators. Normally, one pump per side will operate continuously to maintain pressure in the system. However, the ability of the scram system

to function is independent of the status of the hydraulic pumps at the time the scram signal is received, since the force for rod movement on scram is provided by the energy stored in the accumulators. The second pump starts automatically upon low manifold pressure to increase flow rate for recharging the accumulators following a scram. Both pumps operating have the capacity to recharge all of the scram accumulators within five minutes. The arrangement of the scram power units with respect to pressure regulation, bypass and pressure relief is the same as for the normal drive system with one exception. The exception is that the pressure is regulated at 2,900 psig to the scram pressure header and is relieved at 3,100 psig, the maximum working pressure of the system.

The individual rod scram-drive system is entirely separate from the normal control system, except that both systems use common drain return piping to the oil supply reservoir and short lengths of common piping at each end of the drive piston rod.

3.9.4.2 Applicable HCRDS Design Specifications

Specification No. HWS-6596, titled "Specifications for Horizontal Rod Drive and Hydraulic Controls - 105-N, Project CAI-816," (Ref. 10) imposes the following requirements on the design and fabrication of the HCRDS:

1. compliance with the minimum requirements of applicable sections of the "Joint Industrial Conference Standards" for hydraulic equipment as revised April, 1959;
2. compliance with the National Electric Manufacturers Code and the National Electric Code as applicable;
3. compliance with the following Hanford welding specifications:

    a. Aluminum and Aluminum Alloys - HW-4903-S
    b. Welding Carbon Steel - HW-4926-S
    c. Welding Stainless Steel - HW-4924-S
    d. Qualifying Welders - HW-4900-S

4. a design pressure rating of 2,000 psi on component parts and piping of the hydraulic portion of the HCRDS providing the normal control function;
5. a design pressure rating of 3,000 psi on component parts and piping of the hydraulic portion of the HCRDS providing the scram function or common to both functions.

3.9.4.3 Design Loads, Stress Limits and Allowable Deformation

An experimental testing option was exercised in lieu of establishing a set of stress and deformation allowables. See Section 3.9.4.4 for the description of the testing program.

3.9.4.3.1 Seismic Qualification

A. Scope

The following analysis (Ref. 60) was performed to Seismically Qualify the existing horizontal control rod support structure and the equipment housed at each level. The support structure located in the inner and outer rod rooms left and right side of the process area is a steel structure with wide flange columns and floor beams. The support structure is divided into three levels to house control rod drive mechanisms, thimble housing assemblies, and valve racks, etc. The structural steel columns are connected to the concrete floor (level 1) with anchor bolts and pin connected at the concrete roof level. Floor beams and girders are pin connected at each floor level. On all four sides, the floor beams at each level are connected to the concrete wall allowing the floor to behave as a rigid diaphragm. Field walkdowns and drawing reviews established the following elements as the most critically loaded. 21

(a) Drive assemble on column line E.R. at El. 28 ft - 8 1/8 in., level 3, outer rod room.
(b) Floor beam on column line E.R. between column line 2R-1R at El. 7 ft - 7 7/8 in., outer rod room.
(c) Thimble housing assembly between column lines F-GR and 4R-8R at El. 25 ft - 5 3/8 in., inner rod room.

(d) Column with rod drive assembly loads at 2R-E.R El. 25 ft - 8 1/8 in., outer rod room.

(e) Valve rack/accumulator, drive assembly and chain drive mechanism support connections.

B. Analysis Description and Results

The critical elements as listed above were analyzed for seismic loads using the method "equivalent static analysis" as defined in HPS SDC 4.1 (Ref. 24). The 5% damped design response spectra curves developed for the structure (Ref. 59) were used to compute the seismic load factors. For the critical loading, the stresses and deflections in the X and Y axes were calculated. Allowable stresses are based on HPS SDC 4.1 (Ref. 24). Allowable deflections are based on UNI-3178 (Ref. 50). The following results are noted.

(a) Drive assembly on column line E.R. at El. 28 ft - 8 1/8 in.

The drive assembly weights for all the parts were not available. Available parts such as channel, stiffener plates, roller guide rail, and side rails were calculated and included in the calculations. To allow for the remaining parts field walkdowns were made and one thousand (1,000 lb) have been allowed in the calculations. Photographs taken during field walkdowns are included as Appendix B of Reference 60. The drive assembly was also analyzed assuming an overall weight of six thousand (6,000 lb) based on Kaiser Engineers material receipt dated September 10, 1962. The combined stresses are within the AISC allowable limits. The vertical and horizontal deflections are also within allowable limits as given in Table 2.1 of Reference 50. These limits are meant to apply to rod channels within the core and are conservative for the inner and outer rod rooms. No modifications are needed.

(b) Floor beam on column line E.R. between column line 2R-1R at El. 7 ft - 7 7/8 in.

This is a typical floor beam loaded with three (3) valve rack assemblies and miscellaneous piping. The actual weights for these assemblies were not available. Based on field walkdowns the reasonable weight for each assembly was estimated at 600 lb. Photographs during field walkdowns are included in Appendix B of Reference 60. A weight of 1,000 lb in addition to the floor dead load was allowed in the calculations. The combined stresses are within AISC allowable limits. No modifications are needed.

(c) Thimble housing assembly between column lines F-GR and 4R - 8R at El. 25 ft - 5 3/8 in.

Similarly the weights for all the parts were not available. Based on field walkdowns reasonable weights were assumed conservatively. Photographs taken during field walkdowns are included in Appendix B of Reference 60. A typical rod casing section was analyzed assuming an overall weight of the rod casing as two thousand (2,000 lb) and three thousand (3,000 lb). The combined stresses in the vertical and horizontal direction were within the allowable limits. Similarly deflections in the vertical and horizontal directions are within allowable limits quoted in Table 2.1 of UNI-3178, Volume 1 (Ref. 50). No modifications are required. 21

(d) Column with drive assembly loads at 2R-ER, El. 25 ft - 8 1/8 in.

This column has been analyzed for two rod drive assembly loads (moment about major axes) and a pipe (6-in. dia.) support load (moment about minor axes). The 6-in. dia. water line was supported from the column web on the east side at 2 ft 0 in. from the concrete ceiling. The combined stresses are within the allowable limits. No modifications were required.

(e) Valve rack/accumulator and drive assemble support connections.

The valve rack/accumulator channel support structure and bolted connections have been checked. Similarly drive assembly channel bolted connections including guide rail and roller frame supports have been analyzed. The stresses and loads are within the allowable limits. No modifications are required.

C. Conclusions

The critical structural elements were analyzed based on conservative estimates of system loadings and the conservatism inherent in the equivalent static method. The combined stresses and deflections in the horizontal and vertical directions were well within the allowable limits. The steel floor beams/columns at all levels were connected with 3/4 in. A325 high strength bolts. The critical beam reactions were well in the AISC bolt allowable loads. In addition the structural steel floors (all three levels) were connected to concrete walls on all four sides. Similarly the steel columns were connected to concrete floor and ceiling. As a result, the structural deformations will be limited to an absolute minimum. Therefore, it is concluded that the horizontal control rod support structure is seismically adequate without modifications.

A III/I walkdown was performed to assure that the following equipment would be protected from damage by non-Category I failures:

- Rod drive mechanism in left and right outer rod rooms
- Hydraulic equipment including accumulators, pumps and drive cylinders
- Chain drives
- Valves, controls, piping
- Gun barrel assemblies.

The walkdown team identified 30 potential III/I problems in the Horizontal Control Rod System. One problem was resolved by analysis. A fix was required to resolve 29. The fixes have been completed. 21

3.9.4.4 HCRDS Test and Qualification Program

As indicated below, the program for the design and development of the HCRDS proceeded on the basis that each concept, component and assembly would be thoroughly proven and tested before it would be accepted for incorporation into the system. When the HCRDS went into service, it was known to be a system whose adequacy had already been proven under what were essentially operating circumstances.

The drive system has operated successfully for approximately 25 years to adequately and reliably drive the horizontal control rods as required for reactor operation and shutdown. As a part of each reactor shutdown, the rods are scrammed into the reactor, and the time required to achieve 75% insertion is timed. Rods failing to meet the 75% insertion specified time are infrequent, and these are corrected as they are identified. 15

3.9.4.4.1 Development Testing

The present HCR drive concept was based on development work performed at Hanford by General Electric Company and on test data obtained from the testing on a prototype rod. The test drive, along with a somewhat rudimentary mockup consisting of a simulated gunbarrel and reactor graphite rod channel, were fabricated in early 1958, with test work being initiated later in the year. Using this equipment, sufficient test work was conducted to prove the feasibility of the concept and to accumulate enough data to permit the preparation of a design criteria document and a procurement specification. A purchase order for 87 drive units was authorized in February 1961.

3.9.4.4.2 Testing of First Production Rod Drive

Assembly of the first production horizontal rod drive was completed by the vendor in July 1961. As required by the purchase order, this drive unit was tested at the vendor's plant and then shipped to Hanford where it was extensively tested. The tests at both sites consisted of a complete

functional and operational checkout of the ability of the total rod drive assembly to perform as required by the design criteria. In addition, specific tests of the hydraulic and mechanical components were run to ensure that they were of adequate design and fabricated of materials appropriate to their use (Ref. 10). On the bases of data obtained from the testing at both locations, modifications were made to the first drive unit to correct out-of-specification acceleration and deceleration rates, to eliminate an undesirable delay in rod movement when reversing direction of drive motion and to correct a sealing friction problem involving the cylinder rod packing. Before the final modifications were established, an extensive test program aimed at determining the cause of the problems, the best solutions and the adequacy of those solutions, was conducted. After the changes were proven to be satisfactory, they were incorporated into the formal design and were then made on each drive unit at the vendor's plant.

During the course of the testing of the first production drive unit, additional production-model, rod and drive-associated equipment became available. These items were incorporated into the test mockup as they became available to provide a close simulation of the reactor configuration. The modified mockup included the drive, valve rack, a test gunbarrel assembly, a simulated graphite rod channel and a test rod assembly. The graphite portion of this mockup was capable of being moved or distorted in such a manner as to simulate the distortion of the graphite stack which was anticipated to occur during the life of the reactor. Using this mockup, the drive and test rod were operated approximately 1500 times at scram speed (29 ft/sec maximum velocity). In addition, 700 runs were made with the graphite rod channel distorted in representative five-year steps up to a total calculated to be equal to 25 years of service life (designed scram life - 625 cycles). The rod and drive were also operated numerous cycles at slow (4 ft/min.) and fast (30 ft/min.) normal speeds. The design bases pertaining to rod velocities and scram time were verified during this test phase.

Later in the program, when production equipment became available, the test set-up was again modified. The use of on-reactor type components permitted the construction of a mockup which very accurately simulated the geometrical

features of the reactor as pertaining to the HCRDS. On this mockup, another complete test of the rod and drive unit assembly was conducted. The assembly was life-cycle tested by scramming the unit approximately 800 times with the graphite channel distorted in five-year steps up to a maximum equal to 25 years of service life. The drive assembly and all associated equipment functioned satisfactorily within the requirements of the specification without any component failures (Ref. 11). The drive performed virtually the same regardless of the degree of graphite stack distortion. Also, as the system was used more, better performance, particularly on scram insertion time, was achieved.

3.9.4.4.3 Vendor Testing Prior to Shipping

All HCRDs were tested with a weighted test rod (simulated production rod) at the vendor's plant prior to being released for shipment. This test required that each rod drive, after assembly and inspection, complete 5 runs each at 4 ft/min., 30 ft/min., and at scram conditions.

3.9.4.4.4 Preoperational Testing

The final test of the HCRDS prior to operational use occurred when the operational test procedure (OTP) on the total control rod system as installed was performed. The procedure followed in the performance of this test and the data to be recorded were specified in OTP-T0021, Rev. 2, titled "Control Rod Systems Tests," (Ref. 12). The purpose of this test was two-fold: (1) to demonstrate satisfactory operability and acceptable system performance with respect to adequacy of design equipment and construction and (2) to document the basis for acceptance and provide a record of initial data on the control rod system, including the drive mechanism.

The functional tests required that all rods be individually tested at slow and fast normal speeds and at scram speed. After the individual tests had been completed, testing of the rods at slow, fast and scram speed in groups was carried out. In addition, the performance of the rods and drives under a variety of situations set up to duplicate actual operating circumstances was tested.

Upon the successful completion of these tests as evidenced by the signed-off copy of the completed OTP, the system was accepted for operational use.

3.9.4.4.5 Seismic Testing

Seismic testing of the HCRDS is discussed in Section 3.9.2.

3.9.4.4.6 Change-Out and Qualification Replacement Hydraulic Fluid

Fire-resistant, water-in-oil emulsion, hydraulic fluid is used in the HCR drives. The type of fluid was selected on the basis of a procurement specification (Ref. 13), screening tests, an operational mockup test and an inservice trial period to ensure that the material did not change the performance characteristics of the HCR drives which were initially tested and operated with non-fire-resistant turbine oils.

The screening test of the water-in-oil emulsion hydraulic fluids included the following parameters which are described in Ref. 14:

1. Fire resistance
   a. flash point
   b. fire point
   c. soaked wick test
   d. high pressure spray flammability
   e. flame resistance vs water content
   f. static film flammability

2. Compatibility with system materials
   a. seals
   b. electrical wiring insulation
   c. paint
   d. aluminum and steel parts
   e. trace amounts of turbine oil

The hydraulic fluid judged best from the screening tests was tested in the operational mockup of a single HCR where many operational conditions were simulated. After successfully completing this phase, the fluid then replaced

the turbine oil in the drive system for the left-side HCRs. During the inservice trial micro-organism growth was observed and eliminated through a change to water-in-oil emulsion hydraulic fluid with the same properties and an effective built-in biocide. After further inservice trials this fire-resistant fluid also was installed in the drive system for the right-side HCRs.

3.9.5 Reactor Pressure Vessel Internals

N Reactor is distinctly different from the conventional power reactor. It does not have the conventional pressure vessel and fuel assembly with integral control rods. Therefore, this section deviates from Regulatory Guide 1.70.

The internal components are discussed elsewhere in this NUSAR. The discussion includes description, functional requirements, design data and other pertinent information on the components. The graphite stack moderator, control rods, ball safety system components, thermal shield and neutron shield are discussed in Section 4.5. The fuel elements are discussed in Section 4.2 and the pressure tubes which hold the fuel elements are discussed in Section 5.3. The support structures are discussed in Section 3.8.

3.9.6 Inservice Testing of Pumps and Valves

This section addresses the program for inservice testing for operational readiness of ASME Section III Class 1, 2, and 3 pumps and valves. The detailed program indicating valves and pumps to be tested and the test specifics is detailed in the N Reactor Inservice Inspection and Testing Plan (Ref. 71). The plan was first issued and approved by the DOE-RL June 8, 1983 to ASME Class 1 systems only. The plan was revised and expanded in 1986 and again in 1988. The latest revision (Ref. 71) incorporates the examination and test requirements of Section XI for Class 2 and Class 3 components and systems to form a revised inservice inspection plan for N Reactor. 21

This plan defines the commitment to the DOE-RL made by WHC for fulfilling the requirements set forth by ASME, Boiler and Pressure Vessel Code (B&PVC),

Section XI, Division 1, 1983 Edition, Winter 1983 Addenda (hereafter referred to as Section XI), for Class 1, 2, and 3 components.

Section XI establishes requirements for inservice testing (IST) of nuclear system components. The proper operating characteristics of components, such as pumps and valves, are verified during IST. The integrity and proper functioning of such components are important to nuclear safety.

Section XI applies to reactor designs where the reactor core is enclosed within a pressure vessel. N Reactor differs markedly in its design from pressure vessel reactors in that it is a pressure tube reactor. For this reason, N Reactor cannot fully comply with the requirements of Section XI, for example, those that pertain to the reactor vessel. Where literal compliance with these requirements cannot be accomplished, or would be difficult to accomplish due to operational and ALARA considerations, various international codes and standards have been used as guidance to determine the examinations that appear in this document, using Section XI for guidance. These examinations assure that N Reactor plant safety, as originally established during construction, is being maintained.

3.9.6.1 Testing Procedures

The N Reactor Equipment Maintenance Standards (EMS) provide for the testing and inspection of plant equipment. Testing, inspection and examinations on Class 1, 2, and 3 components required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant," Division 1, are implemented via the EMS program. Compliance with the IST plan for Class 1, 2, and 3 pumps and valves is achieved by implementing EMS inspection and testing procedure UNI-M-130 (Ref. 72).

3.9.6.2 Testing of Pumps

The tests consist of special test procedures for obtaining, through measurement or observation, information to determine the operational readiness of the pump.

The EMS tests consist of: (1) pump hydraulic tests (PHT), for detection of hydraulic changes in the pump and involve measurements of flow rate and differential pressures under standard reference* conditions of temperature and rotative speed and (2) pump mechanical tests (PMT), for the detection of mechanical wear or deterioration and involve measurements of bearing vibration amplitude and bearing or bearing lube oil temperatures and pressures.

Pumps subject to testing under ASME Section XI are the Class 1, 2, and 3 centrifugal and displacement type pumps that are required to perform a specific function in shutting down the reactor or in mitigating the consequences of an accident, which are provided with an emergency power source, shall be subject to the testing requirements of this section. At N Reactor, these are the Class 2 Low-Lift Diesel Pumps (LLDP), the High-Lift Diesel Pumps (HLDP), the Fog Spray Diesel Pumps (FSDP), and the Lift Station Diesel Pump (LSDP) and are listed in Table 3.9-6. 21

Class 1, 2, and 3 pumps that are supplied with emergency power solely for operating convenience are excluded.

3.9.6.2.1 Class 1 Pumps

Class 1 Pumps are those within the reactor coolant pressure boundary listed in Table 3.9-5.

At N Reactor, typically 5 of the 6 Class 1 pumps are operated continuously during reactor operation.

In accordance with the IST Plan (Ref. 71), pumps that are operated continuously during reactor operation need not be run or stopped for IST, provided that the plant log shows that each pump was run at the reference conditions, and the quantities specified were measured, observed, recorded, and analyzed.

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*Specific reference values, or as established by special performance curves permit testing during operation under varying conditions.

3.9.6.2.2 Class 2 Pumps

Class 2 Pumps are those pumps outside the reactor coolant pressure boundary associated with the emergency core coolant system (see Figure 3.9-17) and the fog spray system (see Figure 3.9-18) and listed in Table 3.9-2. Reference values for each pump are initially established in the applicable Pump Inservice Test Plan (Ref. 71). Pump tests are performed in accordance to the IST Plan and implementing EMS procedures.

IST quantities consist of: engine speed (N), inlet pressure (P_i) or outlet pressure (P_o), differential pressure (P), flow rate (Q), vibration amplitude (V), lubricant level or pressure, bearing temperature (T_b).

An IST is run on each pump every three months during normal reactor operation, except for the quantity T_b, which is required to be measured annually. Pumps requiring special test conditions are tested during outages. Instruments used for pump IST are in accordance with Section XI, Article IWP-4000.

All test data is analyzed within 96 hours after completion of a test. The requirements of Section XI, Article IWP-6000 are met.

3.9.6.2.3 Class 3 Pumps

Class 3 pumps are those pumps not included in Sections 3.9.6.1.1 and 3.9.6.1.2. Class 3 pumps are listed on Table 3.9-8.

There are no ASME Section XI requirements for the N Reactor Class 3 pumps.

3.9.6.3 Testing of Valves

This section describes the rules and requirements for IST of certain Class 1, 2, and 3 valves at N Reactor.

Valves subject to testing are the Class 1, 2, and 3 valves, and their actuating and position indicating equipment, that are required to perform a specific function in shutting down N Reactor to the cold shutdown condition or in mitigating the consequences of an accident. Valves are selected for

testing on the basis of safety evaluation, considering their function and location.

Valves exempt from testing are the following, unless there is an associated safety concern:

a. valves used only for operating convenience, such as manual vent, drain, instrument, and test valves, valves used for system control, and valves used only for maintenance
b. external control and protection systems responsible for sensing plant conditions and providing signals for valve operation.

Testing

IST of specified Class 1, 2, and 3 valves is performed in accordance with the testing requirements and frequencies shown in Table 3.9-8. Inservice valve test procedures, and acceptance criteria, are specified in the Inservice Inspection and Test Procedures Manual, UNI-M-130 (Ref. 75). 21

IST of valves consists of:

1. Valve Exercising Tests (VET) for determining that the moving parts of the valve function satisfactorily based on direct or indirect visual or other positive indication.
2. Valve Leak Rate Test (VLRT) for determining valve seat leakage at a specified pressure differential across the valve seat. Permissible leakage rates are specified in Table 3.9-9.
3. Safety and Relief Valve Tests (SRVT) for determining the set (opening) pressure and/or the closing (reseat) pressure of safety or relief valves. Bench testing with suitable hydraulic or pneumatic equipment, or testing in place with hydraulic or pneumatic assist equipment, are acceptable test methods. Valves are also tested for leakage, either during or following SRVT.

VETs and VLRTs meet the appropriate test and corrective action requirements of Section XI, Article IWV-3000. The SRVTs meet the appropriate test and

corrective action requirements of ANSI/ASME OM-1-1981 (this meets the requirements of Section XI, 1986 Edition); additionally, during the pressure interval starting at 90% of the expected set pressure, the rate of pressure increase is maintained below 2.0 psig per second, or whatever lesser rate of increase is necessary to the accurate reading of the set pressure.

For VETs, indirect evidence of valve movement may include the ability of the valve to pass normal flow, as evidenced by normal system operation and makeup rates, and any emergency flow, during emergency conditions under which the valve must function to perform its intended safety purpose.

Valves with remote position indicators are observed during operation tests at least once every two years to verify that valve operation is accurately indicated.

Each valve, after installation and prior to service, is tested as described by this section. These tests are conducted under conditions similar to those used during subsequent inservice tests. Safety and relief valves that will be removed and bench tested during subsequent inservice tests need not be installed prior to the preservice test.

The SRVTs are performed as specified in Section XI, Article IWV-3000.

Valve Categories

a. Section XI, Article IWV-2200, lists three categories of valves that are applicable to N Reactor: A, B, and C.
b. Categories of Valves: Valves listed in Table 3.9-8 are defined in one, or a combination, of the following categories:

    (1) Category A: valves for which seat leakage is limited to a specific maximum amount of the closed position.
    (2) Category B: valves for which seat leakage in the closed position is inconsequential.

(3) Category C: valves that are self-actuating in response to some system characteristics, such as safety/relief valves or flow direction (check) valves. 21

3.9.6.3.1 Class 1 Valves

Inservice testing of Class 1 valves is performed in accordance with requirements of Section XI. Inservice valve test procedures are specified and testing frequencies are established in References 71 and 72. Valves to be tested are listed in Table 3.9-8, Class 1 portion. 21

Reactor coolant pressure boundary valves, except safety/relief valves and the cooling spray line valves to the pressurizer (see Section 5.2.2), which are 2 in. nominal size and under, are excluded from the inservice testing program (see Section 5.2.4.1).

3.9.6.3.2 Class 2 Valves

Class 2 valves are those valves associated with the following systems. The valves to be tested, inspection and testing requirements and frequency are summarized in Table 3.9-8, Class 2 portion. 21

System	Figure
Emergency Core Cooling System	3.9-17
Fog Spray System	3.9-18
Confinement Filter Spray (Backup Coolant Supply)	3.9-24
Graphite and Shield Cooling System (Backup Coolant Supply)	3.9-25

The inspection and test requirements and frequency for Class 2 valves are specified in the referenced Equipment Maintenance Standards.

3.9.6.2.3 Class 3 Valves

Class 3 valves include valves in the following N Plant Systems. Valves to be tested, inspection and testing requirements and frequency are summarized in Table 3.9-8, Class 3 portion.

		EMS Ref.
1.	Graphite and Shield Cooling System (recirculation mode)	123
2.	Rod Cooling Water System (recirculation mode)	218
3.	Circulating Raw Water System	205
4.	Secondary Coolant System	208
5.	Standby Boiler System	209, 210
6.	Demineralized Water System	211
7.	Filtered Water System	216
8.	Radioactive Waste Handling System	621

The inspection and test requirements and frequencies for Class 3 valves are specified in the referenced Equipment Maintenance Standards.

3.9.7 Ball Safety System

The ball safety system is designed to serve as a backup safety mechanism for the HCR safety system. See Section 3.9.4. The system is designed to mitigate the consequences of an event or occurrence during which the HCR safety system is rendered inoperative. The ball safety system also provides nuclear control in lieu of the HCR safety system during certain reactor outage conditions when it is desirable or necessary to remove a portion of the rods for inspection or maintenance. The overall arrangement of the ball safety system is shown in Figures 3.9-26 & 3.9-28.

3.9.7.1 Design Bases

The following requirements as outlined in the original design criteria document (Ref. 16), were applied in the design of the ball safety system to assure that it would provide adequate backup and operational capabilities.

The ball safety system is designed to: | 21

1. Provide reactor shutdown capability sufficient to control any operating or postulated accident situation independent of the horizontal control rod system;
2. Be capable of maintaining the reactor sub-critical for an indefinite length of time;
3. Be capable of inserting the balls rapidly enough to prevent the release of fission products from the fuel elements in the event of an excursion from any credible cause.

Additional design bases involving the operation and maintenance of the system are: | 21

1. The system provides the capability of dropping the balls into any one channel or group of channels with the reactor shut down without affecting the remainder of the channels. | 21
2. Long life and low maintenance as well as ease of replacement and repair are very significant factors in the selection of system components since most work done on the ball safety system is performed in a radiation environment.
3. The ball hopper and gate assembly are removable to facilitate maintenance and to permit clear access to the ball channel. | 21
4. The balls are composed of a base material and neutron-absorbing material which will provide the control required when the balls are exposed to the maximum temperature and environmental conditions present in the ball channels during normal operation and postulated emergency situations. They are designed to not chip, split, corrode or break up during normal operational handling. See Section 4.5.2 for a detailed discussion of the ball materials. | 21
5. All of the system components are designed to permit the free flow of balls and not present any voids or pockets which may retain balls. | 21

6. All components directly associated with the reactor, i.e., affixed thereto, are capable of sealing off the reactor internal atmosphere and of withstanding a maximum internal pressure of 5 psig.
7. The system is capable of being tested in a manner that verifies its functional performance.

The interface between the ball safety system, the horizontal control rod safety system and the reactor scram and shutdown instrumentation are described in Section 7.2.1.1.2.

3.9.7.2 System Description

The N Reactor ball safety system design provides 108 individual vertical channels 3 in. in diameter passing through the active zone of the reactor. Of these channels, 107 are equipped with ball storage hoppers and associated neutron-absorbing balls, hopper release or trip device and ball-recovery equipment. The balls now in use are nominally 0.425-in. in diameter and are made from boronated carbon containing at least 0.09 grams of boron per cubic centimeter as a thermal and epithermal neutron absorber. The general arrangement of the ball safety system is shown in Figure 3.9-26.

Each channel is provided with an individual ball hopper which is located in the top primary shield of the reactor. The layout of the hoppers on the top of the reactor and the hopper numbering system are as noted in Figure 3.9-27. Each hopper contains 2.25 cubic ft of neutron absorbing poison balls, enough to fill the ball exit pipe and active-zone section of the channel with some excess. Shielding plugs filled with high density concrete are used to prevent significant activation or burnout of the balls while in storage and to provide shielding against radiation streaming up the ball channels. A detailed drawing of the hopper assembly is shown in Figure 3.9-28.

Details of the ball trip mechanisms are shown in Figure 3.9-29. A manually-operated toggle valve applies 120 psig air to either side of an air cylinder. The air cylinder compresses the drive spring for the ball release gate, thereby seating the gate. The limit switches signal the positions of the air cylinder piston and the ball release actuating rod.

When a ball drop trip signal occurs, each of the two coils on the holding solenoid is de-energized, causing the armature to drop and release a locking ring which allows a spring with a 250 lb compressive force to open the ball release gate and permit the balls to flow into the ball channel.

The individual ball hoppers are instrumented to transmit information continuously on the position of the gate (closed) and the position of the locking mechanism (unlocked).

The balls are retained in the vertical ball channels by air-operated diaphragm exit valves located in each channel exit pipe beneath the reactor. To remove the balls from the reactor, the exit valves are opened and the balls drain by gravity from the reactor and into the collection hopper. From the collection hoppers they are moved by means of a totally enclosed conveyor to the ball sorter (for removal of residual samarium balls) then discharged into the lower surge hopper from where they flow into the skip hoist which raises them to a level above the top of the reactor. From that point, the balls are returned to the ball hoppers by gravity. Should the balls become wet by inadvertent contact with water, they are dried before being placed in the hoppers to reduce the potential for corrosion of the ball hopper release equipment. The system is maintained under vacuum to control radioactive particulates which might accompany the balls in the form of loose dust particles.

An interlock in the HCR scram safety circuit prevents opening ball channel exit valves when the HCR scram safety circuit is made up. This eliminates the possibility of operating the reactor with the ball channels in such a state that they would not retain the balls if a ball drop should occur.

The air-operated locking cylinders in the hopper gate trip mechanism assembly can be used to lock the ball release gate in the closed position for shutdown maintenance and/or testing purposes. This is done by extending the air cylinder shaft until it seats on top of the release-gate actuating shaft and pushes it to its full down position. As long as the air cylinder remains

pressurized in this position, the release gate cannot open, even when the trip solenoid cell is de-energized. During reactor shutdown the hopper mechanisms can also be deactivated by manually inserting blocking devices in the mechanism. Procedural controls are used to ensure that all blocks are removed before reactor startup.

The mechanical components of the ball safety system are classified as being in Seismic Category I (see Section 3.2.1) and are capable of functioning properly following an earthquake that produces a 0.25 ground acceleration. The seismic classification of the electrical and instrumentation aspects of the system are discussed in Section 3.10. The quality classification of the ball safety system is discussed in Section 3.2.2.

3.9.7.3 System Evaluation

An evaluation of the ability of the ball safety system to mechanically meet and perform as per the requirements of the design criteria as outlined in the design criteria document (Ref. 16), is achieved by reviewing documentation of the pre-startup system checks which were performed and the operation of the ball safety system during the 25 year operating life of N Reactor.

3.9.7.3.1 Pre-Startup Testing

Prior to the startup of N Reactor, all aspects of the ball safety system were thoroughly and comprehensively tested to insure that the system would perform as designed. OTP No. T0013 (Ref. 17) outlines the tests which were conducted, notes the data taken and provides sheets for the recording of the data. The procedure tested the operation of the ball safety system under a variety of circumstances which simulated operating conditions. Equipment tested and tests completed included:

- o System interlocks
- o Individual hopper trip mechanisms
- o Ball drop times
- o Complete ball drop
- o Horizontal control rod safety system backup

- o Ball recovery equipment operation
- o Seismic Testing (Section 3.9.2)

Upon successful completion of these tests, the system was accepted for operational use. Similar ball drops and handling tests were conducted using the replacement boron carbide balls (see Ref. 18). 3

The tests verified that the system components performed their function and that the system met the design criteria. The maximum drop time noted for the original Sm_2O_3 balls was 43 seconds. The maximum boron carbide ball drop time is 31 seconds (see Ref. 18). The system interlocks function as required, the individual hopper trip devices operated effectively in both opening promptly upon receipt of the trip signal and in latching prior to hopper filling, and the interaction with the horizontal control rods safety system responded as designed (Ref. 19). The formal report covering the results of the tests conducted under OPT No. T0013 discusses the status of the ball safety system and states that the "test results show that proper automatic operation is achieved and the balls are inserted at a rate sufficient to fulfill the criteria requirements." The report indicates that the system as designed and installed met the requirements of the function it was intended to serve. 3

3.9.7.3.2 System Reliability

The ball safety system is one of two independent safety systems (the horizontal control rod system is the other) each of which is adequate to provide speed of control and total control (see Chapter 4). The reliability of the system, therefore, must be such as to ensure that it will function as designed if required. The system is routinely inspected and tested as required by the plant EMS program, the Technical Specifications, and the preventative maintenance system. The testing and inspection procedures are specific to the ball safety system and are oriented toward maintaining the system at a level of reliability that will ensure proper functioning of the system under accident conditions. 2

Since start of plant operation in 1964, the ball safety system has been actuated, partially or entirely, during reactor operations and testing because of slow rods, failure of the rods to scram and other inputs from the ball safety circuit. In addition a routine test and inspection program is carried out. This experience provides ample evidence that the ball safety system does function at the established acceptable reliability level.

The functional design is discussed in Section 4.6. Analyses to demonstrate the ability of the ball safety system to safely terminate reactivity transients, without the rod safety system operating, is discussed in Section 15.4.

3.9.7.3.3 Seismic Requalification

The ball safety system was determined to be a system required for return to service of the N Reactor (Ref. 64). Therefore, a reanalysis was performed to show that it can perform its function during and after the Safe Shutdown Earthquake as defined by Hanford Plant Standard SDC 4.1, Rev. 9 (Ref. 24).

A. Scope

The scope of the analysis (Ref. 62) included all components in the system from the trip mechanisms through the exit valves. This includes the ball hoppers, funnels, ball channels, and the electrical duct on top of the reactor. The ball system electrical distribution elsewhere is discussed in Section 3.10. The ball channels were analyzed previously by Impell (Ref. 63) and are included here by reference. This analysis only addresses structural aspects of the system. Active component qualification is part of the Equipment Qualifications (EQ) program discussed in Section 3.11.

The ball recovery equipment is not required to effect a safe shutdown. However, its failures induces stress in the exit valve flanges and the piping between the exit valves and the reactor shielding. Therefore it was considered in this analysis. The loading mechanism is also not required for safe shutdown and while not used is parked where its failure

would not cause damage to any Category I components, therefore, it was not considered in this analysis.

B. Analysis Description and Results

The analysis used five percent damped response spectra developed by EQE (Ref. 59). Each component was modeled as a single degree of freedom system, the frequency was computed, accelerations determined and the resulting forces applied to the system. In each case the components were within acceptable stresses for the loading condition.

The collection housing, conveyor chute, and sorting mechanism were not specifically modeled and analyzed. Instead, overturning of the sorter was assumed such that the largest possible torque would be applied at the collection hopper. Because of the large number of exit valves the resulting stresses in the valve flanges and connecting piping was insignificant. 21

C. Conclusion

All components of the ball safety system necessary to effect a safe shutdown of the N Reactor have a positive margin of safety against overstress or failure due to earthquake load combinations.

A III/I walkdown was performed to assure that the following equipment be protected from damage by non-Category I failures.

1. The ball hoppers and release mechanism must operate.
2. The exit valves must remain closed.
3. Although the release mechanism functions on a loss of electrical power, the cables must be protected from becoming energized by a non-Category I source.
4. It is not necessary that the ball recovery and ball loading equipment remain functional.

The walkdown team identified 6 potential III/I problems in the Ball Safety System. Four of these were resolved by analysis, and two by a fix (Ref. 65).

3.9.7.4 Instrumentation Requirements

The instrumentation requirements of the ball safety system fall into two categories; that which is a part of the safety-related system and that which indicates the mechanical and/or physical status of the system at any particular time.

The ball safety system instrumentation necessary as a part of the reactor trip system required for safe shutdown is described in Sections 7.2 and 7.4.

Information is displayed in the reactor control room and the ball system control room. The data relayed to those locations via instrumentation on the system includes:

1. A light indicating that the individual hopper gates are closed;
2. A light indicating that the individual hopper gates are unlocked;
3. A light and annunciator which indicates when five or more hoppers are unlocked;
4. A light indicating that the drain valves on the bottom of the individual ball channels are open.

Limit switches mounted adjacent to the gate rod and the air cylinder piston rod are used to indicate the hopper status on the display panel. These switches also provide inputs to an "ANY HOPPER GATE OPEN" annunciator located above the nuclear operator's console and to the "5 or MORE HOPPERS LOCKED" logic circuit and annunciator. This latter circuit also provides an input to the reactor trip circuit which prevents the completion of making up of the circuit as long as the "5 or More Hoppers Locked" circumstance prevails.

3.9.7.5 Inspection and Testing

The EMS system described in 3.9.7.3.2 above provides the inspection and testing instructions and procedures which must be applied to the ball safety

system on a routine basis. More specifically, EMS No. 103-N, titled "Ball Safety System," (Ref. 20) defines the system as covered by the EMS, states the EMS requirement, indicates the inspection and test frequencies as applied to the various system components, notes the special equipment needed to perform the inspections and tests, provides a detailed procedure for each inspection or test, specifies the data to be recorded and provides the data sheets to be used.

Typical tests and inspections which are routinely performed on the system are included in the following list. Those items marked with an asterisk are of particular significance in determining the current reliability status of the system and, therefore, their scheduled frequency is dependent upon operating circumstances and the results of inspection and test data evaluation.

- Ball Safety Circuit Trip Checks*
- Slow Rod Ball Backup Circuit Trip Checks*
- 5 or More Hoppers Locked - Rod Scram Test
- Ball Hopper Functional Test (Ball Drop)*
- Exit Drain Valve Functional Test
- Ball Hopper Purges Air Dryer Check
- Breaker Undervoltage Trip Tests
- Relay Inspection
- Hopper Holding Coil Tests
- Slow Rod - Ball Backup Circuit Matrix Tests*
- Annunciator Circuits Functional Tests
- Ball Sample Analysis (reactivity strength)

To ensure that no ball hoppers are unknowingly left in locked status during reactor operation, the startup checklists for both minimum and extended outages included in the Mandatory Operating Procedures require that a local visual check of the ball hoppers be made prior to reactor startup and that this check be substantiated by the display lights on the ball system control panel.

Scheduling of all EMS work is a function of the Maintenance Planning and Scheduling Subsection who routinely receive computer data which indicates when the various inspections and tests are due to be performed. Upon completion of the work, the data sheets are reviewed by responsible representatives of the N Plant Maintenance, N Plant Operations and, periodically, Engineering groups to affirm that the work has been done as required and that appropriate corrective action has been taken and documented. The completed and signed sheets are retained in a central location by the Maintenance Records and Procurement group for equipment history record and audit access.

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TABLE 3.9-1

ASME CLASS 1 SYSTEMS/COMPONENTS

System	Class
27 Fuel Rupture Detection	
This system starts where the sample tubing connects to the primary loop. It includes the tubing, heat exchangers, turrets and sample reinjection system.	1
High pressure sample lines to heat exchangers.	1
33 High Pressure Injection Water	
IWV-551 check valves and piping to the primary loop, pressurizer cold spray and primary pump seal water.	1
37 Reactor Coolant	
The reactor coolant system includes the equipment used to circulate and control the primary coolant. All elements compromising the reactor coolant system pressure boundary are included in this system. This also includes the primary coolant pressure retaining boundary in the steam generators.	1

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TABLE 3.9-2

ASME CLASS 2 SYSTEMS/COMPONENTS

System	Class
12 Compressed Air	
Diesel engine starting air including accumulators, check valves upstream of the accumulators and piping to diesels.	2
13 Confinement	
Hydraulic systems that control Zone I confinement valves.	2
Air systems for confinement inflatable seals and backup confinement vent valves from check valves upstream of the accumulators.	2
The fog spray system includes the two diesel-driven pumps, three high pressure raw water pumps, piping to 105 and 109 fog spray heads, and piping to connection valves of systems which are backed up or serviced by the fog spray system.	2
16 Demineralized Water	
Water supply to the suction of the high pressure injection pumps (HPIPs), including the demineralized water storage tank.	2
17 Diesel Oil	
The diesel oil system starts at the first point of diesel oil handling and ends at the engine-driven pumps at the point of use. Bypass and return lines are also part of this system.	
Diesel fuel oil from the day tanks.	2
18 Drain Systems	
Drain piping and components which pump or transport radioactive effluents from confinement to the disposal facility.	2
22 Emergency Cooling Water	
Emergency cooling water is the "last ditch" water system to and from the primary system. It includes the diesels, pumps, tanks, piping and valves in 181, 182 and 105, and ends at its control valves into and out of the primary system. This system also includes the recirculation pumps and lines for the last ditch piping.	2
Emergency cooling system to Reactor.	2

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TABLE 3.9-2 (cont.)

System	Class
22 Emergency Cooling Water (cont.)	
Dump line to and including dump tank.	2
Drain line from dump tank to vent pipe near 1312-N.	2
27 Fuel Rupture Detection	
Heat exchanger cooling system.	2/4
28 Gas	
Helium piping and components used to provide a gas atmosphere for the reactor which is within confinement.	2
Gas systems from accumulators, for actuation of ECCS valves and for inflatable confinement seals.	2
29 Graphite Cooling	
Piping and components from RWS-2 backup through the reactor including piping to the dump valves.	2
33 High Pressure Injection Water	
The system starts with the pumps in 182-N and ends at the connection to the primary loop. The recuperative heat exchangers are part of the system. It also includes the related piping and controls. The high pressure injection water pumps supply the primary loop with makeup water and also supply the primary pump seal water.	
High pressure injection pumps and piping to the IWV-551.	2
40 Secondary Cooling System	
Includes the secondary side of the 12 steam generators to exterior of the confinement walls.	2

TABLE 3.9-3

ASME CLASS 3 SYSTEMS/COMPONENTS

System	Class
04 Boiler Steam Generator	
Equipment used to generate high pressure steam is included in this system. The boiler proper and boiler external piping and joints as described in ASME Section I P-58.	3
10 Circulating Raw Water	
All raw water systems and equipment except the emergency raw water, fire system and fog spray systems are part of System 10. This system includes the screens, pumps, piping controls, instrumentation and tanks. The system starts at the river and ends at the river or the junction of raw water line and the system it feeds or backs up.	3
16 Demineralized Water	
The system demineralized water system starts at the first block valve on the filtered water influent to the 163-N. It includes all piping, tanks, heaters, pumps, controls, etc., in the 163-N which transport, contain or control demineralized water. The breakpoint between Systems 16 and 09 is at the mixing tees adjacent to the acid and caustic tanks. System 16 includes the supply piping to the demineralized storage tank itself, and the feed lines to the suction of the low pressure injection pump sets. It also includes the alternate supply piping which feeds the suction lines of the afterheat removal and secondary loop makeup pumps. It further includes the recirculating polishing piping which takes off from the discharge of afterheat removal fill pumps 1 and 2 and feed back to 163-N.	3
23 Filtered Water	
This system starts at the suction of the coagulator feed pumps in 183-N and it takes in the chemical mix tank, the coagulators, filters clearwell, storage tank, pumps and piping. The filtered water system provides water for the demineralized system, sanitary system and for several other purposes. Demarcation between the systems is at pump suctions or at manifolds that distribute the water to other systems.	3/4
26 Fuel Oil	
Boiler fuel oil supplies from day tanks.	3

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TABLE 3.9-3 (cont.)

System	Class
29 Graphite Cooling	
The pumps, heat exchangers, piping and tubes in the 105-109 buildings compose the graphite cooling system.	3
32 Rod Safety System	
Rod cooling system includes a closed loop and is designed to ANSI B31.1 Code with the exception of rods and hose connectors.	3
36 Low Pressure Injection Water	
The low pressure injection system starts with pumps in 182-N, supplies the graphite coolant system in 109, backflush and other services in 105, instrument backwash in 109, and water to the emergency seal water steam turbine-driven pump. It includes recirculating piping, in addition to the main piping and valving instrumentation and other related equipment.	3
40 Main Steam and High Pressure Condensate	
Main steam and high pressure condensate.	3
Includes the 16 dump condensers, condensate pumps and lines, surge tanks, main steam lines to WPPSS point of responsibility, feedwater lines from WPPSS and all other main steam lines exterior to the confinement structure.	3

TABLE 3.9-4

CLASS 4 SYSTEMS/COMPONENTS

System	Class
01 Afterheat Removal	
The afterheat removal system starts at the afterheat storage tank inlet. It includes the 182-N afterheat removal pumps, secondary loop makeup pumps and the related piping to 109-N and 184-N Buildings. In 109-N it includes the 14-inch and 12-inch MW supply headers, the surge tank low level makeup stations, the 14-inch lines to MWV-203, the 10-inch MW-1 header, and the 6-inch branch feeders to the steam generators and dump condensers. It also includes the extension of the 12-inch emergency backup line to the deaerator-heater and the 16-inch normal makeup to the deaerator-heater in the 184-N Building.	4
The remaining components of the system 04 outside of ASME Section I jurisdiction but within ANSI B31.1 jurisdiction.	4
08 Chemical Decontamination	
Includes suction and return piping, pumps and piping to, but not including valves at injection points of the primary and graphite cooling systems.	4
09 Chemical Handling	
Includes the facilities for unloading, handling, storage and injection of gas, liquid and dry chemicals used for water treatment. The system begins with piping to unloading and ends at the junction of the injection line and the line of the system being fed. Chemicals and facilities for decontamination are included in the chemical decontamination system (08).	4
12 Compressed Air	
Air equipment for general service.	4
17 Diesel Oil	
Diesel oil system covers all equipment for handling, pumping and storage equipment for the diesel oil and ignition oil system. It includes the equipment in 166, 181, 182, 184 and 184 Annex, but not the ignition oil burners which are part of System 04. Refer to Table 3.9-2 for Class 2 Components of this system.	4

TABLE 3.9-4 (cont.)

System	Class
18 Drain Systems	
Drain piping and components which pump or transport radioactive effluents within confinement.	4
24 Fire Protection	
Includes fire pumps and distribution piping. (The PSI does not cover fire sprinkler systems and fire protection equipment covered by Codes and Standards of the National Fire Protection Associations.)	4
26 Fuel Oil	
This system includes the pumps and piping for fuel oil and the standby boiler fuel oil supply and return lines.	4
28 Gas	
The gas system includes the helium unloading facilities, the storage tanks, the compressors, heat exchangers, filters, piping and instrumentation.	4
All other gas storage vessels and piping.	4
35 Medium Pressure Steam	
The piping, valves, etc., that are used to distribute 55 psi (MP or HP) steam for heating and miscellaneous services make up the System 35. It includes the condensate return lines (COND but not COND-1, 2, 3, 4, or 5) related to the system. Medium pressure steam is distributed over the entire area, but does not include the secondary steam system. The system starts after the steam is reduced to 55 psi and steam or the condensate return entrance to the deaerator storage tank. Standby boiler atomizing steam is part of System 04.	4
39 Sanitary System	
Sanitary water system piping, valves, etc.	4
40 Main Steam and High Pressure Condensate	
Condensate lines COND-3, COND-4, and COND-5.	4

TABLE 3.9-4 (cont.)

System	Class
43 Water Sampling and Analysis	
All equipment and lines used to sample and analyze process water for quality and/or contents are part of this system. The sample lines for System 43 start after the last block valve in the system being sampled. The system ends where the lines connect to drains and where the signal wires connect to equipment controlling other systems. Such equipment as hot water quality lab, radioactive sampling system, nonradioactive sampling system, nonradioactive sampling system and process water sampling system are included in System 43.	4
Nonradioactive sampling lines under 1-inch size and 400 psi.	4

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TABLE 3.9-5

TESTING OF CLASS 1 PUMPS

Pump Service	No. of Pumps	Inspection/Test Requirements and Frequency - Months PHT	PMT	EMS (HW-73000) Reference
Reactor Coolant Recirculation	6	3*	3*	206
High Pressure Injection	5	3*	6*	221

*also monitored during operation.

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TABLE 3.9-6

TESTING OF CLASS 2 PUMPS

Pump Service	No. of Pumps	Inspection/Test Frequency - Months PHT	PMT	EMS (HW-73000) Reference
Diesel Driven ECCS Low Lift	3	---	3	201
Diesel Driven ECCS High Lift	3	1	3	201
High Pressure Raw Water	3	12	3	605
Diesel Driven Fog Spray	2	12	3	605

TABLE 3.9-7

TESTING OF CLASS 3 PUMPS

Pump Service	No. of Pumps	Reference Values Vib.-in.	P-PSI	Flow-gpm	Insp/Test Req. & Frequency-mos. PHT	PMT	EMS (HW-73000) Ref.
Circ. Raw Water	4	.0050	47	102,000	6	3 (motor)	205
Boiler Feedwater	3	.0006	715*	1,300*	12	3	209
Graphite and Shield Recirc.	3	.0009	*	*	12	12	123
Control Rod Recirc. Cooling	4	.0006	*	*	12	6	218
Afterheat Removal Fill	3	.0006	100*	2,500*	12	3	208
Dump Condenser Condensate	16	.0006	70*	2,200*	12	3	208
Low Pressure Injection	3	.0005	*	*	12	3	211
Demineralized Booster	4	.0009	270*	600*	12	3	211
Filtered Water Clearwell	3	---	34*	7,500*		-	206
High Pressure Filtered Water	4	.0006	232*	1,200*	12	6	216
Low Pressure Filtered Water	3	.0006	112*	3,700*	12	6	216
Turb.-Gen. Surface Condenser Condensate	2	---	87*	650*	12	-	116
Secondary Loop Makeup	2	.0005	105*	450*	12	3	208
Drive Turbine Condensate	12	.0009	---	---	--	3	206

*On EMS performance curves

TABLE 3.9-8

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 1 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
BDV-203	4	Gate	MO	Blowdown from PCS Header	1	A	---	(a)	---
BDV-219	4	Angle	Cylinder	Blowdown From Spill Cooler	3	A	12	(a)	---
BDV-222-1R	1X1.5	**	---	Recup. HE No. 2	1	A,C	---	---	60*
BDV-222-2	1X1.5	**	---	Recup. HE No. 1	1	A,C	---	---	60*
CV-1	8	Check	---	ECCS Inlet (NC)	8	A,C	12	(a)	---
CV-2R	4	Check	---	Safety Flow Control Legs (NC)	16	A,C	12	(a)	---
CV-3	18	Check	---	In PCR Lines (NO)	16	A,C	12***	(a)	---
CV-5	18	Check	---	In PCS Lines (NO)	16	A,C	12***	(a)	---
IWV-252(-1to-5)	6	Gate	MO	Injection to PCR Line	5	B	12	---	---
IWV-253	8	Globe	Diaphragm	Injection to PCR (NO)	3	B	12	---	---
IWV-254	6	Check	---	Injection to PCR Lines	5	B,C	12***	---	---
IWV-267	4	Gate	MO	Seal Water Inlet to PCP's	5	B	12	---	---
IWV-281-4R	1X1.5	**	---	Recup. HE No. 1	1	A,C	---	---	60*
IWV-286	3	Check	---	ESW Lines to PCP Seals	5	A,C	12	24	---
IWV-551	8	Check	---	HPIP Discharge (NO)	5	A,C	12	(a)	---
IWV-6215	6	Check	---	Injection to PCR Line, Cell 6	1	B,C	12	---	---
IWV-6252	6	Gate	MO	Injection to PCR Line, Cell 6	1	B	12	---	---
IWV-6267	4	Gate	MO	Seal Water Inlet to PCP #6	1	B	12	---	---

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(a) This valve will not be tested unless and until plant modification is performed.
* See paragraph 10.4.h.
** Pressure relief valve.
*** To accomplish this test, special recirculation is required to be in effect. Performance of this test is contingent upon special recirculation being scheduled for other reasons.

TABLE 3.9-8 (cont.)

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 1 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
IWV-6286	3	Check	---	EWS Line to PCP #6 Seal	1	A,C	12	24	---
PCRV-202	26	Gate	MO	On 26" PCR Lines	5	B	12	---	---
PCRV-204	18	Globe	MO	Steam Generator Bypass Line	5	B	12	---	---
PCRV-206	6	Gate	MO	Decon. Feed to 26" PCR	5	A	---	(a)	---
PCRV-210	2X3	**	---	On 26" PCR Lines	5	A,C	---	---	60*
PCRV-213	4	Gate	MO	Recup. & Spill Clr Lines to PCS	2	A	---	(a)	---
PCRV-6202	26	Gate	MO	On 26" PCR Line, Cell 6	1	B	12	---	---
PCRV-6204	18	Globe	MO	Steam Gen. Bypass Line, Cell 6	1	B	12	---	---
PCRV-6206	6	Gate	MO	Decon. Feed to 26" PCR, Cell 6	1	A	---	(a)	---
PCRV-6210	2X3	**	---	On 26" PCR Line, Cell 6	1	A,C	---	---	60*
PCSV-201-2	18	Gate	MO	Filtered Water Inlet to PCS	1	A	---	(a)	---
PCSV-201-3	18	Gate	MO	Filtered Water Inlet to PCS	1	A	---	(a)	---
PCSV-202	24	Check	---	PCP Discharge (NO)	5	B,C	12	---	---
PCSV-203	24	Gate	MO	Supply to Inlet Header	5	B	12	---	---
PCSV-210	6	Gate	MO	Decon. Return from 24" PCS	5	A	---	(a)	---
PCSV-6202	24	Check	---	PCP #6 Discharge (NO)	1	B,C	12	---	---
PCSV-6203	24	Gate	MO	Supply to Inlet Header, Cell 6	1	B	12	---	---
PCSV-6210	6	Gate	MO	Decon. Return from 24" PCS, Cell 6	1	A	---	(a)	---
PRV-215	4	Gate	MO	Decon. Inlet to Pressurizer	1	A	---	(a)	---

(a) This valve will not be tested unless and until plant modification is performed.
* See paragraph 10.4.h.
** Pressure relief valve.

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TABLE 3.9-8 (cont.)

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 1 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
RV-2	6	**	Pilot	Reactor Coolant System Relief	2	A,C	---	---	60*
RVTV-202/R	1-1/2	**	---	Pressurizer Relief	1	A,C	---	---	60*
V-3	8	Cone	Cylinder	ECCS Inlet - Corridor 5	8	A	12	(a)	---
V-4	10	Cone	Cylinder	ECCS Dump - Outlet Pipe Space	8	A	12	24	---
V-5-1R	10	Gate	Cylinder	Outlet Header Flush - OPS Left	1	A	---	(a)	---
V-5-2R	10	Gate	Manual	Outlet Header Flush - OPS Right	1	A	---	(a)	---
V-14	6	Angle	Diaphragm	Purity Blowdown	2	B	12	---	---
V-18R	6	Gate	Cylinder	Inlet Header Flush	2	A	---	(a)	---
V-28	8	Gate	Cylinder	Auxiliary Coolant From GSCS	2	A	---	24	---
V-43	4	Cone	AM	Stop Valve for Relief Valve	2	B	12	---	---

(a) This valve will not be tested unless and until plant modification is performed.
* See paragraph 10.4.h.
** Pressure relief valve.

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TABLE 3.9-8 (cont.)

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 2 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
CAEV-500	1-1/2	Check	---	LLDP #1 and #2	2	B,C	12	---	---
CAEV-504	3/4	Check	---	LLDP #3	1	B,C	12	---	---
CAEV-505	3/4	Check	---	LLDP #3	1	B,C	12	---	---
CAEV-573	1-1/2	Check	---	HLDP and FSDP	5	B,C	12	---	---
DWV-820	36	Check	---	Demin Tank to Silo	1	B,C	12	---	---
FWV-812	30	Check	---	FW Tank	1	B,C	12	---	---
FWV-817	30	Gate	MO	ERW Tank to HLDP	1	B	12	---	---
FWV-1003	2	Check	---	Confinement Filter Spray	1	B,C	12	---	---
FWV-3096	2	Solenoid	---	Confinement Filter Spray	3	B	12	---	---
GCDV-1	12	Gate	---	GCS Dump Valves	2	B	12	---	---
GCDV-2	12	**	---	GCS Relief Valve	1	B,C	---	---	60*
GCDV-7	6	Check	---	GCS Dump Valve	1	B,C	12	---	---
GCSV-303	8	Check	---	GCS Pump Discharge Valve	3	B,C	12	---	---
GCSV-316	12	Butterfly	Air	RW Supply to GSCS	2	A	12	24	---
RV-1	4	**	---	Corridor 5, V-4 Header	2	B	12	---	---
RWSV-17	4	Butterfly	Air	Fog Spray Station	1	A	12	24	---
RWSV-56-4,7,10	6	Butterfly	Air	Fog Spray Station	3	A	12	24	---
RWSV-59-2,3,5,6,8,9	8	Butterfly	Air	Fog Spray Station	6	A	12	24	---
RWSV-205	12	Check	---	RWS to GSCS	1	B,C	12	---	---

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* See paragraph 10.4.h.
** Pressure relief valve.

TABLE 3.9-8 (cont.)

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 2 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
RWSV-234	12	Gate	---	GSCS Backup	2	B	12	---	---
RWSV-551-1	30	Check	---	LLDP Bypass	1	A,C	12	24	---
RWSV-553-11	24	Check	---	LLDP #2 to Fog Spray	1	A,C	12	24	---
RWSV-553-21	24	Check	---	LLDP #1 to Fog Spray	1	A,C	12	24	---
RWSV-560	24	Check	---	FS Discharge	2	A,C	12	24	---
RWSV-562	30	Check	---	HLDP Discharge	3	A,C	12	24	---
RWSV-810	30	Check	---	ERW Supply to Fog Spray	2	B,C	12	---	---
RWSV-3003	4	Control	---	BU Supply to 117-N Filters	2	A	12	24	---
RWV-206	6	Gate	Air	FS Control Valve - Cells	5	B	12	---	---
RWV-6206	6	Gate	Air	FS Control Valve - Cell #6	1	B	12	---	---
RWV-207	8	Globe	Air	PG Fog Spray Control Valve	2	B	12	---	---
RWV-504	30	Check	---	LLDP #1, #2 Discharge Valve	2	A,C	12	24	---
RWV-518	20	Check	---	LLDP #3 Discharge Valve	1	A,C	12	24	---
RWV-519	20	Check	---	LLDP #3 Discharge	2	A,C	12	24	---
V-60	16	Check	---	ECS Check Valves	2	B,C	12	---	---
V-61	24	Check	---	ECS Check Valve	1	A,C	12	24	---

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TABLE 3.9-8 (cont.)

INSERVICE TESTING OF N REACTOR CLASS 1, 2 AND 3 VALVES

Class 3 Valve Number	Size-Inches	Type	Valve Operator	Service/Location	No. of Valves	Category	Test Requirements and Frequency (Months) VET	VLRT	SRVT
AEV-131	0.75	*	---	TG, Air Ejector, Relief	3	B,C	---	---	60**
AEV-214	3	*	---	DT, Air Ejector, Relief	5	B,C	---	---	60**
AEV-6214	3	*	---	DT, Air Ejector, Relief	1	B,C	---	---	60**
GCRV-307	1	*	---	Surge Tank, Relief	1	B,C	---	---	60**
HPV-162	2.5	*	---	TG, Air Ejector, Relief	1	B,C	---	---	60**
HPV-212	4	*	Diaphragm	MSH Relief	40	B,C	---	---	60**
HPV-6212	4	*	Diaphragm	MSH Relief	8	B,C	---	---	60**
RWSV-806	72	Check	---	CRW Discharge, Check Valve	4	A,C	12	24	---

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* Pressure relief valve.
** See paragraph 10.4.h.

TABLE 3.9-9

PERMISSIBLE LEAKAGE RATES

Valve	Permissible Leakage Rate (gpm)	Valve	Permissible Leakage Rate (gpm)
GCSV-316	5	RWSV-560	100
IWV-286	10	RWSV-562	100
RWSV-17	50	RWV-504	100
RWSV-53	5	RWV-518	100
RWSV-56	5	RWV-519	100
RWSV-59	5	V-4	10*
RWSV-206	200	RWV-803	50
RWSV-551	100	V-28	1
RWSV-3003	10		
V-61	10	RWSV-553	100

*This applies to any one valve or combination of valves on the same header.

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3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

Significant changes in design criteria and methods for seismic qualification of equipment in nuclear power plants have evolved since the start of the N Reactor. Since initial operation, sufficient seismic reanalysis has been performed to assure safety of operation, but a major redesign and upgrading of structures and systems was not attempted. Following the Chernobyl accident in 1986 and the ensuing safety reviews of the N Reactor, a seismic Task Force was formed and a major program of seismic reanalysis and upgrading was initiated in 1987.

The scope and need for seismic upgrades of systems and components was identified in the Westinghouse Safety Assessment of N Reactor (WHC-EP-0035) (Ref. 2).

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In January of 1988, a program leading to the restart of the N Reactor was identified (Ref. 3). The program, which is directly comparable to licensed commercial nuclear plant practice, consists of a Return-to-Service (RTS) Program and a Long-Term Service Program. The objective of the RTS Program is to restart and operate the N Reactor for approximately 24 months without undue risk to the health and safety of the public assuming the possibility of a safe shutdown earthquake (SSE). This is done by defining a "success path" to safe shutdown. During those 24 months, the rest of the systems currently designated Seismic Category I (or Long-Term Service systems) would be evaluated and/or upgraded to 0.25 g so that the N Reactor could operate for the rest of its service life without undue risk to the public with respect to seismic events. The program demonstrates that the risk to the public during the 24 months following restart is equivalent to the risk to the public over the lifetime of the plant, assuming that the RTS systems and structures are seismically evaluated and/or upgraded prior to restart and that this risk is equivalent to the risk accepted at commercial reactors. The precedent for such restart programs is from the commercial nuclear industry, specifically San Onofre Nuclear Generating Station, Unit 1 (SONGS-1).

This section provides the criteria and requirements for the RTS approach at N Reactor for seismic issues.

3.10.1 Seismic Qualification Criteria

The criteria used to select which RTS systems, structures, components, instrumentation, power, and controls required during and following a 0.25 g seismic event used the following assumptions:

1. The seismic event does not create a loss of coolant accident (LOCA) in the reactor coolant system needed for ECCS once through cooling, nor does a LOCA occur simultaneously with or during a seismic event.
2. Offsite and onsite AC power may be lost during or following a seismic event.
3. The plant must be capable of being brought to and maintained in a safe shutdown condition following a design basis seismic event.

For accident analysis purposes, the NRC requires that offsite power is assumed lost (Criterion 2) if a SSE occurs because the typical electrical distribution system is not designed for an SSE. Therefore, commercial nuclear plants rely on an emergency onsite AC power system. Because N Reactor does not have nor does it require, an emergency onsite AC power system, only those systems capable of operating without AC power are considered for use in the success path to safe shutdown.

At the N Reactor a safe shutdown condition due to a seismic event is cold shutdown, i.e., the reactor is at 0% power, and the reactor is depressurized and being cooled by the emergency core cooling system (ECCS).

To satisfy the three criteria above, i.e., the SSE does not create a LOCA where mitigation systems are required, there is a total loss of AC power,

and assure and maintain safe shutdown, the following systems were selected to be seismically evaluated and/or upgraded to 0.25 g prior to restart:

- o Emergency Core Cooling System and trip functions (once through path only, 181-N thru to 1316-N Standpipe)
- o Ball Safety Systems (BSS) (Hoppers to Exit Drain Valves)
- o Horizontal Control Rod (HCR) System (Scram Hardware Only)
- o Confinement, Steam Vent Valves, Weatherheads, and Vacuum Relief Valves
- o DC Power for RTS Systems
- o Monitoring for Safe Shutdown
  - Startup Range Flux Monitor (one channel)
  - Intermediate Range Flux Monitor (one channel)
  - ECCS Pump Speed
  - V-3, V-4 Valve Position Indicator
- o Seismoscope
- o Engineered Safety Features Actuation System.

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The systems, structures, and components designed to Seismic Category I requirements are summarized in Section 3.2 in Table 3.2-1.

In order to bring N Reactor to a safe shutdown condition and maintain it (Criterion 3), four safe shutdown functions must be accomplished. They are as follows:

- o Reactivity control
- o Reactor coolant inventory control
- o Reactor pressure control
- o Decay heat removal.

Reactivity control (scram) after an SSE can be brought about by either the BSS or the HCR system. The seismoscopes can directly initiate the BSS, and subsequently, the HCR system. A total loss of AC will also initiate the HCR system. Either system can be manually initiated. Even though either system is capable of performing the reactivity control function, both were selected

as part of the RTS Program. This is because redundancy in the scram function was judged desirable.

It was required for the RTS Program that each drive (84 total) of the HCR system be seismically evaluated and/or upgraded prior to restart, this included: the scram pilot valve, the two scram valves, the nitrogen accumulator, the hydraulic piston, the chain-link drive, and the HCR. For the BSS, per ball hopper (107 total) the following were evaluated and/or upgraded prior to restart: the ball hopper, the two ball trip mechanisms, the ball channel, and the exit drain valve. The reactor trip system necessary for the RTS program includes the portion from the seismoscopes, and the manual trips for HCR and BSS. The DC power system necessary to support the HCR and BSS was also to be seismically evaluated and/or upgrade prior to restart.

After an SSE, assuming total loss of AC power, only the ECCS is assumed available to provide core coolant makeup water to the reactor. The normal makeup supply (high-pressure injection pumps) is dependent on AC power, which may be lost during an SSE. After the V-4 valves open and the system blows down, the ECCS provides 300-psig inlet water to the reactor block through the ECCS inlet header. All normal pressure control systems (i.e., pressurizer, dump condensers or steam generators) are assumed disabled by a loss of AC power.

To maintain the reactor in a safe shutdown condition, decay heat must be removed. The ECCS was also selected for this function for the RTS program because it can provide unlimited direct cooling to the reactor for safe shutdown.

To confirm that safe shutdown functions are being maintained, the operator needs to verify that the core is shut down and flow is being provided to the core. Therefore, a single channel of startup and intermediate range flux, speed indication for all high lift and low lift pumps, and position indication for all V-3 and V-4 valves were required to be seismically evaluated and/or upgraded prior to restart.

The reactor coolant pressure boundary (RCPB) and the mechanical portions of the ECCS necessary to be seismically evaluated and/or upgraded prior to restart are discussed in Section 3.8.

To assure that the ECCS would be properly controlled and operated with a total loss of AC, the DC power system necessary to support ECCS operation needed to be seismically evaluated and/or upgraded prior to restart. The ECCS pump diesels are automatically air-started on receipt of a reactor scram initiated by the seismoscopes, or by the engineered safety features actuation system (ESFAS). The V-3 and V-4 valves are opened automatically by the ESFAS. An ESFAS signal will result from any two of the following conditions:

- o Primary pump low speed (4/6 trip logic)
- o Primary coolant low flow (2/4 trip logic)
- o Primary coolant high temperature (2/3 trip logic)
- o Primary coolant low or high pressure (2/3 trip logic).

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In the case of a small pipe break in the balance of the Reactor Coolant System (RCS) the operator must manually cause the V-3 and V-4 valves to open if the RCS pressure drops below 645 psig with a concurrent seismoscope scram. An Emergency Response Guideline will cover this situation for the 24 months of plant operation after restart, pending verification that the confinement system will fully function during the SSE.

The above initiation logic must be seismically evaluated and/or upgraded prior to restart.

The structures that house these systems necessary for safe shutdown after an SSE are: buildings 181-N, 182-N, and portions of buildings 105-N and 109-N that contain the success path systems. These structures need to be qualified for a 0.25 g earthquake, or evaluated to assure that failure of structures and components not designed for a 0.25 g earthquake will not affect the operation of the RTS safe shutdown systems (see Section 3.8).

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

A listing of the RTS components and their system boundaries was prepared. This listing was based on the criteria used to generate the list of equipment requiring environmental and seismic qualification for the N Reactor Equipment Qualification (EQ) Program (see Section 3.11), and modified to identify only those equipment necessary to bring the plant to a safe shutdown using the above criteria. The seismic operability of these components is addressed as part of the EQ Program. The list was also used to verify seismic operability for three-over-one (III/I) interactions.

Prior to restart, the seismic integrity and functionality of the RTS systems needed to be assured to bring the N Reactor to a safe shutdown and maintain it should an SSE occur during the 24 months after a restart. The following describes the extent of the seismic evaluations addressed.

1. The RTS components would be walked down to assure that there are no non-seismic Category I components that could fall and damage the RTS components, commonly called III/I.
2. The anchorages, i.e., anchor bolts, baseplates, support members for major components would be evaluated. Major components are piping, cable tray, panels, rack, pumps, tanks, valves, etc.
3. Walkdowns would also trace the physical routing of each wire run through every junction box, conduit, pull box, cable tray, structural opening, sleeve, ladder, or relay panel from end point to end point.
4. The structures (181-N, 182-N, 109-N and 105-N) that house the RTS systems would be seismically evaluated and/or upgraded to assure that they would not affect operation of the RTS system, or evaluated to assure that the failure of structures not designed to 0.25 g would not affect operation of the RTS systems.
5. The analysis for the piping and structures would be updated from the original analysis by using current analytical methods.

6. Seismic functionality of the RTS systems and components would be provided by assuring that the anchorage of RTS systems and components is adequate and that III/I walkdown problems are addressed. This is in keeping with the NRC resolution plan for Unresolved Safety Issue (USI) A-46 (Ref. 4).
7. Seismic equipment qualification is addressed as part of the N Reactor EQ Program under the Safety Enhancement Program (see Section 3.11), which has a completion schedule separate from the RTS Program. The approach and schedule for seismic EQ is in keeping with what is being applied to the commercial nuclear industry for plants of the same vintage as N Reactor.

Although equipment qualification is primarily addressed in the EQ Program (see Section 3.11), certain critical components were seismically qualified as part of the Accelerated Safety Enhancement Program (ASEP) (Ref. 5) to provide added assurance. They include the steam vent valves, the V-4 valves, and, in response to the Westinghouse Independent Safety Review (Ref. 2), assurance that postulated failures of mercury-wetted relays and switches would not affect safe shutdown. 21

In summary, seismic functionality of the components in the RTS systems would be provided by assurance that the anchorages of RTS components are adequate, assurance that seismic interactions are addressed, and reliance on the inherent ruggedness of power plant components.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

The seismic re-evaluation of electrical and instrumentation systems for the RTS and ASEP programs utilized the methods and procedures and included the site specific response spectra defining the maximum earthquake potential (SSE) of the N Reactor site developed in 1987 (Ref. 6) and discussed in Section 3.7 and 3.8.

Supplemental design analysis guidelines for the evaluation of Seismic Category I instrumentation and electrical distribution systems were prepared (Ref. 7). Applied loads to be considered in the evaluations included:

- o Seismic inertia
- o Gravity
- o Pressure
- o Thermal
- o Interface reactions (i.e., nozzle loads).

Thermal loads were considered only if operating temperatures exceeded 150°F.

The guidelines stated:

> "For seismic inertia loading components and component supports will be evaluated using simplified models and conservative hand calculations (more accurate analysis procedures to eliminate field modifications may be used as special case by case applications). The first approach is to use the peak spectra acceleration of the applicable response spectra multiplied by 1.5 to obtain the maximum equivalent static seismic load."
>
> "As required, a mode/frequency analysis used to determine seismic loads. If the natural frequency of the component is in the rigid range (>30H_Z); an equivalent static analysis with ZPA acceleration can be determined based upon the fundamental frequency using 1.5 times the highest 'g' level at or above this frequency."
>
> "Use damping as recommended by Regulatory Guide 1.61. Increased damping may be used when justified by test/data experience. For example, conduit may use 12% damping. In this case apply the 5% spectra 'g' levels without the 1.5 multiplier discussed above. In any case a 'g' level less than the ZPA value is not allowed."

"For combining earthquake directions the three orthogonal components of seismic motion (loading) should be considered. Combination by square root sum of squares or other combinations allowed by Reg. Guide 1.92 is acceptable. Typically in piping/conduit analyses one need only consider the worst direction loading."

"The amount of contributory mass to be applied at each support should be one-half the weight of the uniformly distributed weight between two adjacent restraints acting in the same restraining direction, plus a proportional amount of any concentrated weight between those two restraints. This contributory mass is then multiplied by the appropriate spectra acceleration to obtain seismic support loads. In support analysis, all three orthogonal directions of seismic motion must be considered." 21

Allowable stress was defined for concrete expansion anchors, components, and component supports.

Results of the seismic re-evaluation program are summarized below.

3.10.3.1 ECCS Trip System - Original Seismic Analysis

Two electrically independent ECCS actuating circuits are provided, either of which, when energized, will cause emergency cooling water to be provided to shutdown the reactor.

Each of the ECCS actuating circuits is actuated by any one of the following:

- o Manual pushbutton activation
- o Any combination of two of the following:
  - a. Low flow rate:
    trip signal - 2 of 4

b. Reactor coolant high or low pressure:
trip signal - 2 of 3
c. Reactor coolant high temperature:
trip signal - 2 of 3
d. Reactor coolant system low pump speed:
trip signal - 4 of 5

- Zone I confinement high pressure (trip signal - 2 of 4) and reactor coolant high or low pressure (trip signal - 2 of 3)

Each of the ECCS actuation circuits energize the master ECCS trip relays that start the ECCS diesel pumps (HCR actuating circuits also start the ECCS diesel pumps), open V-3 and V-4 valves for single-pass cooling, and re-trip the reactivity control systems.

The relays used in the ECCS actuating circuits have very little mass in the moving contact; therefore, it is considered that these relays will not change their contact position due to seismic forces produced during the specified earthquake. The relays should function normally to open or close contacts depending on the sensing instrument actuation regardless of the seismic forces. Once either of the ECCS master trip relays has been actuated (two of four relays are in a position to start the ECCS), the diesel drive will start and the V-3 and V-4 valves will be opened for ECCS diesel pumped water flow; any change in the master trip relay circuit will not have any effect on the emergency safety action, i.e., the coolant flow will continue on single pass through the reactor.

The pressure switches, pressure transducers, and pump speed indicator switches, which could be required to function during a seismic event to provide shutdown emergency cooling, have been reviewed. Seismic motion causing closing of the contacts would be in a safe direction, i.e., safe shutdown on emergency cooling, and it is considered unlikely that the motion would prevent the devices from operating as designated.

A trip of the ECCS trip circuit results in (1) energizing V-3 and V-4 control air solenoid valves causing the V-3 and V-4 valves to open to allow single

pass emergency cooling and (2) starting of the high lift and low lift pump diesel engine drive by de-energizing air solenoid valves. The V-3 and V-4 solenoid control valves are classified Seismic Category I. Seismic qualification testing of these V-3 and V-4 control solenoid valves is described in Section 3.9.2. The test proved that the valves could be activated and would not inadvertently actuate during exposure to a range of acceleration frequencies and intensities more severe than those caused by the specified earthquake. The diesel drive starting air solenoid valves are normally energized in a closed position. It is considered unlikely that seismic motion would prevent the valve from functioning, and if seismic motion were to change the position it would be in the safe direction of opening the valve and starting the diesel engines.

3.10.3.1.1 Seismic Reanalysis of the ECCS

The logic sequence required to safely shutdown the N Reactor after an SSE and maintain it in a safe shutdown condition using the ECCS was evaluated. This involved the seismic qualification of the Category I raceway for the ECCS.

Seismic qualification of the Category I raceway, which was defined to include both conduit and cable tray, was a threefold activity. The first was to develop and approve a wire run number listing which lists the circuits (Category I Cabling) required to be seismically supported to requirements for the RTS program. This listing was provided by the RTS program (Ref. 3). Second was to develop and approve the schedule for raceway used in supporting the Category I cable. This was accomplished by drawing review and walking down the cable runs to determine the raceway used. Third was to qualify the Category I Raceway to ensure that the supports are not compromised by the Hanford SSE. 21

The wire run number in general describes the electrical circuit, the number of conductors, specific conduit or tray used for routing ("via") and the "to-from" (beginning and end point) of the conductor or cable. Since most systems terminate in Room 6 or the Control Room, much of the raceway becomes

common to all systems as the cables approach Room 6. Because of this, cable tray and conduit internal to Room 6 were treated as a special case.

All 239 tray supports in Room 6 and the conduit were seismically qualified separately, both of which are documented in Reference 10. Eight new supports were required, 68 repairs and 24 modifications were made. In addition the stability of seismic Category I electrical panels was reviewed and any concerns were resolved. This work was documented in Reference 11. Therefore, once the wire run enters Room 6, no further tray raceway qualification was required for the various control systems.

The entire length of raceway was walked, photographs taken and the raceway qualified either by analysis or by Design Analysis Guidelines Acceptance Criteria (Ref. 7) summarized above in 3.10-3. In cases where the raceway could not be qualified by analysis, fixes were generated that would seismically qualify the raceway. All calculations and analysis were independently reviewed and checked. For ease and convenience in walking the raceway, the ECCS was divided into subsystems. The wire run numbers from Reference 1 were then separated into the various subsystems. The subsystems chosen were:

- Three low-lift pumps, located in Building 181-N, which supply ECCS water to the high lift pump.
- Three high-lift diesel pumps, located in Building 182-N, which supply ECCS to the V-3 valves.
- The V-3 valves that, when open, allow ECCS flow to the reactor core. These valves are located in Building 105-N, Corridor 5.
- The V-4 valves that, when open, allow depressurization of the reactor coolant system and allow ECCS flow through the core. These valves are located in the outlet pipe space in Building 105-N.
- The Engineered Safety Features Actuation System (ESFAS) which includes the following:

    - Primary coolant pump low speed (rotation sense)
    - Primary coolant low flow

- Primary coolant low or high pressure
- Primary coolant high temperature
- Confiner Pressure Trip System.

o The ECCS Logic Circuit Power, which essentially covers the interfacing circuits in Room 6 and between Room 6 and the Control Room, completes the subsystem designations chosen for the ECCS.

The ECCS is activated in the following manner. The diesel pumps are automatically air-started on receipt of a reactor scram initiated by the seismoscopes, or by an ESFAS signal. The low and high-lift diesel pumps also start with actuation of the Horizontal Control Rod (HCR) scram circuit. The V-3 and V-4 valves are automatically opened by the ESFAS. Further description of the ECCS and its operation is found in Sections 6.3 and 7.3 and in Reference 3.

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The analysis and Raceway Qualification and required fixes are fully described in Reference 9. A short description of the analysis results for each subsystem is provided below with a summary table of fixes required and implemented.

Low-Lift Diesel Raceway

The low lift diesel circuits are routed in conduit in the 181-N Building. The analysis indicates that supports 5 and 6 require the removal of rods that support Seismic Category III conduit below. No Seismic Category III over I problem results from this rework.

High-Lift Diesel Raceway

In April 1987, Kaiser Engineers performed seismic analysis for the High-Lift Diesel Control Separation Project No. 230490 - Design Change 87125 which included the conduit for 182, 125-Vdc batteries. The original analysis was performed to horizontal and vertical "G" levels of 0.69 and 0.46, respectively. Subsequently, response spectra analysis was performed and response spectra was developed (Ref. 7) that had "G" levels of 3.4 and 3.0 horizontal and vertical. The original analysis was revised and included. No structural rework was required.

The remaining conduit analysis for the indication circuits connecting the master terminal box and junction boxes showed that support 1 and 10 required rework. All other supports were determined to meet the seismic criteria.

A separate seismic evaluation was made of the circuits that are routed from the 181 and 182-N Buildings in buried underground duct banks to the 109 Building. The analysis results of the typical duct bank indicate the stresses are within the allowable stress values indicated in SDC 4.1, Rev. 9. The duct bank system is capable of withstanding a SSE and will remain functional during and after the SSE. In the 109 Building, the circuits are routed in a cable tray to Corridor 1 in the 105 Building. Both of the duct banks and the 109 cable tray were seismically qualified.

The analysis for the 105 Building Corridor 1 supports indicated that the beams were locally overstressed and required the addition of gusset plates. All other supports were found to be structurally adequate.

V-3 Valve Raceway

There are two sets of actuation circuits for the V-3 valves. The "old" circuits run in conduit from the valve actuators through the north wall into a cable tray. Analysis of the conduit showed the supports to be adequate.

The circuits are routed from the conduit into Tray 23 through Room 31, 30, 29, 27, and 2. This tray routing was qualified in the HCR task (Section 3.10.3.3). No support fixes were required in these rooms.

From Room 2 the circuits are routed through the electrical tunnel and the supports required reinforcing to be seismically qualified. This analysis and redesign is discussed in the seismoscope discussion (Section 3.10.3.7).

New circuits at the V-3 valve actuators were routed in conduit from eight locations in Corridor 5 through Corridor 2 to Room 8. The circuits were then routed to the junction boxes and control panels in Rooms 8 and 9. The analysis of all new conduit supports indicated they were seismically qualified.

V-4 Valve Raceway

V-4 valve actuation circuits are routed from the outlet pipe space valve actuators in conduit to a junction box then to Room 9 via embedded conduit. Analysis of the conduit supports showed that the supports were seismically qualified and the embedded conduit was acceptable as is.

Cable routing indicated that the circuit wiring is routed from conduit to trays 123 and 135 in Room 8 and across Corridor 2 to Room 6. The supports for these trays were qualified in the BSS task, discussed in Section 3.10.3.2. and were acceptable as is.

Summary

Twenty-eight supports were determined to require modification and six "fix packages" were prepared and implemented. The conduit and all other supports required for the ECCS actuation circuits were determined to be seismically qualified as is. 21

A separate III/I walkdown (Ref. 8) was also performed in the 105-N/109-N Building of these same ECCS actuation circuits. The purpose of the walkdown was to assure that no failure of non-seismic equipment and components would adversely affect the ECCS instrumentation.

The walkdown and evaluation identified a total of sixty-five potential problems. Of these, eighteen required fixes and the remaining forty-seven were resolved by analysis and/or application of the Generic Acceptance Criteria (Ref. 7) resulting in seismic qualification of the ECCS instrumentation and electrical circuitry, conduit, cable trays, and supports reviewed.

Graphite and Shield Cooling System and Backup Cooling Trip System

The trip system for actuation of the graphite and shield cooling system backup cooling supply is fail-safe, and seismic motion results in opening of the circuit input switches and relays which is in the safe direction.

The N Reactor GSCS is designed with an emergency once-through cooling mode and has the capability to back up the ECCS. Backup supply for the GSCS is provided via the high pressure raw water system (RWS-2) or the diesel driven fog spray pumps. Indication of low bulk flow and/or high bulk temperature discharge temperature in the system initiates opening of both a 12-in. dump valve in the system return line and a 12-in. valve in the raw water supply line. A more complete description is provided in Sections 5.5, 6.3 and 7.3.

A III/I walkdown of this backup system to the ECCS was also conducted because the GSCS sensors, sensor lines, circuits and cabinets for actuation of once-through emergency cooling of the moderator and shields from the sensors to Room 6 must be protected from damage by non-seismic Category I failures. The actuation circuits to the RWS-2 and dump valves were not included in the walkdown because they are fail safe and open on loss of electric power.

The III/I walkdown (Ref. 8) identified a total of 25 potential problems. Of these, 5 required fixes and the remaining 20 were resolved by analysis and/or application of the generic acceptance criteria.

From the above it has been concluded that with the indicated fixes completed prior to restart the instrumentation for the ECCS and its backup system is seismically qualified.

3.10.3.2 Ball Safety System - Original Evaluation

The HCR system is backed up by the BSS which in turn is activated by failure of the rods to enter the reactor on demand, flux levels higher than preset values following a rod scram, seismic activity or ECCS actuation. The BSS has two electrically independent trip circuit channels (A and B) with two master trip relays in each channel. A BSS trip requires a trip of one master trip relay in channel A and a trip of one master trip relay in channel B in order to actuate the safety action for release of the neutron-absorbing balls into the reactor. The individual circuit relays and the master trip relays are of the same type as used for the rod safety circuit, and seismic motion would cause failure in a safe direction, i.e., to open the circuits and release the neutron-absorbing balls to shut down the reactor.

Because of the fail-safe design of the BSS trip circuit, as noted in the previous paragraph, and because the BSS trip circuit design provides for reactor shutdown in the event of low-level seismic activity, instrumentation and sensors other than the seismoscopes are not classified Seismic Category I. The BSS master circuit relays, which operate the BSSs, are designated Seismic Category I.

A BSS circuit trip de-energizes solenoids which trip the ball hopper latching mechanisms allowing the neutron-absorbing balls to flow into the reactor. Once the latching mechanism has been tripped, further relay action has no effect on ball release. The ball hopper solenoids and associated trip mechanism are designated Seismic Category I. Seismic qualification testing of the hopper trip mechanism is described in Section 3.9.2. The qualification testing proved that the components remained functional during exposure to a range of accelerations, frequencies and intensities more severe than those caused by the SSE.

3.10.3.2.1 Seismic Reanalysis of the BSS

The BSS instrumentation and sensors were seismically qualified and upgrade to Category I as documented in Reference 16.

Wire run lists were prepared and the system raceway walked down, analyzed and evaluated. It was concluded that the BSS Seismic Category I Raceway containing actuation circuits and 125 Vdc power supply are seismically qualified for the routing from the ball drop solenoids to Room 6. No fixes were required for the Ball Safety Circuits Raceway. 21

The separate III/I walkdown team identified a total of twenty-seven problems. Of these, ten required fixes and the remaining seventeen were resolved by analysis and/or application of the generic acceptance criteria (Reference 8).

3.10.3.3 Horizontal Control Rod Trip System (HCR) - Original Evaluation

The reactor trip system (RTS) circuit is designed fail-safe in that loss of electrical power to either of two independent channels will actuate the HCR system, and failure of either channel will not prevent the system from

functioning to shut down the reactor. Sensor input circuit relays into the RTS circuit are of a design such that the seismic motion results in opening of the RTS circuit which is in the safe direction. The RTS master circuit relays are of a type that would not fail to open during seismic motion if de-energized by a trip signal, and if the seismic motion were to impact the operation of the relay it would be in the safe direction of opening the relay and shutting down the reactor. These relays were subjected to shock and vibrational testing at the factory. The RTS master circuit relays which operate the HCR and BSSs are designated Seismic Category I.

A trip of the RTS results in de-energizing individual and solenoid control valves which release hydraulic fluid stored in accumulators to insert the horizontal control rods. The valves are designated Seismic Category I. Seismic qualification tests of the valves and valve racks is described in Section 3.9.2. The qualification testing proved that the components will remain functional during exposure to a range of acceleration frequencies and intensities more severe than would be caused during the SSE.

3.10.3.3.1 Seismic Reanalysis of the HCR Trip System

The HCR Scram System necessary for RTS are the manual trips and the DC Power Supply System necessary to support the HCRs (Ref. 3). Other system inputs that provide rod scram are as follows:

- Intermediate Range Nuclear Instrumentation (RTS) - Section 3.10.6
- Ball Safety System Backup (RTS) - Section 3.10.3.2.
- Power Range Nuclear Instrumentation - Section 7.2.2
- Flow Monitor System - Section 7.2.2
- Zone Temperature Monitoring - Section 7.2.2
- Reactor Coolant Pressure Monitor (ECCS) (RTS) - Section 3.10.3.1
- HCR Cooling Water Flow Monitor (ECCS) (RTS) - Section 3.10.3.1.

As indicated above, the HCR Scram Electrical Control System logic interfaces with the other RTS Systems which are identified above with appropriate references. A complete description of all the HCR Scram System is provided in Section 7.2.

Wire run lists were prepared and system conduit, raceway, and cable trays identified and the HCR trip circuitry was walked down, analyzed, and evaluated.

It was concluded that the cable tray and conduit in Rooms 20, 37, 34, 33, 32, 31, 30, 29, 27, and 2 were determined to be seismically qualified as is. In Rooms 170, 176, Corridor 4 and in the electrical tunnel a total of 31 support fixes/modifications were required. The fixes were installed and the HCR raceway has been seismically qualified. 21

Concurrently a III/I walkdown was conducted of the HCR system and circuits that must be protected from damage by non-Category I failures.

This walkdown and evaluation (Ref. 8) resulted in identification of 11 problems of which 3 required fixes that were subsequently implemented. The remaining 8 were resolved by analyses and/or the application of the criteria listed in Section 3.10.3.

3.10.3.4 Confinement Trip System - Original Evaluation

The pressure-actuated controls that initiate confinement are made up of two circuits, either of which when de-energized will start the confinement sequence.

A trip of the confinement electrical control circuit results in (1) de-energizing hydraulic control valves which actuate various confinement isolation valves, (2) de-energizing solenoid-controlled confinement isolation backup valves, (3) shutdown of supply and exhaust fans and (4) other ventilation system changes required to achieve maximum confinement.

The hydraulic solenoid control valves for operation of the confinement isolation valves have been reviewed. Because of their design and redundancy, it is judged unlikely that seismic motion would prevent their operating. Contrariwise, seismic motion would tend to overcome any static friction associated with start of valve movement.

The fog spray system activation circuit consists of two fail-safe circuits, both of which have to be in the open condition to actuate the fog sprays. The circuitry and relays are such that seismic motion results in opening of the circuit which is in the safe direction.

3.10.3.4.1 Seismic Reanalysis of Confinement Trip System

An evaluation (Ref. 22) for the effects of a SSE event on the 100-N Facility 105-N and 109-N Building steam vent valve actuation components and electrical raceways was performed to support the N Reactor safety enhancement effort (Ref. 5) and the RTS Program (Ref. 3). The evaluation pertained to the two (2) 105-N Building and to the thirteen (13) 109-N Building steam vent valve installations which are Seismic Category I engineered safety systems.

An analysis and evaluation of the steam vent valve actuation system components was performed and the results were applied to assess structural adequacy for the effects of an SSE event. Actuation system elements addressed in the qualification effort included conduits, cable trays, and raceway supports. Steam vent valve actuator components were qualified by test. Seismic accelerations, material stress-related allowable information, and acceptance criteria were provided by the report "Design Analysis Guidelines for Evaluation of Seismic Category I Electrical Distribution System Components," summarized in Section 3.10.3.

Dynamic test and seismic analysis results were applied to qualify the steam vent valve assembly. The test article included all electrical, hydraulic, and pneumatic items that are necessary to assure operability. The tested assembly comprised the valve stack, the hydraulic and pneumatic controllers, the actuators, and the hydraulic accumulator. Results of the test and

analysis were provided by Reference 12, "Test Report of CVV-206-8 Qualification Test."

The steam vent valve attachment to the confinement pressure boundary was evaluated for the effects of an SSE event combined with confinement pressure (Ref. 13). Results of the analysis were applied to show that the attachment was not compromised by the subject loading.

An analysis and evaluation of steam vent valve actuation system elements were performed and the results were applied to assess structural adequacy. Actuation system elements addressed in the qualification effort included conduits, cable trays, raceway supports, and 105-N building pneumatic and hydraulic supply accumulators. This effort was arranged to address major sections of the actuation system such as the pneumatic actuation component raceway supports. Calculations were compiled into seven (7) groups as follows. 21

- o Pneumatic actuation raceway supports in the 109-N Building interior
- o Hydraulic actuation system raceway supports (CVV-206-1 through -11) in the 109-N Building interior
- o Hydraulic actuation system raceway support (CVV-206-1 and -2) in the 109-N Building interior
- o Hydraulic and pneumatic actuation system raceway supports on the 109-N Building roof
- o Hydraulic and pneumatic actuation system in the 105-N Building
- o Inter-support conduit qualification
- o Inter-support raceway qualification.

The results were applied to qualify the actuation system for the effects of an SSE event.

A total of six (6) raceway support modifications were identified and installed; five (5) in the 109-N and one (1) in the 105-N building. They include the following fix-types:

Fix-type	No. Fixes
Add new support	3
Strengthen support	3

The conclusion drawn from this structural evaluation is that the Seismic Category I steam vent valve actuation components and the associated electrical raceways are qualified for the effects of a SSE event provided with the approved raceway support modifications implemented. No modifications to the actuation components, electrical raceways, and electrical conduits are required. The walkdown, the as-built, and the analysis and evaluation information that document this conclusion are contained in Appendices A through C of Reference 3.10-22.

The RTS Program required that the seismic qualification of Category I Electrical Raceway for the 105-N Confiner Pressure Trip System be verified.

The 105-N Confiner Pressure ECCS Trip System consists of pressure sensors located in four separate locations; Room 302, Room 403, Corridor 4 and Room 404. Eight redundant pressure switches monitor 105-N Zone I space pressure and feed redundant signals into two Zone I high-pressure circuits that require two-out-of-four coincidence to trip. These in turn feed two confiner trip circuits. The confiner trip in conjunction with a signal from the reactor coolant loop pressure trip circuits, through time-delay relays, actuates the ECCS. The confiner trip is actuated by a two-of-four coincidence and sends a trip signal to the master relay trip circuitry of its respective trip system. The confiner trip cannot be bypassed. Typical settings are Zone I high pressure, 25 in. H_2O, and reactor coolant system low pressure, 645 psig with 5-second time delay.

The confiner trip was established specifically for protection in event of the occurrence of a small to moderate size break in one of the 16 supply

risers to the reactor upstream of the check valves. Additional description is provided in Sections 6.2 and 7.3.

A wire run list was developed, system raceway identified, and the physical routing of each wire run through every junction box, conduit, pull box cable tray, structural opining, sleeve, ladder or relay panel from end point to end point was traced and verified. The analysis and evaluation was to qualify the raceway, non-Category I components whose failure could prejudice the Category I raceway were not included in this analysis. Nor did the analysis included qualification of devices such as relays, switches, etc. Seismic qualification of equipment was performed under the Equipment Qualification Program (EQ) discussed in Section 3.11. The analysis did however, cover anchorage of this equipment.

The results of this evaluation (Ref. 21) was that one fix package was identified and implemented and it was concluded that the Category I raceway for the 105-N Confinement System wire runs identified were seismically qualified. 21

Three III/I walkdowns were performed as described below.

A III/I walkdown of the Confinement Actuation System components that must be protected from damage due to failure of non-seismic Category I structures and components was performed (Ref. 8).

The III/I walkdown team identified a total of eighty-two potential problems, of these, twenty-two required fixes and the remaining sixty were resolved by analysis and/or application of the Generic Acceptance Criteria (Section 3.10.3).

Thirteen of the fixes were to Seismic Category I cable trays. All recommended fixes were subsequently implemented.

A III/I walkdown and evaluation specifically directed to the steam vent valve actuation components was also performed (Ref. 21). The conclusion drawn

from this qualification effort was that the non-Category I items will not adversely affect the Category I systems during and subsequent to a SSE event provided that the approved modifications are implemented. A total of five (5) modifications were recommended for Building 105-N systems and a total of twelve (12) modifications were recommended for Building 109-N. These modifications have been implemented (Ref. 23).

A III/I walkdown of the Filter Spray Actuation System was also performed (Ref. 8). The Filter Spray Actuation System consists of nine temperature switches which activate the confinement system filter sprays in the event of high filter temperatures that could result from deposition of large quantities of fission products. There are three temperature switches for each of cells A, B and D in Building 117. High temperature indication energizes a latching relay which in turn energizes a relay to supply power for the filter spray control valve for that cell. A more complete description is provided in Sections 6.5 and 7.3.

The III/I walkdown team identified a total of thirty problems. Of these, three required fixes and the remaining twenty-seven were resolved by analysis and/or application of the Generic Acceptance Criteria (Section 3.10.3). The recommended fixes were implemented.

3.10.3.5 DC Power for RTS Systems - Original Evaluation

The seismic qualification criterion used for Seismic Category I electrical equipment is that electrical equipment, when subjected to the vertical and horizontal acceleration of an earthquake with a maximum ground acceleration of 0.25 g, shall perform as follows:

1. Equipment shall retain its structural integrity.
2. Equipment shall be capable of performing its design function.

Electrical Equipment

Seismic qualification testing of Seismic Category I circuit breakers is documented in Reference 1. The tests were carried out by mounting accelerometers on breakers in simulated cabinets and then subjecting the

equipment to cyclic accelerations. The applied accelerations were in excess of the earthquake ground acceleration of 0.25 g. These tests demonstrated that the breakers were stable under severe shock levels.

Seismic Category I type electrical equipment was analyzed for support of buses, circuit breakers, relays and for support of the panels and cable trays to the building structures. This equipment includes the following:

Room 9	125 Vdc batteries and supporting racks for A and B batteries
Room 174	125 Vdc battery C and supporting rack
Room 8	125 Vdc switchgear for batteries A, B and C
Room 6	125 Vdc distribution panels with circuit breakers - includes distribution panels CF, CE, CD and CN.
Room 6	Control cabinets for support of magnetic relays, arc suppression diodes and wiring - includes the following panels: PR-36, 43, 45, 46, 47, 48, 49, 50, 51, 54, 60, 61, R1, R2, S1, S2, W1 and W2.

Batteries were inspected for battery rack support to walls and floor, barriers between cells, and objects that could fall from above the battery cells within the battery room. New battery racks were installed in 1977. These racks were designed for UBC Zones II, III and IV activity which is greater than that which would be experienced during the SSE.

Battery switchgear was inspected for bus and breaker support and for structural support to the building concrete floor. The switchgear located is adequately supported by bolted connections to the room floor and switchgear bus, and breakers are adequately supported within the steel switchgear enclosure.

The cable trays in room 6 are supported by steel hanger rods suspended from the structural steel beams in the ceiling of room 6.

The 125 Vdc distribution panels in room 6 were inspected for bus and breaker support and support of the distribution panels to the walls and building structural steel. DC distribution panels, which contain 10 ampere circuit breakers for electrical supply to the ECCS, fog spray and confinement activation circuits, have been upgraded for seismic resistance. Panels in room 6 which contain logic actuation relays, diodes and electrical wiring are judged based on static analysis of the anchors to be adequately anchored to the concrete floor. Anchor bolts are 3/8-in. size and are spaced on 24-in. centers on back and front steel channels for support of the panel.

3.10.3.5.1 Seismic Reanalysis of DC Power for RTS Systems

The RTS program requires that portions of the three safety related 125 Vdc, 32 Vdc and 24 Vdc systems identified as Emergency Power System (EPS) in the wire run listing be seismically qualified. The systems required by the RTS Program are the following:

Voltage	Battery	Location
125 Vdc	A & B	105-N Room 9
125 Vdc	A	181-N
125 Vdc	A	182-N
32 Vdc	A	105-N Room 174
24 Vdc	D & E	105-N Room 6

The 125 Vdc A & B systems in Building 105-N supply power to the ECCS logic relays, solenoid valves for V-3 and V-4 operation, and the No. 1 and No. 2 low and high lift diesel speed controls and air start solenoids. Low Lift Diesel No. 3 start system and speed control are powered by 125 Vdc batteries in the 181-N Building and High Lift Diesel No. 3 Start System and speed control are powered by 125 Vdc batteries in the 182-N building. The 32 Vdc batteries power the startup range and intermediate range flux monitoring instrumentation.

The 24 Vdc batteries D and E power the display instrumentation for CVV 23 and 24 open/close status, V-3 and V-4 open/close status, and the seismoscope

trip indicating lights. A more complete description of the above systems is provided in Reference 1 and in Section 8.3.

The emergency power system includes the raceway in the 105-N Building from Room 174 for the 24 Vdc batteries extending into Room 9 picking up the 125 Vdc batteries and continuing through Room 8 across Corridor 2 and into Room 6. Embedded conduit in concrete between Rooms 174, 9, and 8 is considered seismically qualified. Portions of wire run in Room 6 were qualified in References 26 and 27. The 181-N and 182-N circuits were qualified in Reference 28. The 182-N Building interior circuits raceway are qualified in DC 87125 which provided IEEE Class 1E separation for the control circuits and was seismically qualified. The 182-N battery supply circuit for High Lift Diesel No. 3 is routed in conduit embedded in concrete and is considered seismically qualified. Four trays identified that contain wire run numbers were qualified in Reference 29. Room 6 conduit was qualified in Reference 27 and cable trays in Reference 26. The remaining conduit in Rooms 174, 8, and 9 was analyzed and evaluated (Ref. 20). It was concluded that all of the conduit in the Emergency Power System was seismically adequate as is. 21

3.10.3.5.2 Seismic Category I Electrical Distribution

The RTS program (Ref. 3) requires that all systems and structures in the logic sequence required to safely shutdown the N Reactor after an SSE and maintain it in a safe shutdown condition are shown to be seismically adequate or upgraded prior to restart. This section documents the seismic qualification of Category I Electrical Raceway for the RTS Systems. The RTS Systems and components documented in this report are:

- Seismoscopes and actuating circuits
- Ball Safety System
- Horizontal Control Rod System
- Emergency Core Cooling System
- Engineered Safety Features Activation System (ESFAS)
- Low Level and Subcritical Flux Monitors and actuating circuits
- Emergency Power System.

All wire runs for the above systems included in Volume 2 of Reference 3 are covered by the following analysis.

Where detailed analyses led to the conclusion that the raceway did not meet seismic Category I requirements fix packages were prepared. In all 23 fix packages were required. These fix packages involved approximately 70 supports not including repairs. All identified fix packages in Reference 12 were completed (Ref. 23) and the Category I raceway for the RTS wire runs identified is seismically qualified.

A III/I walkdown of the Battery Power Supply system was performed to assure that the 125 Vdc systems battery breakers, cabinets, wiring trays and conduit are protected from damage by non-Category I failures. The analysis and evaluation included the two battery rooms located in the 105-N Building, Rooms 6 and 9 (Ref. 8).

The III/I walkdown team identified a total of 11 potential problems. Of these, 7 required fixes and the remaining 4 were resolved by analysis and/or application of the generic acceptance criteria (Section 3.10.3). All fixes were implemented.

Previously, in 1987, in response to ASEP Issue No. 71 (Ref. 5), the two battery rooms "A" and "B" located in 109-N were evaluated for seismic adequacy.

Equipment other than the batteries that are within or near the battery rooms include:

- Free-standing locker cabinets (one in each room)
- Two power panel boxes attached to Room B walls
- Wall-mounted eyewash stations
- Ceiling-mounted exhaust fans
- Two breathing air bottle racks next to Room B
- Breathing air piping and valves attached to Room B walls.

The battery rooms were installed in the 109-N Building in 1970. The breathing air system was installed in 1984. Neither the battery rooms nor the breathing air bottle rack system have been previously evaluated for SSE adequacy.

The evaluation (Ref. 30) concluded that the battery enclosures were inadequate and that the Turbine Bay was not seismically adequate. It noted that circuit breakers 5 and 8 connected to the batteries in Room A and B in Panels DA1 and DA2 were required for Safe Shutdown. Also, Category I cables which carry power from the 105-N Control Room and Room 6 are routed in cable trays along the ceiling of Corridor 19 and the ceiling of the 109-N Building basement where they penetrate the south wall of the Turbine Bay into underground ducts. This portion of the 109-N Building is not seismically adequate and the supports for the cable trays are not adequate to assure that the trays would not fail during the SSE. 21

As a result of this evaluation a design change was initiated and implemented that would eliminate the potential impairing of the control capabilities of High Lift Diesels #1 and #2. The present 109-N DC supplies (circuits DA1-5 & DB1-5) were deleted by this design change and the circuits resupplied from existing safety-related batteries 105-N A & B. To eliminate the potential impairing of the control capabilities of all High and Low Lift Diesels additional cable tray supports were provided as required by this design change.

As discussed in Section 3.2, the Turbine Bay has been reanalyzed and structural modifications have been designed which will seismically qualify the 109-N Turbine Bay Structure. This was required because of the need to seismically qualify the RWS-2 piping system to functioning of the fog spray system and the once-through graphite and shield cooling system. This modification is part of the RTS program and must be completed prior to restart of N Reactor.

This modification provides additional assurance that the Category I confinement circuits and ECCS circuits will remain functional.

3.10.3.6 Monitoring for Safe Shutdown

The RTS program identified the following four system inputs as required prior to restart:

- o Startup Range Flux Monitor - one channel
- o Intermediate Range Flux Monitor - one channel
- o ECCS pump speed
- o V-3, V-4 valve position indicator.

The ECCS pump speed and the V-3, V-4 valve position indicator circuits were reviewed, analyzed, and qualified as part of the Category I actuation circuits qualification discussed in Section 3.10.3.1.1.

The IRFM consists of a gamma compensation ion chamber which is fixed in place and is powered by dual high voltage power supply modules. The SRFM consists of a fission chamber which is part of an integral hydraulically operated assembly that can be inserted into the reactor. The hydraulic system is not required to function for RTS qualification. A more complete description of these systems is provided in Sections 7.2 and 7.6. The IRFM and SRFM modules are located in Room 511 of the 105-N Building on top of the reactor. Wire run numbers were developed and the system raceway identified and each wire walked down and traced from beginning to end.

The raceway begins as conduit from both the IRFM and SRFM on top of the reactor and the circuits are routed to embedded conduit to Corridor 2 and continue to Room 6. Other SRFM circuits are routed in conduit to Room 503 and in tray to Room 6.

Power circuits for the IRFM and SRFM were qualified with the Emergency Power System (Ref. 9). The circuits that run in cable tray are routed with the ball circuits to Room 6 and are qualified in Reference 16.

As a result of the seismic reanalysis and evaluation, the conduit in Rooms 506, 503, 5, and Corridor 2 were determined to be seismically qualified.

The Flux Monitor Raceway was also qualified after replacement of a rod hanger (Ref. 18).

A III/I walkdown to identify and fix hazards due to non-Category I failures was conducted (Ref. 19). The conduit in Room 6 was determined to be seismically qualified upon installation of beam clamp retainers which was done. 21

3.10.3.7 Seismoscope - Original Evaluation

The seismoscopes were designed specifically to detect seismic motion. See Section 3.7.6.1. Relays which are activated by the seismoscope switches "lock in" on momentary closure of the switch and require manual reset to return the system to normal. Seismic motion would not cause the relays to reset.

3.10.3.7.1 Seismic Reanalysis of Seismoscope

The seismoscope instruments are located in Room 178 of Building 105-N. The raceway begins as conduit from the seismoscope in Room 178. The conduit exits Room 178 through the floor and the cable is then routed in tray along the ceiling of Rooms 4, 48 and 5; across Corridor 2; into the electrical tunnel (Room 47) and into Room 6 via tray.

The raceway was walked down and the wire run numbers traced and verified. The result of the analysis required 10 fixes to supports to qualify the raceway. It was determined that the conduit in Room 178 and tray in Rooms 4, 5, and Corridor 2 were seismically qualified as is. Support modification was required for the tray in the electrical tunnel. The 10 fixes identified were installed and the Seismoscope Seismic Category I Raceway is seismically qualified to resist an SSE. 21

A III/I walkdown of the Seismoscope System was conducted to identify potential non-seismic hazards.

The III/I walkdown team identified a total of eighteen problems. Of these, seven required fixes and the remaining eleven were resolved by analysis

and/or application of the Generic Acceptance Criteria (Section 3.10.3). Also in this group, four seismic Category I electrical tray problems were identified (Ref. 8).

3.10.3.8 Engineered Safety Features Actuation System (ESFAS)

An analysis and evaluation of the ESFAS circuits and raceway was performed (Ref. 9). It was concluded as follows:

Primary Coolant Pump Low Speed Raceway

The low speed trip is fail safe and the circuits of interest are in Room 6 and qualified as discussed in Section 3.10.3.5.

ECCS High/Low Coolant Pressure and Low Coolant Flow

The ECCS high/low coolant pressure monitoring tubing system originates from primary coolant headers in the outlet pipe space which is typical for both the left and right sides. The three tubes are routed in and securely fastened to a structural channel that is fastened to the wall and routed to the penetration to Rooms 5 and 37, respectively, and then to the pressure switches in Rooms 3 and 36. The pressure monitoring tubing was seismically qualified by field walkdown which determined that the tubing was adequately attached to the structural channel and the channel was securely fastened to the wall.

The ECCS coolant flow monitoring tubing initiates from four venturi meters in the inlet cooling connector piping on both the left and right side of the inlet pipe space. Penetration from the inlet pipe space is through Rooms 46 and 180, respectively, and then to pressure switches in Rooms 3 and 36. Seismic qualification of this system was performed by Kaiser Engineers for Project H-679 which is included in Appendix A (Ref. 9).

In Rooms 3 and 36 the ECCS electrical circuits initiate at the pressure switches. Tray in Room 3 crosses Corridor 2 and into the electrical tunnel and to Room 6. Electrical circuits are routed from Room 36 in Trays 48, 80 and 22 to Room 6. Room 36 cable tray required 8 support bracket modifications. The electrical tunnel required 10 bracket supports and Tray 22 was seismically qualified with the routing of Tray 13 as discussed in

Sections 3.10.3.1.1 and 3.10.3.3.1. All of the remaining raceway was qualified in Appendix A of Reference 9 and the analysis indicated the supports were acceptable as is.

ECCS High Temperature Thermocouples

Six ECCS high temperature T/Cs (three per side of reactor) are located in the primary coolant return headers in the outlet pipe space at the 40 ft elevation. The circuits are routed in conduit to Tray 142 and then split; 5 circuits in conduit through Rooms 501 to Tray 109, 87, 36 to Room 6 and one circuit from 142 through conduit to Room 507 and Tray 158, 162, conduit and Tray 66 to Room 6. The analysis indicated that six conduit beam clamps in Room 501 required replacement. All of the remaining raceway were seismically qualified in Appendix A of Reference 9.

21

3.10.3.9 Seismic Evaluation of Portable Emergency Lights

In response to ASEP Issue No. 75 (Ref. 5), the portable emergency lights were evaluated for SSE loading. The lights are located in Buildings 105-N, 182-N and 181-N. The location of each portable emergency light is identified and shown in Table 3.10-1. There are additional lights in other parts of the plant, but only the ones noted in Table 3.10-1 were determined to present a potential hazard to equipment required to function after the SSE. In Building 105-N they are located in the control room and Room 6 which is the control room basement. There are four lights in the control room, one in the stairwell and nine in the basement. In Building 182-N there are five portable emergency lights, two located in the stairwells and three in the main floor. This building does not have any portable emergency lights in the basement. Fixed existing emergency lights in the basement are powered from the emergency diesel generator on the main floor. In Building 181-N there are three portable emergency lights, one mounted on the south wall, one on the north wall, and one in the pump house annex.

Five different types of portable emergency lights are installed in the areas evaluated for Category III over Category I hazards listed in Table 3.10-1. More than 50% of these are Teledyne "Big Beam" lights. According to the KEH report, approximately 90% of the portable emergency lights throughout the

plant are Teledyne "Big Beam." There is a maintenance program to replace existing lights with new Teledyne "Big Beam" models. They are very light (20 lb) compared to the other portable emergency lights which weigh at least 40 lb with the 106 lb ELS portable emergency lights being the heaviest.

Several different support bracket configurations were found for supporting portable emergency lights. The two most common throughout the plant are Types I and II. The brackets are mounted to concrete, concrete block, steel columns and equipment inside the building. The support brackets have a 1/2 in. vertical lip to prevent the portable emergency light from sliding, but this provides a pivot point to allow overturning.

A static load method was used to evaluate the portable light fixtures and support brackets. It was done for Teledyne "Big Beam" and Ratelco with support brackets I and II. The analysis was performed using 1.5 times the peak response spectra of the EQE report at 7% damping (for bolted steel structures). The resultant loads are 1.63 g horizontal and .90 g for vertical.

Calculations were made to determine the stability of the light in the support bracket for two different types of portable emergency lights.

Teledyne "Big Beam" was one of the portable emergency lights for which stability was calculated. The calculations show that the light will slide on the bracket if it is not restrained during the SSE. The 1/2 in. vertical lip will prevent the portable emergency light from sliding, but it also provides a pivot to allow the fixture to overturn. Ratelco is the other portable emergency light for which stability calculations were prepared. The Ratelco fixture weighs 43 lb and also was found to slide and overturn.

From the results of the analysis and evaluation (Ref. 31) it was concluded that certain lights in the 105-N Control Room, Room 6, 181-N Building, and the 182-N Building presented a potential Category III over Category I hazard to equipment required to function after the SSE. In particular, lights mounted above the electrical panels in the control room could fall on the

panel and damage equipment. Also, lights mounted over or on battery enclosures could fall and damage the batteries. The four fixes listed below were recommended and have been implemented (Ref. 23).

1. Perform a pull test on the portable emergency light support bracket. The pull test should be 1.63 times the weight of the bracket plus the light.
2. Perform a pull test on the light relay fixture if it is mounted right above the portable emergency light. The pull test should be 1.63 times the weight of the light relay fixture. 21
3. Replace all support brackets that fail the pull test with a Teledyne Model 23 bracket. If the portable light is not a Teledyne "Big Beam" light, it should be replaced by one. Bolt all bracket installations on concrete blocks through the wall and plate on the other side. Implement a four-fastener attachment.
4. Install a seat belt type restraining strap to each of the support brackets to hold the portable emergency light in place.

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TABLE 3.10-1

EXISTING PORTABLE EMERGENCY LIGHTS

LOCATED IN CRITICAL AREAS (1)

Location	Bldg.	Lighting Unit Type
Control Room South Wall	105-N	1311 Ratelco
Control Room South Wall	105-N	1311 Ratelco
Control Room West Wall	105-N	1311 Ratelco
Control Room North Wall	105-N	F Exide
Stairway to Room 6	105-N	Teledyne "Big Beam"
Room 6 South Wall	105-N	Teledyne "Big Beam"
Room 6 South Wall	105-N	Teledyne "Big Beam"
Room 6 South Wall	105-N	ELS 12 SC 450
Room 6 North Wall	105-N	1311 Ratelco
Room 6 East Wall	105-N	Teledyne "Big Beam"
Room 6 West Wall	105-N	Teledyne "Big Beam"
Room 6 SE on Column	105-N	Teledyne "Big Beam"
Room 6 SW on Elect Panel	105-N	Teledyne "Big Beam"
Room 6 NW on Column	105-N	ELS 12 SC 450
Eastside Stairway	182-N	200X Electric Cord Co.
Westside Stairway	182-N	Teledyne "Big Beam"
Chlorine Room NW Corner	182-N	Teledyne "Big Beam"
Switchgear Enclosure on Col.	182-N	200X Electric Cord Co.
On Comm. Panel Control Room	182-N	Teledyne "Big Beam"
North Wall	181-N	Teledyne "Big Beam"
South Wall	181-N	Teledyne "Big Beam"

(1) 105-N Control Room, 105-N Room 6, 181-N Emergency Pumphouse, 181-N Annex, and 182-N

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3.11 Environmental Design of Mechanical and Electrical Equipment

This section provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features and reactor protection systems have been designed and qualification documentation generated to ensure acceptable performance in all environments in which the equipment is, or to which it may be potentially, exposed. Sections 3.11.1 through 3.11.5 discuss the original design and equipment qualification considerations. Section 3.11.6 discusses the present N Reactor Equipment Qualification (EQ) Program Plan developed in 1987. This program is in progress with completion of required Return to Service (RTS) items scheduled for completion prior to restart of N Reactor. 21

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Engineered Safety Features 2

The engineered safety features (ESF) of N Reactor consist of the confinement system, emergency core cooling system, control room habitability system, confinement atmosphere filter system and the emergency drain system. In general, these systems were designed with the components required to operate during normal or accident conditions located outside of the confinement system. This has facilitated maintenance of these components and also improved the reliability of operation. The components located inside the confinement system, such as redundant sensors, filters, spray nozzles, etc., are considered passive in nature except for the V-4 valves. However, the harsh environmental atmosphere during an accident was taken into consideration in the design of the components. See Chapter 6 for more detailed descriptions of the engineered safety systems. 2

The major equipment and components of the engineered safety system are listed in Table 3.11-1 with letters to identify their locations. The normal and accident environmental conditions for these lettered locations are specified in Table 3.11-3.

3.11.1.2 Reactor Protection System

The reactor trip system (RTS) is the protection system that acts to limit the consequences of abnormal conditions by automatic shutdown of N Reactor. The RTS automatically inserts neutron-absorbing horizontal control rods (HCR) into the reactor and, in the unlikely event that HCRs fail to enter the reactor within prescribed time limits, drops neutron-absorbing balls by gravity directly into the reactor core. Either the rod system or the ball system by itself has sufficient capacity to shut the reactor down safely and maintain it subcritical.

The RTS consists of the HCR safety circuit and the ball safety circuit. The electrical circuit of the HCR safety circuit is responsive to primary sensing devices which monitor safety-related reactor operating parameters. The systems which are monitored include:

- Neutron Flux
  - Intermediate range
    - Fast period
    - Excessive flux
  - High range
    - Excessive flux
    - Very high average power
    - Power-rate-of-rise

- Reactor Coolant
  - Pressure tube high or low flow
  - Pressure tube outlet temperature (zone temperature monitor)
    - System very low pressure
    - System extremely high pressure
  - Pressurizer extremely low water level

- Secondary Coolant
  - Main steam header high pressure
  - Surge tank extremely low water level

- Miscellaneous
  - Emergency core cooling system actuation
  - Rod cooling water low flow
  - Electrical power failure
  - Circulating water system low flow
  - Ball safety circuit actuation
  - Rod scram solenoid power loss
  - Ball hopper solenoid power loss

The ball safety circuit is responsive to sensing devices which monitor the following systems:

- Seismoscopes
- ECCS Actuation
- High Flux Level After Scram
- Reactor Supercritical After Rod Scram
- Failed Rods (HCRs not meeting insertion time limit)

With exception of individual pressure tube flow (which trips on a 1/1003 high or low flow), all other sensors are redundant and have trip logic. Abnormal conditions affect several circuits. Therefore, failure of one circuit's sensors to detect an abnormality would not result in failure of the reactor trip system.

The reactor protection system is required to shut down the reactor at onset of the accident and is not required to operate in the course of the accident or afterward. The environmental conditions of the accident would not affect the system. See Chapter 7 for more detailed description of the reactor protection system.

Components and sensors of the reactor protection system are listed in Table 3.11-2 with letters to identify their locations. The normal and accident environmental conditions for these lettered locations are specified in Table 3.11-3.

3.11.2 Qualification Tests and Analyses

N Reactor was designed and placed into operation prior to the existence of the General Design Criteria of 10 CFR 50 and the formalized guides and standards of the Regulatory Guides and IEEE Standards. The designer of each system was responsible for determining the environmental conditions of that system. Preliminary studies were performed to establish general environmental conditions to be expected inside the confinement system (Ref. 1, 2 and 3). Operating experience from other Hanford reactors was utilized to a large extent in determining these conditions.

Qualification tests for environmental conditions were incorporated in the procurement specifications.

Typical examples of environmental tests are listed below:

- 600 Volt Wire and Cable (Ref. 4)
  Heat and Radiation Insulation - Nuclear radiation resistance tests were made by the General Electric Company. Test specimens of all organic materials were submitted for a gamma irradiation tests of 5.5×10^7 roentgen.

  Physical and electrical properties at elevated temperatures of all organic materials were tested by the seller in accordance with the Insulated Power Cable Engineers Association Standard (IPCEAS), S-19-81.

- Resistance Temperature Detectors and Cable (Ref. 5 and 6)
  Radiation exposure of 5.7×10^7 roentgen integrated of 6.0 MeV average gamma (200 R/h).

  Temperature exposure of 750°F for 1000 hours without a reduction of 50% of insulation resistance.

Moisture exposure to ambient relative humidity of 20 to 100% and sprayed water. The seller had the option of performing the tests himself or by a "Certified Testing Laboratory."

- o Reactor Nuclear Instrumentation (Ref. 7)
  Extremes of the temperature and relative humidity in the reactor control room were specified from 65 to 95°F and relative humidity from 30 to 85% with no forced air interchange in the vicinity of the instrument panels. The instrumentation was assembled and tested under these conditions by the seller before acceptance.

- o Confinement Seals
  Reference 8 presents materials, methods of installation, inspection and testing of confinement seals recommended for N Reactor. These seals were developed in the onsite laboratory where the radiation and physical properties were tested.

- o ECCS Dump Valves (Ref. 9)
  These valves are one of the few components of ECCS that are located inside the confinement zone and are required to operate during and after the accident. The environmental conditions specified for these valves were 40-150°F, normal conditions and 230°F, accident conditions. Accumulated gamma radiation dose of 5×10^7 roentgens was specified. The valve was required to function when sprayed with boiling water and induced vibrations from two phase flow. The seller was required to submit test procedures and results to demonstrate qualification of the valve and operator.

- o Horizontal Control Rod Drive and Hydraulic Controls (Ref. 10)
  The HCRs are located in the secondary confinement zone and hence the environmental conditions are not as severe as Zone I. The design environmental conditions were 50 130°F temperatures, 75% maximum relative humidity and gamma radiation level of 8 roentgens per hour. Qualification tests were conducted before acceptance of the drive systems. Reliable actuation was the prime criterion of the test.

2

3.11.3 Qualification Test Results

The majority of qualification tests were incorporated in purchase orders and not formally issued as documents. Prior to acceptance of the equipment and material, inspectors at vendor plant or onsite established that the ordered material met the purchase specification. Where special qualifications were required, such as radiation resistance, samples were subjected to onsite testing and evaluation before authorization of the purchase orders.

Since startup of N Reactor the surveillance and EMS testing programs have monitored the environmental condition of the material and equipment. A large scale replacement program of the stainless steel tubing of the pressure tube flow monitoring system was initiated in 1970 due to stress corrosion failure. The stress corrosion was induced by the high humidity condition inside Zone I and the presence of small amounts of chloride in the piping insulation materials. Sections of the right side flow monitoring tubing were replaced with Inconel 600 material. Leaks were repaired in the reactor coolant system, and the high humidity conditions have been reduced inside Zone I. Flow monitor sensing line leakage is infrequent at the present time.

An investigation program was initiated in 1977 to determine the condition of the insulation material of the electrical wire and cable inside Zones I and II (Ref. 11). No deterioration or degradation was found. Cable samples were submitted for physical testing by Sandia Laboratories under a program sponsored by DOE.

3.11.4 Loss of Ventilation

The reactor control room and cable spreading room air conditioning system are operated by ac electrical power. Loss of the power would also result in interruption of ac power to control room instrumentation thereby significantly reducing the heat load to these areas. For accidents activating the ECCS, only instrumentation required for ECCS diesel engine control would be required to be operable, and this equipment would not be impacted by significant increases in ambient temperatures.

Other areas containing operating equipment during the design basis accident included the 181-N, 182-N and 105-N annex buildings which house operating diesel engines. The diesel engines have their own integral cooling system; intake air is ducted from outside the building and the exhaust vents outside the building. Loss of ventilation therefore would not adversely affect their operation. 2

3.11.5 Estimated Chemical and Radiation Environment

The estimated chemical and radiation environment of the various locations where ESF and RTS equipment is located are listed in Table 3.11-3. These estimates are based on the postulated accidents of Section 15.

As stated earlier, the design philosophy of N Reactor was to locate all possible ESF and control equipment outside the confinement zone. Components inside the zone were designed for the extreme environmental conditions of the accident.

3.11.6 N Reactor Equipment Qualification Program Plan

As part of the N Reactor RTS program a comprehensive EQ Program Plan was developed.

The N Reactor EQ Program objectives are to: 21

- Generate and maintain evidence by identifying, planning, and completing tests and/or analyses that demonstrate the equipment can meet or exceed safety performance requirements, on demand, while being subjected to normal, anticipated operational transient, DBA, and SSE conditions.

- Identify surveillance and maintenance requirements to ensure that the qualification is maintained after the initial qualification is complete.

These objectives apply to all equipment currently installed in N Reactor and all equipment to be installed in the future associated with the following safety functions:

- o Emergency core cooling (including emergency diesels)
- o Reactivity control
  - Horizontal Control Rod System
  - Ball Safety System
- o Confinement, which includes isolation, pressure reduction, and fission product release
- o Graphite and shield cooling
- o Post Accident Monitoring.

The equipment that is qualified by the current scope of the EQ Program includes equipment required to achieve the safety functions described above.

The equipment that will be environmentally qualified is the equipment which is needed to mitigate the effects of DBAs which create harsh environments. The accidents that create harsh environments are:

1. Main steam line or header break (Chapter 15.1.2)
2. Small reactor coolant system line or branch line break (Chapter 15.6.2)
3. Single pressure tube and connector events (Chapter 15.6.4)
4. Loss of coolant accidents (Chapter 15.6.5)
5. Hypothetical accident (ECCS does not actuate) (Chapter 15.6.6).

The equipment to be seismically qualified will be equipment which is needed to safely shut down and maintain a safe shutdown operating mode after an SSE occurs.

Appendix A of the Plan contains the requirements and methods that will be used by the EQ Program. These requirements and methods are modeled after those used in the United States commercial nuclear power industry. These requirements are based on the "Final Rule" expressed by the Nuclear Regulatory Commission (NRC) in 10 CFR 50.49. The final rule is the result of an

evolutionary process over a considerable period of time. Selection of the "Final Rule" as the primary technical basis of the qualification requirements and methods is based on the following considerations:

1. The final rule is applicable to "old plants" as well as "new plants"
2. The requirements are based on the latest available guideline documents
3. It represents the "base standard" used in the commercial nuclear power industry against which equipment capability can be measured
4. The requirements in the final rule reflect the ability of current technology to address such topic as synergistic effects and ionizing radiation dose rate effects.

Appendix B documents the environmental and seismic conditions for which N reactor equipment will be qualified. Equipment will be environmentally qualified for normal, abnormal, testing, accident, and post accident conditions. The environmental parameters for which the equipment will be qualified are temperature, pressure, humidity, chemical effects, radiation, and submergence. Normal plant vibration and seismic vibration motion will be used to qualify the equipment included in this program for dynamic loadings. 21

Appendix C is a list of the types of equipment to be considered by the EQ Program. This list is included in the N Reactor automated Equipment History List and contains unique combinations by equipment type, manufacturer, and model. The distribution of equipment types is also described in Appendix C. This list will be modified as reactor system design changes are performed on the reactor.

Currently, a significant quantity of the equipment installed in N Reactor that will be qualified does not have a unique tag to identify the equipment. The N Reactor EQ Program will attach tags to all equipment that will be qualified that currently does not have a unique tag number.

The EQ Program will also develop a study to identify locations of potential high energy line breaks that will create harsh environments in the vicinity

of equipment to be qualified. The study scope will (1) develop the final scope, (2) identify potential break locations near equipment to be qualified, (3) characterize the resulting cone and environment of the pipe break jet, and (4) identify equipment that will be qualified for the conditions developed by the jet that is encompassed by the resulting cone. The results of this study will be used to assess the modification needs of the EQ Program service conditions. The EQ plans will be revised accordingly. This task will allow the revision of this plan based on new data developed by the N Reactor Probabilistic Risk Assessment (PRA) and new accident analyses that identify components important to risk not included in Appendix C, different environmental conditions, or both.

The open issues identified during the development and review of the N Reactor EQ List will be added to the EQ Program for resolution and a report will be written to document the resolution plan.

A current status report is being compiled which will document work completed and also identify those items which must be completed prior to a restart of N Reactor.

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4.3.2.3.3 Inlet Water Temperature Coefficient

The inlet water temperature coefficient relates a change in neutron multiplication to a change in the inlet water temperature. As the inlet water (reactor coolant) temperature increases, the water density decreases and the fuel temperature increases. Thus, the inlet water temperature coefficient incorporates both the Doppler effect and the water temperature effect in one coefficient which is much easier to use as the inlet water temperature is continuously measured. Both of these effects are negative for all operating and accident conditions; hence, as the inlet water temperature increases, the reactivity status of the reactor decreases.

Figure 4.3-26 shows the inlet water temperature coefficient as a function of average fuel exposure. The total reactivity effect for a given inlet water temperature is obtained by multiplying the coefficient by the difference between the inlet water temperature and 68 F (the base temperature).

A void coefficient is not explicitly defined for N Reactor, the effect is included in the inlet water coefficient and the power coefficient. Nevertheless, N Reactor was deliberately designed to have a negative void coefficient. This was achieved by the selection of a tight lattice pitch (8 in. by 9 in.) and an effective graphite density of 1.3. The fuel to moderator atom ratio is such that N Reactor is slightly under moderated and becomes more so as the coolant is voided. A complete loss of coolant to the core is almost equivalent to the negative reactivity insertion realized by insertion of all the control rods. 21

The void coefficient relates a change in neutron multiplication to a change in average coolant density. This is an extension of the water temperature coefficient and would take effect should the coolant reach the saturation temperature at which point the average coolant temperature would remain constant but the average density would continue to decrease due to the formation of voids. The void coefficient would continue to be in effect and reach its maximum value when coolant is completely vaporized. As the coolant density decreases the thermal utilization increases and neutron energy spectrum hardens. The negative contribution from the hardening of the neutron spectrum predominates over the positive contribution from the increased thermal

utilization (i.e., the void coefficient is negative) over the range of 0% to 100% void in the coolant. An increase in the coolant void fraction, therefore, results in a proportional reactivity decrease for all operational and accident conditions. Figure 4.3-52 shows the void coefficient for Mark IV fuel in N Reactor as a function of coolant density.

4.3.2.3.4 Power Coefficient

The power coefficient relates a change in neutron multiplication to a change in the reactor power level. As the reactor power level increases the fuel temperature, the outlet water temperature and the graphite temperature increase The first two temperature increases are incorporated into the power coefficient while the latter is considered independently as the graphite temperature coefficient. The power coefficient includes the water temperature increase fro the inlet to the outlet and corresponding decrease in water density. Thus, the power coefficient incorporates both the Doppler effect and the water temperature effect into one coefficient. Again both effects are negative for all operating and accident conditions, and reactivity decreases as the power level increases.

Figure 4.3-27 shows the power coefficient as a function of average fuel exposure. To obtain the total reactivity effect, the power coefficient is multiplied by the reactor power level, expressed in megawatts.

4.3.2.3.5 Graphite Temperature Coefficient

Approximately 87 percent of the N Reactor core is graphite moderator. In a thermal reactor such as N Reactor, the neutrons tend to reach thermal equilibrium at the temperature of the moderating material. As the graphite temperature increases, the average energy of the thermal neutron increase and an increase in reactivity is observed.

4

This change in reactivity is due to variations in the neutron spectrum and in the moderation-to-absorption ratio of the reactor coolant. As the graphite temperature increases, the fission rate in the inner fuel element increases while the fission rate in the outer fuel element decreases. The overall effect is a slight increase in the total fission rate of the fuel as can be seen in Figure 4.3-28. Figure 4.3-29 shows the total reactivity effect as a function of average graphite temperature. The curve was developed for an average fuel burnup corresponding to 9 percent Pu-240 production mode. The curve has been upgraded periodically with actual reactor data.

Because of the massive nature of the moderator, the graphite temperature coefficient acts slowly. That is, for a step increase in reactor power, approximately 30 minutes are required for the graphite temperature increase to attain 63 percent of the resulting temperature increase.

4.3.2.4 Control Requirements

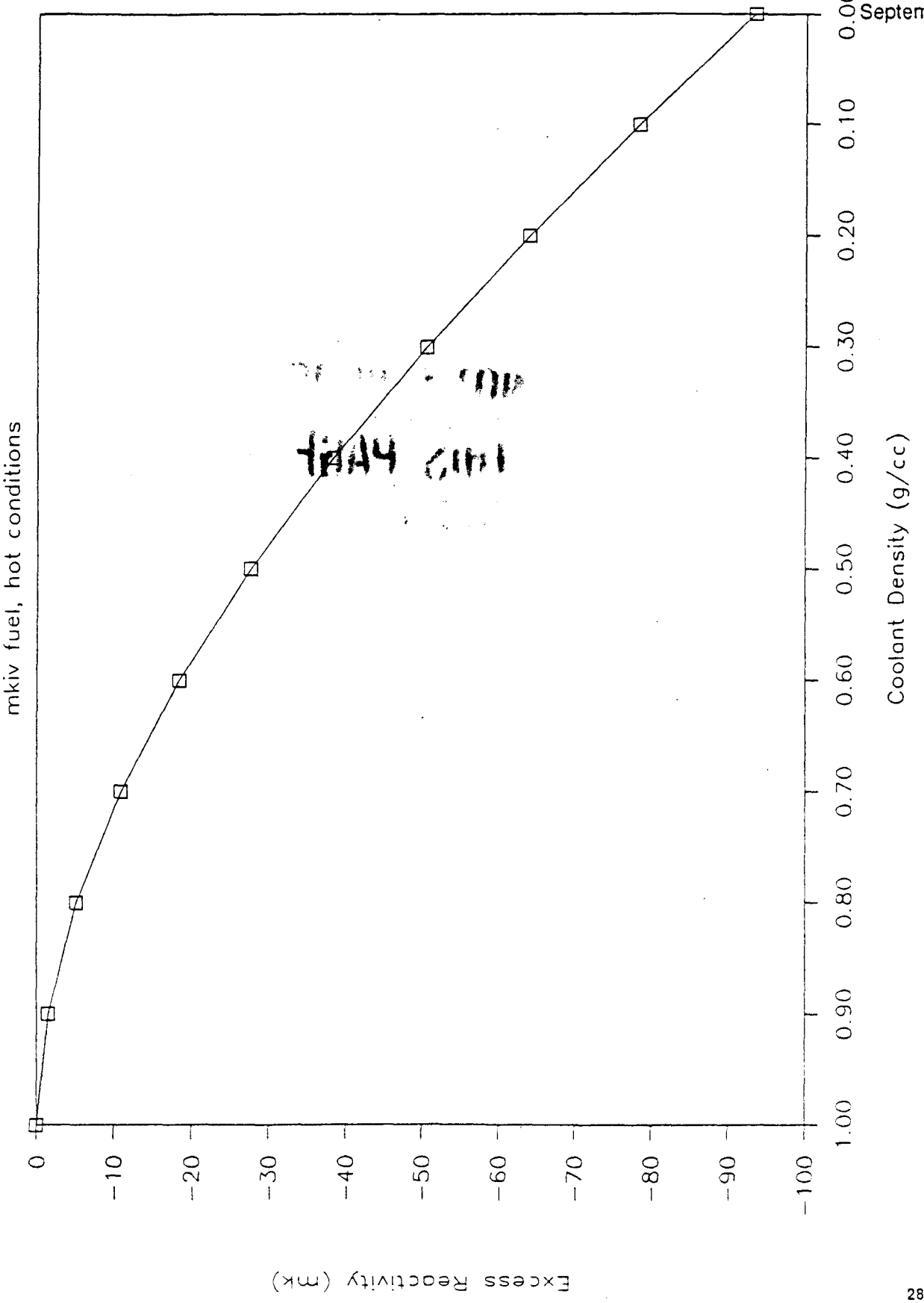
The excess reactivity and the distribution of the excess reactivity among the different reactivity contributions for various core conditions are summarized in Table 4.3-7 and Table 4.3-8.

Reactivity Shutdown Analysis

The ability to shut down the reactor under hot operating conditions is shown in Table 4.3-8 for various core conditions using both the HCR and ball safety systems. Furthermore, Table 4.3-8 displays the ability to shut down the reactor during reactor startup at the point of minimum control available from the HCR system. Figure 4.3-30 shows how the power level, inserted HCR worth and HCR

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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Excess Reactivity versus Coolant Density (mkiv Fuel, Hot Conditions) Figure 4.3-52

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5.2 Integrity of Reactor Coolant Boundary

The reactor coolant and associated special quality pressure systems components were procured, fabricated, and constructed in accordance with Hanford Works Specification HWS-6897, "Furnishing, Delivery and Erection of Reactor Piping in 105-N and 109-N Buildings," dated April 1, 1960, and Amendments 1 (July 1960) and 2 (August 1961). See Section 5.2.3.1 for a discussion of Hanford Works Specifications. Basic to these specifications were the following sections of the ASME Boiler and Pressure Vessel Code and the Power Piping Code:

Section I - Power Boilers, 1956 and 1959
Section VIII - Unfired Pressure Vessels, 1956 and 1959
ASA B31.1.0 - American Standards Association Power Piping Code, 1955

In 1959, as today, the ASME Boiler & Pressure Vessel Codes (B&PVC) and ASA B31.1 (currently identified as ANSI B31.1) represented minimum industry standards for design, fabrication, and erection of pressure systems. The ASME Code Sections in existence at that time were:

Section	Title
I	Power Boilers
II	Material Specifications
III	Boilers of Locomotives
IV	Low Pressure Heating Boilers
V	Miniature Boilers
VII	Suggested Rules for Care of Power Boilers
VIII	Unfired Pressure Vessels
IX	Welding Qualifications

The need and desire for more stringent design criteria--including seismic analysis provisions, tighter material and welding specifications, and a high degree of assurance that all specifications were met to achieve the quality desired for a nuclear pressure system--was recognized by the N Reactor designers.

Accordingly, Hanford Works Specifications were developed that were far more stringent than the minimum code guidance in effect and, for that time, is believ to have represented the most detailed and highest quality requirement specifications in existence for a steam generating plant.

In 1969, in light of advancing materials technology, HWS-8732, "Seam-less Carbo Steel Pipe for Critical N Reactor Piping Systems," was issued to provide for procurement of improved materials for the maintenance and repair of reactor pressure systems. Adoption of these specifications permitted the use of contemporary engineering materials possessing improved properties. Quality requirements imposed by this specification also exceeded the minimum industry code required standards. Along with HWS-8732, repairs or modifications remaine subject to the provisions of HWS-6897.

See Chapter 10 for discussion of secondary coolant (steam) system.

5.2.1 Compliance with Code and Code Cases

N Reactor was built in accordance with AEC Manual Chapters. It has been operat and maintained in accordance with AEC, ERDA, and DOE manual chapters, directive and orders. DOE Order 5480.4 currently prescribes the ASME Boiler and Pressure Vessel (ASME B&PV) Codes as the governing standard. N Reactor was designed and fabricated according to provisions exceeding Code provisions then in effect tha did not specifically address nuclear components.

Complete adherence to the construction Codes could not be accomplished. Severa methods were used to resolve design enigmas: (1) Code Cases, (2) discussion, justification, and approval of code deviations described in formal design crite approved by the AEC, and (3) design and hydrostatic testing to qualify fittings to UG-35, Section VIII, 1959 edition, and ASA B31.1, 1955 edition. Following initial startup, certain deviations from Code were made when authorized by the General Manager or the Assistant General Managers for Technical and Operations (this varied according to the time frame). Variations from the codes are list in Appendices 5.2A and 5.2B. Code Cases are briefly described in Appendix 5.2

As noted above, HWS-6897 has more stringent requirements than the minimum ASME B&PV Code provisions that were available to obtain the design sufficiency and quality desired for N Reactor nuclear pressure systems. Section III of the current ASME B&PV Code is comparable to HWS-6897. The material, procurement, fabrication, and erection of N Reactor pressure systems under HWS-6897 compares very closely with present-day Section III provisions. 14

In comparing Section III with HWS-6897 requirements, some differences were noted. These differences, and the need for consistency, were the principle reasons for using the original specifications for repairs and replacements where practicable. This practice is sanctioned by Section XI, "Rules for Inservice Inspection of Nuclear Plants Components," of the ASME B&PV Code. However, availability, lead-time, and cost considerations have resulted in an increasing use of Section III requirements for procurement of repair and replacement items.

Repair and replacement activities comply with Section XI requirements as implemented by the Pressure Systems Instruction, SD-NR-OMM-003 REV0 (Ref. 18). Reference 17 established the policy governing the use of the ASME B&PV Code applications to N Reactor pressure systems, identified the selected versions of the ASME code that would be applied, and described a program of progressive implementation of Section XI of the code. 21

N-Reactor has a number of unique design features that make applying Section XI a very challenging task. Some of the unique features of the N-Reactor are:

- Greater use of carbon steel piping in safety-related systems than at commercial nuclear power plants; including the first use of carbon steel as the primary material in the class 1 reactor coolant system 14

- Much greater quantities of piping and components in safety-related systems than at commercial nuclear power plants and, therefore, much greater quantities of welds

- Difficult accessibility in many locations.

The original Inservice Inspection Program was expanded in the mid 1970's to partially fulfill the requirements of Section XI as applicable to N Reactor. The current ISI Program is documented in WHC-CM-5-15, Rev. 0 (Ref. 4) and is based on the requirements set forth in the ASME B&PV Code Section XI, Division 1, 1983 Edition, Winter 1983 Addenda. Section XI is based, to a large degree, on present-day commercial power reactors and the associated materials of construction. Section XI needed some interpretation in rectifying stated requirements with N Reactor design and the associated inspectability limitations.

Currently, Reference 17, "Policy Governing the Use of ASME Boiler and Pressure Vessel Code," is applicable. Code deviations are approved by DOE in accordance with instructions in "Pressure Systems Instruction" (Ref. 18).

5.2.2 Overpressurization Protection

The overpressurization protection for the systems described herein were designe in accordance with the ASME B&PV Code Sections I and VIII (1959) and ASA B31.1, 1955. Code Case 1271N was included as part of Sections I and VIII. The reacto coolant system consists of 1003 pressure tubes, six recirculating pumps, and twelve steam generators in parallel, a pressurizer, and interconnecting piping. The design conditions were nominally 600°F and 1825 psig (see Table 5.2-1). However, 900 lb ASA valves were used, and the system parameters are actually defined as the pressure-temperature points in the area on a pressure-temperatur diagram bounded by (1) the zero value axes, (2) the 1825 psig pressure curve, (3) the pressure temperature curve of ASA 900 lb Class Carbon Steel Pipe Flange and Flanged Fittings (ASA B16.5), and (4) the pressure-temperature saturation curve for water (Figure 5.2-1). The system relief valves and pertinent instrumentation are shown in Figures 5.2-2 and 5.2-3.

5.2.2.1 Design Bases

The N Reactor pressure relief system was designed to meet the intent of Sectio I & VIII of the ASME B&PV Code and Code Case 1271N and the ASA B31.1, "Americai Standard Code for Pressure Piping," The Code Committee and the designers (Ref. 2) of N Reactor realized that nuclear facilities, and particularly N Reactor, were unique among fired and unfired pressure vessel

facilities. Design requirements provided an integrated system that would provide for safe operation and continuity of cooling with adequate protection from over-pressurization. The design bases included the following considerations.

1. The reactor was considered a pressurized-water nuclear system from the standpoint of pressure relief, and Case 1271N was considered applicable.

   Therefore, the pressure relief system was designed to meet the overpressure restrictions of Sections I and VIII as modified by Case 1271N. The pressure relief system was required to limit any excessive pressure to 10% greater than the design pressure. Also, a minimum of two relief valves were required.

2. A fundamental part of the design philosophy was the use of a reactor control system of power setback and scram to reduce system pressure. This is based on: (a) the magnitude of the maximum power that might be generated during a reactor excursion and hence the size of the relief system, (b) the necessity of maintaining a coolant supply during all emergencies (loss of coolant through a relief system could abrogate the ability of pumps to recirculate coolant and would require the use of the ECCS), and (c) the avoidance, if possible, of the releasing, either controlled or uncontrolled, of large amounts of radioactive material to a second system or the environs. This philosophy was covered in the 1965 ASME B&PV Code, Section III, Article 9, N-910-2, subsequent to the design and startup of N Reactor. 2

   "N-910-2. It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients, besides those pressure relieving devices installed to limit overpressures. The installation of pressure relieving devices in accordance with the rules of Article 9 may not in itself provide complete protection against either excess pressure or temperature transients to which the vessel may be subjected. Rules governing the application of

protective systems which may complement the pressure relieving devices, however, form an essential part of the overall safety system and the requirements of this article apply to the devices and principles employed to secure the required protection.

"The provisions to meet the requirements of N-910.1 and all applicabl rules of this article are to be the subject of a summary technical report prepared by or on behalf of the owner setting out the degree of overpressure protection provided for each vessel or group of vessels by the pressure relieving devices, their schematic arrangemen in the system, the capacity for relief and dissipation related to the thermal and overall system characteristics based on both the assumptions made and the analysis of that transient condition which dictated the maximum pressure relieving requirements, all having regard to the pressure and temperature limitations set out in design specifications (N-141).

"A copy of this summary report shall, if requested, be made available to the enforcement authority having jurisdiction at the point of installation."

Refer to Table 5.2-2 for a list of reactor trips that protect against overpressure.

3. High pressures and high temperatures caused by failure of the heat dissipation system would cause actuation of the ECCS. The PCS system woul relieve the high pressure and reduce system pressure to the point that the ECCS could start its function.

4. The safety valves would be provided to handle transient high pressure situations resulting from sudden valve closures, transient power surges, and overinjection. The relief system would be provided to eliminate the need for the actuation of the dump system and once-through cooling for most transient excessive pressure situations. The safety valves would be

sized to relieve energy input to the reactor coolant at shutdown conditions and would reseat at a pressure sufficiently high to prevent bulk boiling in the reactor coolant system.

5. A pressurizer surge tank system would be provided that, in combination with a bleed-feed control, would handle all normal volumetric changes of the reactor coolant system without excessive pressure surges. Because the pressurizer would have a heat source and be isolatable, it would be provided with a pressure relieving device.

6. Pressure relieving devices would also be provided on any section of the reactor coolant system that could be isolated and subject to possible heat addition or pressure surge.

The pressure relief devices located on the reactor coolant system are the RV-2 safety valves (2 each), pressurizer safety valve, RTV-202, (1 each); cell safety valves, PCRV-210 and 6210, (5 each and 1 each, respectively); and the spill cooler and recuperative heat exchanger safety valves BDV 209-IR, 222-1R, and 222-2R (3 each). Pressure relief devices are described in Tables 5.2-3 through 5.2-6. 2

Events or transients that the system was designed to accommodate are as follows:

1. Control rod withdrawal/ejection
2. Makeup injection rate too high
3. Shut off or reduce reactor coolant bleed
4. Loss of heat removal capability such as the loss of electrical generator driving turbines, dump condensers, or the recirculation pumping capacity.

5.2.2.2 Control Rod Withdrawal/Ejection

Control rod ejection is a postulated occurrence that is not considered possible. Tests (Ref. 3) of the reactor moderator and pressure tube design configuration have been conducted and (pressures) forces are not developed sufficiently to eject even a single rod in close proximity to a pressure tube

break. Single or multiple rod withdrawal would result in several, almost immediate (1 s), trips. Such withdrawals would be followed by a high power rate-of-rise, excessive neutron flux, very high average flux level, and high zone temperature monitor indications. The resultant scram from the listed trips (see Table 5.2-2) would occur before significant pressurization.

5.2.2.3 Injection Rate Too High

Up to 7,200 gpm could be injected into the reactor coolant system from the high pressure injection system. Bleed system leaks and sampling system flows would compensate for part of the input. The lower setpoint RV-2 valve (1,700 psig) would adequately relieve any excess input.

5.2.2.4 Shutoff or Reduced Reactor Coolant Bleed

Reduced injection controlled by the pressurizer controls should compensate for any reduced bleed; however, as in 5.2.2.3 above, the RV-2 valve would adequatel relieve the excess system pressure.

5.2.2.5 Loss of Heat Removal System

The loss of heat removal could occur in three major ways:

1. Electric load rejection (dual purpose mode)
2. Loss of recirculation pumps
3. Loss of dump condenser cooling (in plutonium-only production mode).

The energy input to the system from the nuclear reaction is approximately 4,000 MWt. This would exceed the capacity of the RV-2 valves, which have a nominal capacity of 15% of 4,000 MWt or 600 MWt. A reactor scram would be incurred from the various reactor protection systems:

1. Load rejection
   - o High steam pressure
   - o High reactor coolant system pressure
   - o High pressure tube outlet temperature (ZTM)
   - o Extremely low surge tank water level.

2. Loss of recirculation pumps
   - o Flow monitor.

3. Loss of dump condenser
   - o Condenser circulating water, low flow
   - o High reactor coolant pressure
   - o High main steam header steam pressure
   - o High tube outlet temperature (ZTM).

2

Table 5.2-2 summarizes the reactor trips that protect against overpressurization. Section 7.2 presents a design evaluation of the various reactor trip systems.

5.2.2.6 Mounting of Pressure Relief Devices

The RV-2 valves were installed in accordance with ASA B31.1. They have undergone a hot trip test and were actuated once by a pilot system failure during operation. No damage to the system occurred. The pilot system failure is described in Section 15.6.1. Repairs and modifications are performed in accordance with the Pressure Systems Instruction Manual (Ref. 18)

21

The other safety/relief valve mountings are not pertinent to this section. Their discharge would only occur during isolation of the equipment they are protecting from the reactor coolant system.

2

5.2.2.7 Applicable Codes and Classification

The applicable codes are ASME B&PV Code, Sections I and VIII, Code Case 1271N and approved Design Criteria HW-64508, "Project CAI-816, 100N Design Criteria Code Compliance - NPR Piping Systems and Pressure Vessels" (Ref. 2).

2

5.2.2.8 Material Specifications

Materials used in the safety/relief valves were selected to meet the expected environmental and service conditions. The materials are described in Tables 5.2-3 through 5.2-6. Materials for other portions of the reactor are described in Section 5.2.3.

2

5.2.2.9 Testing and Inspection

Refer to Table 5.2-7 for the initial tests of the RV-2 valves.

The RV-2 safety valves and other safety/relief valves are tested annually. The pressure relief valves are tested in accordance with the N Reactor ISI and Testing Plan (Ref. 4) and latest revision of the Equipment Maintenance Standard for the RCS (Ref. 15). The detailed description, testing, and inspection of the various instrumentation systems is covered in Chapter 7.

5.2.3 Reactor Coolant Pressure Boundary Materials

This section discusses the material specifications, coolant compatibility, and in-service inspection requirements for N Reactor RCS carbon and stainless steel pressure boundary materials except for the zircaloy pressure tubes that are covered in Section 5.3 and the Inconel-600 steam generator tubes discussed in Section 5.4.2.

5.2.3.1 Material Specifications

The nuclear safety significance of RCS pressure boundary component integrity wa a paramount consideration during the design of the system. ASTM material specifications were selected to provide the basic requirements for RCS components in recognition of their demonstrated adequacy for conventional power plant components with temperature-pressure ratings more severe than those of the N Reactor design. The ASTM specifications were then supplemented with substantial additional testing requirements as deemed necessary to insure adequate material quality and RCS system operational safety margins.

Hanford Works Specifications (HWS) were written for each major component or class of components when grouping by general service and design conditions, or physical configuration, was possible. The HWSs served as the overall procurement specifications for all major plant components as well as RCS components. Included in each HWS were the basic applicable material specifications, welding and testing requirements, and cleaning and packing requirements.

Further delineation of special fabrication, testing, and quality requirements were provided in the N Standards sections referenced by the HWS specifications. The N Standards sections were prepared by the architect-engineer, Burns and Roe, Inc., with assistance from the General Electric Company. The overall procurement specifications for RCS components were also developed in conjunction with and reviewed by consultants with expertise in the areas of pressure piping design and fabrication. The consultants and their associated areas of expertise are listed below:

Dr. E. O. Bergman, C. F. Braun, President of ASME, Chairman of Committee Initiating ASME Section III Code for Nuclear Pressure Vessels; ASME Code Authority

H. S. Blumberg, M. W. Kellog, Chief Metallurgist, Sabin Crocker, Detroit Edison, Ebasco Piping Authority

E. F. Potter, General Electric; Welding Authority

J. G. Schneeman, X-Ray Products Corp; Non-Destructive Testing Authority

The final material specifications for RCS components provided for (1) appropriate and comprehensive requirements beyond minimum piping practices that assure the nuclear safety of the plant, (2) clear, concise statements of all requirements, and (3) requirements commercially feasible to obtain.

The procurement requirements for RCS components also specified detailed requirements for the development and maintenance of records files for each component. The fabrication of all major components was monitored by buyer representatives, and interpretations of radiographs were made by buyer inspectors.

Table 5.2-8 lists the material specifications utilized in RCS pressure boundary components. The applicable HWSs that are the source of respective supplementary material requirements are also referenced. For examples, Tables 5.2-9, 5.2-10, and 5.2-11 list the supplementary requirements applicable to the carbon steel piping, cast steel valves, and wrought carbon steel fittings, respectively. These tables contain a compilation of the supplementary requirements obtained from the respective HWS and the N Standards sections referenced therein. Table 5.2-12 lists the various N Standards sections developed for use in establishing supplementary quality requirements of N Reactor RCS components and other systems as applicable. These same specifications apply to the N Reactor high pressure injection system as covered in this section.

Section XI of the ASME B&PV Code is followed with respect to repairs and replacements. The original materials specifications must be met for replacement components unless reconciled in accordance with the Pressure Systems Instruction (Ref. 18). Additional Hanford Standards have been prepared to permit the use of contemporary engineering materials such as SA-333 seamless tubing, SA-350 LF2 forgings, and SA-352 LCB cast steel (Ref. 5).

5.2.3.2 Compatibility With Reactor Coolant

The reactor coolant is demineralized water with ammonia added for pH control. Radiolysis of the coolant not only produces hydrogen and oxygen from the breakdown of the water but also hydrogen and nitrogen from the radiolysis of the ammonia. A small, but measurable, concentration of hydrazine results from the radiolytic breakdown of ammonia. Normal equilibrium concentrations of hydrogen in the coolant are about 30 to 50 cc/liter, and oxygen concentrations are generally maintained at less than 5 ppb. Hydrazine can be added during shutdown for oxygen control; this is seldom needed unless the reactor coolant system is opened for maintenance. Reactor coolant chemistry limits are listed in Table 5.2-13.

As shown in Table 5.2-8, the principal materials making up the reactor coolant pressure boundary are carbon steel, Inconel 600, Zircaloy-2, and 304 stainless steel. Corrosion rates of the carbon steel in the high pH, low oxygen reactor coolant are less than 0.001 in/yr (Ref. 6), and corrosion rates of the other materials are considerably less. Chloride and fluoride concentrations are restricted to less than 150 ppb to protect the 304 stainless steel and Zircaloy-2 components. With the exception of the reactor coolant pumps, insulated pressure boundary components are all ferritic carbon steel which is not subject to stress cracking as the result of inadvertent wetting and the associated potential for leaching halide solutions onto the component surfaces.

One instance of stress corrosion cracking of a carbon steel RCS pressure boundary component has been observed. A maintenance effort in 1975 involved a weldment at the low pressure side of the V5-2 pressure boundary valve, a 16" flush gate valve on the outlet header manifold. Piping insulation was inadvertently not replaced following completion of the work. The weldment did not receive a post-weld heat treatment (PWHT). Initiation of cracking was attributed to thermal fatigue of the surface from identified leakage impinging on the valve surface. Crack propagation was attributed to the presence of a nitrate ion, subsequently verified as present in the leakage, which can promote stress corrosion cracking of carbon steel. The nitrate ion formed from radiolysis of ammonia in the high gamma fields which exist in the outlet connector spaces immediately above this valve. The V5-2 has been replaced. Records were searched for other areas where post-weld heat treatment may have been inadvertently omitted. Some seal weld locations were identified and, along with other specific RCS locations known to have been uninsulated at the time, were subjected to extensive NDT which failed to reveal any indications. Conditions unique to the V5-2 valve (heavy weldment with no PWHT) were determined to have resulted in this isolated case of stress corrosion cracking.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness criteria for N Reactor's RCS components were based on the most advanced specifications available during the design of N Reactor in 1959 and 1960. The criteria were applied to all ferritic materials for which pressure-boundary materials were fabricated including the bolting used as Grayloc coupling fasteners on inlet and outlet connector tubing.

All ferritic materials used in the RCS pressure-retaining components were required to exhibit a minimum of 15 ft-lb energy absorption at 10°F in Charpy V-notch tests using Type A specimens as defined by ASTM A-370. The 15 ft-lb energy temperature was then generally accepted as representing the nil ductility transition (NDT) temperature. Fracture Analysis Diagram (FAD) procedures developed by the Naval Research Laboratory (Ref. 7) were then utilized to assure that operation of the system above nominal stress levels is undertaken only above the Fracture Transition Elastic (FTE) temperature. The FTE temperature for the thin-section steel encountered in the N Reactor RCS was defined as NDT +60°F.

Impact specimens from piping were oriented with the longitudinal specimen axis parallel to the direction of maximum working and the base of the notch perpendicular to the surface. Both the welded metal and heat-affected zone of all repair or fabrication welds in ferritic components were also qualified to the same requirements. Weld metal samples were taken with the long axis transverse to the weld, one specimen surface not greater than 1/16 in. below the component surface, and the notch parallel with the surface of the weld. All Charpy impact testing was conducted in accordance with the recommendations of ASTM E23-56T, including equipment calibration procedures. Acceptability was based on the average of three tests meeting or exceeding the minimum requirement. No individual result below 10 ft-lb was permitted.

All RCS pressure boundary components fabricated from ferritic materials met the Charpy impact requirements at a temperature of 10°F with the exception of the A 212 Gr B plate comprising one-half of the tube sheet of steam generator 4B. The subject component required a test temperature of 24°F before exhibiting 15 ft-lb energy absorption. The pressure-temperature limits originally restricted RCS pressures to a maximum of 365 psig (σ_h <4,000 psi) for coolant temperature below 70°F if the cell 4 steam generators were isolated from the system. With the cell 4 steam generators in communication with the RCS, pressures greater than 365 psig were prohibited for coolant temperature below 84°F.

In mid-1976, a pressure boundary valve was replaced (see Section 5.2.3.2). The new valve was purchased to the original specifications. However, a temperature of between 10°F and 24°F was required to meet the Charpy energy absorption requirement. As a result, pressure-temperature limits for the entire RCS were revised to restrict pressures to less than 365 psig whenever the temperature is below 84°F. For Primary Coolant System pressure above 365 psig, the pressure-temperature limits are specified in Technical Specifications 3/4.1.3 in which additional PCS pressure-temperature limits for the pressurizer are also specified. 14

Samples of N Reactor seamless and seam welded reactor loop piping have undergone Dynamic Tear testing. The NDT temperature for these samples was confirmed to be between 0 and 10°F.

5.2.3.3.2 Control of Welding

1. Cold Cracking for Low Alloy Steel

   a. The RCS pressurizer internal vessel surface is clad with type 304 stainless steel sheet. The tube sheet areas of the ten original steam generators, the spill cooler, and the recuperative heat exchanger were buttered with 1/4 in. (after machining) of type 304 stainless steel. The steam generator tube sheet buttering was provided with buttered Inconel-600 transition strips between the type 304 material and the unclad areas of the carbon steel manifold. All of the 14

original ten steam generators have been refitted with Inconel-600 tubes The tube sheet areas on these original ten steam generators have had th stainless steel replaced with Inconel-600 cladding. The eleventh and twelfth steam generators, added subsequent to reactor startup, were provided with a buttered Inconel 600 cladding.

The welding process used cannot be ascertained in the respective certified vendor records. It is believed that the single electrode submerged arc process was used in the cladding processes. The ASTM A 212 Gr B FB plate used was produced to fine grain size. The claddinç processes were qualified using the same process, materials, and materiz thicknesses employed in the actual design. The main feature of the qualification tests which examined the potential for underclad crackinç in the base metal was the requirement for two-side bend tests replacinç the end bend tests required by ASTM A 264. The side bend test affordec ample opportunity to examine for underclad cracking in the fused-HAZ regions of the base material.

The clad was stress relieved with the respective vessel at 1100°F to 1150°F for 4 1/2 to 6 h, depending on the vessel in question. A nitro atmosphere was provided during this operation and maintained until temperatures were below 400°F.

During retubing of the original ten steam generators, the original Typ 304 stainless steel clad was removed prior to depositing the Inconel clad. Dye penetrant inspection of the base metal did not reveal any evidence of underclad cracking from the original cladding process. These manifolds were reclad with Inconel under the same requirements a the original process.

b. RCS Carbon Steel Piping. Cold cracking is not generally encountered in the welding of plain carbon steels with carbon contents below 0.3%. No evidence of such cracking was noted during the extensive development, qualifications, and acceptance testing to which the

N Reactor plain carbon steel RCS components were subjected. Cracks of the subsurface shrinkage or hot-tear variety were, however, noted in the longitudinal seam welds of 16 in. and greater diameter piping. This section covers the (1) nature of observed cracks, the testing and analysis to examine their significance to the fatigue life of the piping, (2) special controls instituted to ensure that the defect size in the final product did not compromise the integrity of the RCS, and (3) significance of the cracks to other fabrication and field welding processes.

Approximately 80 tons of the 1,100 tons of seam-welded RCS piping had been fabricated when the presence of a subsurface crack was discovered in a longitudinal weld. The discovery was made during dye penetrant examination of a pipe-to-fitting weld preparation site on a pipe end. Test specimens, cut from production seam welds of pipe made by both involved manufacturers, revealed similar defects. The involved automatic submerged arc seam-welded pipe had been accepted to all originally defined specifications at the time the defects were noted.

The observed subsurface defects were generally perpendicular to the fusion line and penetrated into the weld metal from that point. The majority of the cracks were along the grain boundaries established by dendritic solidification of the weldment (i.e., intergrannular in nature and typical of hot cracking) (Ref. 8, 9). Hot cracking is attributed to the formation of low melting point constituents in the weldment and tearing at these locations during or just after solidification under the influence of shrinkage stresses or weld zone constraint. While hydrogen embrittlement can be a major factor in hot cracking of alloy steels, preheat, heat input and cooling rate are the major factors involved in plain carbon steels (Ref. 10).

The original fusion welded pipe specifications subjected all pipe joints prepared for welding and weldments to visual, magnetic particle, liquid penetrant, and macroetch inspection. All weld joints

received 100% radiographic examination. The inability of the prescribed examinations to reveal the defects was investigated. It was found that larger defects could be detected by angled X ray examinations which aligned the beam closer to parallel than normal to the weld fusion plane. Smaller flaws were detectable by shear wave ultrasonic inspection, while the smallest ones were evident only with destructive cross-sectioning and macroetch or microscopic examination. The guided side bend test emerged from an evaluation of several destructive methods evaluated as the only readily standardized, reliable means of detecting and assessing the defects (Ref. 8).

The initial evaluation focused on defining the extent and magnitude of defects in the fraction of piping completed at the time of their discovery. Extensive bend tests were taken and statistically analyzed. Table 5.2-14 presents the results of this evaluation (Ref. 8) in which a limited number of side bend test indications to 1/2 in. were observed. Means of reducing the size and frequency of the weld defects was simultaneously undertaken.

Thermal contraction stresses during weld solidification, coarse columnar weld grain structure, and formation of low melting point constituents by involvement of the base metal in the fusion process, were identified as the major causes resulting in formation of the weld defects. A "best effort" program by the two involved pipe manufacturers was directed at controlling welding variables to minimize the above effects. The amount of weld deposit was minimized by redesign of the groove preparations. The amount of weld deposit per pass was reduced as was the profile of the individual weld beads to the parent metal. The preheat temperatures were also increased as a means of controlling thermal stresses. Additionally, excess weld metal removal and fairing of the welds into the base metal at the OD and ID of the pipe was implemented to eliminate the weld toes as nucleating sites for fatigue cracks.

Procedures were developed to repair the pipe produced with the original weld procedures. The repair procedures were based on the defect locations noted in the destructive testing. One manufacturer milled the inside surface of the weld and replaced it with multiple passes with the other air-gouged two grooves along the outside surface of the weld and closed each with a single pass to produce acceptable welds. All pipe accepted under the revised specifications was required to show no side bend defects greater than 3/64 in. on three adjacent side bend specimens taken from the opposite ends of alternate pipe sections. Only one defect between 1/32 in. and 3/64 in. was permitted in the three adjacent samples, and only five defects exceeding 1/64 in. on three adjacent samples was acceptable. In addition, 10% of the pipe was sampled at the midpoint using three adjacent side guided bend specimens evaluated to the same requirements.

The resultant level of weld defects was reduced by about a factor of 4, as shown in Table 5.2-14. The second column of this table includes the results of pipe produced to the revised welding procedures and rejected. These results were used in calculating the final defect frequency-size distribution in the constructed RCS piping as shown in Table 5.2-15.

A series of fatigue tests was conducted to examine the effects of weld flaws on pipe integrity for defect sizes synonymous with the 3/64 in. bend test rejection point and the larger defects which were statistically projected to exist (Ref. 9). These tests were conducted at Hanford and the Southwest Research Institute under three different conditions: (1) reversed load bending, (2) internal pressurization of free rings, and (3) pressurization of vessels. These forms of testing permitted evaluation of various wall stress gradients, various mean stress components, biaxial loading conditions, and also permitted maximizing stresses applied to interior and subsurface defects in the respective methods.

The reversed bending tests were used primarily to examine the notch sensitivity of ASTM A 212 Gr B carbon steel and the fatigue sensitivity of N Reactor pipe. Specimens obtained from typical pipe sections including weld defects equal to or slightly exceeding the guided bend test specifications, did not show a reduction in fatigue strength compared to unflawed welds in the same material (Ref. 9, 11 and 12). The resultant fatigue strength of the pipe also complied with the proposed requirements for the ASME Section III Nuclear Vessel Code (Ref. 13). Stress amplification factors developed for various defects further showed that the factors for acceptable weld defects were comparable to those applicable to surface roughness discontinuities within the limits of that permitted by the specifications.

Unidirectional load pressurization tests were conducted by sealing 2-in. long ring sections of pipe between pressure plates with the use of an O-ring seal and leather backing rings. Under this type of loading, marginally acceptable weld flaws again showed no reduction in fatigue strength in comparison with unwelded material.

Biaxial tests were conducted at room temperature using two 7-ft long vessels fabricated from sections of the N Reactor piping. The first vessel utilized a rejected pipe section which had shown side bend tear indications to 1/2 in. length. This vessel failed after 22,000 cycles of pressurization near the yield point at a flaw nucleated weld defect. In contrast, specimens fabricated to the revised requirements experience failure from weld defect nucleated flaws after 100,000 cycles at near-yield stress values. The second vessel, which was fabricated from acceptable pipe, received over 37,000 cycles near the yield point before a failure occurred in the base metal which had nucleated from surface pits. After repair, a second base metal failure occurred at 48,000 cycles.

The fatigue testing program demonstrated that welds accepted to the revised specifications, including the additionally imposed side bend test requirements, did not reduce the fatigue life of the base metal. The tests were conducted at or near room temperature with stresses approaching yield. A large margin of safety was demonstrated in comparison with the design bases 1,000 scram cycle fatigue life of the RCS. The piping met proposed fatigue life requirements for nuclear components.

The vessel failures showed no evidence of impending brittle fracture. This observation includes one test where the test vessel temperature was reduced to -23°F whereupon a fatigue flaw was cycled at near section yield stress values to a condition of excessive leakage. The flaw attained a length of 3 in. on the inside wall without initiating a brittle failure. In a second demonstration, at 50°F, an existing base metal fatigue crack was subjected to a combined hydrostatic and local impulsive load. The impulsive load was generated using an internal explosive squib. The majority of the crack extension was brittle as expected. The propagation of the crack was limited to the impulsive loading zone. It did not occur in the zone affected only by near-yield stress hydrostatic loading.

c. Preheat and Interpass Temperatures. The N Reactor "Welding Specification for Radioactive System Service," N Standard Section N7, was largely based on Section IX of the ASME B&PV Code. For carbon steel thicknesses greater than 3/4 in., a minimum preheat temperature of 200°F was required. No preheat was required for root passes performed by the inert-gas-shield tungsten arc process except that no welding was permitted at base metal temperatures below 60°F. A maximum interpass temperature of 600°F applied to all welding processes. Preheat, as required, was an essential variable in the procedure qualifications as per Section IX.

2. Electroslag Deposits, Control of Welding Variables and Metallurgical Tests for Procedure Qualifications

Control of electroslag weld properties was achieved via procedures and impact tests of the weld metal to assure weldments met base metal requirements. Dye penetrant testing of completed welds also afforded protection against undesirable effects from weld solidification patterns. Macroetching, conducted in support of longitudinal weld development and processing, further demonstrated that this type of problem was properly controlled.

Welding variables were closely controlled to achieve the quality established by procedure qualifications. Preheat temperatures were measured either by "Tempilstiks" or thermocouples and post-weld heat treatment by thermocouples located in the anticipated hottest and coldest areas of the weldment. Weld preparation data, tack welding process, welding process, number of weld passes, size and type of electrode, ampere setting, base materials, inspection methods and results, repair welding data, preheat and postheat treatment requirements and conditions, and welder identification were all controlled and recorded for each shop and field weld of the RCS.

All welding procedure qualifications for high pressure piping included Charpy Impact test requirements for the weld metal and heat-affected zone (HAZ). Weld metal specimens were derived with the specimen axes transvers to the weld axes with the notch parallel with the surface of the welds and in the uppermost surfaces of the specimens. The uppermost specimen surfaces were not to be more than 1/16 in. below the weld surfaces. Heat-affected Zone samples were derived with the long axes parallel to the direction of maximum working, to the extent possible, with the base of the notch perpendicular to the surface. Specimens were cut as nearly as possible from the 1/4T location.

3. Limited Accessibility Welding

Shop fabrication was planned to maximize the amount of welding which could be performed in the horizontal position. Other positions, including regard for accessibility provisions, required specific qualification.

5.2.3.3.3 Nondestructive Examination

Seamless carbon steel pipe used in the N Reactor RCS was subject to ultrasonic inspection requirements for both radial defects extending longitudinally and transverse defects extending circumferentially. For transverse defects, the calibration notch was sized at 3% of the wall thickness, approximately 1 in. long, and not wider than twice the depth. The calibration notch was cut to provide smooth sides extending into the test section transverse to the pipe axis. The test was conducted with ultrasonic pulse-reflection equipment coupled to a shear wave angle-searching unit. The scans were toward the pipe ends with the search unit motion from the pipe end to two diameters past the center. The scans were one crystal width apart, and 100% surface coverage was required. The radial test defect was dimensioned to the same requirements as the transverse defect and cut into the test sample with the sides normal to the pipe surface and extending parallel to the pipe axis. One notch was cut from the OD and one from the ID surface. The scanning was conducted in two opposite circumferential directions.

All forgings received both magnetic particle and ultrasonic inspections after final heat treatment. Ring forgings were tested by shear wave; other shaped forgings by longitudinal wave from all surfaces. Shear wave calibration and rejection was to a 3% wall thickness notch depth approximately 1 in. long and with a width not greater than twice the depth. Rejection for longitudinal wave tests were established at 1 mc frequency for indications exceeding 5% of the end reflection indication.

The fusion-welded pipe received ultrasonic examination of the base metal and weld radiography. For the piping and forging thickness involved, the calibration notches also meet the requirements contained therein.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

The N Reactor RCS pump casings are fabricated from ASTM A 351 Gr CF8M austenitic stainless steel and are the only RCS pressure boundary materials fabricated from austenitic stainless steel. The following discussion is therefore pertinent only to the RCS pump casings.

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

1. The RCS pump casings were solution heat-treated and water quenched to provide an unstabilized structure throughout. The heat treatment was conducted at 2,000 + 50°F and the castings were held at that temperature for 1 h for each 1 in. of thickness. Quenching in water was specified to achieve rapid cooling of the casting to below 800°F. Each casting included provisions for at least two intergrannular stress corrosion cracking samples. Following the heat treatment, the corrosion tests were conducted for 72 consecutive hours in a boiling solution of 10 g $CuSO_4 \cdot 5H_2O$, 10 ml H_2SO_4 (1.84 s.g.), and 90 ml H_2O. The specimens were then bent 150 degrees at a radius not greater than 1.5 times the width. There was no evidence of cracking or other forms of disintegration by either the naked eye or examination under 30x.

   Major repair welds were made on the pump casings as necessary before heat treatment. The pump casings received minor repair welding and were welded to A-216 WCB transition pieces on the inlet and outlet nozzles after heat treatment. The transition piece weld preparations were buttered with stainless steel. All welding was controlled to minimize the heat input and thus the potential for sensitization. No preheat was utilized unless required to establish a minimum component temperature for welding of 60°F, and the interpass temperature on the transition piece-to-pump casing weld was limited to 400°F. The interpass temperature of the buttering process was limited to 200°F.

The pump castings received radiographic inspection over 100% of their volume. Class I requirements of ASTM E71 were specified for thicknesses of 1 in. or less; Class II requirements for thicknesses greater than 1 in.

All surfaces of the finish-machined castings, including as-cast surfaces, received liquid penetrant inspection. All surfaces were required to be free of cracks or tears. Linearly disposed porosity was rejectable if the average center-to-center distance between adjacent indications was less than 3/16 in. Weld faces, exterior surfaces of welding ends prepared for field welds, repair welds, and adjacent base metal were required to meet the following:

Size of Indication (in.)		
Dye Penetrant	Fluorescent Penetrant	Number per Square Inch
≤1/16	≤1/32	10
>1/16 x ≤1/8	>1/32 x ≤1/16	2
>1/8	>1/16	none

2. The RCS pump castings were cleaned by a pickling process following heat treatment. The pickling process was followed by a thorough water rinse. Normal care was exercised in protecting the castings from stress corrosion contaminants during processing, storage, shipments, and installation.

3. The CF-8M austenitic, cast, stainless steel is not cold-worked, thus the provisions for such material are not addressed.

The specifications for N Reactor RCS recirculating pump components made from austenitic stainless steel fulfill the current code requirements with one exception. Weld procedure qualification did not require an intergrannular corrosion test. Heat input was controlled, however, and in-service inspections have demonstrated the acceptability and adequacy of the welding procedures utilized.

5.2.3.4.2 Control of Welding on Austenitic Stainless Steels

1. Weld repairs on the pump casings, welding of the buttered carbon steel suction, and discharge nozzles to the pump casings were made in accordance with N Standards Section N7 and ASME B&PV Code Section IX. The procedure qualifications and individual welder qualifications included evaluation of side bend test specimens, macroetch specimens, and tensile specimens and were made to Military Standard 248 to which the vendor welders were qualified. Military Standard 248 was being used for naval reactor component fabrication at the time.

2. Welds and Weld Repairs - Welds and weld repairs were subject to visual, radiographic, and liquid penetrant examinations. The pressure retaining components were examined over 100% of the volume radiographically. The casings were required to meet Class I requirements of ASTM E71 for wall thicknesses of 1 in. or less, and Class II for wall thicknesses greater than 1 in. Welds and weld repairs were inspected radiographically with no cracks or zones of incomplete fusion permitted. Porosity was controlled to the requirements of Appendix IV of ASME B&PV Code Section VIII. Elongated slag was controlled as follows for total slag inclusions in any 6-in. of weld:

Weld Thickness	Acceptance Limit
Up to 1/2 in.	0
>1/2 to ≤1	1/8 in.
>1 to ≤2	1/4 in.
>2 to ≤4	3/8 in.
Greater than 4 in.	1/2 in.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

The inservice inspection program is administered and conducted by Westinghouse Hanford for DOE in accordance with the "N Reactor ISI and Testing Plan" (Ref. 4). The plan is based on ASME B&PV Code Section XI, 1983 edition through Winter 1983 Addenda and is approved by DOE.

5.2.4.1 Boundary Subject to Inspection

WHC-CM-5-15 (Reference 4), through a colored diagram, illustrates the class 1 pressure boundary as currently defined by the Technical Specifications, and a similar illustration of classes 2 and 3 have been prepared and are in review. 21

5.2.4.2 Accessibility

N Reactor was not designed specifically to enhance inspectability of RCS components. Design considerations given to maintenance needs, however, afford adequate accessibility for inspections. Pressurizer and steam generator inspections are made by the originally designed manways and hand holes. Pump inspections are made when insulation is removed and the pumps are disassembled for overhaul. Pipe and component weld inspections require removal and subsequent replacement of wrapped-type insulation. 14

5.2.4.3 Base Point Inspection Data

Initial component installation and shop weld acceptances were based on radiographic examination. All volumetric, surface, and visual examination techniques are used for the inservice inspections. While acceptance records exist for initial installations, a complete base data record was not established. For certain welds, the vendors were assigned responsibility for maintaining the radiographs for a period of 5 yr. This time period elapsed prior to the initial reviews of inservice inspection requirements. It was found that some of the initial records had been destroyed. Inspections made under the inservice inspection program have been maintained in compliance with applicable requirements. Although, the N Reactor base point inspection program did not comply with ASME B&PV Code Section XI requirements for preoperational examinations, subsequent initial inspections have provided adequate base data. 14 14

5.2.4.4 Examination Techniques and Procedures

Required examinations are identified in WHC-CM-5-15 (Ref. 4) and conducted in accordance with ISI Equipment Maintenance Standards (EMSs). In addition to the standard examination techniques, various unique examinations, both destructive and non-destructive, are performed on process tubes and Graylock connector studs. 21 14

5.2.4.5 Inspection Requirements and Intervals

Reference 4 lists the frequency for each examination method for each component subject to inspection. The acceptance criteria for examination is provided by the appropriate ISI EMS procedure

5.2.4.6 Evaluation of Examination Results

Evaluation of results is based on Article IWA 3000 and IWB 3000 of ASME B&PV Code Section XI. Indications that do not pass examination acceptance standards are reported by Non-Conformance Report (NCR). Final acceptance is then by repair or replacement of the component containing the indication, or by analytical evaluation to show that indication growth during subsequent service is acceptable.

5.2.4.7 Replacements

Replacements (or modifications) are made in accordance with the requirements of the Pressure Systems Instruction, UNI-M-15, which implements ASME B&PV Code Section XI. Thus, replacement components meet original specifications or the differences are technically reconciled.

5.2.5 Leak Detection Through the Reactor Coolant Pressure Boundary

Design of N Reactor's coolant system included a high-planned bleed rate as summarized below:

- o One spill cooler line (approximately 40 gpm warming flow)
- o Eighteen (1/2 in. OD) coolant sample lines for water quality monitoring (approximately 20 gpm).

Thus, the design bleed rate from the reactor coolant system is about 60 gpm.*

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*1 1,003 coolant sample lines (1/4 in. OD) for fuel cladding failure monitoring (approximately 240 gpm); The rupture monitor reinjection system (see Reference 16) makes up the loss and has reduced this bleed rate to zero.
2 Two coolant purity bleed lines (approximately 25 gpm, these are not used during operation).

There are approximately 12,000 valve packing glands, nozzle gaskets, and replaceable seals that are potential sources of unplanned coolant leakage. Leak inspections are conducted during each scheduled reactor refueling outage. Should any RCS leakage through a pressure-containing member (i.e., valve body or pipe) be found, it must be repaired in accord with established procedures prior to reactor startup. Leakage detected from packing glands, gaskets, and seals are evaluated on a case-by-case basis. Factors evaluated include leak rates, potential for damage to components, and potential for the leak rate to increase. Leak repairs are scheduled according to priorities established by the evaluation. Normally, leaks above 2 gpm at the 300 psig inspection pressure are repaired prior to reactor startup.

The unplanned leak rate from the 12,000 potential sources averages about 300 gpm. This leak rate is more than a factor of three lower than the maximum tolerable sustained rate of 1000 gpm as established by the maximum makeup capacity of the demineralized water plant. Coolant leak rates, however, do not compromise the reactor coolant pressure boundary because of the following:

1. The inspection frequency of the leak detection program discussed below in 5.2.5.1 | 14

2. The high pressure injection system, as discussed in Section 9.3.4, is capable of supplying coolant at a rate ten times greater than the experienced bleed and leak rate combined. This capability eliminates the need for emergency core cooling actuation except for postulated accidents as covered in Section 6.3 and Chapter 15. See Chapter 11 for the radiological consequences of reactor coolant leakage.

The N Reactor leak detection program incorporates both frequent inspections during scheduled reactor outages and instrumentation to detect abnormal leakage when the plant is operating. The methods and effectiveness of the leak detection program are covered in this section.

5.2.5.1 Visual Inspections

A visual leak inspection of the reactor coolant system is performed with the system pressurized to 1,600 psig during each scheduled reactor refueling outage. Scheduled outages occur on the order of approximately every 30 to 45 days. Thus, the reactor coolant pressure boundary is inspected from six to eight times per year.

Visual inspections identify each leak by leak rate, location, and type. Any leak with structural integrity implications is administratively required to be repaired prior to startup. Seepage through a crack in a pipe or valve body would be of that type. Administrative controls also cover repair of leak sources that are not detrimental to facility safety.

The visual inspection method of leak detection is effective to much lower than 1 gpm. Scheduled inspections are reinforced by inservice inspections that are performed on a less frequent basis. The inservice inspection requirements are specified in WHC-CM-5-15 (Ref. 4) and are administered through Inservice Inspection and Test Procedures (Ref. 15).

5.2.5.2 Pressure Tube Flow Monitoring

During power operation, the pressure tube flow monitor serves to detect leaks in both the inlet connector and pressure tube and for detection of coolant flow restrictions for each reactor fuel column. Each flow monitor is set to scram the reactor before any leak in the inlet connectors or pressure tubes upstream from the fuel column could cause inadequate flow for the prevailing pressure tube power level. In addition, the flow measurement for each pressure tube is routinely analyzed for normalcy by a minicomputer that determines whether the flow has deviated from the established operating range by a statistically significant amount, but within the set trip points. This flow deviation check would identify a leak of 12 gpm or less which corresponds to a 6% flow change on a tube operating with a normal flow of up to 200 gpm. The printed measurements are read out and analyzed periodically.

The leak monitoring function of the pressure tube flow monitor also provides assurance that a leak on any one of the 1,003 pressure tubes will immediately result in a reactor scram if flow drops to a preset trip point. Trip settings provide a reactor scram prior to departure from nucleate boiling in event of an individual or general tube flow reduction or a leak in a pressure tube or its associated flow connectors. Flow monitor trip settings are discussed in Section 4.4.6.

5.2.5.3 Moderator Gas System Moisture Monitoring

During power operation, the exterior surfaces of all 1,003 pressure tubes are surrounded by helium which is circulated and regulated by the moderator gas system. See Section 9.3.6. A very small pressure tube leak, on the order of 50 gal/d (less than 0.04 gpm), would be identified as a leak into the reactor based on water collection in gas dryers and in drip legs. Before obtaining a measurable leak rate, high-gas dew point readings would have been noted. In a larger tube leak, an alarm would be actuated on high-gas system pressure, and other alarms would be actuated by high water level indications in moisture collecting drip cells of the reactor moderator gas system.

2

Since 1963 when N Reactor began operation, no pressure tube leak has occurred. There is no indication from surveillance program results of any increased likelihood of such a leak (see Section 5.3). However, the sensitivity of the leak detection system has been demonstrated with a leak in each of two different graphite cooling tubes that occurred at different times.

5.2.5.4 Spill Cooler Leak Monitoring

The secondary-side cooling water for the spill cooler portion of the reactor coolant system is normally nonradioactive raw water that is routed to the 1325-N facility. This practice was a precautionary step to preclude leakage of radioactive primary water being returned directly to the river.

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A diversion valve and diversion line were added by DC-87024 (Ref. 19) allowing the raw water effluent line to be diverted to the RWR line. To prevent uncontrolled radioactive release to the river, sample of the spill cooler raw water is continuously monitored for gamma radioactivity, by the Liquid Effluent Radiation Monitor (LERM), which provides an alarm at a set activity level. On

receipt of an alarm, during startup and during shutdown, the diversion valve is turned to the 1325-N facility. During normal full power operation with no detectable leakage and no alarm present, the flow is diverted to the river via the RWR line. If leakage is detected either by receipt of an alarm or by grab samples, the diversion valve is turned to the 1325-N facility. In addition, grab sample of spill cooler raw water can be obtained via SP-94. Sensitivity of the automatic leak detector system varies inversely with the raw water flow. The raw water flow is automatically varied with the spill rate. Thus, the leak detection sensitivity varies inversely with raw water flow. The LERM alarm can detect levels corresponding to a 0.08 gpm leak rate at the normal raw water flow rate of about 100 gpm and to a 2 gpm leak rate at the design maximum flow rate of 2,780 gpm. A separate grab sample is collected three times a week at SP-94 to back up the LERM.

5.2.5.5 Steam Generator Leak Monitoring

Samples from the cell steam generator secondary side are routinely taken and the radioactivity measured with a multi-channel analyzer. Since the reactor does not continue to operate with a confirmed cladding failure on any fuel element, coolant activity is normally low (see Chapter 11). Therefore, fluorine-18 is a good isotope to track, and its 0.51 MeV photon is used as the reference energy peak.

A leak of about 0.02 gpm can be detected during power operation, and positive identification can be made at approximately 0.16 gpm. However, experience has shown that a leak in steam generator tubing must be about 0.16 to 0.66 gpm at full operating pressure before it can be specifically located by post-shutdown visual leak inspection efforts.

5.2.5.6 ECCS Dump Line Temperature Monitoring

Thermocouples monitor coolant temperatures in the ECCS dump line downstream from the V-4 reactor dump valves. This provides an indirect means of detecting leakage from the reactor coolant system past the V-4 valve seat during power operation.

5.2.5.7 High Pressure Injection Rate Monitoring

Normal injection rates reflect the design coolant bleed-off the supply of seal-water to the reactor coolant pumps, total leakage from the system, and occasional flow to pressurizer sprays. Although the injection rate varies, abnormal makeup increases are logged and investigated as potential leaks. Technical Specifications (Reference 14) require that the maximum high-pressure injection water makeup rate shall be 1,900 gpm. If the limit is exceeded, the reactor shall be shut down at the normal rate or faster. 14

Considering the normal fluctuations that occur in the injection rate, the sensitivity of this leak detection method is about 100 gpm. The method is not of the same order of sensitivity as the other leak detection methods. It provides a real time effective means of detecting leaks anywhere in the reactor coolant system. The flow rate of such leaks can be readily accommodated by the reserve capacity of the injection system. The injection system total capacity is over 6,000 gpm, about a factor of ten greater than the capacity typically provided for a LWR. The reserve capacity is about 5,000 gpm.

An example of a small pipe break that was detected by this method was the failure of a spill cooler tube of 5/8-in. OD. The tube parted, creating a double-ended leak path of reactor coolant to the low pressure cooling water side of the spill cooler. The reactor coolant system injection increased by 200 gpm and was immediately noted. Total injection rate remained below the prompt shutdown administrative limit; however, after the leak location was identified the reactor was shut down for repair. Significantly, this small break was accommodated without any reactor coolant system upset or reduction in coolant to the reactor fuel. It used less than 5% of the reserve injection capacity to maintain reactor coolant system inventory.

5.2.5.8 Testing and Calibration

The instruments discussed above are tested and calibrated according to Equipment Maintenance Standards (Ref. 15). 14

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16. "105-N Fuel Rupture Monitor Sample Water Reinjection," Project 516.

REFERENCES - 5.2 (cont.)

17. UNI-719 REV1, "Policy Governing the Use of ASME Boiler and Pressure Vessel Code," September 28, 1984.

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TABLE 5.2-1

REACTOR COOLANT SYSTEM DESIGN PARAMETERS

Reactor Coolant System

Design Pressure, psig	1,825*
Design Temperature, °F	600°F*
RV-2 Valve capacity, each lb/sec.	1300
RV-2-1 Valve setpoint, psig	1700 ± 15
RV-2-2 Valve setpoint, psig	1775 ± 15
Reactor trip points (See Table 5.2-2)	

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*See Section 5.2.1 and Figure 5.2-1 for pressure-temperature envelope because of use of ASA 900-pound valves.

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TABLE 5.2-4

CELL SAFETY VALVES

2" x 3" PCRV-210, PCRV-6210

Body	ASTM A-216 Gr WCB
Seat Surface	Stellite #6
Flow Rate, gpm	600 @ 2000 psig
Set Pressure, psig	1,825

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TABLE 5.2-5

PRESSURIZER SAFETY VALVE

1 x 1 1/2" RVTV-202

Body	ASTM A-216 Gr WCB
Seats	Stellite #6
Flow Rate, lb/h	17,500
Set Pressure, psig	1,825

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TABLE 5.2-6

RECUPERATIVE & SPILL COOLER HEATER EXCHANGER SAFETY VALVES

1 x 1 1/2" Valves BDV-209-1R, 222-1R, 222-2R & IWV-281-4R & 281-6R

Body	ASTM A-216 Gr WCB
Seat Surface	AISI 304 STN STL
Flow Rate, lb/h steam	≈19,010
Set Pressure, psig	1,800

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TABLE 5.2-16

INSERVICE INSPECTION REQUIREMENTS

Examination Area	Method	Frequency[4]
1. Pressurizer		
a) Vessel Cladding	V - internal patch (36 sq in)	Once/10A
b) Nozzle-to-Shell Weld (N-1, N-2, N-3, etc.)	V and UT or RT	Three/A
c) Heater Connection Weld	V - external surface	10 percent/A
2. Steam Generator Cells		
a) RC Pump Casings	V - internal	One case/6Q
b) RC Pump Studs (pressure-retaining bolts)	V	One pump/6Q
c) Valve Bodies (PCS 24 in - PCS 18 in)	V - internal	As Repaired
d) Valve-to-Pipe Welds (PCS 24 in - PCR 18 in)	UT or RT	One cell/6Q, each size - both ends
e) Elbows (longitudinal and traverse welds)	UT or RT	One cell/6Q, one elbow per generator
f) Steam Generator Tubes[1]	EC	100 tubes/A
g) Tube Sheet-to-Shell Weld	V and UT or RT	Once/10A
h) Inlet/Outlet Header Flange Bolting	V and UT or RT	One generator/6Q
3. RCS Piping		
a) Grayloc Studs	Destructive	20/2A
b) Connector-Riser Welds[2]		
Front	V and UT or RT	10/A
Rear	V and UT or RT	10/A
c) Pressure Tubes[3]	Destructive	1977
d) Nozzles and Rolled Joints	R & V	6/A
e) Pipe Hangers and Supports	V	10 percent/A
f) Snubbers	V	25 percent/A
4. Other		
a) RCS Hydro (1600 psig)	V with consideration for personnel safety	As required by TS 4.17.2.1

[1]Destructive examination required on a representative sample of tubes when removed from steam generator.
[2]Inlet and outlet riser.
[3]The destructive examination frequency for pressure tubes is based upon previous results. The results are due one year after removal.
[4]WHC-CM-5-15 (Ref. 4) is controlling.

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equipment was verified before use on the reactor and the details of the verification are discussed in Reference 56.

The eddy current system examined the inner surface to about 35 percent of the tube wall. Ultrasonic methods can examine the entire tube wall. The inner 35 percent is adequate for four main reasons:

1. Like thick-walled vessels, the inner surface of the pressure tube has higher tensile stress, thus the inner surface has more significance.

2. The inner surface has the highest concentration of stress risers resulting from wear due to fuel charge/discharge and in-resident contact with the fuel.

3. The inner surface will embrittle faster because of the higher hydrogen absorption rate on the water side of the tube. Hydrogen tends to embrittle zirconium alloys.

4. The inner surface was the most difficult to examine during preservice inspections.

Five pressure tubes were removed from N Reactor in 1987 for the destructive evaluation of flaw indications found during the NDE of 152 tubes. Five of the indications in three tubes were found to be fabrication induced flaws with no indication of in-service growth. All other indications were minor surface anomalies such as fretting corrosion pits, shallow gouges or scratches. The details of these investigations and the findings are discussed in References 11, 12, 13, 14.

5.3.4.2.3 Visual Inspections

Visual inspection examines the inner surface of the pressure tubes. As a part of the pressure tube surveillance program, a minimum of nine pressure tubes per year are visually inspected by either CCTV or borescope to search for

visual evidence of mechanisms which could limit pressure tube life or cause premature failure (Reference 16). This visual evidence includes cracks, circumferential or deep axial scratches, deep wear marks, blisters, and any significant changes in the type of anomalies generally observed in the tubes: 20

1. Long, shallow axial scratches from charge/discharge.

2. Fretting pits from spacer supports, predominately in the downstream unfueled region, but also found in the upstream unfueled region.

3. Shallow wear from fuel supports.

4. Stain-like indications originally observed in Tube 1941. These are generally observed at mid-length of fringe tubes.

Nine selected tubes per year are sufficient based on past experience of six tubes per year plus 50 percent increase to investigate high-distortion areas.

Fourteen pressure tubes were examined visually by CCTV during the 1987 Standdown: 1173, 1941, 1974, 2061, 2873, 2952, 2954, 3053, 3057, 3063, 3162, 3166, 3255, 3351 (Reference 16). The CCTV camera was pushed through the pressure tubes. The inspections along with commentary by inspection personnel were recorded on video tapes. The commentary on these tapes provided the axial tube location of the camera at regular intervals and at areas of interest. 20

The 1987 visual inspection of fourteen N Reactor pressure tubes has revealed no unacceptable indications (Reference 16). Stain-like surface marks were observed in the mid-length area of tubes 1941, 1974, 2873, and 3057. After Tube 1173 was removed for evaluation of flaw indications found by eddy-current and ultrasonic inspections, a stain-like mark was observed in the mid-length area of the tube. This area was sectioned and examined. No unusual concentrations of hydrogen or metallurgical changes were found and it was 20

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thickness, and surface defects. Mechanical destructive tests include fracture toughness testing, tensile testing, and burst testing. 20

As of January 7, 1988, sixteen pressure tubes have been removed; eleven of these pressure tubes have been removed for examination and destructive testing and five tubes have been removed for characterization of flaw indications. Nine tubes were removed between 1963 and 1984. The other seven tubes were removed in 1987. References 7, 8, 9, 25, and 26 present the updated results of recent postirradiation examinations. The results of these examinations and tests are included in the detailed discussion of material properties that follows. Historically, only small changes have been observed in the pressure tubes and graphite stack over a several-year operating cycle. 20

5.3.4.3.1 Tensile Properties

Tensile properties have been measured on specimens from cold-worked Zircaloy-2 pressure tubes that have operated in Canadian CANDU reactors. In one of these reactors (NPD), the tube operated with a hoop stress of 72 MPa (10.5 ksi) and a center temperature of about 504°F (262°C). In the other reactor (Pickering), a tube operated with a hoop stress of 91 MPa (13.2 ksi) and a center temperature of about 522°F (272°C). For comparison, N Reactor tubes operate with a hoop stress of 10.8 ksi and a center temperature in a range of approximately 400°F to 550°F. The data and trends of the tensile tests at 536°F (280°C) are presented in Table 5.3-1 and Figure 5.3-3 (from Reference 27). Some reported strengths for Hanford N Reactor tubes have also been plotted for comparison. Both data sets show an increase in strength with irradiation. 20 20

In N Reactor, three tubes (Tube Nos. 1173, 2954 and 3053) were removed and used in the PIE program during 1987. Tube 1173 was manufactured by Allegheny/Ludlum Company. Tubes 2954 and 3053 are both manufactured by Harvey Aluminum Company. Both of these tubes had accumulated high levels of exposure, with Tube 3053 having the highest time integrated power in the reactor. Tube 3053 was also of interest because it represented a minority 20

of the Harvey tubes that contained 18 percent cold work, rather than the more common level of 30 percent cold work for that manufacturer. Both were selected to avoid high levels of silicon and carbon that may have previously produced anomalous results and atypical behavior (Tube 2755, References 25, 26). Tube 1173 was selected due to availability, but was of interest because of large distortions that are known to occur in the fringe areas. As a consequence, Tube 1173 became the first tube to be tested that represented a position outside the "Spike ring" (see Section 4.3.1.2) region of the core.

Testing of these three tubes was completed in 1987, see Reference 9.

A total of twenty-one tensile strength and ductility measurements were made using flat tensile specimens 0.10 inch in thickness with a 1.0-inch-long gauge section as recommended in ASTM Standard E8-83. The specimens were oriented in the tube axial direction, consistent with previous tests. Tests were conducted at room temperature, 350°F, and 550°F. Fast neutron fluences of the tube sections evaluated ranged from 0 to 5.9 x 10^{21} n/cm^2 (E > 1 MeV). Three of the tests on Tube 2954 (one at each temperature) were performed at higher (100 times) rates than are normally used in tensile testing to determine the effect of higher loading rates that might be due to a pressure tube rupture event.

The 1987 tensile test data are consistent with the previously generated database and extend the range of fluence covered to 5.9 x 10^{21} n/cm^2. The tensile results are shown in Figure 5.3-4, where the 1987 data may be seen to conform with the earlier data. With a total of 75 data points now available, it is possible to construct a predictive equation for tensile strength in ksi:

$UTS = a + bT + c\ln(\phi + .01)$, where

$a = 123.1$ T = temperature, °F

$b = -0.076$ ϕ = fluence (10^{21} n/cm^2)

$c = 6.70$ Std. Deviation = 6.5

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5.3.4.3.3.2 Crack Growth Model, Deterministic Analyses

A deterministic evaluation of N Reactor pressure tube service life, with regard to the brittle fracture failure mode, has included predictions of growth of postulated cracks in the tubes. The deterministic analysis presented herein was carried out using a crack growth rate equation of:

$$\frac{da}{dn} = 2.45\ (10)^{-14} \times (\Delta K)^{2.6} \qquad \text{(EQ 5.3-3)}$$

where a is crack depth in inches n is the number of cycles, and ΔK is the cyclic stress-intensity factor range in $psi(in.)^{1/2}$.

This equation was developed by applying an additional factor of 25 to the test data reported in 1983 by Picker and Pickles (Reference 33). The factor of 25 was based on engineering judgment. The data represent crack growth rates observed in tests conducted in a water reactor environment. The tests were conducted at various cyclic frequencies ranging from 4 cycles per minute to 0.002 cycle per minute.

5.3.4.3.3.3 Crack Growth Model, Probabilistic Analyses

Additional assessments of crack growth data were made during the probabilistic brittle fracture analyses (Reference 34). The resulting best estimate equation is based upon a least squares fit of the data and is:

$$\frac{da}{dn} = (1.17 \times 10^{-7} - 1.29 \times 10^{-8} f)(\Delta K)^{2.6} \qquad \text{(EQ 5.3-4)}$$

where f is frequency in cycles per minute and the other symbols are the same as eq. 5.3-4.

A comparison with the deterministic Equation (5.3-3) may be made by converting the ΔK units of Equation 5.3-4 to $psi(in.)^{1/2}$. The resulting equation is:

$$\frac{da}{dn} = (0.185 - 0.0204\ f)\ 10^{-14}(\Delta K)^{2.6} \qquad \text{(EQ 5.3-5)}$$

Comparing Equation 5.3-5 to Equation 5.3-3 indicates that the equation used for the deterministic analyses (Equation 5.3-3) gives crack growth rates that are higher than the best estimate (Equation 5.3-4). The difference is a factor of approximately 13. The equation used in the deterministic analyses represents a conservative upper bound of the data, which is appropriate for this method. The probabilistic method, however, utilizes a mean value of the data (Equation 5.3-4), with an assumed probabilistic distribution (in this case log normal) about the mean. 17

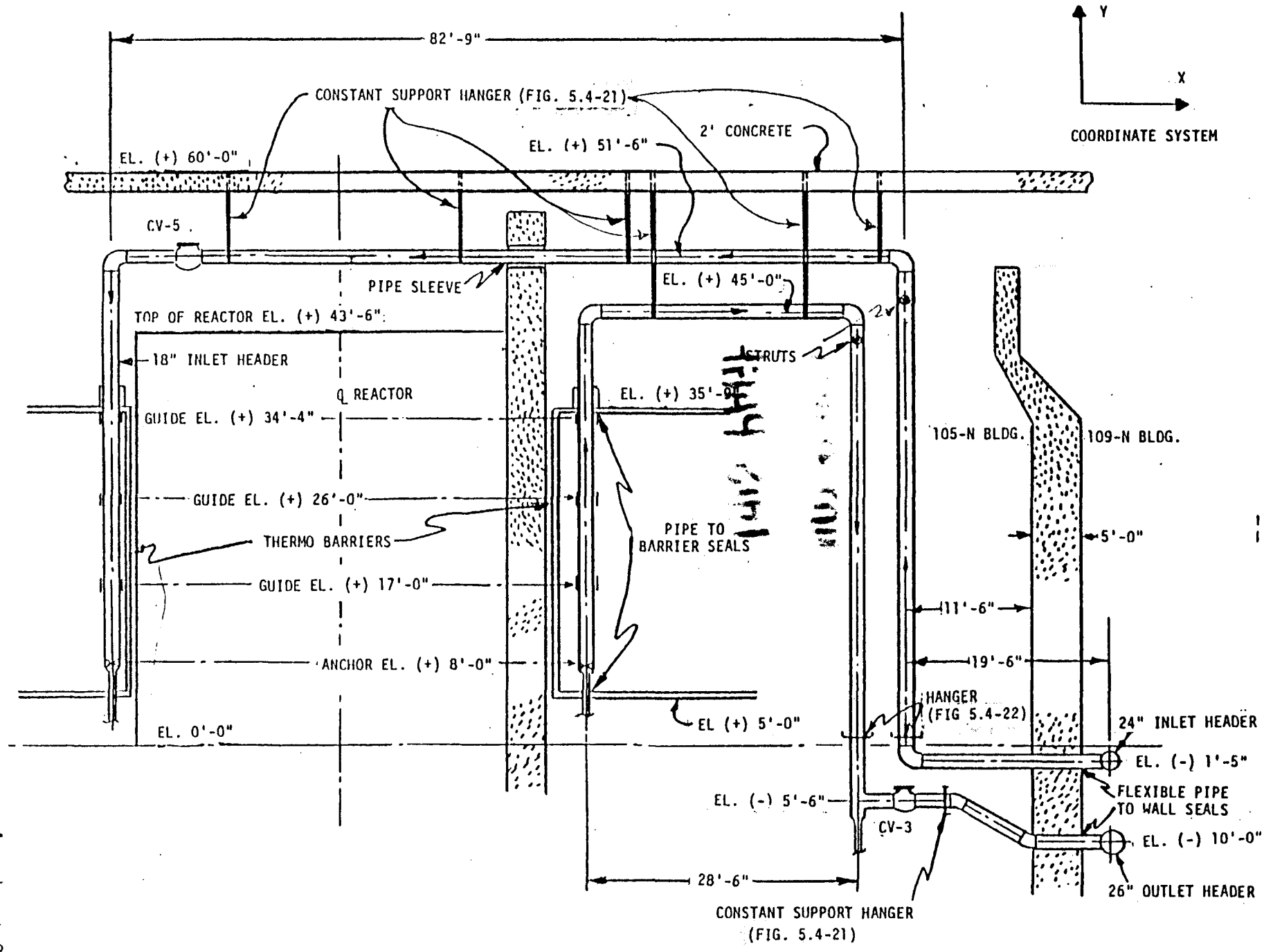
5.3.4.3.4 Delayed Hydrogen Cracking/Hydride Blistering 20

In delayed hydrogen cracking (DHC), hydrogen atoms dissolved from hydride in the zirconium alloy matrix migrate up a tensile stress gradient and precipitate at the point of highest stress. If sufficient hydride accumulates (forming platelets) and the stress is high enough, the hydride platelet cracks. The conditions for DHC are:

- o Zirconium hydrides in the matrix
- o A large tensile stress

Once initiated, delayed hydrogen cracking has the potential for relatively rapid propagation rates controlled by a diffusion process. Failures in Canadian reactors have been associated with: 1) excessive tensile residual stresses at rolled joints and 2) hydride blistering at a local cold spot on the tube. Detailed discussions of DHC and associated damage mechanisms are presented in References 26 and 35. 17 20

Axial cracks in CANDU reactor pressure tubes caused two notable failures. The first failure, in 1974, produced leaks in the nonirradiated, unfueled sections. They were discovered prior to cracks achieving the size necessary to cause unstable tube failure. Hence, leak-before-break (LBB) occurred. This failure was attributed to delayed hydrogen cracking (DHC) in a region of high residual stresses (nozzle-rolled joints) of Zr-2.5% Nb tubes. The second failure occurred in August 1983 and was also attributed to DHC from hydrides 17

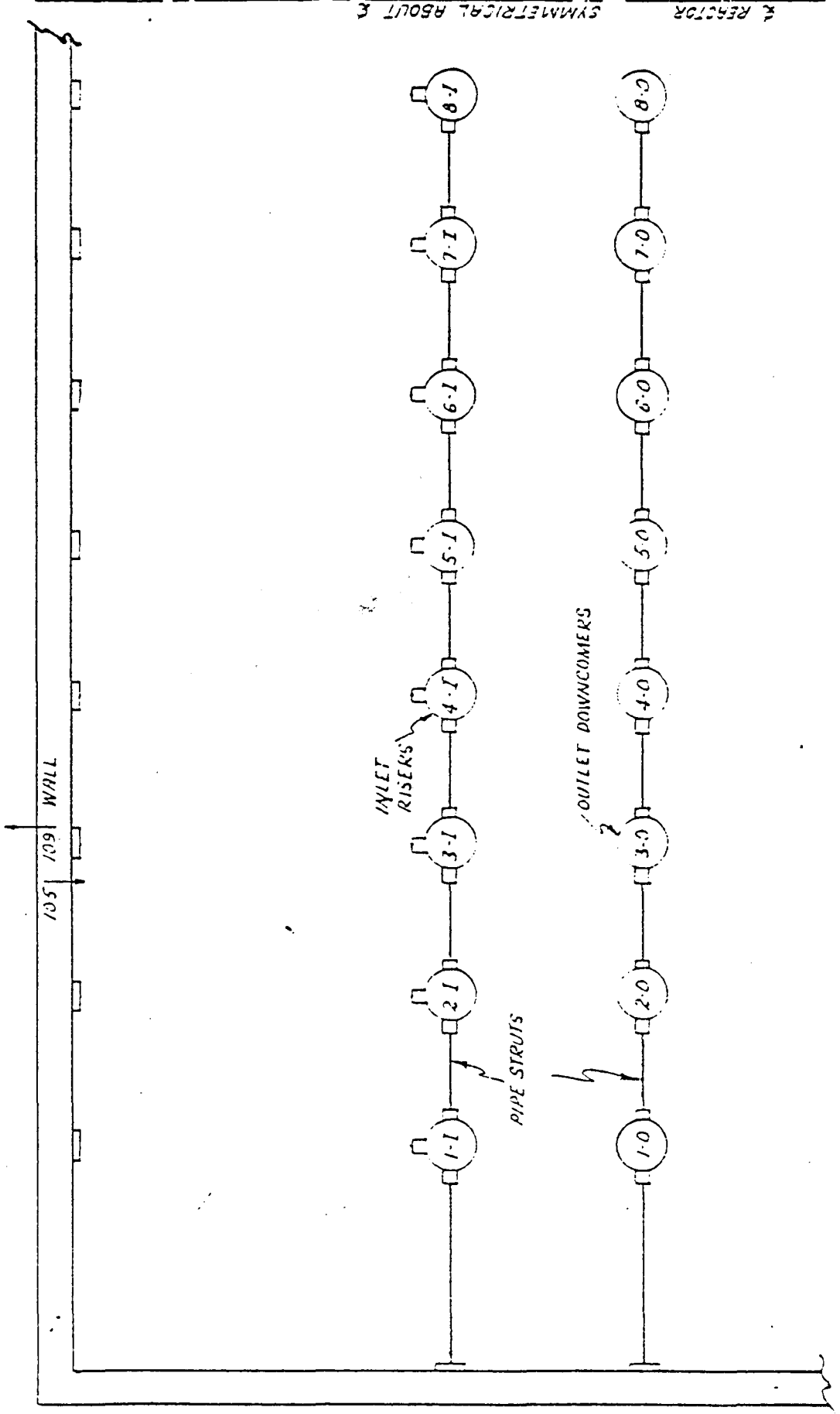


REACTOR COOLANT PIPING
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8.0 ELECTRICAL POWER

N Reactor electrical power for operating and shutdown conditions is supplied by offsite and onsite ac power sources. Considerable redundancy has been incorporated into both of these power systems, and either the offsite or the onsite ac power source is capable of providing for normal reactor shutdown on recirculation cooling. N Reactor loads supplied by these ac power sources are not safety-related in that they are not required for safe shutdown with ECCS cooling, and they are not classified as Class IE Seismic CategoryI systems (see Section 3.2.1). 21

Safe reactor shutdown using emergency core cooling system (ECCS) cooling of the reactor does not require either onsite or offsite ac power. Coolant pumping for the ECCS is supplied by completely independent (independent from normal reactor operational pumping systems) diesel-driven pumps. See Section 6.3. Electrical power required for ECCS actuating devices (solenoid valves and logic magnetic relays) is supplied by onsite dc battery-powered systems.

The definition of offsite ac power sources as used in the NUSAR refers to the ac power supplied by the Hanford 230 kV Loop from the BPA power transmission system and includes the ac distribution system (A Bus) within N Plant supplied from the offsite sources.

The definition of onsite ac power sources as used in the NUSAR refers to N Plant turbine generator (TG) ac power source, and the ac distribution systems normally supplied by the TG (B Bus).

Electrical powered safety related loads and their power sources are treated in Sections 8.1.3 and 8.3.2.

8.1 INTRODUCTION

N Reactor electrical equipment receives power from an offsite ac power system (Hanford 230 kV Loop) and an onsite ac power system (184-N building turbine generator). These two ac power sources are electrically independent, and each power source has a separate electrical distribution system. Either the offsite

ac power source or the onsite ac power source will provide for the orderly, safe shutdown of N Reactor utilizing normal recirculation coolant and will maintain the reactor and engineered safety features in a safe shutdown mode.

The onsite dc distribution systems for supply to dc loads are normally supplied by rectifiers which are powered from either the onsite or the offsite ac power source with batteries floating on the line to pick up the load in the event of ac power loss.

The primary power sources that may be used to energize the N Reactor electrical systems are:

1. the offsite power source from the Hanford 230 kV Loop through the 151-N substation 50 MVA transformer (A Bus normal source, see Section 8.2.1);
2. the onsite steam generation system (B Bus normal source, see Section 8.3.1.1);
3. the offsite power source from the Hanford 230 kV Loop through the 151-KW substation 50 MVA transformer and 3 miles of overhead 13.8 kV transmission line. (Alternate offsite power source for either A Bus or B Bus, see Section 8.2.1.);
4. the Hanford 230 kV Loop through the 151-D substation 31 MVA transformer and 2 miles of overhead 13.8 kV transmission line. (This power source is used as an alternate to K-N 13.8 kV power source when the K-N power source is not available, see Section 8.2.1.); and
5. the dc battery system through the dc distribution systems; these sources provide dc power to certain engineered safety features for safe shutdown of the reactor and confinement. See Section 8.3.2.

These electrical power systems are independent of the power generated by WPPSS from N Reactor secondary system steam.

8.1.1 Offsite Hanford 230 kV Loop Power System and Interconnections

The offsite Hanford 230 kV Loop supplies power to four substations for reactor areas and one substation for chemical separation areas (200-E and 200-W). See Figure 8.2-2. The Hanford 230 kV Loop is normally supplied power from two 230kV buses at the BPA Midway substation. Power can also be supplied into the Hanford 230kV Loop from the BPA Ashe Line. Any one of the three offsite power sources can provide adequate power for normal operation or shutdown of the reactor. See Figure 8.2-1.

Power lines which supply the BPA Midway substation are interconnected to the Pacific Northwest power grid system, which includes transmission systems at 500 kV, 230 kV and 115 kV. See Figures 8.2-4 and 8.2-5. 7

8.1.2 Onsite ac Power System

The onsite ac power source (see Figure 8.3-1) is supplied by N Reactor steam turbine generator with adequate capacity to supply all of B Bus power loads. The steam supply to the generator can be supplied by either an oil-fired boiler steam system or by nuclear-generated steam from N Reactor.

8.1.3 Electric Powered Safety-Related Loads

The majority of N Reactor safety systems and engineered safety feature systems do not require electric power to perform their design safety functions. The majority of the systems required for safe shutdown and confinement capability are designed such that no electrical energy is required to provide the

safety action. The energy required to perform the safety function in these systems except as noted below is provided by: (1) diesel engines with direct-driven pumps, (2) gravity acting on a fixed mass, (3) compressed springs, (4) air accumulators, (5) hydraulic accumulators, (6) gas cylinders or other means of stored energy.

Those engineered safety features, that require electrical energy to perform their safety function and their electric power sources are as follows:

Engineered Safety System Loads	Electric Power Source*
o Logic relays for actuation of emergency core cooling system (ECCS)	125 Vdc Battery A - 105-N 125 Vdc Battery B - 105-N
o Solenoid valves for release of stored gas for opening pilot valve in ECCS (V3 and V4 valve operation) (Two coils on one pilot valve)	125 Vdc Battery A - 105-N 125 Vdc Battery B - 105-N
o Confinement actuation relays, timers and pilot valves for actuation of the confinement system (two independent systems)	125 Vdc Battery A - 105-N 125 Vdc Battery B - 105-N
o ECCS and fog spray diesel engine drive speed control motors	125 Vdc Battery - 181-N 125 Vdc Battery - 182-N 125 Vdc Battery A - 105-N 125 Vdc Battery B - 105-N
o Inverter for supplying ac power to temperature instrumentation for ECCS actuation	125 Vdc Battery A - 105-N 125 Vdc Battery B - 105-N

*A spare battery and charger located in 105-N building are available to power either A or B dc Bus.

8.1.4 General Evaluation of Electrical System Adequacy

The power systems for N Reactor that provide for reactor core cooling, confinement and other features for safe operation and shutdown of the reactor are as follows:

1. Two electrically independent and redundant ac power systems, offsite (A Bus) and onsite (B Bus), both of which are electrically independent from the WPPSS Hanford Generating Project, provide for dual energy sources for reactor operation. Either the offsite (A Bus) or the onsite (B Bus) ac power system will provide energy for normal reactor shutdown and provide electrical power for adequate recirculation cooling for the shutdown condition. Automatic shutdown of the reactor occurs upon loss of voltage on either ac power source. In the event of loss of both ac power sources, the reactor emergency core coolant is supplied by a direct-driven diesel pumping system, and two redundant dc power sources provide for speed control of the diesel-driven pumps, for opening V-3 and V-4 valves and for surveillance of the reactor shutdown status.

2. Two redundant reactor shutdown safety systems, controlled by 125 Vdc power sources, provide for insertion of neutron absorption material into the reactor based on engineered fail-safe techniques. Loss of the dc power source will actuate the safety system for shutdown of N Reactor.

   One safety system uses hydraulic power to insert horizontal control rods into the reactor. The hydraulic power is supplied by a separate independent hydraulic accumulator for each rod. A backup safety system uses gravity to drop neutron absorption balls into the reactor. The balls are stored in hoppers located over vertical ball channels in the reactor, and the balls are released by a mechanism using stored spring energy.

Both reactor shutdown safety systems require 125 Vdc electrical energy to hold an electrical armature in a position to prevent the safety action. Loss of dc power causes spring energy to change the armature position and, thereby, release the stored energy for actuation of the safety action.

The horizontal control rod reactor shutdown safety system has two independent safety circuits, channel A and channel B. Either safety circuit, channel A or channel B, can initiate the shutdown safety action.

The backup ball safety system has two safety circuit channels, A and B. Both ball safety circuit channels are required to initiate the shutdown safety action. Loss of dc energy in both safety circuit channels A and B will cause the neutron absorbing balls to flow by gravity into the reactor for shutdown.

3. N Reactor confinement system is provided 125 Vdc electrical energy from two redundant dc systems for operation of safety-related devices for actuation of confinement.

4. Two offsite electrical power sources, 151-N substation and K-N line, are supplied to the N Reactor ac distribution system. These offsite power sources are physically independent of each other. A single power system fault concurrent with a single safety-related component failure (breaker or relay) in either offsite power system will not cause a failure of both offsite power sources.

5. The electrical power supply systems and components important for safety for N Reactor exceeds the requirement of 10 CFR 50, Appendix A, Criterion 17 in that (1) the onsite electrical generating equipment (the TG set and steam supplies) is operating during the power mode of operation (2) the electrical supplies for systems that provide for safe shutdown and emergency cooling are in service and backed by battery supplies and (3) no sequencing or transfer of electrical supplies is required during emergency conditions.

The electrical power supply systems important for safety for N Reactor meets the requirements of Criterion 18 in that the dc power systems are in continuous service during the reactor mode of operation and are, therefore, being constantly monitored and tested for availability for supplying the electrical energy necessary for the safety action.

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8.2 OFFSITE POWER SYSTEM

The offsite ac power system supplies normal operation and shutdown loads. There are no safety-related loads and the system is not required for safe shutdown on emergency cooling. Furthermore, the offsite power system can provide shutdown recirculation cooling without reliance on either the onsite power system or the emergency cooling system.

8.2.1 Description (Figures 8.2-0, 8.2-1, 8.2-2 and 8.2-3)

The Hanford 230 kV Loop supplies offsite power to the N Reactor 151-N sub-station. A single 230 kV line connects the 151-N substation to the Hanford 230 kV Loop at switching station A-23. The Hanford 230 kV Loop receives power from the BPA Midway substation or from Ashe Line. See Figures 8.2-1 and 8.2-2. Hanford 230 kV Loop line No. 1 is supplied from BPA Midway substation Bus No. 3, Hanford 230 kV Loop line No. 2 is supplied from BPA Midway substation Bus No. 1, and Hanford 230 kV Loop maintenance line No. A3 connects between the 251-W substation and the tie between KE (A-7) and KW (A-9) substations. The BPA Midway substation consists of three 230 kV buses connected by normally-closed bus sectionalizing breakers. Each BPA Midway bus section is supplied by a line from Priest Rapids generating station. BPA Midway buses No. 1 and No. 3 are also supplied by lines which connect to the Grand Coulee 230 kV substation and to 230 kV substations in the lower Columbia River power system. The Grand Coulee and lower Columbia River substations are interconnected to the BPA 500 kV transmission system. The BPA 500 kV and 230 kV transmission systems interconnect to all of the Columbia and Snake River hydroelectric generating stations and northwest steam generating plants and are also interconnected to California and British Columbia power network systems. BPA 1977 Main Grid and BPA 1977 Walla Walla Grid drawings show power system connections. See Figures 8.2-4 and 8.2-5.

The Hanford 230 kV Loop lines are supported on wood pole H-frame structures with aluminum crossarms, and on steel towers. The routing of the two Hanford 230 kV lines from the BPA Midway substation to the Hanford Reservation boundary is parallel for 1 mile with a 600-foot separation. The lines are routed separately to form a loop throughout the Hanford Reservation boundary.

The electrical power transmission conductors are a mixture of 636 MCM/795 MCM/1272 MCM-ASCR (aluminum cable with steel core). The 1272 MCM line is from midway to the Ashe tap at While Bluffs. The lines are routed separately and form a loop within the Hanford Reservation boundary.

The Hanford 230 kV Loop is routed near each of four reactor area substations and one chemical processing area substation (200-E and 200-W) (Figure 8.2-2) and is connected to the BPA Ashe 230 kV line. There are currently five operating Hanford 230 kV Loop substations each containing three 230 kV breakers providing breaker and one-half protection for line sections. Two other substations, 151-N and Ashe, (Figure 8.2-1) contain only one single breaker for line termination. The Hanford 230 kV Loop is normally operated with all 230 kV circuit breakers closed forming a closed 230 kV Loop and with two single line taps, one for 151-N substation and one for Ashe line. The Hanford 230 Kv Loop also contains a maintenance line (A3) from the 251-W substation to between the KE and KW substations. The Hanford 230 kV Loop extends from BPA Midway Substation 230 kV Bus No. 1 to BPA Midway Substation 230 kV Bus No. 3. Each of the Hanford 230 kV Loop sub-stations has three breakers and a bypass switch that may be closed to provide closed loop operation while performing maintenance on any of the three sub-station breakers. Each line 230 kV circuit breaker is rated 3-phase, 60 hertz 230 kV, 1600 ampere continuous duty and 10,000 MVA momentary interrupting duty.

The Hanford 230 kV Loop in addition to supplying power to N Plant and other DOE substations, normally supplies power to Ashe and White Bluffs sub-stations. However, in the event that both Midway power sources for the Hanford 230 kV Loop are lost, power will flow into the Hanford 230 kV Loop from the Ashe line with sufficient power for normal operation or shutdown of N Reactor.

The flow of power from the Ashe line into the Hanford 230 kV Loop is supplied from 115 kV lines connected to the White Bluffs substation. One 115 kV line at the White Bluffs substation is connected to the BPA Franklin substation, and the other 115 kV line is connected to BPA Benton switch. See Figure 8.2-3.

The 230 kV transmission line lengths from N 230 kV tap switch A23 (see Figure 8.2-1) to the three 230 kV power sources into Hanford 230 kV Loop are as follows:

16 miles	-	N 230 kV Tap (switch A23) to Midway Bus No. 3,
19 miles	-	N 230 kV Tap (switch A23) to Midway Bus No. 1, going through 251-W substation,
13 miles	-	N 230 kV Tap (switch A23) to Ashe 230 kV line (switch A22),
16 miles	-	N 230 kV Tap (switch A23) to Ashe 230 kV line going through 251-W substation,
18 miles	-	Ashe line (switch A22) to Midway Bus No. 1, and
19.5 miles	-	Ashe line (switch A22) to White Bluffs Substation.

The Hanford 230 kV Loop line traverses over mostly level desert terrain. Range fires are controlled by onsite tanker fire trucks and other heavy equipment used for fire control.

The following is a listing of 500 kV transmission lines which cross over Hanford 230 kV lines and the K-N 13.8 kV line. The 500 kV lines are supported by double insulator strings on structures adjacent to 230 kV line crossings.

500 kV Line	230 kV Crossings	K-N 13.8 kV Crossing
Hanford-Lower-Monumental	2	1
Hanford-John Day	3	1
Hanford-Ostrander	3	1

The preferred offsite power source to N Reactor consists of the 13.8 kV A Bus supplied from the Hanford 230 kV Loop at substation A1 through a 30/40/50 MVA transformer, rated 230-13.8 kV and located within N Reactor security-fenced enclosure. See Figure 8.2-6.

An alternate preferred offsite power source to N Plant is supplied from the 100-KW Area 13.8 kV substation bus. Power is transmitted over K-N line by 3 miles of overhead transmission line consisting of double circuit three-phase aerial cable with 477 MCM-aluminum spaced aerial cable supported

on wood poles. The 13.8 kV power system at 151-KW substation is supplied from the Hanford 230 kV Loop at substation A7 through a 50 MVA transformer. The K-N 13.8 kV power source can be switched by bus tie breakers at N Reactor to either the 13.8 kV A Bus or the 13.8 kV B Bus. The K-N 13.8 kV line has a transmission capability of 15 MVA at 5 percent voltage regulation.

A second alternate offsite power source is available by manual operation of aerial line switches. See Figure 8.2-1. This power source is supplied from a 13.8 kV Bus at 151-D substation. The power from the 151-D substation is transmitted over 2 miles of overhead three-phase circuit of 477 MCM-ASCR open wire supported on wood poles (designated the D-N line). The D-N line has approximately half the capacity of K-N line. The line can be used by opening an aerial switch on the K-N line and closing an aerial switch into the D-N line at N terminal.

8.2.2 Inspection and Testing

The Hanford 230 kV Loop transmission lines are readily accessible for inspection and testing and are patrolled at a minimum frequency of once every 30 days. Service and testing of substation equipment is performed on a routine basis in accordance with plant and system maintenance procedures. Figure 8.2-1 shows switching stations that can be used to bypass a substation and maintain a closed Hanford 230 kV Loop. This permits testing and maintenance of equipment within the substation.

The maintenance, inspection and testing procedures are designed to assess the condition and deterioration of insulation and the condition of protective systems. Protective relays are tested and calibrated at 1-year intervals.

8.2.3 Safety Features

8.2.3.1 Lightning Protection and Bird Guards

The Hanford 230 kV Loop is protected from lightning strikes by a continuous overhead static conductor and continuous counterpoise grounding system. The

K-N 13.8 kV line is protected from lightning by grounding of the neutral conductor supporting the spaced aerial cable.

Bird guards, consisting of wire quills installed on 230 kV structures immediately over the suspension insulators, prevent large birds from resting in a position where bird excrement can cause an arc from the 230 kV conductor to the grounded portion of the 230 kV structure. Bird guards are installed on all structures between substations A9 to A4 and A1 and on other line sections where bird excrement has been observed on insulators and 230 kV structures.

8.2.3.2 BPA Midway Substation Relaying

Each BPA Midway bus section is protected by bus differential relaying with breaker failure backup protection. In the event of a bus fault on any BPA Midway bus section, the differential relaying system will trip all line breakers connected to the faulted bus and all connecting sectionalizing breakers. In the event a sectionalizing breaker fails to clear the fault, a breaker failure backup relay system will trip all line breakers and will trip the sectionalizing breaker in the adjacent bus to interrupt the fault current flow to the faulted bus. Sectionalizing and line breakers tripped by differential relaying clear the fault current in five cycles and breaker failure backup breakers clear the fault current in 13 cycles. Each BPA Midway bus is normally operated with all sectionalizing breakers closed forming one continuous bus section.

8.2.3.3 Offsite 230 kV and 13.8 kV System Relaying

Phase to phase faults on the Hanford 230 kV Loop are detected and a trip signal initiated by two independent relay systems. Each Hanford 230 kV Loop line section between terminal breakers is protected by a phase directional carrier current relay system. The system consists of carrier current transmitters and receivers at each line section terminal location and line wave traps and couplers in B phase 230 kV conductor at each terminal end. Each line section has a fixed carrier frequency band width so that frequencies of one line section do not interfere with carrier frequencies of any other Hanford 230 kV Loop line section.

A second independent phase current relay system consists of directional impedance distance relaying. This relay system compares the line voltage to the current flow into a line section. Since impedance is a function of line length, these relays can be set to a given line length. Each impedance relay has three separate movements that can be adjusted to an impedance value for the line length. Zone I impedance relay element is normally set to cover 80 percent of the line section length, and timing is set for instantaneous or one cycle relay operation. Zone II relay element is set for the total line section length plus 50 percent of the adjacent line section length, and timing is set to operate the line breakers for clearing a fault in 20 cycles. Zone III relay element is sensitive to phase to phase faults anywhere within the Hanford 230 kV Loop. The tripping time for Zone III is approximately 45 cycles. Zone II relay is a backup to Zone I relay, and Zone III relay is a backup to Zone II relay.

Line to ground faults are also detected and breaker trips initiated by two independent relay systems. The primary ground fault relay system consists of directional ground current relays that control carrier current transmitters, the same as for the phase to phase carrier relay system. Relay time is instantaneous (less than one cycle) with fault clearing time a maximum of five cycles. The secondary and backup ground fault relay system consists of a directional ground relay, with local breaker tripping and fault clearing time set for 20 cycles.

The relay system provided for Ashe line section involves a three terminal line section which extends from substations A4, A8 and Ashe. Phase directional carrier current relaying transmitted on B phase conductor, the same as in other sections of the Hanford 230 kV Loop, is provided. Ground directional carrier current relaying is also provided, the same as in other line sections of the Hanford 230 kV Loop. A second independent carrier current relay system is transmitted on A phase conductor. For phase faults this relay system is identified as permissive overreach transfer trip. A directional impedance distance relay set to reach beyond the zone of the three terminals operates a carrier signal that allows tripping when all three terminals indicate fault current in the proper current direction. For ground faults, an instantaneous directional ground overcurrent relay with

permissive overreach transfer trip operates a carrier signal for tripping. Phase fault and ground fault trip signals are transmitted over the same carrier frequency and phase conductor. A backup directional ground relay with time delay will trip each separate local terminal breaker if the ground current is not cleared by carrier transfer trip in five cycles.

All line section breakers in the Hanford 230 kV Loop are provided with a single automatic reclosure with synchronous check protective features. When a line fault occurs both line section breakers trip in five cycles and automatically reclose in 5 to 7seconds. These reclosing times apply to both phase or ground faults except that the BPA Midway feeder line breaker for the Hanford 230 kV Loop will delay reclosing for 1 minute if the line fault involved all three phases.

The K-N 13.8 kV intertie line is protected by pilot wire differential relaying with pilot wire tripping permissive only when fault current is flowing in the line. This system provides protection for phase or ground faults. Each terminal end of the line has phase and ground fault timed overcurrent relaying. One automatic reclosure is provided for the KW line breaker with hot line relay check. Automatic reclosure at the KW terminal is permitted only when the line has no voltage source from N Reactor. The 13.8 kV breaker for K-N line at Nterminal is manually closed.

8.2.4 Power Distribution System

8.2.4.1 13.8 kV Distribution System A Bus

The 13.8 kV distribution system (see Figure 8.2-9) consists of 15 kV metal enclosed switchgear, with air circuit breakers rated for 500,000 kVA interrupting duty. Two bus tie breakers allow power connections to the K-N 13.8kV line or power connections to the onsite 13.8 kV B Bus. The A Bus switchgear supplies power to two 3500 hp CRW river-pump motors, provides a feeder circuit to WPPSS for startup and standby of the steam generator plant and supplies power to the A Bus 13800/4160 volt transformer for N Reactor plant operation.

Protective relaying on the A Bus 13.8 kV power system includes:

1. Bus differential current
2. Time phase overcurrent
3. Directional time overcurrent
4. Directional time overpower
5. Time ground overcurrent

The 13,200 volt, 3500 horsepower CRW river-pump synchronous motors are protected by special types of relaying. Overloads up to 250 percent of motor full load current rating are annunciated. Overloads above 250 percent will cause an inverse time overcurrent relay to trip the motor circuit breaker. A phase to phase fault current greater than the motor starting current causes instantaneous overcurrent relays to trip the motor circuit breaker. All ground faults are detected by inverse time ground relays that initiate tripping of the motor circuit breaker. Current balance relays trip the motor circuit breaker when the unbalance between phase currents exceeds a fixed percentage. Out of step operation of the synchronous motor is detected by a power factor relay, which initiates a trip of the motor field circuit breaker. For high speed power transients, a power directional relay trips the field circuit breaker when the motor input power falls below a fixed percentage. Tripping of the motor field breaker causes the motor to operate as an induction motor until voltage is sufficient to resynchronize the motor. The motor is protected from overheating when operating in the induction mode by thermal overload relays that trip the motor circuit breaker. Differential relaying protects the motor from phase or ground faults.

The switchgear for 13.8 kV "A" Bus is located in 151-N building. The power for opening and closing the breakers is supplied from a 125 volt dc battery system. See Figure 8.2-6.

The normal operating load for 13.8 kV "A" Bus is 13,000 kVA with a maxium 23,000 kVA connected load.

The 13.8 kV power source from the 151-N substation transformer and the K-N line source transformer at the 151-KW substation are grounded through a

zig-zag transformer and a four ohm grounding resistor which limits the ground fault current to a maximum of 2000 amperes. This 2000-ampere maximum ground current is adequate to operate ground sensitive relays, thereby, providing fast clearing of all ground faults on the 13.8 kV system. 7

8.2.4.2 4160 Volt Distribution System A Bus

The 4160 volt A Bus distribution system (see Figures 8.2-8 and 8.2-10) is supplied from a 10,000/12,500 kVA, 13,800-4160 volt transformer with 5 kV metal enclosed switchgear for distribution of 4160 volt power. Switchgear for 4160 volt A Bus is located in one-half of 153-N building and is physically isolated by concrete walls from the B Bus switchgear (see Figure 8.2-7) in 153-N building. The 4160 volt switchgear breakers are closed and opened by 125 volt dc power supplied from a 125 volt dc lead-acid battery bank with a separate 125 volt dc rectifier for battery charging duty. Power supply to the battery charger is from the 480 volt A Bus power panel "A" (see Figure 8.2-15) in the 153-N building. 7 7

The 4160 volt feeder breakers are rated 1200 ampere continuous current duty and 150 MVA interrupting rating. Feeder breakers supply power to various 4160 volt motors and to 4160-480 volt transformers for building low voltage power distribution.

Protective relaying on the A Bus 4160 volt power system includes:

1. Bus differential current,
2. Time phase overcurrent,
3. Time ground overcurrent,
4. Current balance,
5. Instantaneous phase overcurrent, and
6. Motor differential current for 2000 HP HPI pumps.

7

The following is a listing of loads and transformers directly supplied by A Bus 4160 volt switchgear:

Building	Motor HP & Use	4160-480/277 Volt Transformer Capacity
105-N		2-1500 KVA
109-N		2-1500/1725 KVA
163/183-N		1000/1150 KVA
151-N		300 KVA
182-N		2000/2300 KVA
184-N		750 KVA
109-N	1-350 HP Decon Pump	
184-N	1-800 HP Boiler Feed Water Pump	
184-N	1-350 HP Boiler Forced Draft Fan	
182-N	3-2000 HP High Pressure Injection Pumps	
109-N	6-400 HP Reactor Coolant Shutdown Pumps	

One additional 800 hp boiler feedwater pump motor normally supplied from B Bus (see Section 8.3.1.3) may be supplied on an alternate basis from A Bus by removing the drawout 4160 volt breaker from B Bus and inserting the breaker in A Bus. Two of the reactor coolant pump motors and one forced draft fan motor shown above as normally being supplied by A Bus may be supplied on an alternate basis from B Bus by breaker positioning. These operations are used only for shutdown conditions.

The 4160 volt transformer secondary is wye-connected and grounded through a 2.4-ohm resistor which limits the ground fault current to 1000 amperes. The 1000 ampere ground fault current provides selective ground relaying for fast clearing of any 4160 volt system ground with minimum damage to equipment.

All 4160 volt motors are protected by three types of relaying. Overloads up to 250 percent current rating of the motor are annunciated. Overloads above 250 percent current will cause a motor trip based on inverse time current characteristics. A phase to phase short circuit current greater

than motor starting current will cause instantaneous overcurrent relays to trip the motor circuit breaker. All ground fault relays are inverse time and will trip the motor circuit breaker. Ground relays are set below the motor full load current rating since no selectivity is required with other relays. High pressure injection pumps are protected by differential relaying.

8.2.4.3 480 Volt Distribution System A Bus (Figures 8.2-11, 8.2-12, 8.2-13, 8.2-14, 8.2-15, 8.2-16)

Power is supplied at 480/277 volts to buildings by 4160-480/277 volt transformers (Transformer ratings and location as shown in Section 8.2.4.2.) Transformers are three-phase, wye-connected with a solidly grounded neutral. All 480 volt switchgear is provided with a main secondary incoming breaker and 480 volt feeder breakers for supply to motor control centers and in some cases to large 480 volt motors supplied directly from the switchgear feeder breaker. A 480 volt bus tie breaker is supplied for connecting the 480 volt A Bus to the 480 volt B Bus. Bus tie breakers are interlocked so that the bus tie breaker cannot be closed unless one incoming line 480 volt breaker, either A Bus or B Bus, is open. Bus tie breakers are closed only for shutdown maintenance. All nonsafety-related but necessary for normal shutdown power loads are redundant with equal units on the 480 volt A Bus and 480 volt B Bus. This ensures the ability to perform a safe shutdown with either A Bus or B Bus in the event of the loss of either the offsite or the onsite primary power source.

Most 480 volt breakers are protected by static overcurrent trip devices in each phase with long-time, short-time or instantaneous current characteristics. The 105-N building incoming line breakers are protected by inverse time induction overcurrent relays operating on 480 volt current transformers as well as static trip type over current devices. 7

Some nonsafety-related, but necessary for normal shutdown, 480 volt loads are supplied from a 480 volt distribution panel with a power source from both 480 volt A Bus and 480 volt B Bus. Power is supplied to these loads from a panel through an automatic transfer switch so that only one power source is the normal source and upon loss of the normal power source voltage,

the transfer switch will automatically switch to the alternate power source. Transfer switching time is approximately three cycles.

Motor, lighting and instrument supply transformers are supplied from 480 volt motor control centers.

8.2.4.4 120 Volt ac Systems A Bus (Figure 8.2-11A)
A nonsafety-related instrument power source is supplied from 112.5 kVA transformers rated 4160-120/208 wye. Two identical systems, one for A Bus and one for B Bus, are supplied to 120/208 three-phase distribution panels. All of N Plant nonsafety-related instrument power requirements are supplied from this dual 112.5 kVA power source.

Noninterruptable 120 Vac is supplied to the ball drop system timers and the power calculator panels through DC to AC inverters in the event that normal 120 Vac is lost.

One instrument power system supplies 120 volt ac power to the 184-N Plant service boiler ignition and control system. See Figure 8.2-15. This system is supplied by a motor generator set in the 184-N building and is rated at 6.25 kVA, 120 volt, 60 cycle single phase. The M-G drive motor operates at synchronous speed 1800 rpm and is supplied from an automatic transfer switch with input power sources to the transfer switch from 184-N 480 volt A Bus and from 184-N 480 volt B Bus. The M-G set has sufficient inertia so that the transfer time for the motor power source from B Bus to A Bus will not affect the generator 120 volt ac power output.

A second nonsafety-related instrument power system supplies 120 volt, 60 hertz power to instrumentation for control of reactor heat dissipation. See Figure 8.2-11A. Power for this system is supplied from the 105-N 120/208 volt distribution system and is transmitted through resonant frequency-type voltage regulators rated at ± 1 percent for load variations of 0 to 100 percent and line changes of ± 10 percent. Two identical voltage regulator systems each rated 50 kVA, one supplied from A Bus and one supplied from B Bus, supply power to an automatic transfer switch. The output from the transfer switch is distributed from a circuit breaker panel to nonsafety-

related instrument power loads in 105-N and 109-N buildings. The automatic transfer switch transfers from the normal A Bus to the alternate B Bus upon a percentage drop in voltage on the A Bus power source.

8.2.5 Analysis

8.2.5.1 Stability Considerations

8.2.5.1.1 Fault Studies

BPA stability study 76A726 (Figures 8.2-17 through 20) performed in July 1976, provides swing curves for power supply lines into Midway substation for fault conditions occurring on BPA Midway substation buses. Fault condition studies are as follows: 7

1. Swing 110 - Three-phase fault simulated on the BPA Midway No. 2 bus section. At five cycles on 60 hertz base, the bus differential relay trips the sectionalizing breaker between bus No. 2 and bus No. 1, trips the sectionalizing breaker between bus No. 2 and bus No. 3 and trips all line breakers on bus section No. 2. All BPA Midway connected lines are stable. 7

2. Swing 115 - Three-phase fault simulated on BPA Midway bus No. 1 with an assumed failure of the sectionalizing breaker between bus No. 1 and bus No. 2 to open. At five cycles on 60 hertz base, the bus differential relay trips all breakers on bus No. 1. At 13 cycles, the breaker failure relay trips the sectionalizing breaker between bus No. 2 and bus No. 3 and also trips all line breakers on bus No. 2. At 20 cycles, all Priest Rapids generators are tripped. All remaining BPA Midway connected lines on bus No. 3 are stable. 7

3. Swing 116 - Three-phase fault simulated on BPA Midway No. 3 bus section with an assumed failure of the sectionalizing breaker between bus No. 2 and bus No. 3 to open. At five cycles, the bus differential relay trips all breakers on bus No. 3. At 13 cycles, the breaker failure relay trips the sectionalizing breaker between bus No. 1 and bus No. 2 and trips all line breakers on bus No. 2. At 20 cycles, all Priest

Rapids generators are tripped. All remaining BPA Midway connected lines on bus No. 1 are stable.

4. Swing 117 - Three-phase fault simulated on the BPA Midway-Vantage 230 kV line with an assumed failure of the line breaker on BPA Midway-Vantage line to open. At 13 cycles, both sectionalizing breakers and all line breakers on bus No. 2 are tripped. At 20 cycles, all Priest Rapids generators are tripped. All remaining BPA Midway connected lines on bus No. 1 and bus No. 3 are stable.

The swing curves shown for the four fault conditions provide line relative angle in degrees vs. time and bus voltage vs. time.

Stability is indicated by the curves which show the diminishing magnitude of the abscissa function as time progresses. Priest Rapids Station generator swing curves are not shown for fault conditions where breaker failure relaying trips BPA Midway line breakers in 13 cycles since Priest Rapids station relaying automatically trips all generator line breakers due to the severity of the fault. These fault studies were performed by BPA using digital computer transient analysis with the total BPA 2000 Bus system simulated for the fault condition. None of these simulated faults would create a power system disturbance severe enough to cause the actuation of the 1400 megawatt braking resistor to be applied at BPA Chief Joseph substation site. See Section 8.2.5.2.

The Swing studies of simulated faults associated with Midway feeders and line breakers showed that after the fault is cleared there would exist one stable Midway 230 kV bus for supply of the Hanford 230 kV Loop.

8.2.5.1.2 Natural Causes

Since construction in 1943 there has never been a 230 kV system failure within the Hanford 230 kV Loop from natural causes such as lightning, wind or ice. A range fire on one occasion caused a single Hanford 230 kV line section fault. This type of fault is normally cleared by 230 kV system relaying with the loss of only one 230 kV line section from the loop.

8.2.5.2 Availability Considerations

The transmission features which minimize the probability of power failure due to faults on the Hanford 230 kV Loop, the BPA grid, or interconnections are: 7

- The Hanford 230 kV Loop is interconnected to the BPA grid system through Midway Buses 1 and 3 and the Ashe line; therefore, failure of two of these power sources would not interrupt offsite power to N Reactor.

- The multiple paths in the 230 kV Loop minimizes the effect of failure of individual lines. 7

- All Hanford 230 kV Loop lines have continuous overhead ground wires which minimize the probability of faults being initiated by lightning strikes, and bird guards to keep birds off the insulators. 7

- Use of breaker failure backup relaying on the BPA Midway bus, on all lines attached to BPA Midway bus and on all Hanford loop line sections minimizes the probability of fault clearing spreading to adjacent zones.

- Use of automatic reclosing assures restoring power lines to service in minimum time for momentary disturbances.

- The BPA interconnections to the Hanford 230 kV Loop, Midway Buses 1 and 3 and Ashe line, are supplied by multiple feeders from varied power sources with line breakers to prevent propagation of line faults to the buses.

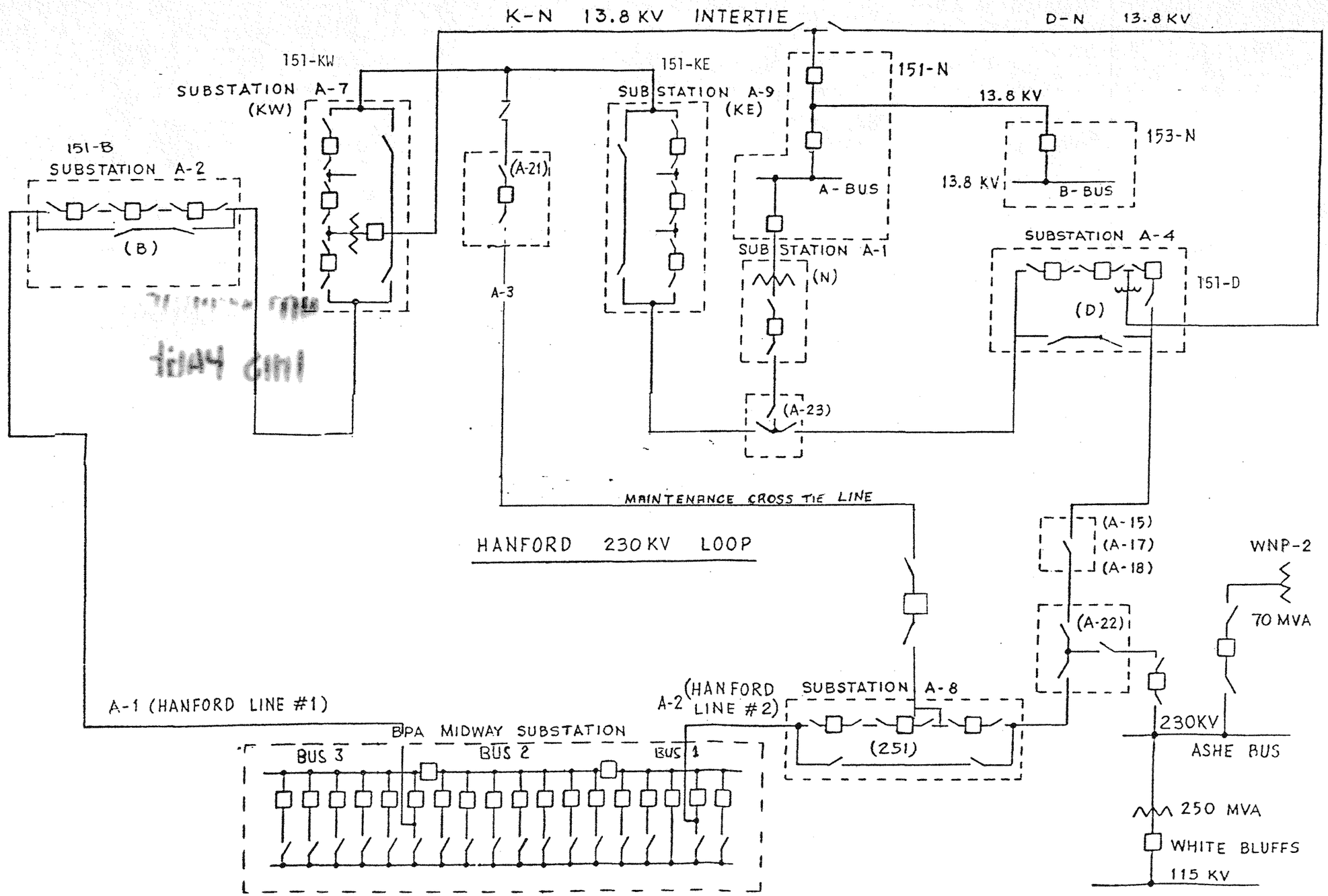
The BPA Midway buses 1 and 3 and Ashe line feeders are interconnected to the BPA Northwest Power Pool Main Grid. This grid is inherently stable because of the following:

- The recent BPA installation of the 1400 megawatt dynamic braking resistor at Chief Joseph substation is applied by remote control from a BPA Control Center. The electric brake is used to slow down the power system change in hertz due to clearing a severe three-phase fault. The

electric brake will prevent the breakup of the Pacific Northwest power system allowing time for high voltage breaker tripping to remove excess generating capacity from the power system and a restoration of normal 60 hertz operation.

- BPA power system security schemes are designed to prevent power system disturbances from developing into regional blackouts. Underfrequency load shedding relays operate automatically to drop selected system loads when generation and load balance is severely upset. During heavy import power schedules, steps are taken to increase the capacitor compensation on the Pacific Intertie lines, the spinning reserve is increased to match the ac Intertie import level and ac Intertie out-of-step relays are adjusted to increase the stability limit at which the relays will trip.

- BPA control centers are equipped with dynamic system map boards, computer displays and extensive supervisory control and communication systems. These information systems provide the operating personnel with information for decisions to prevent the blackout of the power systems under the supervisor's control. The power grids in the United States which have experienced blackouts have been predominately supplied by steam-driven generators. The majority of the generation in the BPA system is supplied by hydro-generators which do not require under-frequency tripping as normally required by steam-driven generators. Therefore, a power system supplied by hydro-generation can sustain a more severe disturbance. Unlike steam generating plants, hydro-generators can be rapidly put back in service when generators are tripped off the line by a system disturbance (Ref. 1 and 2).

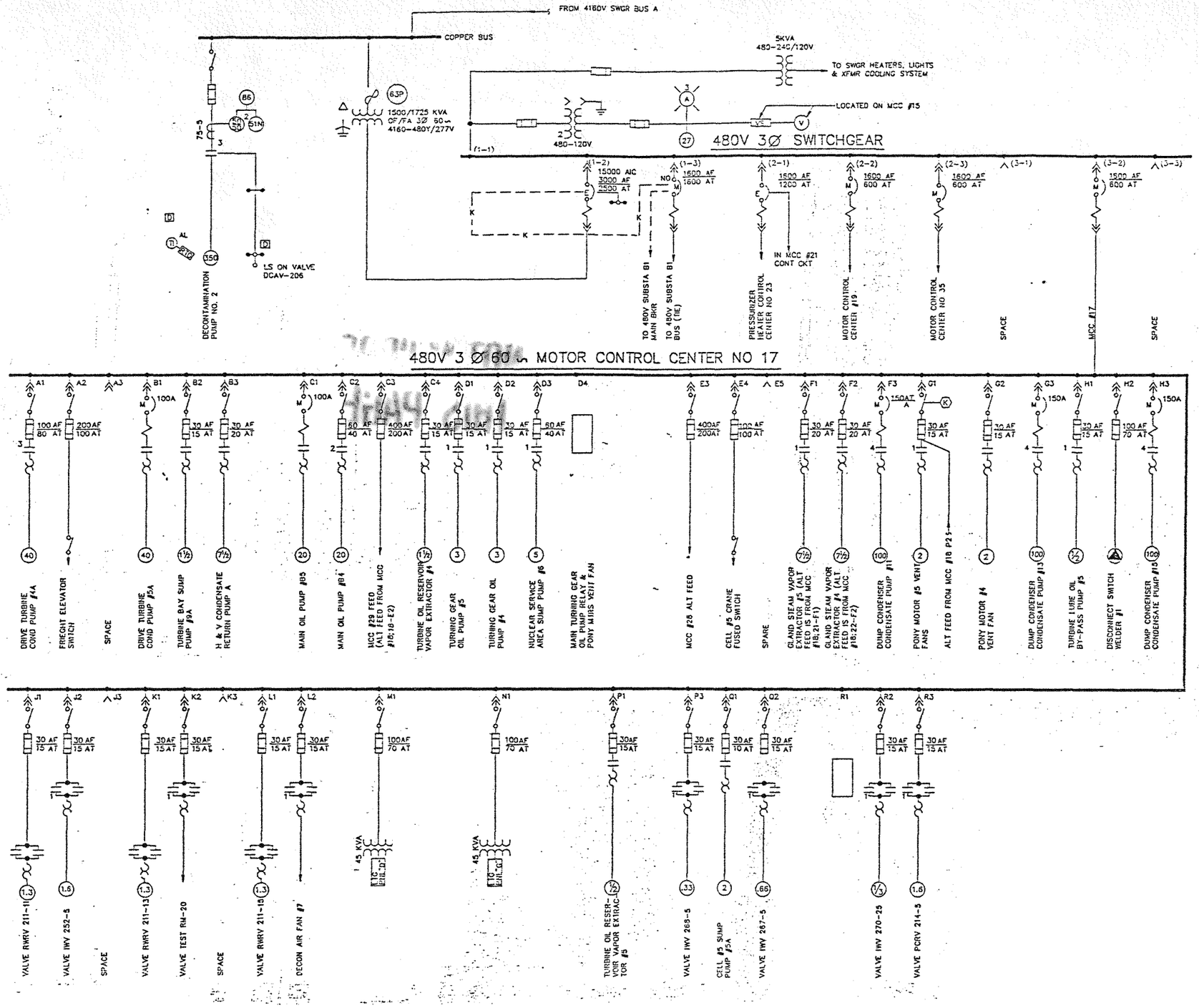


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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Hanford 230 KV Loop Transmission System Figure 8.2-1

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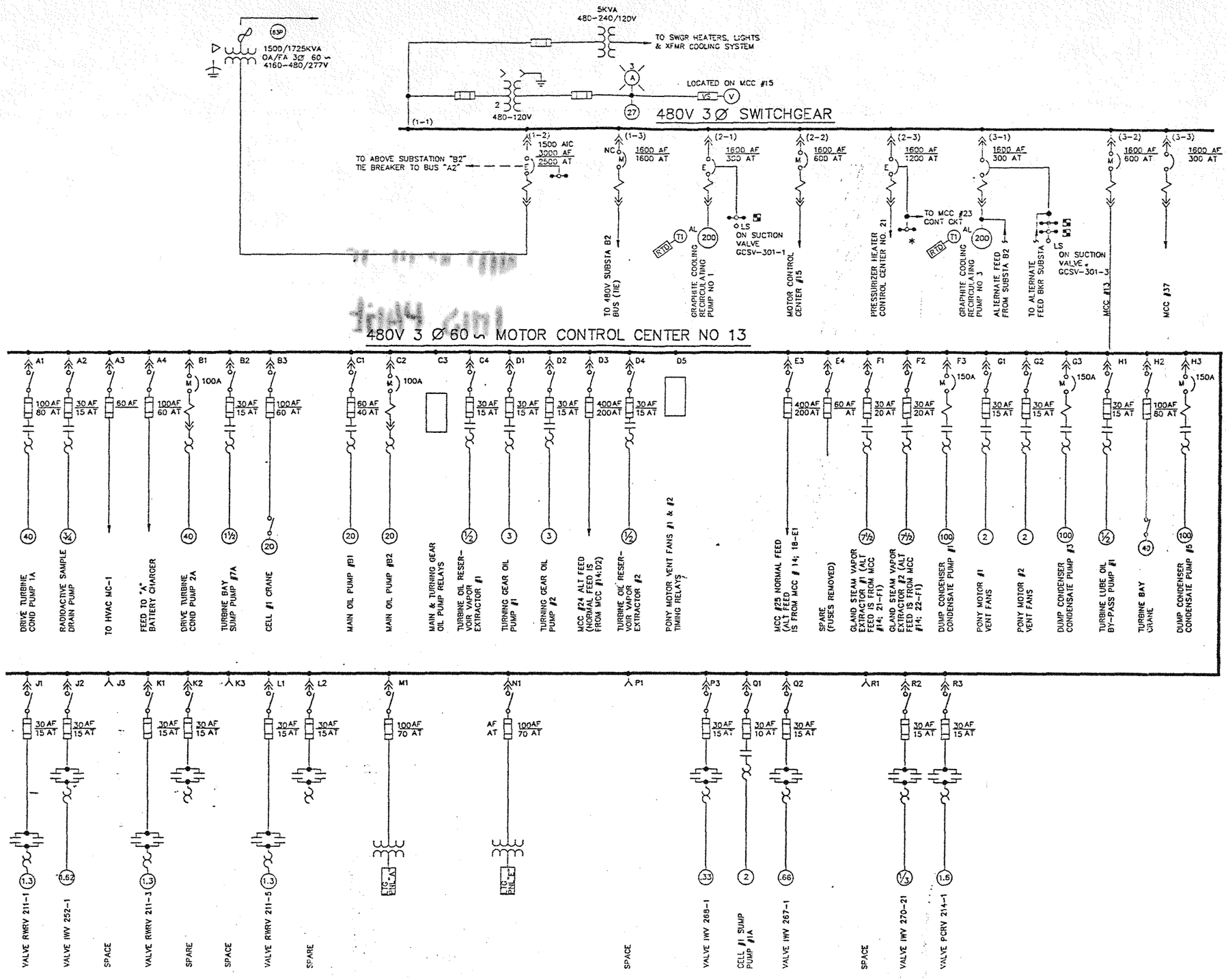


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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	480 Volt Motor Control Centers
	Figure 8.2-12

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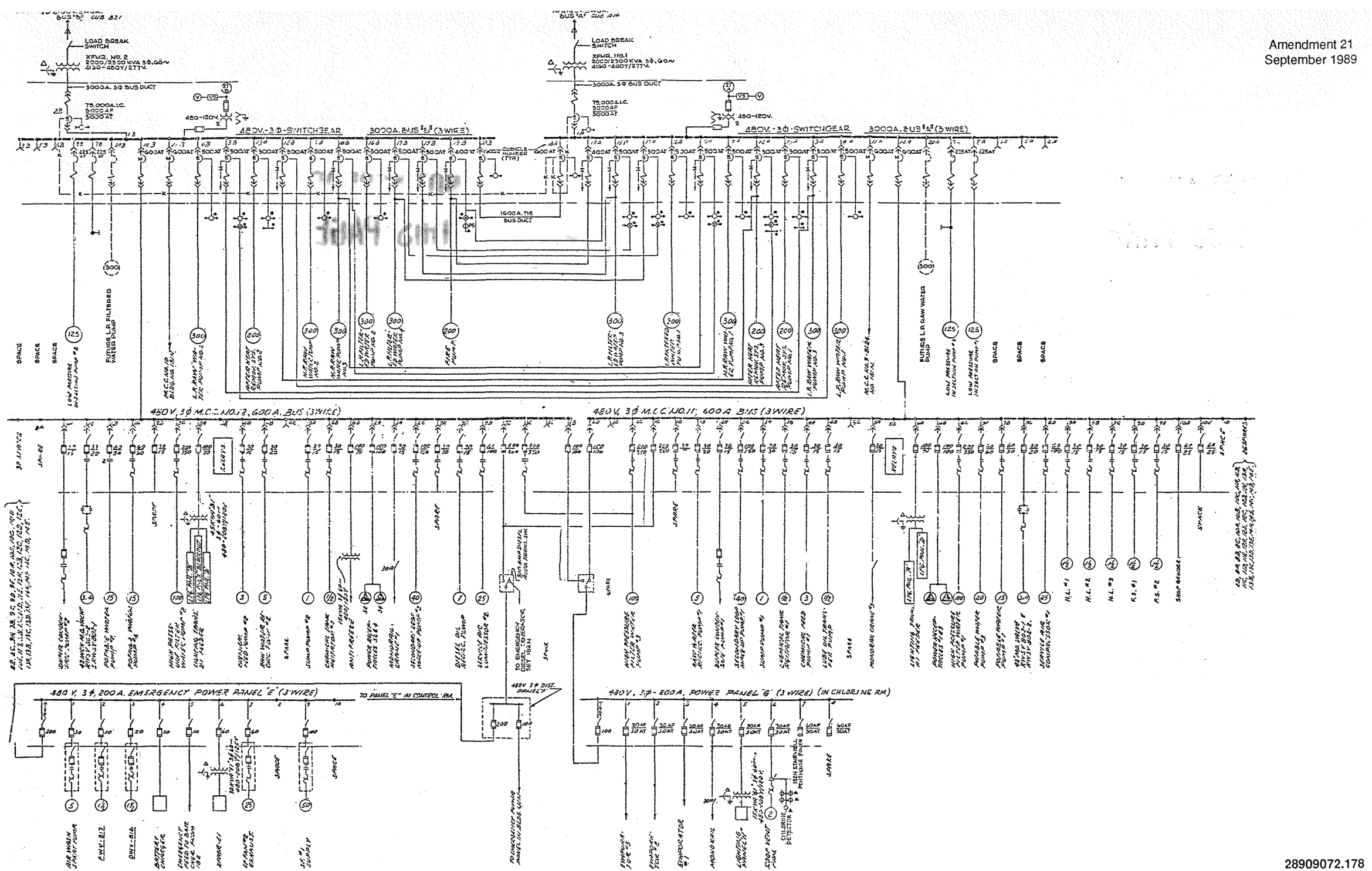


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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	480 Volt Motor Control Centers
	Figure 8.2-12a

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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	480 Volt Motor Control Centers
	Figure 8.2-14

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8.3 ONSITE POWER SYSTEM

8.3.1 AC Power System

The onsite ac power system supplies normal operation and shutdown loads. There are no safety-related loads, and the system is not required for safe shutdown on emergency cooling. Furthermore, the onsite power system can provide shutdown recirculation cooling without reliance on either the offsite power system or the emergency cooling system.

8.3.1.1 Description (Figures 8.3-0 and 8.3-1)

The onsite ac power system (B Bus) consists of a steam turbine-driven generator (TG) supplying power to a 13.8 kV bus separated from the offsite ac power system (A Bus) by double bus tie breakers. The TG has a 5 stage turbine rated at 15,625 kW, at 3600 rpm, based on a saturated steam supply of approximately 135 psig. The turbine is directly coupled to a two pole, 60 hertz, 13.8 kV hydrogen-cooled generator rated at 22 MVA at 0.85 power factor. 7 21 21 7

Steam is supplied to the turbine from plant service oil fired boilers, one rated 575,000 pound per hour and two each rated 200,000 pound per hour or by steam produced in N Reactor secondary system. See Chapter 10. When the TG is operated in parallel with an offsite power source, the maximum load to be supplied by the offsite power source is limited to 7 MWe. This limitation is based on boiler load carrying capabilities during boiler transient load increase (Ref. 1). 7

The generator dc excitation power is supplied by a 90 kW, 250 volt dc direct connected exciter. An auxiliary exciter of the same dc rating is powered by a 440 volt, three-phase 150 hp, 1750 rpm induction motor. Either one of these exciters can be switched for supply of dc power to the generator. An amplidyne voltage regulator system controls the voltage of the dc exciter generator which, in turn, controls the ac voltage output of the 22 MVA generator.

Turbine speed control is accomplished by conventional steam governor control. The TG is protected from overspeed by governor or manual actuation of quick-closing steam supply stop valve.

The 13.8 kV generator is wye connected and grounded by an 8 ohm resistor which limits the maximum ground fault current within the generator to 1000 amperes.

Protective relaying on the 22 MVA generator includes the following:

1. Differential current relaying,
2. Loss of field relaying,
3. Time overcurrent with voltage restraint relaying, and
4. Time negative sequence current relaying.

8.3.1.2 13.8 kV Volt Distribution System B Bus

Power from the generator is supplied to 15 kV metal enclosed switchgear with air circuit breakers rated for 500,000 kVA interrupting duty. See Figure 8.3-1. Bus tie breakers permit power connections to the offsite K-N 13.8 kV line or power connections to the offsite 13.8 kV A Bus. Bus tie breakers cannot be closed unless the two electrical systems are synchronized through the operation of automatic synchronizing equipment (Ref. 1 and 2).

B Bus supplies power to two 13,200 volt 3500 hp CRW river pump motors and provides power to the B Bus 13,800-4160 volt transformer for N Plant Operations.

Protective relaying on the 13.8 kV switchgear feeder breakers includes:

1. Bus differential current,
2. Time phase overcurrent,
3. Time ground overcurrent, and
4. Instantaneous phase overcurrent.

In addition to the above relaying, the two bus tie breakers have two types of directional power relaying for protection of the generator. One is a

very inverse time power directional relay set to limit the maximum amount of power which can be exported from the generator. These relays are set at approximately 30 percent of the turbine rating. A very high speed power directional type of relaying is set to trip the B Bus tie breaker when the generator is operating out of synchronism following a power system fault. This relay is instantaneous and is set to operate at approximately 70 percent power rating of the drive turbine.

Special types of protective relaying required for the 13,200 volt, 3500 hp CRW motor is the same as described for A Bus motors in Sections 8.2.4.1.

The switchgear for B Bus is located in 153-N building. Power for opening and closing the breakers is supplied from a 125 volt dc battery system with rectifier for charging the batteries. The battery bank and rectifier is located in 153-N building and is isolated by a concrete wall from the switchgear and batteries in 153-N building for A Bus.

The normal operating load for 13.8 kV B Bus is approximately 10,000 kVA with a 16,000 kVA connected load.

8.3.1.3 4160 Volt Distribution System B Bus

The 4160 volt B Bus distribution system (Figure 8.3-2) is supplied from a 10,000/12,500 kVA, 13,800-4160 volt transformer with 5 kV metal enclosed switchgear for distribution of 4160 volt power. Switchgear for 4160 volt B Bus is located in one-half of 153-N building and is physically isolated by concrete walls from the A Bus switchgear in 153-N building. See Figure 8.2-7. The 4160 volt switchgear breakers are closed and opened by 125 volt dc power supplied from a 125 volt dc battery bank with a separate 125 dc rectifier for battery charging duty. Power supply to the battery charger is from the 480 volt B Bus power panel B in the 184-N building (see Figure 8.3-5).

The 4160 volt feeder breakers are rated 1200 ampere continuous current duty and 150 MVA interrupting rating. Feeder breakers supply power to various 4160 volt motors and to 4160-480 volt transformers for building low voltage power distribution.

Protective relaying on the B Bus 4160 volt power system includes:

1. Bus differential current,
2. Time phase overcurrent,
3. Time ground overcurrent,
4. Current balance,
5. Instantaneous phase overcurrent, and
6. Motor differential current for 2000 hp HPI pumps.

The following is a listing of loads and transformers directly supplied by B Bus 4160 volt switchgear:

Building	Motor hp and Use	4160-480/277 Volt Transformer Capacity
105-N		2-1500 kVA
109-N		2-1500/1725 kVA
163/183-N		1000/1150 kVA
NE Substation		300 kVA
182-N		2000/2300 kVA
184-N		750 kVA
109-N	1-350 hp Decontamination Pump	
184-N	2-800 hp Boiler Feed Water Pumps	
184-N	3-350 hp Boiler Forced Draft Fans	
182-N	2-2000 hp High Pressure Injection Pumps	

In addition to the above, two 400 hp reactor coolant pump motors, one 2000 hp high pressure injection pump motor and one 350 hp forced draft fan motor, normally supplied by A Bus (see Section 8.2.4.2), may be supplied on an alternate basis from B Bus by removing the 4160 volt breaker from A Bus and inserting the breaker in B Bus. Furthermore, one of the boiler feed pump motors shown above as normally being supplied by B Bus may be supplied on an alternate basis from A Bus by breaker repositioning. These operations are used during reactor shutdown. See Figure 8.3-2.

The 4160 volt transformer secondary is wye connected and grounded through a 2.4 ohm resistor which limits the ground fault current to 1000 amperes. The

1000 ampere ground fault current provides selective ground relaying for fast clearing of any 4160 volt system ground with minimum damage to equipment.

All 4160 volt motors are protected by three types of relaying. Overloads above 250 percent current will cause a motor trip based on inverse time current characteristics. Overloads up to 250 percent current rating of the motor are annunciated. A phase-to-phase short circuit current greater than motor starting current will cause instantaneous overcurrent relays to trip the motor circuit breaker. All ground fault relays are inverse time and will trip the motor circuit breaker. Ground relays are set below the motor full load current rating since no selectivity is required with other relays. High pressure injection pump motors are protected by differential relaying. 7

Bus tie breakers are not provided; therefore, the 4160 volt A and B Buses are independent both physically and electrically.

8.3.1.4 480 Volt Distribution System B Bus (Figures 8.3-3, 8.3-4, 8.2-11A, 8.2-11B, 8.2-13, 8.2-14 and 8.2-16) 7

Power is supplied at 480 volts to buildings by 4160-480/277 volt transformers with transformer ratings and location as shown in Section 8.3.1.3. Transformers are three-phase, wye connected with a solidly grounded neutral. Four hundred eighty volt switchgear is provided with a main secondary incoming breaker and 480 volt feeder breakers for supply to motor control centers and, in some cases, to large 480 volt motors supplied directly from the switchgear feeder breaker.

A 480 volt bus tie breaker is supplied for connecting the 480 A Bus to the 480 B Bus. Bus tie breakers are interlocked so that the bus tie breaker cannot be closed unless one incoming line 480 volt breaker, either A Bus or B Bus, is open. Bus tie breakers are closed only for shutdown maintenance. All critical power loads are redundant with equal units on 480 volt A Bus and equal units on 480 volt B Bus. This ensures the ability to perform a safe shutdown with either A Bus or B Bus in the event of the loss of either the offsite or the onsite primary power source.

Most 480 volt breakers are protected by series overcurrent relays in each phase with both short time and long time current characteristics. The 105-N incoming line breakers are protected by inverse time induction overcurrent relays operating from 480 volt current transformers.

Some nonsafety-related but necessary for shutdown 480 volt loads are supplied from a 480 volt distribution panel with a power source from both 480 volt A Bus and 480 volt B Bus. Power is supplied to these loads from a panel through an automatic transfer switch so that only one power source is the normal source; and upon loss of the normal power source voltage, the transfer switch will automatically switch to the alternate power source. Transfer switching time is approximately three cycles.

Motors, building lighting and miscellaneous control transformers are supplied from 480 volt motor control centers.

8.3.1.5 120 Volt ac Systems B Bus (Figure 8.2-11)
A nonsafety-related instrument power source is supplied from a 112.5 kVA transformer rated 4160 - 120/208 wye. Power from the transformer is supplied to a 120/208 three-phase distribution panel. N Plant nonsafety-related B Bus instrument power requirements are supplied from this 112.5 kVA power source.

N Plant has two power systems that supply all nonsafety-related, but necessary for shutdown, 120 volt ac power instruments. One instrument power system supplies 120 volt ac power to the 184-N plant services boiler ignition and control system. This system is supplied by a motor generator (MG) set in the 184-N building and is rated at 6.25 kVA, 120 volt, 60 hertz single phase. The MG drive motor operates at synchronous speed of 1800 rpm and is supplied from an automatic transfer switch with power sources to the transfer switch from 480 volt A Bus and 480 volt B Bus from the 184-N building switchgear. The MG set has sufficient inertia so that the transfer time for the motor power source from A Bus to B Bus will not affect the generator 120 volt ac power output. See Figures 8.2-15 and 8.3-4 boiler control MG power.

A second nonsafety-related instrument power system supplies 120 volt, 60 hertz power to instrumentation for control of reactor heat dissipation. See Figure 8.2-11A. Power for this system is supplied from the 105-N 120/208 volt distribution system and is transmitted through a resonant frequency-type voltage regulator rated at ± 1 percent for load variations of 0 to 100 percent and line voltage changes of ± 10 percent. Two identical voltage regulator systems each rated 50 kVA, one supplied from A Bus and one supplied from B Bus, supply power to an automatic transfer switch (5K65). The output from the transfer switch is distributed from a circuit breaker panel to nonsafety-related instrument power loads in 105-N and 109-N buildings. The automatic transfer switch transfers from the normal B Bus to the alternate A Bus upon a percentage drop in voltage on the B Bus power source (Ref. 3). 7 7

A safety-related instrument power system supplies 120 volt, 60 hertz power to ECCS instrumentation for reactor power level indicators and recorders and to instruments in the ECCS system requiring an ac power source. This instrument power source is supplied by redundant inverters each operating from electrically independent 125 volt dc battery power sources in 105-N. The power output of each inverter is rated for 1.5 kVA at 0.65 power factor with sola-type voltage regulators in the inverter ac output circuit. Each inverter has an automatic transfer switch in the output ac circuit which will transfer to an alternate 120 volt, 60 hertz ac source in event of failure of the inverter ac output. One inverter is supplied from the 125 volt dc A battery, and the second inverter is supplied from the 125 volt dc B battery in the 105-N building. This instrument power source is used to supply safety-related instrument systems which must function during, and in the event of, the failure of both the onsite ac power source and the offsite ac power source. 7 7

8.3.1.6 Emergency HVAC Standby Power System Control Room Habitability System

The standby AC power source for emergency HVAC consists of a diesel generator set and attendent air starting, fuel supply system, and automatic control circuitry. The diesel generator set supplies power to those electrical loads that are required to ensure functioning of the emergency control room HVAC habitability system as well as the cable spreading/instrument room

(Room 6). Table 8.3-3 identifies the equipment and associated generator loads. The diesel generator set is housed in basement of the 182-N building.

8.3.1.6.1 Diesel Description, Generator Ratings and Capability

The generator is rated at 187 KVA at 0.8 power factor, 150 KW, continuous output. This is sufficient to carry the total automatically connected loads.

The emergency diesel generator unit automatic start signal is initiated after a delay of 5 seconds has elapsed following a power failure. This precaution is to override momentary power interruptions. The diesel generator is capable of operating at full speed, no load, for 8 hours and then accepting the design load in the required sequence stated in Table 8.3-3.

The size of the diesel generator satisfies the requirements of NRC Regulatory Guide 1.9 and conforms to the following criteria:

a. The continuous 8760 hour nameplate rating exceeds the maximum load requirement indicated in Table 8.3-X (later).
b. The diesel generator is capable of starting, accelerating and supplying its loads in without exceeding a 5% frequency drop. The unit is capable of recovering to 98 percent of normal frequency in less than 40 percent nominal load sequence time interval.
c. The diesel is capable of starting, accelerating and supplying its load in the proper sequence without exceeding 20 percent voltage drop at its terminals. The unit is capable of recovery to 90 percent of normal voltage in less than 40 percent of each load sequence time interval.
d. The diesel generator is capable of starting, accelerating, and running its largest motor at any time after the automatic load sequencing is completed, assuming that the motor had failed to start immediately.
e. The diesel generator is capable of reaching full speed and voltage within 10 seconds after receiving a signal to start, and can be fully loaded within 30 seconds following the start signal.
f. The speed of the unit does not exceed 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal speed, whichever is lower.

Engine Fuel System

An independent fuel oil day tank, having a minimum capacity of 4 hour full load operation, is provided. As a minimum, the tank is provided with a fuel transfer pump, a level control switch, a fuel level gauge, a safety relief valve, and a rigid mounting base. The fuel transfer pump is a positive displacement pump driven by an alternating current electric motor. The motor is in accordance with the Auxiliary Motor section of the specification for the diesel generator unit. The positive displacement pump is capable of lifting fuel 20 feet. The tank level control switches provides automatic high and low level float switches for control of the fuel transfer pump. There are separate alarm contacts for indicating extreme low fuel oil level. All the day tank equipment is mounted on a rigid base and has been installed on the floor adjacent to the diesel-generator unit. The piping between the day tank components and between the oil day tank and engine is in accordance with ANSI B31.1, NFGPA 30 and NFPA 37.

Engine Cooling System

The diesel engine is water-cooled, and cooling system for each engine consists of direct engine-driven jacket water circulating pumps, water temperature regulating valves, unit mounted radiator with blower fan, expansion tank, radiator air discharge duct adapter, and necessary pipe fittings and lagging. The engine cooling water system is completely self-contained and closed-loop. Only the engine jacket water is used for cooling the various engine components.

Engine Lubrication

The engine lubrication system is integral with the engine and is positive in action. It includes direct engine-driven positive displacement lubricating oil pump, lubricating oil filters, all piping, valves.

8.3.2 DC Power System

Safety-related dc systems as used in Section 8.3.2 includes those electrical systems supplying power necessary for Class IE engineered safety feature functions that provide for the capability of shutting the reactor down and supplying it with ECCS. These safety-related power systems may also supply

power to other protection systems and control circuits not required for safe shutdown with ECCS cooling.

8.3.2.1 Description

The dc power system comprises the nonsafety-related and the safety-related dc distribution systems.

The safety-related 125 Vdc distribution systems provide reliable sources of continuous power necessary for Class IE engineered safety feature functions and other protection systems and control circuits. A failure of one system will not result in the loss of the other system.

The nonsafety-related dc distribution systems provide a reliable source of continuous power for nonsafety-related control, dc to ac inverters and emergency lighting.

Safety-Related 125 Vdc Distribution System

The safety-related 125 Vdc distribution system comprises the station batteries, chargers, the 125 Vdc switchgear, distribution panels, cable and raceway that interconnects the power source and the loads.

The safety-related, 125 Vdc distribution system in the reactor building (105-N) is divided into two power channels. A third battery is provided to replace either, should the need arise (substitution of the third battery for either power channel primarily for maintenance purposes is procedurally controlled).

The load groups and their corresponding power channels in the 105-N building are designated as "A" and "B". The third load group and corresponding power channel system is designated as "C". See Table 8.3-1.

The safety-related, 125 Vdc switchgear for each power channel in the 105-N building is a metal-clad, enclosed indoor design and has two-pole, metal frame, air circuit breakers. The 125 Vdc bus is arranged in a two-wire, ungrounded configuration. Continuous bus and breaker ratings are based on the maximum 1-minute demand for any mode of operation. Breaker momentary

and interrupting ratings are based on a bolted fault at the switchgear bus and fault current available from both the battery and charger. Direct acting overcurrent trip devices provide fault protection.

The safety-related distribution panels are standard metal enclosed panel boards with magnetic or thermal trip two-pole circuit breakers protecting the branch circuits. Circuit breaker rating is based on continuous ratings of connected load and fault current available at the panel mains from the 125V switchgear to which it is connected. See Figures 8.3 5, 8.3-6 and 8.3-7.

The safety-related 125 Vdc batteries, switchgear and distribution panels are located in the 105-N reactor building. The 125 Vdc switchgear is located in a switchgear room adjacent to the battery room.

Nonsafety Related DC Distribution Systems

Separate and independent nonsafety-related dc distribution systems, each with its own battery, battery charger and distribution busses serve the support buildings and reactor building. When a loss of onsite power and offsite power occurs, all batteries are capable of carrying the normal load requirements for the time periods, as shown in Table 8.3-2, without dropping below 1.75 volts per cell in the case of lead-acid batteries, or 1.14 volts per cell in the case of nickel-cadmium batteries.

Each battery system is sized to have 25 or more percent excess capacity over that obtained from the duty cycle to account for loss of capacity during battery life. Each battery is tested every 2 years to prove it is capable of performing as specified. Each battery charger has the capacity to continuously carry the steady-state connected loads without assistance from its respective battery. The capacity of each charger is sufficient to restore the battery connected to it from fully discharged state within an 8-hour period, while maintaining normal power for any load condition. Each battery charger is designed to prevent it from becoming a load on the battery due to a power feedback during loss of ac power to the charger.

There are no direct connections or ties between the nonsafety-related systems and the safety-related systems.

Nonsafety-related batteries are maintained in a temperature range of 77 F to 90 F. Ventilation is provided to limit hydrogen buildup to below 2 percent of total volume of the room in which the battery is located.

Protective devices and sufficient indication are provided to isolate and detect failed equipment.

Each nonsafety-related 125 Vdc system is operated ungrounded and is provided with a ground detection scheme of the voltage dividing type that will alarm a ground on either the positive or negative side of the line. The ground detector has the capability of detecting a ground at a resistance well above that at which equipment would fail to function.

Nonsafety-Related Station Batteries

Nonsafety-related station batteries are maintained in the same manner as safety-related batteries as specified in Reference 1.

All batteries are installed on seismic Category I battery racks.

Safety-Related DC Bus for Disconnecting and Protective Devices

The 125 Vdc metal frame air circuit breakers and the 125 Vdc distribution panels interconnect the sources, buses and loads within the given power channels of the safety-related dc power system. All devices are manually operated.

Sufficient surveillance is provided locally and in the control room to indicate the status of each safety-related, 125 Vdc switchgear bus and loads connected to these buses so the operator can determine that the system is operating normally. Power for the surveillance system is normally supplied by nonsafety-related ac power, with backup power supplied from the safety-related dc systems.

Safety-Related DC Electric Circuit Protection System

The safety-related dc distribution system which includes the switchgear and distribution panels have direct acting protective devices that are coordinated so a single fault will not trip more than one protective device.

Safety-Related Battery Chargers

The safety-related battery chargers are supplied 480 Vac from the nonsafety-related 480 Vac buses. Each charger has the capacity to continuously carry the steady state connected auxiliary loads required from station full-load operations or the steady-state post design basis accident load, without assistance from the batteries. This capacity includes the capability to restore the battery from the design minimum charge state within 8 hours, regardless of the status of the plant during which these demands occur. Transient loads that occur during startup, normal operation, shutdown or emergency shutdown may be adequately carried with assistance from the batteries if the transient loadings exceed the rated capacity of the charger. Only the batteries are necessary to power the safety-related loads during an emergency.

Separate and independent regulated battery chargers are provided in the 105-N building for each safety-related station battery. They are connected to the 125 Vdc distribution system buses by circuit breakers. Battery chargers for each 125 Vdc battery are supplied power from the 480 Vac bus of the same power channel as the bus and battery to which it is connected. This assures that a single failure of a 480 Vac bus cannot de-energize the normal supply to more than one 125 Vdc distribution system bus. In addition, one spare charger is provided with a manual transfer switch on both its ac input and its dc output. This allows the charger to be manually connected to either one of two safety-related batteries while remaining entirely in the selected power channel during normal maintenance, service or testing of the normal chargers. In addition, the spare battery charger located in the 105-N building can supply power to either battery by controlled switching procedures.

Each charger in the 105-N building is located in a separate room adjacent to the battery room containing the station battery to which it is connected.

Safety-Related Station Batteries

Each safety-related station battery has been sized to carry the emergency loads following loss of coolant accident concurrent with the loss of ac power for 10 hours* without battery terminal voltage falling below 100 Vdc,

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during any part of the duty cycle. Determination of the emergency load requirements for sizing each battery is based on complete loss of ac power, inverter full load and conservatively estimated 125 Vdc instrument and control load.

Each safety-related station battery is sized to have 25 percent excess capacity over that obtained from the duty cycle to account for loss of capacity during battery life.

Three 125V batteries (A, B and C as shown in Figure 8.3-5 and Table 8.3-1) are provided for 105-N building. They are connected to the 125V switchgear buses in such a way that each bus is provided with one 125V two-wire service.

During normal operation, the battery chargers have the capacity to supply the safety-related 125 Vdc distribution system load demand and, at the same time, maintain the safety-related batteries in a fully charged state. Each battery is floated on the bus, and will assist in meeting unanticipated and uncalculable peak demands that are greater than the charger capacity.

Batteries are inspected and tested on a frequency of 1, 3 and 6 months and 2 years. Equalizing charges are given to each battery by the chargers on a 3 month basis to maintain uniform cell charge. Application of such charges are administratively controlled. Each battery is load tested every two years to prove it is capable of performing as specified in Reference 1.

Each safety-related 125 Vdc system is operated ungrounded and is provided with a ground detection system of the voltage dividing type that will alarm a ground on either the positive or negative side of the line. The ground detector has the capability of detecting a ground at a resistance well above that at which equipment would fail to function.

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*125 Vdc power is only required to initiate certain Engineered Safety Feature Protective Actions. See Section 7.3. Further energy requirements are provided by diesel engines.

The 105-N reactor building safety-related batteries for power channels A and B are housed in a battery room adjacent to the switchgear room. Battery room walls are concrete, having 3-hour fire ratings. See Figure 8.3-8.

All batteries are installed on seismic Category I battery racks. The temperature of all safety-related batteries is maintained between 77 F and 90 F by the building ventilation system to assure battery performance and limit hydrogen buildup.

8.3.2.2 Analysis

Safety Analysis of the Onsite DC Power Systems

The dc power supplies, distribution system and load groups are arranged to provide dc electric power to the safety-related dc loads and for the control and switching of the safety-related systems. Sufficient electrical isolation and redundancy are provided to prevent the occurrence of common failure modes in the safety-related systems. A single active failure in any part of the safety-related dc power system will not prevent the safety-related dc power system from fulfilling its safety function.

"Single failure" used herein means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure.

As described in Subsection 8.3.2.1, the onsite safety-related dc power system provides separate and independent dc power to redundant load groups during abnormal and design basis accident conditions. The redundant load groups comprise engineered safety features and plant protection systems that are grouped in such a way that loss of one group does not prevent the minimum safety functions from being performed. Consequently, a single active failure in any part of the onsite dc power system cannot prevent the onsite dc power system from fulfilling its safety function.

In the event of loss of all ac power supplying the chargers, the batteries will operate in an auctioneering mode as the voltage decays and will carry the load on the 125 Vdc buses as indicated in Table 8.3-2. The load does

not change appreciably from reactor operation to safe reactor shutdown conditions.

The battery chargers are designed to restore the safety-related batteries from the design minimum charge state within eight hours regardless of the plant status.

The station dc system standby power sources in the 105-N building consist of three battery banks, designated A, B and C. The redundant safety-related load groups are each fed by a separate battery and battery charger. There is no provision for automatically connecting one battery-charger combination to any other redundant load group. No single active failure will result in the unavailability of dc power to the engineered safety features or the plant protection system.

The safety-related and nonsafety-related dc electric equipment is designed and located to permit appropriate periodic inspection and testing.

Tests and inspections, to assure the operability and functional performance of the dc electrical system components and the operation of the dc system as a whole, are performed on a routine and continuous basis as outlined by the N Reactor EMS system (Ref. 4). Typical tests and inspections carried out on a scheduled basis include:

1. Ni-Cad battery inspection and floating charge voltage check;
2. Specific gravity, temperature and charging voltage check on lead-acid batteries;
3. Ni-Cad battery cell voltage readings and electrolyte level checks;
4. Lead-acid battery equalizing charge, specific gravity, cell and battery voltage and electrolyte level checks;
5. Battery charger inspection, voltmeter checks and undervoltage device checks;
6. Test of automatic transfer switches;
7. Grounded detector device tests;
8. Annunciator circuit tests;
9. Battery capacity tests;

10. Breaker inspections; and
11. Cable insulation resistance tests. 7

The tests and inspections are performed during reactor shutdown periods under conditions that simulate normal system operation.

The safety-related dc power systems are not exposed to hostile environments, such as high radiation, high temperature or moisture during normal or accident conditions.

8.3.2.3 Cable and Raceway Systems

Description

The cable and raceway system comprises the cables and raceways for the 120 Vac distribution systems, 125 Vdc distribution systems and the instrumentation and control system. The 120 Vac and 125 Vdc cables and raceways that convey power to engineered safety features and plant protection systems are classed as safety-related equipment. All other cables and raceways are considered "associated" circuits and equipment.

To eliminate the possibility of the installation of unqualified cable in a safety-related system, all cable is qualified for application as safety-related equipment even though a portion of it may be applied as associated equipment.

All safety-related dc distribution cable trays, as well as much of the nonsafety-related cable, are coated with flame retardant material, or are protected by sprinklers. See Section 8.3.3. Recent reviews have identified the need to enhance the adequacy of fire protection features in Corridor 19. These actions will be accomplished prior to plant restart. 21

Application

Wire and cable insulation is selected on the basis of obtaining an insulation system that is acceptable for each application. Insulation systems comprise appropriate material or combinations of materials for strand shielding (where required), primary insulation, insulation shielding (where required), tapes, fillers and jackets.

Equipment and Component Reliability

The safety-related equipment and components were procured prior to the certification requirements but have been proven in service since 1964.

The nonsafety-related equipment and components have also been proven in service for the same period of time.

8.3.3 Fire Protection for Cable Systems

Fire protection for plant cable systems is accomplished by one or more of the features:

1. coating of cables and trays with an approved fire retardant coating;
2. water spray from automatic wet pipe sprinkler systems;
3. automatic halogenated fire extinguishing agent system;
4. manually operated water hoselines;
5. automatic alarm to professional fire fighting personnel from ionization type smoke detectors and rate of rise heat detectors.

All cable penetrations through walls and floors between established fire zones have been provided with fire stops.

A complete description of plant fire protection facilities is presented in Section 9.5.1, "Fire Protection System", of this Safety Analysis Report.

8.3.3.1 Design Bases

The level of fire protection required at N Reactor is specified in Chapter VII of DOE Order 5480.1.A. This level of fire protection reflects the privately insured industry classification of "preferred" or "improved risk." Fire protection upgrading has been completed in compliance with the DOE Order (see Section 9.5.1). Factory Mutual Engineering Association (FMEA) has reviewed, while under contract to the Federal Government, existing fire and explosion hazards and prevention and protection systems at N Reactor facilities. Evaluations and recommendations that resulted from these reviews are the subject of Reference 5. More recent fire tests and surveys have indicated that more stringent items were needed to insure fire protection.

These recommendations and the recommendations of the N Area Safety and Fire Protection Engineer, form the basis for fire protection and prevention activities at the N Reactor facilities (see Reference 6). 7

8.3.3.1.1 Fire-Retardant Coating

Fire-retardant coatings are Factory Mutual approved products and are applied to manufacturers' specifications. The retardant coating used is a water-base compound consisting of thermoplastic resinous binders, inorganic noncombustible reinforcing fibers, and flame-retardant chemicals. The test result from Factory Mutual Research Corporation is included in Appendix 8.3A.

8.3.3.1.2 Wet Pipe Sprinkler Systems

Automatic wet pipe sprinkler systems were designed and installed in accordance with the effective edition of National Fire Protection Association (NFPA) No. 13.

8.3.3.1.3 Halogenated Fire Extinguishing Agent System

The halogenated fire extinguishing agent system uses Halon 1301 as the extinguishing agent. This system was designed and installed in accordance with the 1973 edition of NFPA No. 12A.

8.3.3.1.4 Fire Alarm Signalling System

Factory Mutual and Underwriters Laboratory approved ionization and rate of rise detectors are used to provide automatic and manual signalling of fire conditions to the Central Fire Station through local alarm systems. These systems were designed and installed in accordance with requirements of effective editions of the National Fire Codes.

8.3.3.1.5 Fire Stops

Fire stops for cable tray penetrations are designed to retard fire spread between established fire zones. Existing fire stops are of two basic types. Original fire stops were obtained by packing the penetration with mineral wool and form fitting metal or transite plates on each end of the penetration anchored to the wall. The newer fire stops differ in that in place of form-fitted plates, a coating of Flamemastic 71A (See Appendix 8.3A) is 7

applied to the ends of the penetration such that the dry thickness is a minimum of one-eighth inch.

8.3.3.2 Specific Plant Features

An extensive program for coating cable trays with fire retardant has been in effect for several years. Cable trays that have been considered for coating are:

- o those containing cables necessary for safe shutdown,
- o those in groups of two or more,
- o full trays,
- o those trays 12 inches or more in width, and
- o those containing cables having combustible insulation.

All cable trays containing cables necessary for safe shutdown have been coated. Prior to the application of the coating, cables with the potential for overheating have been replaced.

8.3.3.2.1 Reactor Building (105 Building)

The reactor control room (room 145) and the reactor instrument and electrical equipment room (room 6), located immediately below the reactor control room, contain large concentrations of cables and cable trays associated with safety-related dc power systems, protection system circuits, control circuits and other equipment.

The reactor control room has been provided with an automatic Halon 1301 total flooding fire extinguishing system. The actuation mechanism is tripped by a cross-zoned system of ionization detectors which are also linked to the fire alarm system. Rapid dispersal of the Halon into enclosed consoles is accomplished by fans normally used for console ventilation. In order to retard fire spread from adjacent areas, all cable tray penetrations have been provided with the newer type fire stops. See Section 9.5.1.

Room 6 is a spacious area offering clear and uncluttered access to all areas and is provided with an automatic Halon 1301 total flooding fire extinguishing system. Storage of combustible material is minimized and

operation of large mechanical equipment is not permitted. All cables and cable trays have been completely coated with fire retardant. A fire detection system of ionization detectors annunciates in the reactor control room and at the central fire station. A manually operated 1000-lb carbon dioxide system is available from strategic locations providing total room coverage. This system is intended for use by fire department personnel See Section 9.5.1. A Halon automatic fire suppression system for part of room 6 of a type which uses neither water nor toxic gases, was installed. For further information see Section 9.5.1. 21

Lesser, but still significant, concentrations of cables and cable trays are encountered in the main dispersal routes from room 6. These routes are corridors 1, 2, 4, 5, the cable tray tunnel, and the lift station. All cable and cable trays in these areas except corridor 5 and the lift station have been coated with fire retardant. Sections of cable trays in corridor 5 are afforded coverage from automatic wet pipe sprinkler systems. Plans exist to install a heat detection system in cable trays in the lift station, and sprinklers, smoke detectors, and water hose stations in the cable tray tunnel. There are also plans to install sprinklers over cabling, and water hose stations in corridors 1 and 2. 7

8.3.3.2.2 Heat Exchanger Building (109-N)

The turbine bay (0-ft 0-in. elev.), turbine bay basement and corridor 19 (both 16-ft 0-in. elev.) of the 109-N heat exchanger building contain significant concentrations of cables and cable trays.

Cable trays in the turbine bay have been completely coated with fire retardant. Many of the cable trays which do not meet either of the aforementioned criteria have been coated as well. See Section 8.3.3.2. A wet sprinkler system was installed in each steam generator cell in the immediate vicinity of electrical cable runs near the south end of each steam generator cell. 7

Cable trays in the turbine bay basement have not been coated with fire retardant, but have been afforded protection by an automatic wet pipe

sprinkler system. An automatic wet pipe sprinkler system was used in this area due to the difficult access for applying fire-retardant coatings.

All cables and cable trays in corridor 19 have been completely coated with fire retardant and have been provided coverage from an automatic wet pipe sprinkler system.

8.3.3.2.3 Process Buildings

A total sprinkler system has been installed in the Plant Services Boiler (184-N building). The sprinkler system provides adequate fire protection for the building and its contents.

Other process buildings do not contain significant concentrations of cables and cable trays. Protection for these is provided by physical separation and incidental coverage from systems intended for protection of buildings or equipment.

8.3.3.2.4 Potential for a Brown's Ferry Incident at N Reactor

There are several distinct factors that contributed to the spread of fire at the Brown's Ferry Reactor that do not exist at N Reactor. The Brown's Ferry fire started in the cable spreading room, and spread into the reactor zone due to air pressure differential. The air movement through the combustible material in the zone barrier carried the fire into the reactor zone. Carbon dioxide and dry chemicals would not extinguish the Brown's Ferry fire due to the large air supply passing over the burning cables.

At N Reactor, cables that run out from room 6 leave Zone V without any appreciable change in air pressure. Room 6 is in ventilation Zone V which has a positive air pressure slightly above atmospheric. See Section 9.4.1. Since there is no appreciable zone air pressure differential, air movement to supply oxygen to a postulated fire through the wall penetration is reduced.

N Reactor electrical cables, which enter the reactor confinement zone, pass through a pressure fitting junction box which does not contain any combustible material other than the wire insulation. Wire entering the pressure fitting box contains a fire-retardant insulation. Wire which enters

the confinement zone leaves an essentially atmospheric pressure zone which separates the room 6 air pressure zone from the negative pressure reactor confinement zone.

Also, N Reactor cables in room 6 and cables leaving this room are all coated with a fire-retardant coating which will retard flame spreading. Brown's Ferry cables vigorously supported the combustion after the fire started. 7

Further, N Reactor barriers, where cable trays pass through building walls from different air pressure zones, are constructed of noncombustible material that have been covered with a sprayed on fire retardant coating. Barriers at the Brown's Ferry Reactor were easily ignited and burned through, permitting more air and oxygen to be supplied to the spreading fire.

Access to and activities in the N Reactor room 6 are closely controlled by Reactor Operations personnel. Smoking and open flames are not permitted in room 6. Air balancing between different pressure zones is done with smoke generators by trained Westinghouse Hanford Company (Westinghouse Hanford) Operations personnel. Storage of extraneous combustible materials such as drawings, paper, solvent etc. in room 6 or cabinets is minimized. Relays which have covers that could possibly ignite if overheated have been fitted with protective metal covers. Extensive tests were conducted to demonstrate the adequacy of these covers prior to installation. 7 7

Room 6 fire protection systems are described in earlier pages of Section 8.3 and in Chapter 9. As noted in Chapter 9, fire fighting protection services are provided by trained Westinghouse Hanford Operations personnel who are familiar with N Reactor. The fire fighting personnel have developed approved prefire plans which define fire fighting procedures and methods for various locations in the plant including the control room and room 6. 7

The Brown's Ferry fire and associated findings have been reviewed by N Reactor management. The findings which were judged to require corrective action have been implemented. 21

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TABLE 8.3-2

BATTERY LOAD CARRYING CAPACITY

WITHOUT BATTERY CHARGER ASSISTANCE

Building	Battery Designation	Safety-Related	Capacity in Hours	
105-N	A	X	10	
105-N	B	X	10	
105-N	C	X	10	
105-N	D		8	
105-N	E		8	
109-N	A		48	
109-N	B		40	
151-N	A		125	
153-N	A		110	
153-N	B		96	
181-N	A	X	40	21
182-N	A	X	24	21

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9.2.3 Demineralized Water System

The demineralized water system (see Figures 9.2-10 and 9.2-11) includes the equipment for processing water to produce a high purity product and the large storage tank in which the demineralized water is stored. The demineralized water system is operated such that a minimum of 452,000 gallons of water is held in reserve in the tank for ECCS use only. The distribution of the water from the storage tank to the various items of equipment and other systems in which the demineralized water is used, is described in the sections noted below:

- Emergency Core Cooling System - Section 6.3
- Graphite and shield cooling system - Section 5.5
- Secondary loop makeup system - Section 10.4.7.2
- Afterheat removal fill supply system - Section 9.2.6 | 21
- High pressure injection pumping system - Section 9.3.4
- Horizontal rod recirculation cooling system - Section 5.6

The demineralized water plant is not required for safe plant shutdown. However, the demineralized water storage tank is required for safe shutdown.

9.2.3.1 Design Bases

The demineralized water plant is designed to provide a sufficient quantity of high purity water for use in the N Reactor facilities as noted above. The system will produce at a maximum peak rate of 2300 gpm with 1600 gpm being the normal or "long run" production volume. The following list identifies the water quality criteria for the demineralized water plant effluent as currently used at N Reactor:

o conductivity	1 mho/cm (Max.)
o O_2	50 ppb (Max.)
o SiO_2	100 ppb (Max.)

The chloride ion concentration is of such a low concentration that it cannot be measured in the demineralized water. However, the chloride ion is checked for in each system that uses demineralized water to insure that no adverse quantity changes arise.

Storage facilities for 900,000 gallons of demineralized water are furnished in the demineralized water storage tank. In order to assure that those piping systems and equipment pieces served by the water from this tank will not be operated in the temperature - stress region of brittle fracture, the water within the tank is normally maintained at 100 F.

In order to ensure that the water in the demineralized water storage tank does not become contaminated through contact with air, a blanket of inert gas is maintained in the top of the tank.

The demineralized water system tank and piping are constructed as per applicable portions of the codes, specifications and standards issued by the following:

- o American Welding Society (AWS)
- o American Society of Mechanical Engineers (ASME)
- o American Society for Testing Materials (ASTM)
- o American Standards Association (ASA)
- o American Water Works Association (AWWA)
- o Hanford Standards
- o Hanford Works Specifications (HWS)
- o Joint Industry Conference (JIC)
- o American Steel Society Engineering Staff (ASSES)

The specific application of these codes and standards to the demineralized water system is indicated in Reference 10 for the tank and References 11 and 12 relative to the piping.

The design conditions for the storage tank and the piping system are as follows:

- o Demineralized water storage tank - Minimum flow rate to and from the tank - 1350 gpm at 70 F. No exterior insulation on tank.
- o Piping - 250 psig at 150 F.

The demineralized water system is a non-radioactive system and does not require that specific methods of leak detection and leakage containment be in place.

The air supply to the potable water tank is provided by either of two sources: (1) the high lift pump house (182-N building) low pressure air system (SA-3; see Section 9.3.1.4) and (2) the control air system (CAS; see Section 9.3.1.2). Both of these sources are capable of supplying oil-free air to the tank in sufficient quantity to maintain a pressure of 80 psig on the water surface.

The four potable water pumps, located on the main floor of the high lift pump house, have a design rating of 250 gpm each at 200-foot head. To provide flexibility in the event pump maintenance is required during electrical system manipulations, two of the pumps are supplied from offsite power and two, from onsite power.

9.2.4.3 Instrumentation Applications

The instrumentation and controls associated with the potable water system focus on maintaining proper water level within the potable water tank. This is accomplished by using limit switches, located at predetermined elevations in the tank, to start and stop the pump motors as specific water levels are achieved.

Additional instrumentation includes equipment to monitor the potable water pump discharge pressure, filtered water supply pressure and temperature and air pressure in the top of the potable water tank.

9.2.4.4 Design Evaluation

The design of the potable water system provides water that will meet all sanitary requirements. The system is conservatively designed to provide quantities of water adequate to enable proper function of the plumbing fixtures in all parts of the plant.

The filtered water supply for the potable water system originates as raw water obtained directly out of the Columbia River at a point upstream of any components that contain radionuclides. Since the potable water system is independent of any other system within the plant which potentially may contain radionuclides, accidental injection of radionuclides is not credible. The potable water is routinely tested for bacteriological and chemical quality. This procedure is described under Tests and Inspections below.

9.2.4.5 Tests and Inspections

Testing of the bacteriological and chemical quality of the water produced and distributed by the potable water system, is conducted by the Environmental Health Sciences Division of the Hanford Environmental Health Foundation.

Bacteriological quality is monitored as part of a routine program (weekly) of sampling and analysis for coliform bacteria in potable water sources at N Reactor and throughout the total Hanford project. The chemical quality is also monitored by weekly analysis for nitrates in samples from major project drinking water systems as part of the same program. In addition, an expanded annual potable water survey, including an analysis for 11 chemical parameters as specified by the EPA Interim Primary Drinking Water Regulations (Ref. 15), is performed.

The mechanical equipment and instrumentation associated with the potable water system are routinely and systematically inspected and functionally tested to ensure that they will produce a quality potable product. The tests and inspections are carried out as a part of the N Reactor preventative maintenance program. Typical tests and inspections include the following:

1. Calibration checks of all pressure, flow and temperature-indicating instruments
2. Functional tests of control instruments
3. Mechanical inspection of pumps and motors
4. Pump performance tests
5. Safety relief valve inspection and calibration test
6. Third party inspection of the storage tank

9.2.5 Ultimate Heat Sink

9.2.5.1 Design Bases

The Columbia River serves the function of the Ultimate Heat Sink (UHS) for the N Reactor. As the UHS the Columbia River provides adequate cooling water to remove heat from essential nuclear plant equipment for safe and orderly shutdown and to maintain it in a safe condition. The river also provides cooling water for

emergency cooldown following a loss-of-coolant accident without the availablity of offsite power.

9.2.5.2 System Description

The Columbia River is regulated by dams and reservoirs to meet various functional requirements such as flood control, hydro-electric power, irrigation, municipal and industrial supply. The N Reactor heat removal needs represent a very small fraction (<4%) of the total river flow at the licensed minimum flow rate. Section 2.4 provides a description of the Columbia River drainage basin including the dams up and downstream of the N Reactor and the affect of the dams in controlling floods and providing for at least the minimum flow rates for the UHS safety function. Section 2.4.14 details the low river flow and level requirements and the emergency procedures used to mitigate any reduction in river flow below the regulated minimum of 36,000 ft^3/sec.

The low-lift pumps (3) provide cooling water from the river to the various essential safety systems (ECCS, fog spray, graphite and shield cooling). The low-lift pumps are driven by diesel engines to ensure cooling water flow in the event of a loss of offsite power. 21

The low-lift pumps and the intake structure are described in Sections 6.3.2.2 and 3.3 respectively. Adequate flow can be provided with the assumed loss of one low-lift diesel pump.

9.2.5.3 Safety Evaluation

The heat removal method involves a once-through cooling system which initially draws on stored demineralized water and then on untreated Columbia River water to cool the fuel in the reactor and essential safety system components. The cooling water is not recirculated so no heat exchangers or makeup for pond evaporative losses or extensive heat rejection calculations are required. The capability of the river to provide adequate flow is described in Section 2.4.11.6.

The Columbia River is a large river which serves as a reliable heat sink as required by Regulatory Guide 1.27.

9.2.5.4 Testing and Inspection Requirements

The Low-Lift Diesel Pump testing is described in Section 6.3.4.

A technical specification limit of minimum Columbia River level (378 ft, 4 in.) is routinely monitored during operation with a surveillance requirement to calibrate the level indication monthly. 2

9.2.5.5 Instrumentation Requirements

A low river level alarm at the river gauging station one mile below Priest Rapids Dam (17 miles up-river from the N Reactor site) will annunciate in the 105-N control room.

9.2.6 Condensate Storage Facilities (Afterheat Removal System Storage Tank) 2

The condensate storage and makeup system stores and supplies demineralized water to the following systems and/or functions:

- o Secondary loop makeup during normal operation
- o Secondary loop flooding during water-to-water (reactor shutdown) operation (see Section 10.4.7)
- o Normal makeup to the deaerating heaters-Standby Power House (184-N building)
- o WPPSS facility makeup

The system also serves as a receiver for water spilled for level control of the secondary loop, water not needed in the secondary loop after water-to-water operation and condensate returns from various plant sources including the excess inventory from the 184-N building deaerator and also from the WPPSS facility. The condensate storage and makeup system is not a safety related system and is not required for safe plant shutdown under emergency circumstances.

9.2.6.1 Design Bases 2

The condensate storage and makeup system is designed to have sufficient capacity to supply all anticipated normal condensate makeup requirements. Storage facilities for 501,000 gallons of demineralized water and returned condensate are provided in the afterheat removal water storage tank. In order to ensure

that the systems served by the condensate storage or makeup system are not adversely affected by the addition of the makeup water, the contents of the afterheat removal water storage tank is normally maintained at 150°F.

The system tank and associated piping are designed and constructed as per the codes, specifications and standards issued by the following:

- o American Welding Society (AWS)
- o American Society of Mechanical Engineers (ASME)
- o American Society for Testing Materials (ASTM)
- o Hanford Works Specifications (HWS)
- o Hanford Standards

The specific application of these codes and standards to the condensate storage and makeup system is indicated in Reference 16 for the tank and in References 17 relative to the piping.

The design conditions for the storage tank and the piping system are:

- o Afterheat removal water storage tank - free convection within the tank with no flow to or from the tank with the water temperature at 150°F.
- o Exterior insulation on tank.
- o Piping - 150 psig and 200°F.

The condensate storage and makeup system is a non-radioactive system and does not require that any specific method of leak detection and leakage containment be in place.

9.2.6.2 System Description

The afterheat removal water storage tank is located in the tank farm area immediately adjacent to the high lift pump house (182-N building), and the two sets of pumps associated with the condensate storage and makeup system - the afterheat removal fill pumps and the secondary loop makeup pumps - are located in the 182-N building.

Demineralized water, from the demineralized water plant (see Section 9.2.3) as well as condensate from a variety of noncontaminated heaters and heat exchangers throughout the plant, is supplied to the afterheat removal water storage tank (Figure 9.2-14). This water is distributed by the afterheat removal fill pumps to the heat exchanger building (109-N) for use in the secondary loop flooding during water-to-water operation and for secondary loop makeup during reactor operation and to the standby power house (184-N building) as emergency makeup to the deaerating heater (Figure 9.2-15). Water is also drawn from the tank by the secondary loop makeup pumps for use in the 184-N building as the normal makeup water to the deaerating heater (Figure 9.2-16). The afterheat removal water storage tank has a 500,000 gallon capacity. The tank shell, roof and bottom plates are fabricated of ASTM A-283, Grade C steel. The exterior surface is insulated with mineral fiber felt, fibrated asphalt and one layer of 15-pound reinforcing roofing felt. Accessories consist of a spiral stairway providing access to the roof platform, a 16-inch roof vent and a manhole designed to permit personnel entry to the interior of the tank for inspection.

The tank is equipped with circular heating coils which maintain the water at a temperature of 150°F during periods of maximum flow and an atmospheric temperature of -25°F using 55 psig saturated steam. The water temperature is controlled automatically by regulating the steam flow to the heating coils.

(The use of the coils is not always required since the condensate return to the tank is often sufficient to supply enough heat to maintain the tank temperature.)

The three afterheat removal fill pumps, which provide flow from the afterheat removal water storage tank to the 109-N and 184-N buildings, have design ratings of 2500 gpm each at 235-foot head. They are driven by 200 hp electric motors, one of which is supplied from offsite power, one from onsite power, and the third from either power source, depending upon the switching arrangement in effect.

The two secondary loop makeup pumps which provide the flow to the 184-N building, have design ratings of 450 gpm at 242-foot head and are driven by 40 hp electric motors - one powered from offsite power and one from onsite power.

All piping associated with the condensate storage and makeup system is fabricated and installed in accordance with the requirements and codes stipulated in Reference 17.

9.2.6.3 Instrumentation Applications

Instrumentation, adequate to maintain proper control and surveillance of the condensate storage and makeup system, is provided. This includes equipment for monitoring both suction and discharge pressures on all of the pumps, the level and temperature of the water in the storage tank and the instrumentation for controlling the flow of steam to the tank heating coils.

9.2.6.4 Design Evaluation

Trouble-free operation since 1963 has demonstrated that the condensate storage and makeup system is capable of routinely providing the quantity of water required for normal secondary loop makeup, for makeup to the deaerator heater, WPPSS makeup and for secondary loop flooding following reactor shutdown. The ability of the system to receive and store the condensate from a variety of sources, including the WPPSS facility, has also been adequately demonstrated.

The steam vent capacity of the afterheat removal storage tank in a worst-case situation has been evaluated and analyzed. The case used was based on the maximum thermodynamic input which would create a major steam release within the tank. This was determined to be the release into the tank equivalent to the total condensate return from the WPPSS facility when operating at maximum production level. This amounts to 3060 pounds of condensate per second at a temperature of 265°F which develops a steam release rate of 4556 cubic feet of steam per second.

The analysis showed that completely safe tank venting would occur and, therefore, that excess tank venting capacity for all normal process releases is available (Ref. 18).

Three afterheat removal fill pumps are provided in order to assure that two can be maintained in operation or be available for operation at all times. The power supply to operating pumps or to pumps on ready standby is split between offsite and onsite power - one pump on each. This ensures that one pump will remain in or be available for operation in the event of loss of a single electrical ac power source.

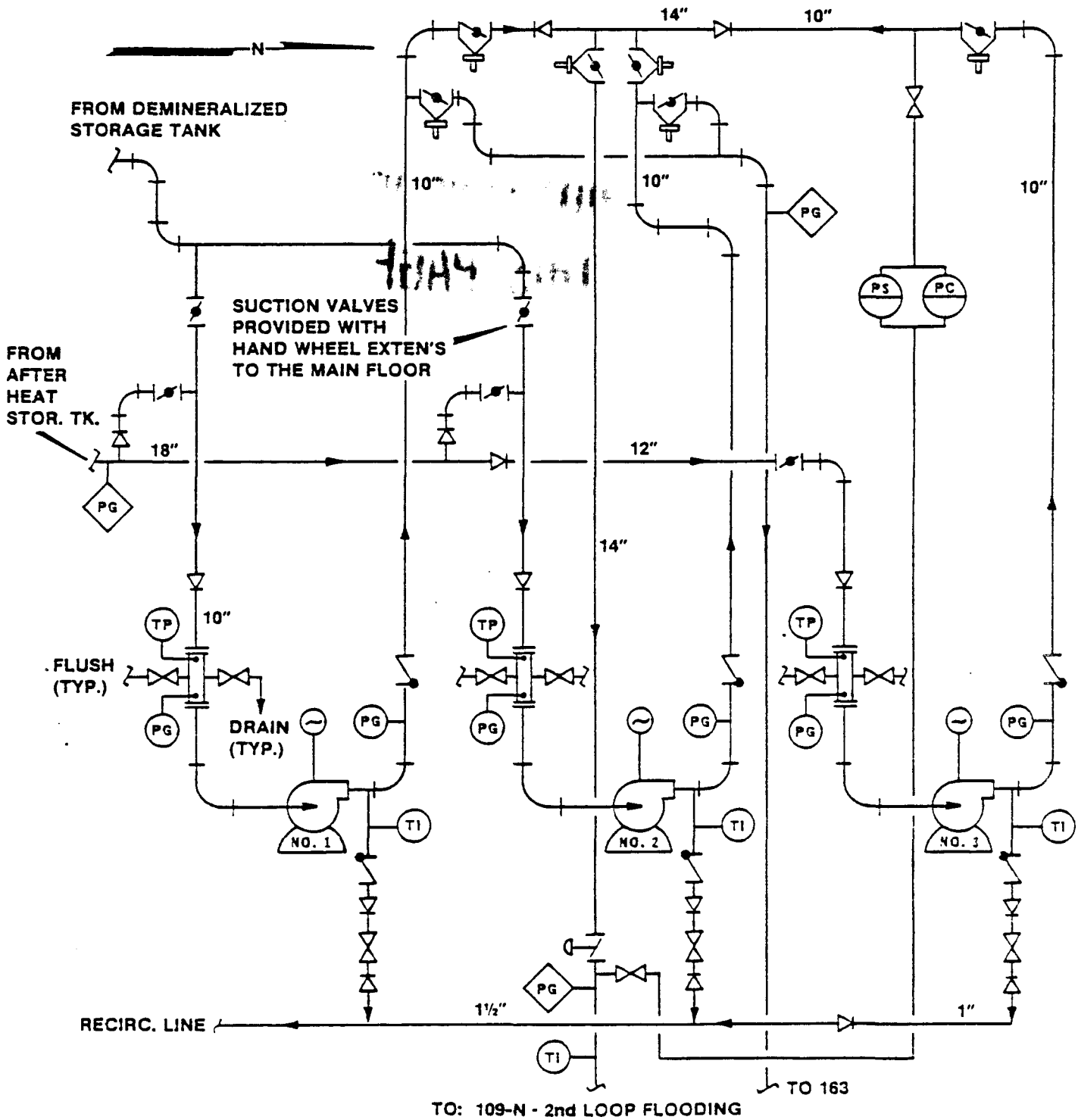
While the normal suction supply to the three afterheat removal fill pumps is from the afterheat removal water storage tank, an alternate source of supply is available. By proper valving, two of the pumps may receive suction supply directly from the demineralized water storage tank.

The environmental conditions which can adversely affect the functions of the condensate storage and makeup system are discussed in Section 3.4 (Flooding) and Section 3.5 (Missile Protection).

9.2.6.5 Tests and Inspections

The following tests and inspections are typical of those performed on the condensate storage and makeup system on a routine and continuing basis in order to ensure that the system is available for continuous service. These tests and inspections are implemented in accordance with the Equipment Maintenance Standards that are described in Reference 19, and cover:

1. Fill pumps - bearing vibration tests
2. Pressure and temperature instrument calibration checks
3. Pump performance tests
4. Internal inspection of storage tank



UNITED NUCLEAR INDUSTRIES, INC. N REACTOR SAFETY ANALYSIS REPORT	AFTERHEAT REMOVAL FILL SUPPLY PUMPING SYSTEM Figure 9.2-15

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9.5 OTHER AUXILIARY SYSTEMS

This section covers: (1) the fire protection system; (2) communications systems; (3) lighting systems; (4) the diesel generator functional system for N Reactor and (5) the diesel fuel storage and transfer system.

9.5.1 Fire Protection System

The N Reactor fire protection system objective is to provide protection for workers, the public and property. To achieve this objective, the system has been designed to ensure that, should a fire start within an N Reactor system that requires reactor shutdown, safe shutdown can be accomplished and maintained until the fire is extinguished. The basic elements of the system are divided into three categories: (1) preventive measures that minimize the probability of a fire; (2) isolation of high fire hazard areas and (3) detection, suppression and extinguishment of fires that could occur. This section covers the design basis, system description, safety evaluation, inspection and testing and personnel qualification and training for the N Reactor fire protection program. Detailed information covering fire zones, fire areas and fire barrier ratings are addressed in References 5 and 15. | 16

9.5.1.1 Design Basis

N Reactor systems, structures and components have been designed and located to minimize, consistent with other safety requirements, the consequences of postulated fires. The fire protection system is designed so that neither inadvertent system operation nor system failure will induce failure of Category-1 safety-related structures, systems and components. See Section 3.2.1 for details on Category-1 items.*

The intent of the N Reactor fire protection system is to comply with a level of protection called "improved risk" as used by the insurance industry. | 21

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*Category-1 items are those reactor structures, systems and components that have been designed to withstand postulated earthquakes producing a horizontal ground acceleration force of 0.25 g.

Improved risk protection requires compliance with the fire protection and loss prevention standards in DOE Order 5480.4, "Environmental Protection, Safety, and Health Protection Standards", DOE Order 5480.1A, Chapter VII, "Fire Protection", and DOE-RL Order 5480.7, "Fire Protection". Generally, an improved risk property is one that would qualify for complete insurance coverage by the Factory Mutual System, the Industrial Risk Insurers and other industrial insurance companies that limit insurance underwriting to the best protected industrial risk class. A higher standard of protection may be justified in certain instances for the purpose of national security, program continuity, or protection of the public. The objectives of the fire protection program at 100-N are that:

1. No threats to the public health or welfare will result from fire.
2. There are no undue hazards to employees from fire.
3. Vital DOE programs will not suffer unacceptable delays as a result of fire.
4. Property damage will be held to manageable levels.

The "maximum possible fire loss" is defined as the maximum loss that could occur in a single fire area assuming the failure of both automatic and manual fire extinguishing actions. The possible fire loss criteria and other requirements for fire protection engineering survey for 100-N Area (Reference 5) are summarized in DOE-RL Order 5480.7.

The major features of the N Reactor fire protection system that have been incorporated as the result of design or design changes include:

1. Noncombustible construction is employed throughout the plant to the extent practical.
2. Fire barriers are used to isolate high potential fire areas.
3. Fire retardant coating is provided for specified runs of electrical cables having combustible insulation that could create a potential fire hazard. See Section 8.3.3.
4. The turbine-generator lube oil systems are protected as described in Section 9.5.1.

5. Fire resistant hydraulic fluid is used in the horizontal control rod hydraulic systems.
6. Heat and smoke detectors are used in selected areas.
7. Elevator and work platform hydraulic systems are protected by confinement systems and/or automatic water spray systems. Additionally, elevators C, D, and W have fire resistant oil installed. 21
8. The fire water and raw water systems provide water to the following systems:
   - o wet and dry pipe sprinkler systems as listed in Table 9.5-1;
   - o deluge systems as listed in Table 9.5-2;
   - o fire hose stations located in critical areas throughout the plant.
9. Fire extinguishers are located throughout the plant.
10. A large, well-equipped, highly-trained fire department force is continuously available within 4 min after notification. A larger backup force is available when needed. 16
11. An extensive testing and surveillance program is conducted.
12. The communications system insures communications with the fire department.
13. Chemical suppression systems provide protection where water could cause impairment of Category-1 equipment.

9.5.1.2 Postulated Fires and Controls

The N Reactor fire protection system uses design aspects that employ separation, noncombustible and fire resistant materials, fire wall divisions, fire stops, fire retardant cable coating, conduit cable and piping, dampers and specific fire detection and suppression systems. Noncombustible fire resistant materials are used whenever possible to minimize fire hazard throughout the plant. 16

The postulated fires that could directly or indirectly affect safety-related systems are those potentially occurring at in-plant oil sources, electrical cables and motors or ordinary combustibles in some cases. These types of fires are respectively categorized as Class B, C and A fires.

Class B fires would involve fuel oil from the diesel drives, pump and turbine lube oil systems, insulating oil from the station transformers, or hydraulic oil from hydraulic equipment. These fires would be characterized by intense heat reaching 1500°F within minutes and generation of considerable smoke.

Class C fires could occur in energized electrical equipment, motors or cables. These fires would initially be characterized as smoldering fires producing large amounts of smoke with little flame or heat generated. Cable fires however, have the potential for severe heat and toxic products release.

Class A fires involve ordinary combustible materials such as wood, paper, cloth and some plastics. These fires are characterized by 1500°F heat and considerable smoke.

Physical separation, fire barriers, and/or redundant fire protection means are provided between associated systems or equipment where combustible material exists. Egress is provided from all plant areas via clearly marked fire exits and passageways. Egress is available via exits to outdoors at grade elevation or through adjacent buildings separated by fire walls from associated potential fire areas.

The principal hazardous chemical that would be released by fires is carbon monoxide. Significant quantities of other chemicals may also be produced, depending upon the composition of combustible materials. Precautions are taken by virtue of the use of emergency breathing apparatus to mitigate the consequences of chemical inhalation from the products of combustion. Self-contained breathing apparatus is available for the fire department personnel to combat fires.

Water for the fire protection of the Category-1 equipment and for the balance of the plant fire protection system is provided by two horizontal centrifugal pumps. Each pump is capable of supplying 2000 gal/min at 70 lb/in^2 which is adequate for any postulated single fire. An electric motor drives the first pump with power provided from two independent sources. The other pump is diesel driven. Both will start automatically, should there be a drop in

line pressure. While the fire protection system has been seismically designed to comply with the National Fire Protection Association (NFPA)-13 requirements, it is not designed in accord with criteria used for Category-1 equipment. Section 3-10.3 of the NFPA-13 lists the requirements for protection of piping against damage where subject to earthquakes. The main features of the requirements are concentrated on flexibility of the piping systems to minimize damages resulting from relative movement of the system and differential movement within and between structures. The backup water supply to the 117-N Building confinement exhaust filter sprays were upgraded to Seismic I (0.25 g) by Ref. 18 to the valve house outside of 117-N.

All equipment in the fire protection system conforms to the standards of NFPA and was originally Underwriters' Laboratory (U.L.) listed or Factory Mutual (FM) approved. All fire detection systems and those deluge valves for Category 2 locations are provided with backup dc power in the event of primary power unavailability. Deluge valves also have manual actuation provisions.

Noncombustible construction materials are used throughout N Reactor and its important support facilities, with limited amounts of combustible finish materials. Fire barriers are used to isolate work areas and rooms.

Operations with potential for fire that need not be near plant processes are housed in separate buildings or fire areas. Table 9.5-3 shows the fire protection system features at locations within 100-N Area.

9.5.1.3 System Description

9.5.1.3.1 General

The fire water is supplied from the area fire main pumping supply system, with a 1-in intertie to the raw water supply (RWS-2) system for pressure maintenance. See Section 9.2.1 and Figure 9.2-3. Suction water for the pumps on both systems is derived from the low-lift, circulating-raw-water supply system, with the emergency low lift diesels providing backup. See Section 9.2.1.

The fire water system is designed to provide 2000 gal/min at 70 lb/in^2. This system is supplied by two automatic horizontal centrifugal pumps. The pumps have ratings of 2000 gal/min each at 161.5-ft head. One pump is driven by a 200 hp, 440V induction motor, supplied with power from either the offsite source or the on-site source of ac power. The other pump is driven by a 200 hp diesel engine equipped with an electric starting motor. Power is delivered alternately and automatically from one of two battery banks on successive starts. A dc generator battery charger and voltage regulator are provided for battery charging.

Normally the system pump suction header is supplied by the circulating raw water supply system; however, the suction header can also be supplied by the 181-N low-lift emergency raw water supply system, the demineralized water storage tanks or the filtered water storage tank.

System pressure is maintained by a 1-in supply line from the 182-N high-pressure raw-water system. Both pump controllers have pressure switches tied in to the pump discharge header, to provide automatic starting on a fire main pressure drop. The electric pump is set to cut-in first.

The fire service water supply source is backed up by the 12-in intertie to the 16-in loop at the Washington Public Power Supply System (WPPSS) Hanford Generating Project. This system's fire service supply is from pumps located on a suction station at the Columbia River about one quarter mile upstream of river pump house 181-N. There are two electric-driven pumps and one diesel driven pump. The pumps are automatic starting, 2000 gal/min, 125 lb/in^2 vertical turbine fire pumps. All equipment and controllers are U.L. listed or Factory Mutual approved.

The WPPSS backup system's electrical power supplies are independent of N Reactor except during startup and shutdown during which the electrical power supplies are provided by A Bus through C1X7. Operations are supervised by the WPPSS control room which has three separate communication channels to N Reactor. The intertie is normally closed with a post indicator valve on

each side of the property line fence. These valves would be opened manually on request by either area's control room operator.

Fire service mains at N Reactor are pressurized at 110 lb/in^2 (yard level), by a pressure regulating valve in the 1-in RWS (2) intertie. The water for fire protection is fed through a 12-in loop. Fire hydrants are approximately 250-ft apart. 16

The wet pipe and dry pipe sprinkler systems have fusible link heads that activate when exposed to excessive heat. The activation temperatures vary with respect to the equipment being protected and the maximum ambient ceiling temperatures.

Automatic total flooding Halon systems are provided in Room 6, the control room in the 105 Building, both the 182 and 184 Buildings' control rooms and the emergency shutdown room. The systems are actuated by cross-zoned smoke detection systems, and have manual tripping capability. 21

Ionization and/or photoelectric smoke detectors are used in rooms 145, 8, 9, 47, and 6 of 105-N and other areas (151-N, 182 Control Room, 184 Control Room, and 163 Control Room and Master Control Center and Room 122 of 109-N).

100-N Area fire alarms and system trouble signals are transmitted to the area Central Fire Station from the local alarm panels. 21

All equipment in the fire protection system conforms to the standards of the National Fire Protection Association (NFPA) and is Underwriters' Laboratory (U.L.) listed or FM approved. The fire detection system and some deluge valves are provided with backup dc power in the event of primary power unavailability; other deluge systems are independently air operated. Deluge valves are provided with manual actuation provisions.

In all cases, fire extinguishing media are selected with due regard for the radiological, metallurgical, physical and chemical compatibility of the extinguishing agents with plant components. 16

9.5.1.3.2 Category-1 Items*

Category-1 items in N Reactor are defined in Section 3.2.1. Table 9.5-1 identifies the wet and dry pipe sprinkler systems that have been installed in accord with the National Fire Protection Association (NFPA) Code No. 13, to provide fire protection. Table 9.5-2 provides similar information on deluge systems that have been installed in accord with NFPA-15. Both of these tables cover the full scope of coverage. Those items that are classified as Category-1 are marked with an asterisk.

Other protective features for Category-1 items include:

1. Two foam extinguishing systems are installed to provide special protection for oil handling and oil storage at the 166-N building and tank farm. The system was designed in accordance with NFPA-11.
2. The buildings are provided with 1-1/2-in diameter hoses arranged so the Category-1 items are effectively covered.
3. A dry standpipe with a siamese pumper connection is provided on the 109-N and 105-N roofs for the Fire Department use. Five hose and reel cabinets are distributed to provide complete coverage in accordance with NFPA-14, Class-1 service.
4. Portable extinguishers are located throughout the plant. Standards for the installation, maintenance and use are in accordance with NFPA-10.
5. Fire partitions are provided between diesel engines in 181-N and 182-N buildings with curbs to limit the spread of oil to adjoining engines, and separate detection and suppression systems for each engine.
6. Control Rooms in 105-N, 109-N, 182-N, 184-N, and 163-N are protected with Halon 1301 fire suppression systems.
7. Building 117-N has been provided with seismically qualified backup raw water supply for the confinement filter spray system (Ref. 18) to the valve house outside 117-N.

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*Category-1 items are those reactor structures, systems and components that have been designed to withstand postulated earthquakes producing a horizontal ground acceleration force of 0.25 g.

9.5.1.3.3 System Interactions

Requirements to ensure that operation of the fire protection system will not produce an unsafe condition by interaction with other reactor systems are:

1. All sprinkler covered areas within the plant are adequately drained to minimize water damage. Detailed evaluation on water runoff control is documented in UNI-1710 (Reference 5). 16
2. For the reactor control rooms and Room 6, Halon 1301 is the primary extinguishing agent because it will minimize electrical system perturbations and leave no residue.
3. The fire pump shutoff head is below system static test pressure.

9.5.1.3.4 Protection of Category-1 Equipment

The following information identifies, on a location by location basis, protection for Category-1 items.

105-N. The 105-N Category-1 locations include the following spaces:

1. Immediately above the reactor
2. Inlet face and "C" elevator (work platform)
3. Discharge face and "D" elevator (work platform)
4. Pipe spaces containing reactor coolant piping
5. Zone-1 ventilation supply and exhaust ducts and plenums
6. Horizontal Control Rod Rooms
7. D.C. electrical equipment room. 21

The locations listed as 1 and 4 above contain minimal combustible materials. Areas listed as 2 and 3 above contain hydraulic fluid to operate the rams on the "C" and "D" elevators. These locations are protected by automatic water spray (deluge) systems and have fire-resistant oil installed to minimize the potential for fire (Ref. 20). 16 21

The "D" elevator ram and associated hydraulic equipment are located next to the discharge pit. A remotely possible fire could arise from the ignition of spraying fire resistant hydraulic fluid caused by a high-pressure leak. 21

To prevent such leaks, components such as foot valves are enclosed with a cover that is equipped with a drain and holding tank. The holding tank has a level alarm to alert control room personnel. In addition to oil containment protection, the fog spray can be activated manually to extinguish a fire. See Section 6.2.2.

The horizontal control rod rooms are Category-1 locations that have fire suppression systems and in addition, the rod control and scram equipment contain a fire resistant emulsion of oil and water.

The Zone-1 ventilation supply and exhaust ducts and plenum exhaust to the 117-N building. This building contains the filtering system for exhaust air and is designed to prevent excessive release of radionuclides to the atmosphere. The filter and diversion valve hydraulics in this building are combustible; thus, an automatic sprinkler protection system has been installed. In addition, a fog spray system is used to protect the filtering system from a postulated fire caused by radionuclide decay heat. (Heat detectors in filter train alarm at high temperatures and trip water spray systems at high high temperatures. Water spray discharges directly on the charcoal filters.) The fog spray system would receive its water supply from filtered water. It is also backed up by new seismically qualified piping from the fire protection water supply (Ref. 18) to the valve house outside 117-N.

109-N. The 109-N Category-1 locations include the following spaces:

1. Steam generator cells
2. Auxiliary equipment cells
3. Pipe gallery
4. Cell access corridor
5. Pressurizer penthouse
6. Emergency shutdown panel room.

The pipe gallery and pressurizer penthouse contain minimal combustible materials. Locations listed as 1, 2 and 4 contain minimal amounts of

combustibles. The primary source of combustibles in the steam generator cells is from the lubricating oil for the reactor coolant pumps. Drip pans and drain lines are provided at each reactor coolant pump to collect oil leakage from the pumps. Oil leakage can drain to collection tanks in the access corridor. Each pump has automatic sprinkler protection. In addition to the automatic sprinkler system, the cell confinement fog spray system can be activated manually. The supply for the sprinkler system was changed from RWS-2 to the Fire Protection System by Project H-639. The emergency shutdown panel room is provided with a Halon fire suppression system. 21

182-N Building

The 182-N building is separated from other buildings. The building contains the following Category-1 items: emergency diesel generator, high lift emergency raw water pumping system, fog spray diesels, high lift diesels and 125V dc battery systems. In addition, the N-Plant fire protection water system originates in this building. Sprinkler additions were made for the emergency fire pump and emergency generator by Project H-639. Spray shielding was provided for the emergency generator. 21

Potential fires in this building could stem from ignition of fuel, should it leak from the diesel lubricating or fuel lines. All the fuel and oil tanks are located outside the building. With exception of the two 360-gal day tanks, all the combustible liquids are stored in underground storage tanks.

Potential fuel oil fires in the building would be Class B fires that could generate temperatures of 1500°F, within 4 min, and considerable smoke. Separate detection and suppression systems for each engine would control or extinguish the fire. Fire partitions between the diesels would minimize damage potential. Drainage piping is provided to convey fire water from the basement area (Ref. 19). 21

The ventilation system can be manually operated to clear smoke from the building. See Section 9.5.8

Electrical Control Systems

Buildings 151-N, 153-N, 105-N, 109-N and 184-N contain the essential switchgear for reactor operation. These buildings are either isolated from other buildings or separated by a fire wall.

1. 151-N

   The incoming power for the 100-N Area is 230 kV from offsite with the main transformer located in the 151-N switchgear yard. The 13.8 kV lines go from the low side transformer terminals of the transformer through metal ducts into the 151-N building. Electrical cables are physically separated and do not exist in sufficient bundles to significantly contribute to a fire. The building interior and contents are largely noncombustible.

   A small Class C fire could occur inside the building. Portable Class C rated hand extinguishers are available for fire fighting. Automatic smoke detectors are provided for early warning.

   The 30/50 MVA transformer, that potentially could initiate a Class B fire, is located in the yard. The transformer is protected by a deluge water spray system that would be actuated automatically by rate-of-rise heat detectors and is backed up by provision for manual actuation.

2. 153-N

   153-N houses the switchgear and supporting equipment for offsite and onsite power. Each power source is supplied by a 13.8 kV/4160V service transformer located outside at opposite ends of the building and isolated from the building by fire walls. The two power sources are separated within the building by a concrete block fire wall. Electrical cables are physically separated and do not exist in sufficient bundles to significantly contribute to a fire. See Sections 8.2.1 and 8.3.1 for information on offsite and onsite power.

   Only a small potential exists for a Class C fire within the building. Portable Class C rated hand extinguishers are provided throughout the

building for fire fighting. The 153-N building is equipped with an automatic heat detection system. 16

3. 184-N

The plant services generator housed in this building is designed to use low pressure steam from the boilers to generate ac power sufficient to supply onsite power plant loads. The generator is a hydrogen-cooled unit designed to operate at 18,750 kW. The unit is completely enclosed to prevent leakage of hydrogen gas. 21

A CO_2 system is used to purge air from the system before operation and hydrogen from the system prior to maintenance.

184-N is protected from Class B fires with automatic and manual sprinkler systems. The building interior is constructed of noncombustible materials and is occupied 24 h daily. The automatic sprinkler systems consist of six open head deluge type and one closed head fusible link wet pipe type. The manual systems are a deluge open head type that can be controlled from the control room and/or from pull stations located on the risers on the main floor of the annex. The manual system would provide water should additional protection be needed. Ventilation is adequate to vent smoke. Water drainage is adequate. The control room is protected by an automatic Halon 1301 system. 16

4. Cable Spreading Room (Room 6)

The cable spreading room, located below the 105-N reactor control room (room 145), contains control and instrumentation cables as well as instrumentation backup batteries. Most cables in the spreading area are coated with Flamemastic 71A. See Section 8.3.3 for detailed discussion of fire protection features provided for Room 6. The fire retardant coating on these cables significantly reduces the combustibility of the cable insulation. Fire stops are provided for all cable penetrations out of this room. 21

A Class C electrical fire is possible in this room. The ionization and photoelectric smoke detectors in this room would detect a fire in the incipient stage, thus alerting the fire department and the control room simultaneously. The automatic Halon 1301 protection system for Room 6 is cross-zoned. The Halon system can also be actuated manually. Portable extinguishers are distributed throughout the room.

The recirculating ventilation system in the cable spreading room adjoins the reactor control room. Four self-contained breathing apparatus are available in the control room should they be required. The ventilation system would be manually controlled either to isolate the system or to exhaust smoke from the room.

5. 105-N Control Room

The control room is located above and adjoins the cable spreading area. There are no heavy power cables in the control room; therefore, the potential for fire due to electrical fault is minimized.

Credible fires of consequence in the control room would be small Class C or Class A fires producing smoke with little flame or heat.

Smoke detectors are located in the control room to alert the control room operators of a fire condition. Since the control room is continuously manned, fire fighting could be rapidly initiated while maintaining safe plant shutdown capabilities. Means of performing plant shutdown activities outside the control room have been provided and are discussed in Section 7.4.1.3 and Section 3.1.1, Criterion 19.

Portable extinguishers are provided in the control room for fire fighting. Self-contained breathing apparatus are also available to assist should fire fighting be required. An automatic Halon-1301 System is provided. Should a fire develop such that normal fire fighting procedures are not feasible, the Halon 1301 System following a 30-s delay, would automatically discharge. The 30-s pre-action alarm feature allows operating personnel time to observe the fire and evacuate

non-essential personnel if required. The manual system can be actuated at any time. Operating features are the same as the Room 6 system. A manually operated abort switch is provided, if manual extinguishment is feasible. The ventilation system is interlocked to the Halon system, and should not be manually operated, until a fire is extinguished.

6. 105-N Electrical Cable Tunnel

All electrical cables feed Category-1 equipment through Room 47, which is the electrical cable tunnel. The room is made of noncombustible concrete, and fire stops provide barriers between Room 47 and the cable spreading room (room 6). The electrical cables are coated with a fire retardant coating which reduces the combustibility of the insulation. Automatic wet pipe sprinkler protection, early warning smoke detectors, and manual water hose stations are provided.

At the opposite end from Room 6, Room 47 opens into Room 2, a valve and equipment room. Automatic sprinkler protection and fire retardant coating are provided for electrical cables in Room 2 through Corridor 5 and Corridor 4. The postulated fire in Room 47, Room 2 or the above corridors would be classified as a Class C fire. This type fire could be extinguished with a portable hand extinguisher, water hose stations, or by the automatic sprinklers provided. Drainage is provided to direct fire water from the affected area.

Both outer rod rooms contain hydraulic power packs for the rod systems. The fluid is a water/oil emulsion type safety fluid, which is considerably less combustible than ordinary hydraulic fluid. However, it can burn when atomized and/or exposed to sufficient heat. Each room contains a 180-gal reservoir. The pumps are curbed. A large spill would enter drain system No. 14 and be routed to the lift station, which has sump oil detection and annunciation. Trouble in the oil systems will also annunciate in the 105 Control Room. The outer rod rooms are adequately protected on each level with Automatic Sprinkler's Suprotex deluge system.

Quality control for the oil, to assure that its fire resistance has not degraded, consists of:

1. Monthly PM's to check the percent of water in the emulsion.
2. Annual samples sent to the vendor for lab analysis, which includes bacteria checks.

181-N

The circulating raw water system originates at 181-N building and is designed to supply the normal water requirements of 100-N Area. The system is supplied by four pumps. Electric power to pump motors 1 and 3 is supplied from the offsite electrical power source and to motors 2 and 4 from the onsite ac electrical power source.

There are three ECCS low-lift diesels housed inside building 181-N to provide an additional water source in addition to the four electric powered pumps. Water for fire suppression can also be obtained from the filtered water tank or from the WPPSS Hanford Generating Project. The fuel oil for these engines creates the potential Class B fire.

The diesels are protected by open-head deluge water spray systems. The deluge systems would either be activated by heat detectors or manually.

Fire protection spray water can be adequately drained from the building.

New Fuel Storage

New reactor fuel is stored in Rooms 181, 182 and 29, the work elevator (platform) room of 105-N.

The fuel is shipped to the 100-N Area in wooden boxes and stored in 182. From here the fuel is moved to room 181 and loaded into tubular magazines. Once the magazines are loaded, they are stored on racks next to the work elevator. See Section 9.1.1.

Potential fires in the fuel storage areas are Class A - wooden boxes, and Class B - hydraulic fluid beneath the work elevator. An open-head-deluge, heat-actuated, system protects the underside of the work elevator. Rooms 181 and 182 are equipped with a wet-pipe sprinkler system. Adequate drainage is available to eliminate sprinkler runoff. The ventilation system is adequate to vent any smoke generated from fires in this area. Water hose stations are also available. 16

9.5.1.3.5 Non-Nuclear Facility Protection

Fire protection systems are provided in various plant areas not required for the safe shutdown of the reactor systems. The protective system features and the areas protected are covered in Table 9.5-3.

The generator step-up, unit auxiliary and reserve auxiliary transformers are located sufficiently far from adjacent buildings to minimize the effects of a fire.

Fire barrier walls are provided between transformers spaced less than 50-ft apart.

The boiler fuel storage tank is located within a diked area and is isolated from any safety-related structures or components.

9.5.1.3.6 Tests and Inspections

Equipment Maintenance Standards have been implemented for testing systems included in the technical specifications at a frequency set by NFPA Codes and DOE. Other systems are tested on a reduced frequency in accordance with DOE requirements. Most tests are accomplished by the Hanford Fire Department. The diesel fire pump is tested weekly via a 35-min operating run. 16 21

9.5.1.3.7 Fire Protection Codes and Seismic Design

The fire protection system has been designed and constructed in accord with the following National Fire Protection Association (NFPA) codes:

Systems Employed	Locations	NFPA Code Nos.
Foam extinguishing systems	Tank farm and 166-N	11
Halon 1301 system	(Several)	12A
Automatic sprinkler systems	(Several)	13
Standpipe and hose systems	Roofs of 105 and 109-N	14
Auxiliary fire alarm signaling system	(Several)	72B

The automatic sprinkler systems that have been installed over engineered safety features equipment are anchored and supported in accordance with NFPA-13 requirements for seismic conditions.

9.5.1.4 Design Evaluation

The fire protection system is designed to detect, automatically extinguish or hold in check, fires that are most likely to occur in N Reactor. Equipment for manually extinguishing all types of fires is accessible from any point in the area.

Two fire pumps, one electric and one diesel, are provided. The electric pump is powered from one of two sources - offsite or onsite ac power. The diesel-engine-driven fire pump can deliver the maximum design basis fire water flow in the unlikely event of both offsite and onsite power loss.

Fire pumps operate below the nameplate specifications, because pump impellers have been modified to reduce maximum available pressure to the fire protection system. Both pumps are rated for 2000 gal/min at 70 lb/in^2.

Fire potential is minimized by the following design features:

1. Noncombustible materials of construction are used throughout all nuclear structures.

2. Hydraulic systems in the rod rooms and 105-N elevators use fire resistant hydraulic fluid. 21
3. Dry pipe automatic sprinklers are used where freezing could occur.
4. Fire barriers are used between diesel engines.
5. New electrical cable installed in critical nuclear safety areas is required to be IEEE 383 rated (see UNI-M-38, Control 207).
6. Electrical cables in the above areas are also coated with a fire retardant flamastic. 16

Fire detection devices are tested for reliability. See Section 9.5.1.3.6.

Each deluge or dry system operates from a separate air supply. Failure of any one system would not make the other system inoperative.

All smoke detection systems are on supervised circuits, and the halon actuation circuit in Room 6 is a supervised Class A circuit (capable of releasing halon even in a single open or ground fault circuit condition).

All fire alarm systems have secondary internal battery power supplies, sufficient to operate the systems for 60 h, in the event of a loss of AC power. 16

Floor drains, in areas containing sprinkler and deluge systems, are designed to accept the maximum flows that can occur by actuation of these systems. No flooding of Category-1 equipment will occur when any of these systems is operating. A remote potential exists for water getting into the cable spreading room. Should the major 8-in fire line break in Room 2, water would flow into Rooms 2 and 12 and into Room 47. The combined floor space in these locations is approximately 14,000 ft^2. Two 8-in drains are located within the approximately 14,000 ft^2 of floor space. The fire line break would actuate both the electric and diesel fire pumps forcing approximately 4000 gal/min into Room 2. 16

Before water could enter Room 6, it would have to reach a height of 4 ft within the approximately 14,000 ft^2 of contiguous floor space, which would

take about 1.6 hours, ignoring the drainage factor. Water could then seep into room 6 at the cable tray penetrations, which are sealed with fire stops. However, before water could reach room 6, the Fire Department would have valved off the flow, and the reactor would be shut down as a precautionary measure. Following such a postulated event, the lines would be repaired and placed back into service.

Table 9.5-3 summarizes the protective features for 100-N Area facilities. The Central Fire Station staff is always available and would arrive within 4 min following notification. Thus, in the unlikely event that both the automatic and N Reactor manual activations systems fail, the postulated fire would be extinguished by the trained Fire Department staff. See Section 9.5.1.5.

9.5.1.5 Personnel Qualifications, Training and Administrative Controls

Personnel Qualifications

The Hanford Site Fire Protection group has one full-time fire protection engineer assigned for support of 100-N Area fire protection systems. The Fire Protection group is independent from the 100-N Area Operations and Engineering Divisions.

The qualifications for the Fire Protection Engineer position are:

1. Equivalent to a college or university degree in Fire Protection or other engineering disciplines such as chemical, electrical, civil, industrial, or mechanical.
2. Not less than 3 yr of experience in responsible charge of fire protection engineering work.

The 100 Area Fire Station is centrally located for the 100 Areas. The station is 10,000 ft^2 in area and houses two 1500-gal/min pumpers, two 1500-gal brush trucks, and one 4500-gal supply truck. Station 2 maintains one attack truck with AFFF foam, halon, and dry chemical capability.

One four-man company responds to all 100 Area alarms with a Battalion Chief in charge. One company from the 200 Area fire station provides backup on all alarms and would respond directly to the fire if necessary.

For brush fires, personnel from one engine company operate two brush trucks which are supplied by the 5000-gal semi-tanker. 21

Training

The fire department has a fully trained rescue squad that is properly equipped for rescue work.

Each firefighter has a minimum of 2-h training per duty day which includes both:

1. rescue and suppression practices, and
2. job related functions and building tours.

In addition to training for fire fighting and rescue work, the fire department staff:

1. conducts apparatus inventory and service daily;
2. checks all masks weekly;
3. checks all ladders monthly;
4. flushes, checks and dries all fire hoses in September, December and March;
5. conducts pressure tests of all fire hoses in June each year;
6. tests all pumps for reliability each May and makes vacuum tests quarterly;
7. checks and makes flow tests of selected project fire hydrants each March and September;
8. provides, when requested, classes and demonstrations to other plant personnel; 21
9. updates all pre-fire plans;

Administrative Controls

In order to reduce the potential for fires, various administrative tools are used to control storage of combustible material (Ref. 21), control welding and cutting processes in safety-related areas (Refs. 22 and 26), provide for adequate housekeeping and cleanliness control (Ref. 23), and establish fire watches when automatic detection/suppression systems are disabled (Ref. 24).

Administrative controls are also established to ensure that fire barriers are not degraded by inadvertent modification (Ref. 25), and that fire doors and dampers (Ref. 27), and portable extinguishers are maintained (Ref. 28). 2

9.5.2 Communication Systems

This section covers N Reactor's communication systems that are designed to provide safe, effective, and convenient operational audio communications between various plant locations and between the plant and locations exterior to the plant.

The communication systems are designed to meet specific criteria, codes, and standards (Ref. 1, 2, and 3).

9.5.2.1 Design Bases

The N Reactor communication systems were designed to ensure adequate communications within all important N Reactor work spaces, N Area locations, other onsite locations and offsite locations even if ac power should fail.

Codes and standards which were applied in the design and construction of the N Plant communication systems are:

- National Electric Code (NEC)
- National Electrical Safety Code (NESC)
- Hanford Works Specification (HWS) - 6971: Specification for Audio communication Systems

- HWS-8002-S: Specification for Thermoplastic Insulated Wire and Cable, Rev. 8-25-59, Type TW 600 Volt
- HWS-6574, Rev. 0: Specification for Confinement Seals
- J-2-2, Rev. 1, Hanford Standards Instrument Panel Finish
- J-2-7, Rev. 0, Hanford Standards Instrument Panel Nameplate
- D-3-10, Rev. 1, Hanford Standards Grounding Methods Conductor Table
- D-3-10b, Rev. 1, Hanford Standards Grounding Methods: Lighting and small Power Loads
- D-3-10d, Rev. 1, Hanford Standards Grounding Methods: Connections to Sheet Metal Enclosures
- D-20-20, Rev. 0, Hanford Standards 15 Amp 120 Volt Receptacle

9.5.2.2 System Description

For a communication systems block diagram, see Figure 9.5-2. Each of the 13 systems are described as follows. In addition, the Patrol Security Net is described.

1. A general plant telephone (GPT) system is provided and maintained by General Telephone and Electronics (GT&E) commercial telephone company with branch headquarters in Richland, Washington. A contract manager for Richland Operations is selected by DOE. The GPT system in 100-N area connects with GT&E system and is a part of the outside commercial telephone system.

   Electronic gear and the main distribution panel are in room 6 of the 105-N building. The automatic exchange is in 100-K Area. The system has a capability of 234 private lines.

2. A private automatic exchange (PAX) telephone and paging system was designed and built by General Dynamics, Stromberg-Carlson Division, Rochester, New York. The telephone exchange portion of this system is the Dialmaster TSB-100-100-10 telephone switchboard. The dialmaster telephone system is a completely automatic, three-wire telephone switchboard, accommodating 96 private telephone lines and 14 links. With these lines and links, 14 two-way conversations between

28 telephones may be made simultaneously. The system provides for dial selective area paging to three areas plus an overriding dial selection all-call to three areas simultaneously. The telephone switchboard paging lines are terminated in audio line amplifiers, which are used to drive power amplifiers for each area. Loudspeakers or public address horns are provided as required throughout the plant.

In addition, the system has capability for conference calls, "camp on" capability (wherein a calling party can wait out a busy line until the called party hangs up, and they will be automatically connected) and executive override. The override feature is included on five lines. These five lines are able to break into the conversation of any called line if that line is busy. To preserve privacy, a brief warning signal sounds on the called line before the interruption becomes effective. The PAX system is also used for broadcasting taped messages. Use of this device disables, or locks out, all other lines for emergency purposes.

In noisy areas, projective-type loudspeakers are used to overcome ambient noise levels as high as 115 dB. For communication from such areas, noise shielding "hear here" booths are installed. Volume controls are provided for bringing the PAX audio levels to an acceptable level for personal comfort.

3. A dial automatic telephone system (DATS) is similar to the PAX system except that it is owned and operated by Washington Public Power Supply System (WPPSS). It is connected to the various power dams and electrical substations in the Pacific Northwest via a microwave system. There are 15 handsets in the 100-N area as follows:

   - o 105-N control room supervisor
   - o 105-N communications console
   - o 109-N water laboratory
   - o 109-N heat exchanger roof

- o 109-N condensate return valves
- o 182-N high lift pumping station
- o 184-N auxiliary steam generating plant
- o 151-N high voltage electrical switching yard
- o A and B shift manager's office
- o C and D shift manager's office
- o 105-N electrical shop
- o 105-N mechanical shop
- o N Plant liasion office
- o Director, Reactor Operations Division
- o N Plant Operations section manager

DATS, being operational rather than commercial, has a somewhat better reliability factor than GPT. It has an 8-h battery backup and uses carrier systems or microwave instead of aerial lines so that it is less susceptible to weather outages. The 200-line switch is located in the 185-N (WPPSS generating facility building) so that calls between N Plant and 185-N do not have to go through circuitous routes for completion.

4. The emergency communication telephone (ECT) has one handset at the communication console in the 105-N control room with the other set at guard station 4. These instruments are hard-wired together, so that when communication is desired no line-finding is necessary at the 100-K area switch.

5. The contaminated area communication system (CAC) is a two-way audio intercommunication system between the 105-N control room and locations inside the 105-N building. It consists of master and remote stations with most of the remote stations being inside contaminated areas. The capacity of this system is a combined total of 57 master and remote stations. The main distribution panel is located in room 6 of the 105-N building.

When a master station calls a remote station, the remote station is able to maintain a communication "hands free" so far as the caller at the remote station is concerned. When a remote station calls a master station, the remote station switch must be depressed which will ring the master station. When the master station response button is actuated, the ringing at the master station will silence. The master station operator then must ask the remote station to identify. After the remote station has identified by station number, the remote station caller must release the calling switch and may talk "hands free".

The CAC master stations are completely self-contained with cables, connectors, speakers, microphones, amplifier, power supply, volume controls and station selector switches for calling remote and other master stations. Master stations control the volume of incoming and outgoing speech.

CAC remote stations are completely self-contained units designed for operation without a separate power source, and are wired with four remote-call-to-master pairs of wire. The stations include a combination speaker-microphone, four large pushbuttons, cable, terminal boards and a water repellent enclosure. The four pushbuttons permit the remote station to call, by voice, the four master stations of the system. A master station legend is provided at each remote station.

6. The heat dissipation contaminated area communication system (HCC) is a two-way audio intercommunication system between the 105-N control room and locations in the heat dissipation building (109-N principally) contaminated work area. The HCC system consists of four master stations and up to 50 remote stations. The HCC system main distribution panel is located in room-6 of the 105-N building. In all respects, it is similar to the CAC system described in five above.

7. The water plant communication system (WPC) is an audio intercommunication system, providing service between the 105-N control room, all heat dissipation buildings and water handling and treatment

buildings, is similar in type and configuration to the CAC system of five above. It has 8 master stations and a capacity for 60 remote stations.

8. The maintenance communication system (MC) is a loud-speaking, all master, common talking system. It consists of 11 circuits, each one having 3 voice channels. The 11 circuits are: instrument, electrical, elevator, control rods and balls, television, gas leak detection, pumping plant, heat exchangers, river pump house, turbo-generator and water treatment.

   Except for four stations, which are panel mounted, all are portable units. Each station has volume control. MC outlets are installed in most work locations. Cables for all MC circuits connect to a panel at the 105-N communication console and can be patched together. The system is powered by power supplies located at the communications console and at other buildings.

9. The elevator communication system (ECS) is a common amplifier, inter-communication system with stations at work areas associated with fuel charge-discharge operations. These locations are: graphic test panel, communications center, W elevator and metal charging machine, C elevator, D elevator, front ready room, rear ready room, fuel discharge pit, metals inspection area, inlet valve operating gallery, rear view window, left transducer room, right transducer room, equipment room 30, room 503, instrument room 180. Each station consists of at least one microphone and one speaker assembly. Some stations consist of more than one microphone and more than one speaker. Electronic mixers are used at stations having more than one microphone. Except for the mixers, all electronic equipment is grouped at the communications rack assemblies located in room 6 of 105-N.

   A graphic test panel, simulating the entire system, is located at the communications rack assemblies adjacent to the ECS electronic components, to facilitate trouble shooting and adjustment of the system.

One ECS station is located at the graphic test panel for system test and adjustment purposes.

A voice-operated relay facility was built into the ECS system to permit effective, noninterfering, voice-controlled operation of all stations. With initial use, it was discovered that the voice-operated relay facility was unacceptable for operational use, chiefly because the varying ambient noise levels at the stations caused intermittent spurious actuation of the voice channel. The voice-operated relay facility was locked out, and the system remains operable with each station having a push-to-talk microphone. The speakers, at a station at which speech originates, are muted to prevent howling from acoustical feedback. Each speaker is equipped with a volume control switch.

10. The ball recovery communication system (BRS) is similar to the ECS system except that it provides communication between ball recovery areas. These locations are: graphic test panel, communications center, top of unit, ball hoist loft, ball control panel PC3, door 506 in room-10, under unit, room-4 and the front ready room. All statements, regarding the effectiveness of the ECS system, are applicable to the BRS system. The BRS system has more stations in Radiation Zones which include the top of the unit and under the unit; these are critical to effective communication in safety ball system handling situations.

    The use of radio transceivers linked with the base station has been helpful at filling in the communication requirements. Also an audio pick-up device has been used under the unit so that during ball drop operation, operations personnel are not required to stand by in the relatively high radiation zone to listen for the sounds made as the balls drop through, but instead can wait in an outer room and listen through the audio pick-up device. From that outer room, they can use BRS or radio transceivers to complete the communication link.

11. The emergency communication system (EMC) fulfills specified requirements (Ref. 3) by use of a mix of subsystems. The details of the

netting of this communication system are shown in Figure 9.5-1. This figure is a Venn logic diagram wherein intersection of areas drawn implies two-way communication in accordance with criteria (Ref. 3). The subsystems are: sound-powered telephone subsystem (DLS), radio transceiver subsystem and PAX paging subsystem.

The DLS subsystem is a common talking, common signaling telephone system. The original system was supplied by Stromberg-Carlson, manufacturers of industrial sound-powered telephone systems among other audio devices. These units employ a transducer in the microphone to convert the sound power from the voice directly to electrical signals without requiring electrical power input with sufficient converted power to carry up to 20 miles to a conventional telephone receiver. These units require a two-pair cable for connection between sets--one pair for signaling and one for voice communication. Signaling is achieved with a hand-crank magneto at each station. Cranking the magneto produces a penetrating howling sound at all stations except the one sending. Where multiple stations are on the same line, a code similar to that used on rural telephone systems years ago is used--a system of long and/or short rings to identify a called station.

The sound-powered nets, which accomplish one-half of the required emergency communication circuits, are:

Net	Between
1	181-N, 182-N, 163-N
2	182-N, 163-N, 184-N, 109-N, 181-N
3	105-N, 163-N
4	105-N, 182-N
5	105-N, 181-N
6	105-N, 109-N
7	105-N, 184-N
8	105-N, WPPSS
9	105-N, Fuel Storage
10	105-N, 153-N, 151-N

Net	Between
11	105-N, Critical Equipment room
12	105-N, Guard station 4

The radio transceiver (TPR) subsystem consists of 19 portable General Electric transceivers, 2 base stations and 5 audio remote control units for use with the base stations. Eighteen of the transceivers are GE PE series with from 2 to 5 watts RF power output. One is a portable or automobile mounted two-way radio of the GE Porta-Mobil II series with 20 watts RF output. The radio tranceiver locations which provide the other half of the EMC required circuits, are:

Location	Number of Transceivers
105-N control room	1
Critical equipment room	1
153-N low voltage switchgear	1
151-N high voltage switchgear	1
Fuel storage area	1
WPPSS	1
Guard station 4	1
182-N water pumping plant	2
109-N heat exchanger building	2
163/183-N water treatment plant	2
184-N backup steam generating plant	2
181-N river pump plant	1
1100-N Manager, N Plant Operations	1
1101-N Director, Reactor Operations Division	1

Two transceivers are located at the places indicated above so that the chief operator can communicate from his local center, not only with the other centers but with personnel under his direct control who he may send outside the control room to perform activities to alleviate a postulated emergency situation. The mobile transceiver is for use

principally by Radiation Monitoring personnel on missions outside the immediate 100-N area.

One of the two base stations is located at the 51-ft level of 105-N building serving as an excellent radio/transmitter for communicating from the control room in 105-N to points up to 30 miles away. The other base station is located at Priest Rapids Dam and is used for the purpose of monitoring and calibrating of the radio telemetry link which transmits level and temperature information for command, control and recording between 105-N control room and the Priest Rapids Dam.

The audio remote control devices are used in conjunction with the base stations so that they may be keyed or monitored from remote positions connected to the base station by a telephone pair. The remotes are located at the 105-N control room, the instrument and electrical maintenance control offices, the mechanical maintenance control office, the critical equipment room and the radiation monitoring office. With this control equipment operating through the base station, calls anywhere in radio line of sight of the base station, which are on the emergency frequency of 167.85 MHz, may be monitored and effective command and control maintained.

The portables and base stations operate at the assigned frequency of 167.85 MHz using narrow band frequency modulation (FM). The hand held portables are fitted with 6-in long spring antennae in lieu of the longer, more cumbersome quarter wave dipole. For use in noisy areas up to 115 dB, headsets and boom mikes are available. Tone coded squelch, called "Channel Guard" is used in the transceivers and base stations so that they may fit in with other users of the assigned radio frequency described in number 13 below.

The emergency use transceivers and base stations are tone coded "A" so that when the selector switch is placed in the A tone position, they will receive only messages from other transceivers coded A and from the control room locations which are normally switched to the A tone code

position for communication with the A coded group of transceivers. By this means, the emergency use receivers encoded A will not have the distraction of monitoring all the routine maintenance and operational calls. Conversely, the B group set up for maintenance and operational traffic, will not hear calls on emergency tone coded Channel A. Therefore, users of transceivers on routine work on B tone must monitor the RF channel before transmitting to make certain they are not interferring with an emergency transmission.

If any transceivers which are on A for an emergency use require contact with users set on B tone, they may switch to B tone and contact those users. The base stations are tone coded A only. Therefore, if a B user requires the greater coverage capability possible by use of the base stations, he must obtain permission from the Control Room Supervisor prior to going to A tone. Priority is yielded always to the emergency user by this method.

The PAX paging subsystem uses paging facilities of the PAX exchange with the additional feature of an uninterruptible power source. In the event of failure of the ac power, the PAX exchange will switch over to receive power from an inverter sourced from emergency battery supplies. Some of the audio power amplifiers will automatically switch from the usual offsite ac to the alternate onsite ac source in the event of offsite ac failure. Most of the amplifiers have a backup battery supply for a minimum of 4-h use after ac power failure. In noisy areas, the PAX paging gain is set high enough to be heard over the ambient noise level.

12. The radio paging system, (RPS) used throughout the Hanford Reservation, provides a means for contacting key personnel designated to be on call on weekends and holidays. If someone on the Hanford Reservation (served by GPT) requires the attention of a person off duty, within about 50 mi, wearing a small "page boy" belt receiver, he may dial 84 plus the two number code for the particular person on the GPT. This call goes to the General Telephone Exchange Central Office, from there to a GE

paging terminal in the Federal Building which normally transmits via UHF to the Motorola paging transmitter located on Rattlesnake Mountain. The Motorola paging transmitter calls the page having the indicated identification number alerting the wearer by causing the paging receiver (page boy) to emit a beep. The wearer then pushes a button on the page boy and receives the short message such as "call telephone number XXX XXXX."

If contact with an individual wearing a page boy is desired from some location offsite, the 105-N Control Room Supervisor or the Hanford Patrol Emergency Office can take the message and send it via the system. There exists an alternate means of transmission to the Motorola paging transmitter on Rattlesnake Mountain. If the GE paging terminal in the Federal Building does not operate, a standby controller which will operate through microwave facilities can be used.

13. A radio transceiver communication system (RTS) - In addition to the radio transceivers used for emergency operational purposes described in 11 above, there are three similar transceivers dedicated to routine operational and maintenance purposes. They are General Electric PE model hand-held transceivers (TPR) operating with narrow band FM at 167.85 MHz, with an RF output power of 2 watts. Belt clips and carrying cases are available for convenience and protection of the transceivers in use. When used in Radiation Zones they are wrapped in plastic bags for protection against contamination. For use in noisy areas up to 115 dB, headsets and boom mikes are used. They are fitted with Channel Guard tone coded squelch, which is normally set in "B" tone as discussed in paragraph 11 above.

14. Patrol security communication net. Two radio transceivers (TPR) are maintained in readiness at N Plant for emergency communication purposes with the Hanford project security communication system. Through these two transceivers, which are battery operated and portable, two-way communication is maintained for postulated emergency purposes between

the 100, 200 and 300 areas patrol headquarters, fire departments, patrol cars and the Patrol Emergency Office.

9.5.2.3 System Evaluation

The reliability of N Reactor's communications system can be evaluated by examining two general groupings of systems used: cabled and wireless. There are five varieties of cabled systems used at N Reactor. These cabled systems can be categorized, from the most reliable to the least reliable, as follows: (1) sound-powered systems requiring no electrical power nor electronic amplifier components, (2) dedicated telephone system with backup ac and/or battery power supplies (eg. PAX and DATS), (3) wired systems having central or distributed amplifiers with no backup power supply (eg. WPC, MC and ECS), (4) wire systems similar to (3) but located in Radiation Zones (eg. CAC, HCC and BRS) and (5) general plant phone.

There are four independent wireless communications systems used in 100-N area. These systems are essentially of the same reliability with the possible exception of the page boy system. The four systems are: (1) the radio transceiver subsystem portion of the N Reactor EMC, (2) a radio transceiver communication system for N Reactor Maintenance and Operations, (3) the security patrol communications net serving for onsite and offsite emergency communications and (4) the radio paging system used for contacting key personnel offsite.

The radio transceivers designated for routine operational and maintenance use have the same factor of reliability as those designated for emergency. In the event of the requirement arising for emergency communication, the routine users will maintain radio silence unless they are required by the control room. Since they operate interchangeably with the emergency radios, they may be brought into the system as required.

Reliability of the radio paging system is enhanced via its backup transmission paths, the Federal Building in Richland, Washington and the transmitter on Rattlesnake Mountain. 100-N Area is one of several locations on the Hanford Reservation using this system.

Reliability of N Reactor power supplies is fully discussed in Chapter 8. The following summary provides identification of each communication's system normal and backup power supplies:

System Abbreviation	Normal Power Supply*	Backup Power Supply(ies)*
GPT and ECT	offsite, ac	offsite battery (dc)
PAX	A Bus, ac	B Bus, ac; battery (dc)
DATS	offsite, ac	offsite battery (dc)
CAC, HCC, WPC, MC, ECS, BRS	A Bus, ac	
EMC	A Bus, ac	B Bus, ac; battery (dc) sound powered subsystem requires no power
Radio Base Station	A Bus, ac	
TPR (3 systems)	Battery, dc	
RTS	Transmitters, ac,	Battery, dc, on receivers

Chapter 15 describes the postulated emergency situations that could impact on N Reactor. Due to the multiple methods of cabled and wireless communications, the availability of backup power and successful operational experience with these systems-loss of all communications, resulting from any of the accident conditions (including postulated tornadoes, severe earthquakes, missiles or fires) is not credible.

9.5.2.4 System Inspection and Testing Requirements

The communications systems used at N Reactor have proven to be reliable. Most of the systems are tested daily via normal operational use. Those systems, or portions of systems, not used daily are functionally tested once each month. In addition, the Westinghouse Hanford portion of the communications system is covered by an Equipment Maintenance Standard (EMS), "Critical Communications Systems", number 606 (Ref.4). The EMS covers

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*A Bus ac is received from offsite sources and is commonly referred to as "offsite power;" B Bus ac is received from the onsite plant turbine generator and is referred to as "onsite power."

functional testing of sound and the power supply. Functional testing is accomplished once each 3 months.

System maintenance follows any component failure. EMS tests may reveal incipient off-standard component conditions, prior to failure, and corrective maintenance is then either accomplished on-the-spot or immediately upon test completion.

9.5.3 Lighting Systems

Normal ac lighting is furnished in all areas of the 100-N complex to maintain sufficient illumination levels for the safe performance of routine operating and maintenance functions. In case of failure of the ac power, emergency dc (and some ac) lighting is provided for personnel safety in most areas with the exception of a few locations which are very rarely occupied.

Lighting systems are not required for the safe shutdown of the reactor.

9.5.3.1 Design Bases

The criteria for lighting levels in the various buildings and areas of the plant were based on illumination levels recommended by the Illumination Engineering Society at the time of N plant construction and a maintenance factor of .6 to.8. Table 9.5-4 gives the average illumination levels in foot candles for the various plant areas as specified in the design criteria (Ref. 1). There are no restrictions on the use of mercury or its components at any locations within the N Reactor complex. Four plant areas are specifically required to be provided with emergency dc lighting (other than portable lighting packs) which will be activated in the event of loss of ac power. These are 105-N Building Zone I inlet and outlet pipe spaces and the 105-N building reactor control and switchgear rooms. All routinely occupied areas in each of the buildings within the 100-N complex are required to be provided with portable emergency lighting battery packs.

9.5.3.2 System Description

The plant lighting system is shown on Figure 9.5-3. The outside area lighting is a 240V ac system, using two types of lights - 1500-watt

incandescent and 1000-watt mercury vapor In addition, all buildings are provided with outside lights for doors and walkways. These are incandescent lights supplied at 120V ac. Normal inside lighting systems are 277V ac, 240V ac and 120V ac. Lighting loads are divided evenly between two independent power systems - offsite power and onsite 13.8 kV turbine generator. Most of the lighting for the 105-N (reactor building) and 117-N building is 277V ac fluorescent. Office space is in general 120V ac fluorescent, with 120V ac incandescent lighting for some corridors and stairwells. The top of the reactor is lighted by 277V ac metallic vapor lights, while the area over C and D elevators are lighted by 240V ac mercury vapor lights. All the other buildings in the 100-N complex have 120V ac lighting systems. In general, high bay lighting is mercury vapor, office areas are fluorescent and low bay lighting is either fluorescent or incandescent.

Emergency lighting is provided by three separate 36V dc batteries and chargers (one for each of the following areas) which are automatically energized whenever there is a loss of normal ac power in the area covered by the particular 36V dc system as required by Reference 1.

1. 105-N building Zone I inlet pipe spaces
2. 105-N building Zone I outlet pipe spaces
3. 105-N building reactor control room and switchgear room

The wiring for these three systems is in totally separate raceways from each other and from all normal lighting and power circuits. All of the 36V dc lamps are incandescent. Emergency lighting is provided in the 182-N and 181-N buildings by the emergency diesel generator set (480V ac) located in the 182-N building. Emergency power panel E, in the 182-N building, is fed from normal ac power or from the diesel generator through an automatic transfer switch which operates on a loss of normal power. Emergency power panel E feeds emergency lighting panel EE in the 182-N building and emergency panel E in the 181-N building, providing emergency energization of normal lighting circuits upon loss of normal ac power in both the 181-N and 182-N buildings as required by Reference 7.

The above mentioned emergency lighting systems will allow access to all safety-related equipment during a loss of normal ac power. In addition to the above mentioned emergency lighting systems, all buildings are required to have emergency lighting packs, which consist of a 12 volt battery, a charger and control unit, and incandescent automotive type lamps for personnel safety during a loss of normal ac power. These packs are energized by the normal lighting power supply with each unit operating independently and coming on whenever the control unit senses a loss of normal voltage. Similar units are used at doors to light exit signs. These units are placed strategically throughout the plant and are required to have sufficient battery capacity to provide adequate illumination for a minimum of 30 min, as described in Reference 6.

9.5.3.3 Design Evaluation

The normal plant lighting provides ample illumination for the conduct of the routine functions required during plant operation. It permits clear definition of all fixed and moving hazard situations, adequate visuals checks of equipment status and evacuation of the plant or from specific areas of the plant if such becomes necessary.

The emergency lighting systems, provided in the vital plant locations noted above, are independent of any other system and will function upon loss of all sources of power to the plant. The dc batteries which supply the four systems, will provide lighting within the area for a minimum of 3 hours. The portable 12 volt battery packs, which are installed throughout the plant, are available to plant personnel in the event of a total loss of power and provide lighting adequate for safe reactor shutdown and the evacuation of personnel.

9.5.3.4 Tests and Inspections

AC lighting systems require no testing since they normally remain energized. All ac lighting is inspected weekly and relamped as necessary to maintain proper illumination levels. The three 36-volt emergency lighting battery-charger units and the 182-N building emergency diesel generator set are tested and inspected on a routine and continuous basis as prescribed by

current Equipment Maintenance Standards requirements (Ref. 7). The 12-volt power pack units are tested as required by Maintenance Planning and Scheduling Report N-48. Typical routine tests and inspections that are made on the emergency equipment are summarized in Table 9.5-5.

9.5.4 Diesel Fuel Storage and Transfer System

The diesel fuel storage and transfer system (DOS) shown in Figure 9.5-4, maintains an adequate supply of diesel fuel oil in storage and supplies it to seven ECCS and fog spray diesel engines located in the river pump house (181-N bldg.) and the high-lift pump house (182-N bldg.) and the diesel powered fire pump and the diesel-driven electrical generator in the 182-N building. The system also supplies ignition oil for the firing of the boilers in the plant service power house (184-N bldg.).

The 105-N lift station sump pump diesel engine has an independent underground storage which is not connected to the DOS.

9.5.4.1 Design Basis

The DOS is designed to supply diesel fuel oil to diesel engine power emergency equipment under normal* and postulated accident conditions. The equipment and piping are laid out such that the failure of any single active component will not result in a loss of more than one diesel engine on the ECCS system or the fog spray system. Redundant engines are provided on each of these systems.

The total storage capacity of the DOS is 465,700 gal. The diesel fuel is contained in four 105,000-gal bulk storage tanks and five day tanks (three - 15,000 gal and two - 360 gal) which are located close to the equipment they service. The supply lines to and from each day tank are sized to provide a 3 gal/min gravity flow to each diesel engine supplied by the tank.

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*The plant service boiler system provides oil-fired boiler steam to the plant turbine generator during reactor startup and shutdown period. See Section 10.4.11.

The DOS day tanks are provided with an automatic "keep full" device which is operated by a level sensor within the tank.

The supply lines from the DOS day tanks to the diesel engines are designed to allow supply from any other tank which might be removed from service for maintenance or repair. Each day tank is also provided with an unloading facility which permits filling of the tank from a fuel oil transport truck.

The design conditions of the DOS are as follows:

Normal operating pressure	15 lb/in^2 (gage)
Normal operating temperature	80°F
Design pressure	75 lb/in^2 (gage)
Design temperature	120°F

The piping is fabricated and installed in accordance with the codes, standards and specifications of the following organizations:

- o American Standard Association (ASA)
- o American Water Works Association (AWWA)
- o Hanford Works Standards (HWS)

The specific application of these codes, standards and specifications is detailed in References 8 and 9.

The piping hangers and supports are designed to support the pipes when they are subjected to horizontal seismic loads of 0.5 g and are spaced to eliminate vibration.

Piping for the 105-N lift station sump pump diesel fuel system is fabricated in accordance with ASTM specifications A106, A53, A181, A338 and A388. The underground fuel tank is fabricated of 1" carbon steel plate.

9.5.4.2 System Description

The DOS consists of an unloading station, four storage tanks, two transfer pumps, one recirculation pump, oil purification centrifuge and five "day" tanks. See Figure 9.5-4.

The unloading stations for No. 6 fuel oil and diesel fuel oil are in the same structure (166-N bldg.). There are four tank truck unloading stations and six tank car unloading stations available.

Four diesel fuel storage tanks are located east of 166-N building and adjacent to the No. 6 fuel oil storage tank. These tanks are 30 ft in diameter, 20-ft high to the roofs edge and have a storage capacity of 105,000 gal each. The bottom 12 ft of the tanks are provided with two layers of fiberglass insulation. Each tank is equipped with a water trap to prevent buildup of water within the tank.

The average use rate of diesel fuel is approximately 60,000 gal/yr. Therefore, during normal operational and shutdown periods only two of the four tanks are utilized for storage purposes to minimize long term storage which might result in instability of the diesel fuel. A foam fire protection system for the storage tanks is described in Section 9.5.1. A dike, 8-ft high with dimensions of 215 ft by 74 ft, is provided to confine any lost fuel from a possible tank rupture or leakage.

The two transfer pumps associated with the system are located in the basement of 166-N building. They are positive-displacement, rotary type with a 20 gal/min rating at 50 lb/in^2 (gage). The motors are 1.5 hp induction type and can be powered by either offsite or onsite ac power. The pumps are used to transfer diesel fuel from the unloading station to the storage tanks, to the centrifuge from the storage tanks and to the storage tanks from the centrifuge.

A DeLaval centrifuge is available in the basement of the 166-N building for use in separating dirt, sludge and water from the diesel fuel if it becomes necessary to do so.

A gravity flow of diesel fuel is provided from the diesel fuel storage tanks to the five day tanks located at the 182-N building.

Three day tanks are underground tanks with a capacity of 15,000 gal each. Two day tanks are above ground tanks with a capacity of 360 gal each. These are all located adjacent to the 182-N building. An unloading facility for direct auxiliary fill from a transport truck is provided for each tank.

A gravity flow of diesel fuel to the nine emergency diesel engines located in the 181-N and 182-N buildings is provided from the three underground day tanks. Day tank No. 1 supplies diesel fuel to high-lift diesel No. 3 and fog spray diesel No. 1. Day tank No. 2 supplies low-lift diesel No. 2, high-lift diesel No. 2 and fog spray diesel No. 2. Day tank No. 3 supplies low-lift diesel No. 1 and high-lift diesel No. 1. The two above-ground day tanks supply the emergency generator and fire pump diesel engines, respectively. See Figure 9.5-4.

The 105-N lift station pump diesel engine fuel is stored in a 5000-gal underground tank which is filled directly from transport trucks. Oil is circulated through a loop containing a small (3-gal) reservoir. The reservoir supplies the diesel engine by gravity feed. The circulating pump is electrically-driven by the engine generator.

Ignition oil for the firing of the Foster-Wheeler and the two C-E boilers located in the 184-N building, is supplied by the diesel fuel system. A 2500-gal capacity day tank is located underground just outside of the 184-N building for this purpose. The tank is supplied with No. 2 diesel fuel from the 166-N storage tanks via the transfer pumps located in 166-N. The supply line to the tank is labeled Diesel Oil Supply (DOS), and the lines to the boiler ignition pumps are labeled Ignition Oil Supply (IOS). The fuel level in the day tank is constantly monitored and recorded in the 184-N building.

The 182-N day tanks are provided with a mechanical fill valve that is operated by a level sensor. Isolation valves for each day tank and each diesel are provided for maintenance access. The supply lines from the three

underground day tanks are interconnected such that any one of the tanks can be removed from service for maintenance without interrupting flows to required systems.

9.5.4.3 Design Evaluation

Electrical power to the DOS transfer pumps is supplied through two separate power sources, so that a single failure will not result in a loss of pumping capacity. A single failure analysis of the system is given in Table 9.5-6 for loss of onsite or offsite power. The transfer pumps do not have a safety related function.

A minimum of 100,000 gal of diesel oil is required to be available in storage within the system. This amount of diesel fuel is sufficient to operate all seven emergency diesel engines at full load for 7 days. Assuming the area is attended following a LOCA, the number of diesel pumps operating can be reduced to four (one low-lift, two high-lifts and one fog spray) and still provide ample coolant to the reactor. In this case, the minimum diesel oil supply required to be maintained in storage would last in excess of 10 days.

In addition to the diesel oil in storage, at least three sources of diesel oil are available locally, and each is capable of delivering diesel fuel to the site within 1 day. Oil is also available by truck or rail from within a 200-mile radius which includes the cities of Seattle and Spokane, Washington and Portland, Oregon.

All internal surfaces of the diesel oil storage tanks have been painted with two coats of sealer to guard against corrosion (Ref. 13) except for the lift station sump pump diesel engine fuel storage tank, which is covered externally with a moisture and flame resistant asbestos coating. The high compatibility between the diesel oil and the systems components has kept corrosion to a minimum during the 14 yr of plant operation.

Winter freeze protection is provided by a layer of fiberglass insulation around the outside of the storage tanks to a level of approximately 12 ft and by blending the normal diesel oil supply (No. 2 diesel oil) with winter

grade (No. 1 diesel oil) diesel oil. The three day tanks supplying the emergency diesel engines, are placed underground to provide additional freeze protection. In addition, each of the four diesel oil storage tanks is equipped with a water trap which is routinely drained, thereby preventing a buildup of water within the tanks.

The system is provided with valves and check valves in locations which permit the tanks so that maintenance or repair work can be performed without taking the system out of service.

The environmental circumstances which may affect the ability of the DOS to perform its functions, are discussed in Sections 3.7 Seismic Design and 3.3 Wind and Tornado Loadings.

9.5.4.4 Tests and Inspections

Inspection and testing of the DOS is carried out on a routine and continuous basis as outlined by the N Plant EMS program. The following lists typical tests and inspections which are routinely performed on the system:

- o Visual inspection of tanks and piping for leaks
- o Internal inspection of day tanks
- o Sample oil - inspection for water; analysis for Cetane Index, temperature, viscosity, pour point, water and sediment
- o Internal inspection of storage tanks
- o Inspection and calibration of storage tank level indicators

The inspection and test procedures are noted in EMS 201 and EMS 621 (Ref. 10).

9.5.4.5 Instrumentation Application

The instrumentation required for the DOS to adequately perform its function is minimal. The storage tanks and the day tanks are provided with level indicators which constantly note the fuel level within the vessel. Each of the day tanks is equipped with a mechanically controlled automatic fill valve which maintains the fuel oil level at the "full" mark.

9.5.5 Diesel Engine Cooling Water Systems

The function of the diesel engine cooling water systems is to provide necessary cooling for the various diesel engines while they are operating. Three different types of systems are employed; (1) the diesel engines which power the high-lift pump house (182-N building) generator and the 105-N lift station pump, (2) the fire pump diesel engine located in the 182-N building and (3) the system for the other seven emergency diesel engines in the river pump house (181-N building) and the 182-N building.

9.5.5.1 Design Bases

Diesel generator cooling system - The diesel engine which drives the 182-N building emergency generator was procured as a package unit which included the engine cooling system, as designed by the vendor, to be adequate to properly cool the 230 hp (1800 r/min) engine when operating on a continuous basis. The cooling system is a water recirculation type which makes use of an air-cooled radiator and fan to dissipate the heat from the coolant (Ref. 12).

Diesel sump pump cooling system - The diesel engine and associated cooling system which drives the 105-N lift station sump pump was designed and procured from a single vendor. Cooling is accomplished by a conventional coolant (water/antifreeze) recirculation, radiator and self-powered fan system. Because the housing structure is unheated and well-ventilated the engine cooling system is equipped with a thermostatically-controlled electrical heater and circulating pump to facilitate starting. Coolant temperature is maintained at 80 F during nonoperating periods. During operation, air passing through the radiator is discharged outside by a duct through the west wall.

Fire pump diesel engine cooling system - The 205 hp (1775 r/min) diesel engine which drives the emergency fire pump located in the 182-N building, was procured as a package unit which included an adequate cooling system designed and provided by the vendor. The cooling system is a recirculation type with the heat from the coolant being dissipated in a water-cooled heat exchanger.

The minimum water flow required is 50 gal/min at 85 F and a maximum pressure of 25 lb/in^2 (gage) (Ref. 11).

Emergency diesel engine cooling system - The cooling system for the seven 1600 hp (750 r/min) diesel engines which drive the emergency pumps (2 low-lift pumps in the 181-N building, 3 high-lift pumps and 2 fog spray pumps in the 182-N building) are designed to meet all of the cooling requirements of the engines on a continuous operational bases. The system includes the equipment and piping necessary for cooling the engine jacket recirculated water, lube-oil and gear-box oil.

The design conditions for the diesel engine cooling water system are as follows:

- Normal operating pressure - 20 lb/in^2 (gage)
- Normal operating temperature - 150 F
- Design pressure - 75 lb/in^2 (gage)
- Design temperature - 200 F

The piping is fabricated and installed in accordance with the codes, standards and specifications of the following organizations:

- American Standards Association (ASA)
- American Water Works Association (AWWA)
- Hanford Water Standards (HAPO)

The specific application of these codes, standards and specifications is detailed in (Ref. 8 and 9).

The piping is hung and supported such that it will withstand horizontal seismic loads of 0.5 g with the hangers and supports located to eliminate vibration.

9.5.5.2 System Descriptions

The emergency generator diesel engine is provided with a closed-circuit, air-cooled radiator type cooling system which is capable of dissipating the heat developed by the engine when it is operating at full load in ambient temperatures ranging up to 125°F. The cooling system equipment consists of a radiator, a lube-oil cooler, thermostats, a fan, a water pump and a lube-oil pump all driven by the engine. Figure 9.5-5, part A., shows the cooling system.

When the engine is operating, the water pump circulates the coolant through the engine cooling jacket, the lube-oil cooler and the radiator where it is cooled by air which is moved through the radiator by the fan. The fan speed is 2300 r/min, and the coolant flow rate is approximately 100 gal/min at an engine speed of 1800 r/min.

The 105-N lift station sump pump diesel engine, except for its size, 175 hp, and speed, 1600 r/min, is very similar to the generator diesel drive described above. The coolant is an antifreeze and water mixture.

The fire pump diesel engine cooling system is of the closed circuit type and includes a circulating water pump driven by the engine, a heat exchanger, a lube-oil cooler and an engine jacket temperature regulating device. The system is shown in Figure 9.5-5, part B.

During engine operation, the cooling water is circulated in the system by means of a centrifugal-type pump which is mounted on the engine and belt driven by the engine. The water circulated through the engine jacket, the heat exchanger, the lube-oil cooler and back to the pump. A thermostatic bypass which automatically controls the engine jacket temperature by allowing some or all of the water to either bypass or circulate through the heat exchanger, is included in the system.

The heat exchanger is provided with raw water from a line which orignates just downstream of the fire pump discharge. The raw water piping between the fire pump discharge and the heat exchanger includes a normally open dc

operated solenoid valve, a strainer and a manual shut-off valve. When the engine is standing by, the solenoid valve is closed, opening when the engine starts or if a dc power loss should occur. A bypass line with a manual flow regulating valve, is installed around the strainer, solenoid valve and regulating valve. A pressure regulating valve is provided at the strainer outlet to protect the heat exchanger against the fire pump high discharge pressure. The pressure is reduced to 25 lb/in^2 (gage).

The diesel engines on the emergency low-lift, high-lift and fog spray pumps are cooled by a closed system at each engine which circulates coolant to the engine jacket and the lube-oil cooler. This coolant, filtered water, is passed through a heat exchanger in which the excess heat is transferred to water supplied from the circulating raw water system.

Low pressure filtered water is supplied to individual 30-gal expansion tanks which are the sources of makeup water to each low-lift, high-lift and fog spray diesel engine jacket cooling system. The expansion tank level is controlled by a level switch which actuates a solenoid valve in the filtered water supply line when water is needed. The tank and piping also provide means for jacket system venting, overflow and draining. The temperature of the jacket coolant is controlled by a thermostatic three-way valve that permits all or part of the water to bypass the jacket water heat exchanger.

The system includes a second heat exchanger which serves to cool the gear-box oil. This oil is routed directly to the heat exchanger, where the heat is dissipated to water supplied by the circulating raw water system. This system is shown in Figure 9.5-6.

9.5.5.3 Design Evaluation

The system that provides the cooling for the diesel engine which drives the 182-N emergency generator, is an integral part of that engine and requires no outside support to ensure its operability. The engine, including the cooling system, is maintained in prime operating condition. Frequent testing of the engine over 14 yr of operation has proven that the cooling system can be relied upon to perform its function without fail.

The engine is provided with automatic safety controls which will actuate both visible and audible alarm systems in the 182-N building control room in the event that the jacket water or the lube-oil temperatures reach a high temperature set point. A further rise in the jacket water or lube-oil temperature will result in an automatic shutdown of the engine.

The 105-N lift station sump pump diesel engine is cooled by recirculating the cooling water through a fan cooled (air) radiator all of which is an integral part of the engine. Because this engine is maintained in prime operating condition, is tested on a monthly basis and is a proven commercial product, it can be relied on to perform when needed. Secondly, its need is not critical on an instantaneous basis, rather, several hours could elapse before its need is critical. Thus, sufficient time is provided for manual starting should it fail to start automatically.

The engine is provided with automatic safety controls which will actuate visible indicators on the lift station engine control panel and a visual and audible alarm in the 105-N control room should either the cooling water or lube-oil temperature reach a high-temperature setpoint or the engine exceed an overspeed trip point. A further rise in either temperature would result in automatic shutdown of the engine.

The fire pump diesel engine is cooled by a closed circuit type system which is actually a part of the engine itself in that the water pump, the heat exchanger and the lube-oil cooler are built into or attached directly to the engine. The cooling system, along with the engine, is maintained in top operating condition. Frequent testing of the assembly during 14 yr of plant operation has proven the reliability of the system.

The raw water supply to the heat exchanger takes off immediately downstream of the fire pump discharge. As a result, the pipe run to the exchanger is very short and is fastened directly to the engine. Whenever the pump is running, cooling water for the heat exchanger is available.

The engine is provided with automatic safety controls which will actuate both visible and audible alarms in the 182-N building control room, should the jacket water or lube-oil temperature reach a high set point. A further rise in jacket water or lube-oil temperature will result in automatic shutdown of the engine.

The cooling systems used on the diesel engines which drive the emergency low-lift, high-lift and fog spray pumps, are designed by the vendor to provide adequate cooling of the engine when operating at full load on a continuous basis. Experience with the engines gained over 14 yr of operational testing, standby usage and simulated emergency tests, have shown the engines and their cooling systems to be extremely reliable. The cooling jacket and coolant circulating pump are an integral part of the engine, while the two heat exchangers (one each for jacket water and engine lubricating oil) are mounted immediately adjacent to and on the same foundation as the engine. The pipe runs to and from the heat exchangers are short and well supported.

The cooling water to the heat exchangers can be supplied from a number of sources. The normal supply is from the 181-N building electrical circulating raw water pumps via the low pressure raw water system. See Section 9.2.1. If, when the reactor is shutdown, these pumps are not in service, the supply is obtained from the diesel driven low-lift pumps in the 181-N building via a line which takes off of the diesel powered low-lift discharge header and supplies the low pressure raw water system. The piping is arranged in such a manner that whenever any of the emergency low-lift pumps is operating, the low pressure raw water line is pressurized. A third source of cooling water for the 1600 hp engines in 182-N is available through manual valving from the discharge headers of the pumps driven by the engines.

The cooling system and water supply piping associated with the cooling of the seven emergency pump diesel engines are designed to ensure that no single component failure in the system will cause the loss of more than one diesel engine. A redundant engine is provided for each set of emergency pumps (low-lift, high-lift and fog spray). The loss of a single diesel engine in

each of the three services will not impair the adequacy of the emergency cooling system or the fog spray system.

These engines are provided with automatic safety controls which will actuate both visible and audible alarm systems in the 182-N building control room in the event of high jacket water or lube-oil temperature.

The environmental circumstances which could adversely affect the capability of the seven large diesel engine cooling systems to adequately perform their function are described in Section 3.7, Seismic Design.

9.5.5.4 Tests and Inspections

The testing and inspection of the seven emergency cooling system and fog spray pump diesel engines, the diesel fire pump engine, the 105-N lift station sump pump diesel engine and the 182-N building emergency generator diesel engine, is performed on a continuous and routine basis by plant personnel. The cooling systems associated with these engines are included in the test and inspection program.

The Plant Equipment Maintenance Standards Program (Ref. 10) currently requires that a full operational test of the emergency diesel engines be performed as follows:

- o 182-N emergency generator - weekly
- o Low- and high-lift pumps - prior to each reactor startup
- o Fog spray pumps - monthly
- o 105-N lift station sump pump - monthly

The fire pump diesel engine is not covered by an EMS test procedure. Instead, a timing device automatically starts the engine once each week. This test is essentially identical with that performed on the other engines under the EMS procedures.

The specific EMS procedures which pertain to the testing and inspection of the diesel engines are Nos. 116, 201, 605 and 621.

During these tests, sufficient data to permit an accurate evaluation of the machines performance are taken. The data are analyzed and appropriate corrective action, if required, is taken.

9.5.6 Diesel Engine Starting Systems

The function of the starting systems associated with the several diesel engines which drive emergency equipment in the river pump house (181-N building) and the high-lift pump house (182-N building), is to start the engines and bring them up to operating speed at a rate which will not cause damage to the engine and yet is consistent with the postulated emergency needs.

Three startup systems are used. The fire pump (see Section 9.5.1) and the emergency generator diesel engines, both located in the 182-N building, and the 105-N lift station sump pump diesel engine make use of a dc electric starting system for both normal (or test) starts and for emergency starts. The dc power is supplied by batteries.

The seven diesel engines which power the emergency pumps (two low-lift pumps in the 181-N building, and two fog spray and three high-lift pumps in the 182-N building), are started with a high pressure compressed air system. This system includes an accumulator tank, bottled nitrogen cylinders, piping and the starting control equipment on the engine.

9.5.6.1 Design Bases

The diesel engines which drive the emergency fire pump, the emergency generator and lift station sump pump, are standard units and each was procured from a vendor as a package. This package included the starting system. The starting system is designed by the vendor to be adequate to start the engine either manually, by pushbutton on the engine control panel or automatically. In the case of the generator, an automatic start will result from a loss of normal ac power supply to the 182-N building, the fire pump diesel will start on loss of pressure in the area fire main system and the lift station

sump pump diesel engine will start when sump water level reaches a preset float switch. The engines are equipped with wet type storage batteries which provide the dc power for starting the engines. The batteries for emergency fire pump and emergency generator diesel engines are sized to provide a minimum of four, 15-s duration, diesel cranking starting cycles (Ref. 10 and 11). The batteries for the sump pump diesel engine is sized and tested for four 30-s cranking cycles.

The seven 1600 hp diesel engines which drive the emergency pumps, make use of high pressure air as the engine starting medium. The system is designed to provide sufficient air for at least two engine starts in the event of main air supply loss. The engine starting system is provided with 600 lb/in^2 (gage) air from the air accumulator, by two separate lines - one for manual starts and the other for automatic starts.

The starting air system extends from (and includes) the starting air accumulator to the starting air timing valves on the engine cylinders and includes all of the piping, valves and accessories between. The piping, valves, etc. are fabricated and installed according to the codes, standards and specifications issued by the following organizations:

- American Standards Association (ASA)
- American Water Works Association (AWWA)
- Hanford Works Standards (HWS)

The specific application of these codes, standards and specifications is detailed in References 8 and 9.

The design conditions of the starting air system are as follows:

- Normal operating pressure - 615 lb/in^2 (gage)
- Normal operating temperature - 80°F
- Design pressure - 630 lb/in^2 (gage)
- Design temperature - 100°F

The system piping is hung and supported in such a manner as to survive horizontal seismic loads of 0.5 g and are spaced to eliminate vibration.

9.5.6.2 System Description

The diesel engines which drive the emergency fire pump and emergency generator located in the 182-N building and the 105-N lift station sump pump, are provided with 24-volt dc starting motors as in integral part of the engine assembly. These motors are supplied with 24-volt dc power from heavy duty, lead-acid type storage batteries which are mounted on a rack immediately adjacent to the engine. Heavy duty cables connect the batteries to the starting motor and the control equipment.

The engines can be started manually from the engine control panel which is located alongside the engine. Automatic startup of the fire pump diesel will be initiated if low water pressure (normal pressure - 115 lb/in^2 (gage); diesel pump is actuated at 85 lb/in^2 (gage)) occurs in the area fire system main, the emergency generator diesel engine if the normal building ac power supply is lost and the lift station sump pump diesel engine if the sump water level reaches the 449-ft elevation. The engines will attain full rated speed in not more than 5 s after the start impulse is received.

Figure 9.5-7 shows the electrical wiring for the fire pump diesel engine. This is also typical of the wiring for the emergency generator and sump pump diesel engines.

The high-pressure air system used to start the emergency low-lift, high-lift and fog spray pump diesel engines, is supplied with air from the high pressure air system (CAE) described in Section 9.3. Air at a nominal 600 lb/in^2 (gage) is piped to each individual engine air accumulator which is 24 in. in diameter and 30-in long and contains enough air to ensure at least two engine starts.

A check valve just upstream of the accumulator prevents backflow in the event of a line break. From the accumulator, two lines run to the air valves which starts the engine. One line, in which a manually operated quick opening

valve is installed, goes directly to the starting air valves and is used for manually starting the engine. The second line bypasses the manual valve and includes a Flexflo valve which is controlled by a normally open dc powered solenoid operated pilot valve, for automatic engine starting. Figure 9.5-8 shows this system.

The engines are started by admitting the high pressure air directly into certain engine cylinders through the starting air manifold. The starting air manifold is supplied with high pressure air by either or both of the lines noted above. The air enters the cylinders from the manifold through a timing valve and a check valve. The timing valves are operated by a camshaft which controls the timing of the air admission so that each cylinder receives an impulse in proper sequence. The check valve prevents backflow of air from the cylinder and closes off the starting air passage when the engine is in operation.

Instrumentation constantly monitors the air pressure in the supply lines to each engine. A visual and audible low pressure alarm is received in the 182-N building control room if a low pressure condition should occur. Figure 9.5-8 shows this the starting air system to a single engine.

9.5.6.3 Design Evaluation

The mechanism which starts the emergency fire pump and emergency generator, both located in the 182-N building and the 105-N lift station sump pump, is specifically designed by the vendor to accommodate the requirements of the engine which they service. The system is essentially self-contained in the sense that it requires no support from outside systems to perform its function (other than the starting impulse). An automatic battery charging device is included which ensures that the batteries are maintained fully charged at all times. When the engines are operating, a generator driven by the engine keeps the batteries charged.

These engines, including the starting system, are maintained in prime operating condition. Frequent testing of the engines over 14 yr of plant

operation has proven that the starting systems can be relied upon to perform its function without fail.

The starting system on the large diesel engines which drive the emergency low-lift, high-lift and fog spray pumps, is designed by the vendor to ensure quick and reliable startup of the engines when either a manual or automatic starting impulse is received.

Subsequent to the volcanic ash eruption of May 18, 1980, bottled nitrogen, stored in cylinders, was installed as a backup to the high-pressure air system on the high-lift and fog spray diesels.

The starting air supply system is so arranged that the accumulator on each of the three high-lift diesel engines receives its supply from a different starting air receiver. The same applies to the accumulator on the low lift and fog spray engines. Check valves in the lines prevent loss of air from the accumulator back to the supply system. Each accumulator contains enough high pressure air to ensure at least two engine starts. Isolation of the accumulators is such that the loss of any one accumulator will not result in the loss of more than the single diesel engine served by that accumulator. A redundant diesel engine-pump assembly is provided for each set of emergency pumps (low-lift, high-lift and fog spray) to ensure that failure of one engine to start in each of the three services will not adversely affect the adequacy of the emergency cooling system. The reliability of the air powered starting system on this engine is demonstrated by the fact that since startup of plant operations in 1963, only three single diesel engine startup failures have been documented in the more than 8000 test starts.

The environmental conditions which could potentially impair the capability of the diesel engine starting systems to perform their function are described in Section 3.3, Wind, Tornado and Volcanic Ashfall Loadings; Section 3.5, Missile Protection; and Section 3.7, Seismic Design.

9.5.6.4 Tests and Inspections

The inspection and testing of the seven emergency cooling system and fog spray pump diesel engines, the 105-N lift station pump diesel engine, the diesel fire pump engine and the 182-N building emergency generator diesel engine, is performed on a continuous and routine basis by plant personnel. The starting systems associated with these engines are included in the test and inspection program.

The Plant Equipment Maintenance Standards Program (EMS) (Ref. 10) currently requires that a complete operational test of the emergency diesel engines be performed as follows:

- o 182-N emergency generator - weekly
- o Low- and high-lift pumps - prior to each reactor startup
- o Fog spray pumps - monthly
- o 105-N lift station pump - monthly

The fire pump diesel engine is not covered by an EMS test procedure. It is on a timer device which causes the engine to startup and operate automatically once per week. This test is essentially identical with that performed on the other engines under the EMS procedures.

The specific EMS procedures which pertain to the testing and inspection of the diesel engines are numbered 116, 201, 605 and 621 (Ref. 10).

9.5.7 Diesel Engine Lubricating Systems

The function of the system used to lubricate the emergency fire pump diesel engine and the diesel engine on the emergency generator (both located in the high lift pump house - 182-N building), the 105-N lift station sump pump diesel engine and the seven 1600 hp diesel engines which drives the emergency high-lift, low-lift and fog spray pumps, is to provide a reliable source of lubrication to all working parts of the engines during their operation.

The fire pump, emergency generator and sump pump diesel engines are each a complete unit supplied by long time vendors of this type of equipment from their standard lines. The lubrication system is an integral part of the engine.

Each of the seven 1600 hp diesel engines is equipped with a lube oil system which is independent of the system associated with any other engine. A lube oil transfer system, which services all of the engines, is also provided. This facilitates the addition of makeup oil to the individual engine system if required and/or the complete change-out of the oil from an engine if necessary for maintenance purposes or if the oil becomes contaminated.

9.5.7.1 Design Bases

The working parts of the 182-N building emergency generator, the fire pump and the 105-N lift station sump pump diesel engines are lubricated by a force feed system which is an integral part of the engine design. The oil pump is driven by the engine and provides the total force for circulating the oil. The following design conditions apply:

	Fire Pump Diesel Engine	Generator Diesel Engine	Sump Pump Diesel Engine
Oil Specification (above 30 F Ambient)	SAE40W	SAE40W	SAE 30
Oil Pressure at Full Load	30-50 lb/in^2	55-100 lb/in^2	55-100 lb/in^2
Oil Capacity	36 qt	26 qt	32 qt

The lube-oil system associated with the low-lift, high-lift and fog spray pump diesel engines, is designed to provide oil under pressure to lubricate the various working parts of the engine. Two oil pumps are included in the system. One engine driven pump circulates the oil when the engine is operating. The second pump is electrically driven and provides oil circulation when the engine is standing by. When running at full load and speed, oil is supplied to the engine at 55 lb/in^2 (gage) and a temperature of 160°F, and the temperature of the oil as it leaves the engine is 190°F.

The lube-oil pump circulates the oil at the rate of 115 gal/min. The capacity of the lube-oil system is 150 gal of SAE40W oil.

The system is designed to assure that no single component failure within a lubrication system will cause the loss of more than one diesel engine. The redundancy of equipment provided and the total pumping capacity available in the diesel powered emergency pumps is such that the loss of a single diesel engine will not impair the adequacy of the emergency core cooling system or fog spray system.

9.5.7.2 System Description

The lube-oil system on the 182-N building emergency generator diesel engine the 105-N sump pump diesel engine and the fire pump diesel engine consists of gear-type oil pump which is driven by the engine, an oil pan or sump which is part of the engine crankcase and the oil distribution header and lines within the engine. Filters, strainers and screens are located throughout the system to provide for the proper cleaning of the lubricant. The system pressure is controlled at a nominal 40 lb/in^2 (gage) on the fire pump diesel engine and 60 lb/in^2 (gage) on the emergency generator and sump pump engines.

References 11 and 12 provide detailed description of the lube-oil systems in the emergency generator diesel engine and the fire pump diesel engine respectively. The lube-oil system of the sump pump diesel engine is nearly identical to the other two small diesels.

The engines are provided with instrumentation which monitors the lube-oil pressure and actuates both visual and audible alarms in the event the oil pressure reaches a low-pressure set point. A further decrease in the lube-oil pressure would result in an automatic shutdown of the engines.

Each of the 1600 hp diesel engines, which drive the emergency low-lift pumps in the 181-N building and the emergency high-lift and fog spray pumps in the 182-N building, is equipped with a lube oil system which is totally independent of the system on any other engine.

The operating system consists of a lube-oil pump, strainer, filter, oil cooler, pressure regulating valve and an oil sump tank plus the associated piping necessary to direct the lube oil to the working parts which require lubrication. The lube-oil pump is driven by the engine and takes over the pumping function whenever the engine is operating. The pump draws the oil from the sump tank and forces it through the strainer, the oil cooler, the filter and the pressure regulating valve and into the engine lube-oil piping system. The oil enters the engine at a nominal pressure of 55 lb/in^2 (gage).

After flowing through the various lubrication paths throughout the engine, the oil is collected in the sump tank and is recycled. Instrumentation which monitors the temperature of the oil leaving the engines and the temperature of the cooled oil as it re-enters the system, is provided. The pressure of the oil as it enters the engine is also monitored.

An electrically-driven, standby lube-oil pump is provided for each of the engines. This pump operates continuously when the engine is in standby status. The function of this auxiliary pump is to keep the lube oil circulating through the engine, thus, maintaining it in a lubricated condition; ready for quick emergency startup if needed.

A third lube-oil pump is included in the system. This is a manually-operated lever action pump which can be used for priming the oil system, if necessary, to supply lubrication to the engine. It is capable of pumping 15 gal/min when actuated at the rate of 50 double strokes per minute.

To facilitate the addition of makeup lube oil and/or the complete change-out of the lube oil to individual 1600 hp diesel engines, the 181-N and 182-N buildings are each provided with a lube-oil transfer or supply system. This consists of a transfer pump, a lube-oil storage tank, a waste oil storage tank and the piping required to service the two emergency low-lift diesel engines in the case of the 181-N building system and the three emergency high-lift and two fog spray diesel engines included in the 182-N building system. The lube-oil storage tank is equipped with facilities which permit

oil to be unloaded from a truck into the tank, and the waste oil tank is set up so that it can be drained to a truck for disposal of the used oil.

9.5.7.3 Design Evaluation

The lube-oil system, provided by the vendor of the emergency fire pump, the emergency generator and sump pump diesel engines, are specifically designed to accommodate the requirements of the engine which they service. The system is specific to each engine and requires no support from outside systems to perform its function.

These engines, including the lube-oil system, are maintained in first class operating condition. Frequent testing of the engines for more than 14 yr of plant operation has proven that the lube-oil system can be relied upon to perform its function as designed.

The lube-oil system with which each of the diesel engines that drive the emergency low-lift, high-lift and fog spray pump is equipped, is designed to provide adequate lubrication to the vital working parts of the engine on a continuous use basis. The system is complete within itself and once the engine is in operation, does not depend on any outside system (other than cooling water to the oil cooler, see Section 9.5.5) to perform its function. As a result, the failure of any single component in a lube-oil system, would only affect the operation of the engine that is serviced by the system in which the failure occurs. The redundancy of equipment and the total available pumping capacity included in the diesel powered emergency pumps are such that the loss of a single diesel engine would not impair the adequacy of the emergency core cooling system or the confinement system.

The instrumentation on the system provides ample warning that lube-oil temperatures or pressure are approaching out-of-limit status, thereby providing time to take corrective action.

The environmental conditions which could adversely affect the capability of the diesel engine lube-oil systems to perform their function are described in Section 3.7, Seismic Design.

9.5.7.4 Tests and Inspections

Inspection and testing of the emergency fire pump diesel engine, the emergency generator diesel engine, the sump pump diesel engine and the seven 1600 hp diesel engines which drive the emergency low-lift, high-lift and fog spray pumps, is performed on a routine and continuous basis by plant personnel. The lube-oil systems associated with these engines are included in the test and inspection program.

The Plant Equipment Maintenance Standards Program (Ref. 10) currently requires that a complete operational test of the emergency diesel engines be performed as follows:

- o 182-N emergency generator - weekly
- o Low- and high-lift pumps - prior to each reactor startup
- o Fog spray pumps - monthly
- o 105-N lift station sump pump - monthly

The fire pump diesel engine is not included in an EMS test procedure. However, this machine is on a timer device which causes the engine to startup automatically once per week and operates for approximately 45 min. During the time it is running, the engine is closely observed and checked by operation personnel.

The specific EMS procedures which pertain to the testing and inspection of the diesel engines are Nos. 116, 201, 605 and 621 (Ref. 10).

9.5.8 Diesel Engine Combustion Air Intake and Exhaust System

It is the function of the diesel engine combustion air intake and exhaust systems to provide each of the ten diesel engines, which power emergency equipment, with clean air taken from the atmosphere and to remove the effluent gases from the engines and exhaust them to the atmosphere.

The combustion air intake system, designated DI, includes and extends from each of the ten intake cleaner-silencers to the ten diesel engines-seven in

the high lift pump house (182-N building), two in the river pump house (181-N building) and one in the 105-N lift station. The exhaust system, designated DE, extends from each of the diesel engines to its corresponding muffler.

9.5.8.1 Design Bases

The DI and DE system on each diesel engine are designed to be independent of any other engine and to preclude the possibility of the intake air being affected by exhaust gases from any of the engines or by other potential chemicals which may be stored or generated in their vicinity. This independence is such that a component failure in the intake or exhaust system of one engine will not adversely affect the operation of any other engine.

The intake and exhaust piping are designed to provide an air flow rate to and from the engines as follows:

	At Full Load	
	ft^3/min	r/min of
o 1600 hp emergency pump diesel engines	6000	750
o Fire pump diesel engine	1000	1775
o emergency generator diesel engine	600	1800
o Lift station sump pump diesel engine	550	1800

Other design parameters of the system for the 1600 hp engines are as follows:

	DI	DE
o Normal operating pressure	Vacuum	5 lb/in^2 (gage)
o Normal operating temperature	80°F	600°F
o Design pressure	75 lb/in^2 (gage)	75 lb/in^2 (gage)
o Design temperature	100°F	700°F

The environmental conditions which influence the design of the system are discussed in Section 3.3, Wind and Tornado Loadings, Section 3.4, Flood Design and Section 3.5, Missile Protection.

No instrumentation or controls are required to ensure proper operation of the system.

9.5.8.2 System Description

The combustion air intake system for each of the five 1600 hp diesel engines, which drive the emergency high-lift and fog spray pumps, consists of air cleaner-silencers, located on top of the 182-N building, and the 16-in diameter pipes through which the air is conveyed to the engines which are located in the basement of the building approximately 40 ft below the roof level. The air cleaner-silencer is a cylindrical tank which is perforated around the center for air entrances. Internally the tank contains an oil bath and filter arrangement through which the intake air must pass. The oil is entrained in the air and is then removed together with the dust and dirt particles in the filters.

The air intake system for each of the two 1600 hp diesel engines which drive the emergency low-lift pumps in the 181-N building, is identical to that described above except that the cleaner-silencer units are located alongside the building at ground level. Also, this system contains dry filter to guard against volcanic ash. The 16-in diameter intake pipe then runs from the cleaner up the outside of the building to roof level and back down inside the building to the diesel engines.

The DE system for the 1600 hp diesel engines consists of a 14-in diameter pipe which conducts the engine exhaust fumes out of the building to the roof where the mufflers are located. The muffler is a horizontally-mounted tank, 10-ft long and 31-ft in diameter, with internal baffles to disperse and silence the noise of the exhaust. After the gases leave the muffler, they are exhausted to the atmosphere via a short section of 14-in pipe which directs them upwards from the roof and away from the intake units.

A flexible connector, located within a few feet of where the intake air pipes and the exhaust pipes enter and leave the engines, is included in the piping runs. This device permits movement of the piping in relation to the engine, isolates vibration and eliminates the need for close alignment of the components.

The combustion air intake system on the three smaller diesel engines (the 205-hp fire pump, the 230-hp emergency generator and 175-hp sump pump), is the oil bath-filter type. Each unit uses room air and is installed on the machine itself and is so constructed as to also function as a silencer. Room air in the case of the sump pump diesel is essentially outside air as its housing structure is unheated and about half of the north side is a screened opening. The exhaust system on these three engines is similar to that on the larger engines in that the exhaust pipe carries the fumes from the engine to the building roof where, after passing through a muffler, they are discharged to the atmosphere. Figures 9.5-9 and 9.5-10 show the intake and exhaust system installations in the 182-N and 181-N buildings, respectively.

9.5.8.3 Safety Evaluation

Both the combustion air intake and the exhaust system for the diesel engines are of adequate size to provide the air necessary to ensure proper engine operation for as long as may be required. The air cleaner-silencer is designed to permit cleaning during operation, thereby eliminating the need to shutdown the diesel engines for maintenance of the air cleaner should long term operation become necessary.

Figure 9.5-9 shows the physical relationship of the air intakes to the exhaust pipes on the 182-N building roof and Figure 9.5-10 shows the same configuration for 181-N. In both cases, the spacing is such that the likelihood of sufficient exhaust fumes being taken in with the combustion air to cause engine ignition problems is prevented. In addition, the stub pipe from the exhaust muffler directs the effluent gases up and away from the air intakes. There are no gases, gas-producing or finely divided materials stored in the vicinity of either the 181-N or 182-N buildings or

the 105-N lift station. Also, no high or moderate energy piping systems are located in close proximity to either building. The potential for flooding is discussed in Section 3.4.

The location of the inlet air cleaner-silencers, as shown on the figures noted above, is sufficiently remote or well protected from other items of equipment or stored materials that the possibility of the air intake being restricted due to the misplacement of such items is eliminated. Snow, ice and rainfall present little problem in this area, since the annual amounts received are very low. However, if snow or ice should accumulate, the fact that the air intakes are readily accessible permits easy clearing of the intake assemblies.

The intake of particulates is controlled by the air cleaning and filtering capability of the air inlet assembly. Since this device can be cleaned with the engines running, continuous operation of the engine is assured. Low barometric pressure within the normal range, has no effect on the operation of the diesel engines.

9.5.8.4 Inspection and Testing Requirements

Inspection and testing of the emergency fire pump diesel engine, the 182-N building emergency generator diesel engine, the 105-N lift station sump pump diesel engine and the seven 1600 hp diesel engines which drive the emergency low-lift, high-lift and fog spray pumps, is performed on a continuous and routine basis by N Plant personnel. The combustion air intake and exhaust systems associated with these engines are included in the test and inspection program.

The Plant Equipment Maintenance Standards Program (Ref. 10) currently requires that an operational test of the emergency diesel engines be performed as follows:

- 182-N emergency generator - weekly
- Low- and high-lift pumps - prior to each reactor startup
- Fog spray pumps - monthly

o 105-N lift station sump pump - monthly

The fire pump diesel engine is not included in an EMS test procedure. Instead, a timer causes the engine to start automatically once per week and operate for approximately 45 min. During these operational tests, the engines, including the combustion air intake and exhaust systems, are carefully inspected by operations personnel. The specific EMS procedures which pertain to the testing and inspection of the diesel engines are No. 116, 201, 605 and 621 (Ref. 10).

Periodic inspection of the condition of the oil and the sludge buildup in the intake air cleaner is performed on a routine basis. Sludge removal and oil addition or total replacement are carried out when the inspection indicates such action is necessary.

9.5.9 Plant Service Boiler Fuel System

The plant service boiler fuel system shown in Figure 9.5-11 maintains an adequate supply of No. 6 oil in storage and supplies it to three steam boilers. The boilers are discussed in Section 10.4.11 and the ignition oil supply is discussed in Section 9.5.4. The system includes a full unloading station, fuel oil pump house, bulk storage tank, day tanks and heating system.

9.5.9.1 Design Basis

The plant service boiler fuel oil system is designed to supply the three plant service boilers with a continuous supply of No. 6 fuel oil under all temperature and load conditions. The equipment is designed to accept the oil at an unloading station and move it through a pumping station to a bulk storage tank. The pumping station also moves the oil from the storage tank or unloading station to day tanks adjacent to the 184-N plant service boiler house.

Total storage capacity of the fuel oil storage is 1,445,000 gal. The single bulk storage tanks volume is 1,375,000 gal; the two day tanks each have 35,000 gal capacity. The unloading system is designed to handle 600 gal/min

of oil at 60°F. The transfer system is designed to handle 600 gal/min at 120°F.

The steam heating system is designed to keep the transport lines warm and to heat the bulk storage tank to 90°F and the day tanks to 120°F at -10°F ambient.

The day tanks are provided with a level readout in the 184-N control room and an annunciator for preset maximum and minimum levels.

The piping and tanks were fabricated and installed in accordance with appropriate sections of codes, standards and specifications of the following organizations.

- o American Petroleum Institute
- o American Society of Mechanical Engineers
- o American Standards Association
- o American Society for Testing Materials
- o National Board of Fire Underwriters
- o Hanford Works Standards

The specific application of these codes, standards and specifications is detailed in References 13 and 14.

Design conditions for the fuel oil system are as follows:

	Oil Piping	Steam Heating
Normal operating pressure, lb/in^2 (gage)	10 to 180	150
Normal operating temperature, °F	120 to 200	375
Design -- Pressure, lb/in^2 (gage)	200	175
Test -- Pressure, lb/in^2 (gage)	300	263
Design -- Temperature, °F	250	400

9.5.9.2 System Description

The fuel oil system consists of an unloading station, one storage tank, two transfer pumps, chemical mixing equipment, two day tanks and a steam heating system. See Figure 9.5-11.

The fuel oil unloading station is located west of the 166-N building. It is a long narrow concrete trench containing six tank car and four tank truck unloading connections. Fuel oil from tank cars or trucks is unloaded through these stations into a 12-in line which extends to the 166-N building and forms the suction manifold for the transfer pumps. The station is equipped with a steam system for heating the transfer lines and for connection to the heating coils in shipping equipment.

The two transfer pumps are located in the 166-N basement. Their purpose is to move oil from the unloading station to the storage tank to the day tanks at 184-N or between tanks. They are horizontal, screw type pumps rated at 300 gal/min at 111-ft head. Electrical power is supplied from an offsite ac source to one pump and an onsite ac source to the other. The motors are rated at 30 hp, 480 volts. Other equipment used with these pumps includes the pump suction heater and the duplex strainer. The heater's function is to maintain free flowing oil at approximately 120°F. The strainer's purpose is to remove any debris from the oil. It is rated at 600 gal/min at 120°F. There is also a level switch in the pump suction manifold which prevents the pump from operating without load when a tank empties.

Chemicals are added to No. 6 fuel oil to inhibit combustion deposits and corrosion. A 125-gal tank and air-driven pump are located in the SE corner of the 166-N basement. Chemicals are mixed in the tank and pumped into the 12-in fuel oil system line during the time fuel oil is being transferred to the day tank. Diesel oil is used to effect dissolution of the chemicals in the mix tank.

The fuel oil storage tank is of metal construction, 70 ft in diameter and 48-ft high to the edge of the conical shaped roof. It is columnar supported with one beam in the center and five more in a circular arrangement

supporting the roof between the edge and center. Storage capacity of the tank is 1,375,000 gal. The tank foundation is a concrete ring extending 12-in above ground. Fill inside of the ring is compacted oiled sand. Surrounding the tank is a dike 136 ft by 215 ft with banks 14-ft high to contain the oil in case of tank rupture or leakage.

The two fuel oil day tanks at 184-N are 20 ft in diameter, 15 ft 2 in. in height and have a storage capacity of 35,000 gal each. They are located north of the 184-N building and are surrounded by a rectangular concrete wall 40 ft by 80 ft and 6-ft high. The tank foundation is a circular concrete ring filled with compacted oiled sand. Piping is arranged so that the day tanks can be filled and emptied separately or simultaneously.

Fuel oil viscosity change with temperature requires maintaining the oil at a temperature so it will flow freely. Temperature control is accomplished by steam heating and insulation. From 184-N to 166-N the fuel oil piping is adjacent to hot steam pipes. The piping in these particular runs are covered with insulation (Gilsulate) to keep the fuel oil as warm in transit. The storage tanks are provided with internal heating coils and are also insulated.

Diesel oil is required for ignition during startup of the boilers. The diesel oil system is discussed in Section 9.5.4. The fire protection system is discussed in Section 9.5.1.

9.5.9.3 Design Evaluation

The fuel oil system has provided a dependable fuel supply to the three steam boilers. The use of two transfer pumps, one supplied by onsite ac power and the other by the offsite ac power, is a reliable feature of the system.

The storage tank inventory is maintained nearly full by the fuel oil supply contractor. Total storage capacity is 1,445,000 gal and is not expected to fall below 1,000,000 gal. This is considered to be an adequate inventory since the annual use rate is 5,000,000 gal.

In addition to the oil in storage and the truck supply from the oil storage terminal south of Pasco, No. 6 oil is also available for truck or rail shipment from Portland, Oregon, or Seattle, Washington.

The system is provided with valves and check valves in locations which permit isolation and/or draining specific sections of piping, pumps and tanks so that maintenance or repair work can be performed without removing the system from service.

The postulated environment circumstances that could affect the ability of the fuel oil system to perform its functions are discussed in Sections 3.7, Seismic Design, 3.3, Wind and Tornado Loadings and 3.4, Flood Design. The environment is protected from tank content loss by diking, and from fires by the fire protection system. See Table 9.5-3, page 12 of 12, and Section 2.2.

9.5.9.4 Tests and Inspection

Inspection and testing of the fuel oil system is carried out on a routine and continuous basis as required by the N Plant Equipment Maintenance Standards. Such testing and inspection includes:

- o Visual inspection for leaks - valves, piping and tanks
- o Inspection and calibration of instrumentation - level gauges and temperature indicators
- o Hydro testing of heating coils
- o Internal tank inspections
- o Fuel oil sampling and analysis

9.5.9.5 Instrumentation Application

The instrumentation required for the fuel oil system to adequately perform its function is minimal. The storage tank is provided with a float type level indicator readable at the tank. The day tanks are provided with similar level gauges and with a remote reading gauge in the 184-N control room. In addition, the day tanks are provided with an annunciator that would alarm should the preset upper or lower levels be exceeded. Each of the tanks is provided with four temperature gauges with readout at the tanks. The 2 day

tanks are provided with an automatic temperature controller to maintain the oil temperature at about 120°F. The storage tank temperature is manually controlled. Temperatures and levels are read and recorded manually every 2 hours.

9.5.10 Shuttle Facility

An installation called the shuttle facility occupies one of the horizontal control rod channels which is not used for a reactor control rod. The facility is used for neutron activation of samples in support of DOE testing and experimental programs.

The shuttle facility consists of three separate irradiation test facilities. Two of these are pneumatically-operated and extend only into the reflector of the reactor. The third unit is hydraulically-operated and extends into the reactor core. Capsules containing sample materials are inserted pneumatically or hydraulically into the proper reactor zone. The capsules are retrieved by reverse flow of propelling media.

The three units are constructed of stainless steel tubing and are contained within a single titanium thimble. The pneumatic reflector units are 1 1/2-in OD and the hydraulic unit is 1 1/8-in OD. The capsules are made of either linear polyethylene, titanium or high purity aluminum. Cooling is provided by circulating water from the HCR cooling system within the thimble and around the tubing. In addition, a small continuous flow of HCR cooling system water is provided within the tubing for capsules in the reactor core.

Insertion and retrieval of capsules can be done during the reactor operation. Because the samples are small and limits are placed on the quantity of neutron producing or absorbing elements in the samples, only slight localized reactivity changes result. They do not affect the reactivity balance of the reactor. Operation of the facility is conducted under a Production Test. The coolant flow and temperature instrumentation is the same as that for the HCRs and is part of the reactor safety system as described in Section 7.2.

9.5.11 Loose Parts Monitor System

The Loose Parts Monitor System monitors the incoming primary pipes to each steam generator for loose parts carried through the pipes to the steam generator tube manifold.

9.5.11.1 Design Bases

In mid-November 1985, during a long maintenance outage, operators noticed that a check valve was passing a greater-than-expected backflow. The valve CV3-8 was opened and inspected; the flapper was found lying inside. The three 1 3/8-inch-diameter studs which hold the flapper in place; and their lock washers and keepers were missing.

After determining the possible locations where they might be, from the size of the missing parts and other factors, a search was initiated.

All possible locations for the missing metal parts were inspected, including the most probable location, the steam generators downstream of the valves. The missing parts were not found there and metal parts of the steam generators were also found missing. Some of these missing metal parts were located at other places and some of these missing metal parts were not located.

Loose parts in the primary system may change the piping system or cause a breach in the primary system, creating the possibility of an environmental radiation release. While a loose part monitor cannot prevent loose parts from entering the primary system, it can alert operations personnel to the presence of potentially dangerous loose parts before damage occurs.

Loose parts in the primary system may result in flow blockage with attendant departure from nucleate boiling (DNB), which in turn could result in failure of fuel cladding. Each individual process tube is continuously monitored for flow adequacy and an automatic scram will occur if the flow falls outside a predetermined acceptance band. Analyses have been performed which indicate that up to 75 percent of the process tube flow area can be blocked without a scram and still prevent DNB, or up to 95 percent of the flow area with a scram, and there will still be sufficient cooling to prevent fuel failure.

As an information system, relative to coolant flow to the fuel, the flow monitor system is superior to the Loose Parts Monitoring System (Ref. 17).

The detection system, for practical purposes, only monitors for loose parts which originate between the downstream side of the process tubes and the steam generator inlet manifold. The steam generator tube size is 5/8 inch diameter so that loose parts small enough to pass through these tubes pose little risk to reactor safety.

9.5.11.2 System Description

Specifications of the major components, their characteristics, the system requirements, spare equipment, environmental conditions, specification and marking, maintainability, fabrication, component interchangeability, safety and human engineering are found in Appendix D of DC-86150 (Ref. 16).

The Loose Part Monitoring System (LPMS) has been installed in the 109-N Building. Sensors have been installed on each of the twelve 18-inch steam generator primary inlet pipes. Housed in instrument rack LPM-PNL (IP-100) are the signal conditioners, an event recording device, and trip unit. LPM-PNL is in the turbine bay area adjacent to DT-4. A trip signal annunciates on an existing annunciator panel in the 109-N Bay Area.

The LPMS receives an audio signal from each transducer, records it, stores the time-stamped information on a hard disk, and annunciates a trip. The system provides a hard copy and an audio play back on request. It is sensitive enough to detect a 1/2-inch-diameter sphere or larger steel object in the steam generator manifold.

9.5.11.3 Test and Calibration

Test and calibration of the LPMS will be provided by the Preventive Maintenance Organization and Standard Operating Procedures.

REFERENCES - 9.5

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14. Burns and Roe, Inc., "Mechanical Installation for Standby Power House Building No. 184-N," HWS-6624, September 19, 1960.

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19. UNI-4292, "Functional Design Criteria 182-N Fire Protection Upgrade," Project H-789.

20. MPSR 88091, "Fire Resistant Hydraulic Fluid for C, D and W Power Packs," dated January 11, 1988.

21. WHC-CM-4-3, Standard FP-13, "Flammable/Combustible Liquids," dated March 30, 1988.

22. WHC-CM-5-13, AM-07-07, "Controlled Welding and Cutting Permit."

23. WHC-CM-5-13, AM-10-01, "Housekeeping and Cleanliness Control."

24. WHC-CM-4-3, Standard FP-1, "Fire Watch Requirements," dated March 30, 1988.

25. WHC-CM-4-3, Standard FP-8, "Fire Barrier Penetration," dated March 30, 1988.

26. WHC-CM-4-3, Standard FP-9, "Cutting and Welding," dated December 11, 1987.

27. WHC-CM-4-3, Standard FP-10, "Fire Door/Smoke Damper Maintenance Standard," dated March 30, 1988.

28. WHC-CM-4-3, Guide FP-2, "Portable Fire Extinguishers," dated December 11, 1987.

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

1. Building 105-N

Room Number	Current Use or Contents
1	Ventilating Equipment and Compressor Room*
4	Ball Control Room
7	"R" Elevator Landing -15'
9	Battery Room*
25	Air Lock - RZN
28	Zone I Confinement Valve Room*
29 (partial)	Beneath "W" Platform +5' ready room
31	Gas Facility Operating Gallery
38	Instrument Shop - RZN
39	Toggle Valve Rework Ship - RZN
49 & 49A, 49B, 49C	Miscellaneous Storage, injection pumps and generator
50	Laboratory with 4000 channel analyzer
101	Carpenter Shop
102	Paint Shop
103	Rigging Loft
104	Millwright Shop
105	Fabrication Shop
106	Welding Shop
107	Pipefitter's Shop
108	Tool Crib
109	Warehouse

*Protection provided for Category-1 facilities, systems and components.

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

1. Building 105-N (cont.)

Room Number	Current Use or Contents
110	Supply Fan Room
111	Soiled SWP Clothing
113 & 114	Clean SWP Storage
115	Shower Room
116	Change Room
117	Office
120	Janitor's Closet
121 & 122	Women's Lounge and Lavatory
123 - 130	Offices
131	Change Room
132	Lunch Room
133 & 134	Model Room
135 & 136	Conference Room
137	Electrical Shop
140	Instrument Shop
141, 142 & 146	Offices
153 - 161	Offices
162	Electronics
163	Men's Lavatory
164 - 167	Offices
169	RM Laboratory
172	Lobby and Air Dryer - RZN
173	Storage Room - RZN

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

1. Building 105-N (cont.)

Room Number	Current Use or Contents
174	Battery Room
181	Metal Prep Room
182	Metal and Spacer Storage
183 & 184	Dryer Rooms
187	Lunch Room
188	Change Room
189, 190 & 191	Basin Hot Shop
193 (Partial)	Transfer Area
194	Laundry Storage Area
301	720 Cycle Alternator Room
302	Rear Face Landing
303	Instrumentation and Electrical Storage Room
304	+14' Change Room (off Room 110)
305	+14' Miscellaneous Storage
306	Acid Storage
307 & 309	14' Decontamination Room
308	Incore Flux Monitor Computer
401	Storage Room - RZN
402	Storage Room
403	Technical Contaminated Storage Room
404	Hot Storage Room 28' Level (Left Rear)
501	Shop/Exit - RZN (Shop Only)
502	+40' Ready Room

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TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

1. Building 105-N (cont.)

Room Number	Current Use or Contents
502B	Shop/Exit - RZN (Shop Only)
503	+40′ Change Room
520	Ball System Relay Room* - RZN Southwest Corner
521	Stair Landing/Storage - RZN
601	Landing and Storage - RZN (Southeast Corner - RZN)
602	Hot Storage Room (JAJ)
603	Partial Coverage (Project H-595)
610	Landing and Storage
Corridors	
1	
2	
3	Corridor 3
4	Corridor 4 - Part RZN
5	West End Corridor 5
7	East Part - South of Control Room (Room 145) Including Stair No. 5
8	
11A	
11B	Corridor 11B (Includes Front Face Entry - RZN)
12A	
13	
14A	

*Protection provided for Category-1 facilities, systems and components.

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

1. Building 105-N (cont.)

Corridors	Current Use or Contents
14B	
15	
18	
20	Corridor 20
22	Corridor 22 - Part RZN
23B	Corridor 23B
27A, 27B & 29A	Corridors
29B	Corridor 29B - RZN
501	Corridor
(No Room # Assigned)	+50' Lab Room

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2. Building 109-N

9	Turbine Bay Basement
10	Cell Access Cooridor - RZN
11	Airlock - East End of Cell Access Corridor - RZN
12	Freight Elevator Machinery Room
13	Storage Room - 16'
15, 17 & 18	SWP Change Room
19	-16' Corridor No. 19
20	Soiled SWP Clothing Storage
21	Hot Tool Room
22	Decontamination Room
23	Hot Shop

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

2. Building 109-N (cont.)

Room Number	Current Use or Contents
24	Welding Shop
25	Hot Storage
104	Change Room and Lab Entrance - RZN
116	Sheetmetal Shop and 0' Level
118	Stair No. 7 to Hot Water Quality Lab
164	R&WQC Lab
202	Office
203	+12' Corridor
204	Office
205	Instrument Module Shop
206	Office
208	R&WQC Lab
209	Stair No. 7 to Hot Water Quality Lab
301 & 302	Hydraulic Shop
307	R&WQC Lab
308	Stair No. 7 to Hot Water Quality Lab
402	Cell 6 Heating and Ventilating Penthouse
Stairwell #3	Main Stairwell from 0' Level to Roof
Stairwell #7	Rearface Next to Corridor 7
Cells 1 - 6	Steam Generator Cells
Cell Access Corridor	Corridor

TABLE 9.5-1

WET PIPE SPRINKLER PROTECTION

3. Building 117-N (Stack Filter)*

4. Building 182-N Basement (High Lift Pump House)*

5. Building 184-N and Annex (Power House)

6. Building 1101-N (Wood frame Administration Building)

7. Building 1734-N (Oil and Gas Storage) 16

8. Building 1712-N (Tool Crib)

9. Building 1714-N (Tool Crib)

10. Building 1722-N (Basin Hot Shop)

11. Building 1723-N (Storage) Installed by Project H-595 21

12. Building 1100-N (Offices)

DRY PIPE SPRINKLER PROTECTION

1. Building 109-N

Room Number	Current Use or Contents
402	Cell 6 Penthouse

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*Protection provided for Category-1 facilities, systems and components.

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TABLE 9.5-2

DELUGE PROTECTION

Location	Current Use or Contents
1. Building 105-N	
Room 29	Hydraulics and contaminated storage beneath "W" platform
Room 30	Hydraulic equipment room for work platform and "S" elevator
Room 49	Miscellaneous equipment
Pit	"C" elevator
Pit	"F" front elevator pit
Pit	"R" rear elevator pit
Room 170	Left-side outer rod room*
Room 176	Right-side outer rod room*
2. Building 109-N	(1) Drive turbines and reactor coolant pump oil console units (supply changed from RWS-2 to Fire System by Project H-639) (2) Hydraulic units on roof*
3. Building 181-N	Diesel Engines*
4. Building 182-N	Diesel Engines*
5. Building 184-N	Hydrogen storage shed oil conditioning equipment, tubine oil reservoir, seal oil unit, fuel oil heating and pumping equipment
6. Building 184-N Annex	Combustion engineering boilers, burners and oil heaters (manual only)
7. Building 151-N (yard)	30/50 MVA transformer

*Protection provided for Category-1 facilities, systems and components.

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TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

LEGEND

AS - means automatic sprinkling capability

EFW - means early fire warning via detection and annunciation

None - means that neither AS or EFW is available

NA - means not applicable

105-N Room Number	Current Use or Contents	Protective Features	Remarks
1	Ventilation, Equipment and Compressor Room	AS	
2	Valve and Equipment Room	AS	
3	Transducer Room, Left Side	(b)	(b) Heat detectors
4	Ball Control Room	AS	
5	Fuel Clad Failure Monitor Room Left Side	(b)	
6	Pile Instrument and Electrical Room--Cable Spreading Room*	(a) EFW	(a) Total flooding Halon 1301 with connected reserve
7	"R" Elevator Landing-15'	AS	
8	Electrical Equipment Room	(b)	
9	Battery Room*	(b) AS	Project H-595 installed sprinkler system
10	Outlet Piping Space*	None Required	
11	Emergency access pit	None	
12	Electrical Room	(b)	
13	Outlet Piping*	None Required	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
15	Examination Facility	None	
17	Storage Basin	None	
18	Storage Basin	None	
19	Zone II Exhaust Fan Room	(b)	
20	Not Listed	NA	
22	Reactor Piping & Ball Recovery Area	(b)	
23	"C" Elevator Pit	(d)	(d) Automatic deluge system
24	Lower Hinged Panel Walkway	None	
25	Airlock	AS	
26	Inlet Piping*	None Required	
27	Occupancy Not Listed	NA	
28	Zone I Confinement Valve Room*	AS	
29	Work Elevator Equipment Room	(d)	
30	Equipment Room	(d)	
31	Gas Facility Operating Gallery	AS	
32	Gas Facility Pipe Tunnel	None	
33	Zone III Exhuast Tunnel	None	
34	Zone I Exhaust Tunnel*	None	
35	Pipe Tunnel*	None Required	
36	Right Transducer Room	(b) EFW	(b) Heat detectors

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks	
37	Right Fuel Clad Failure Monitor Room	(b) EFW		16
38	Air Lock Entry to Room 21	(b) EFW		
39	Not Listed	AS	Project H-595 installed sprinkler system	21
40	Filters	None		
41	Pipe Tunnel*	None Required		21
42	Valve Box	None		
43	Pump Well	(b)	(b) Detection installed by Project H-639.	21
44	Zone II Exhaust Tunnel	None		
45	Supply Plenum	None		
46	Left Side--Flow Monitor Valve Room	(b)	(b) Heat detectors	
47	Electrical Tunnel-19' Corridor 47	EFW		
48	Corridor between Rm 4 and Rm 22	(b)		
49	Miscellaneous Storage	None		16
50	Counting Lab	AS		
51	Power Pack Room, Front "S" Elevator	(d)	(d) Automatic deluge system	
52	Air Lock - 15'	None		
101	Carpenter Shop	AS		
102	Paint Shop	AS		

*Category-1 facilities, systems and components. 21

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
103	Rigging Loft	AS	
104	Millwright Shop	AS	
105	Fabrication Shop	AS	
106	Welding Shop	AS	
107	Pipefitter's Shop	AS	
108	Tool Crib	AS	
109	Warehouse	AS	
110	Supply Fan Room	AS	
111	Soiled Clothing Storage	AS	
112	Not Listed	NA	
113-114	Clean SWP Storage	AS	
115	Shower Room	AS	
116	Change Room	AS	
117	Office	AS	
118	Not Listed	NA	
119	Not Listed	NA	
120	Janitor's Closet	AS	
121-122	Women's Lounge and Lavatory	AS	
123	Office	AS	
124-130	Office	AS	
131	Change Room	AS	

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
132	Lunch Room	AS	
133-134	Model Room	AS	
135-136	Conference Room	AS	
137	Office	AS	
138	Not Listed	NA	
139	Not Listed	NA	
140	Instrument Shop	AS	
141	Office	AS	
142	Office	AS	
143	Air Conditioning Room	(b)	(b) Heat detectors
144	Electrical Switchgear Room	EFW	
145	Control Room*	(f)(q) Halon 1301	(f) Total flooding Halon 1301 with connected reserve (q) Remote shutdown capability available should control room become uninhabitable. This capability uses cables which are routed through Rooms 6 and 145, though ECCS manual actuation is independent of Rooms 6 and 145.
146	Office	AS	
147-149	Control Room Gallery	EFW	
150-152	Not Listed		
153-155	Offices	AS	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
156	Planning and Scheduling	AS	
157-161	Offices	AS	
162	Electronics	AS	
163	Men's Lavatory	AS	
164-166	Offices	AS	
167	RM Office	AS	
168	Radiation Timekeeping Supply Crew Office	None	
169	RM Laboratory	AS	
170	Left Side--Outer Rod Room*	(b)(d)	(d) Automatic deluge system
171	Left Inner Rod Room*	(b)	(b) Heat detectors
172	+5' Lobby Left Rear Zone I Entry	(b)	
173	RM Storage Room	(b)	
174	Battery Room--Operations Hot Storage Room	(b)	
175	Discharge Alignment Area	(b)	
176	Right Side--Outer Rod Room*	(b)(d)	(b) Heat detectors
177	Right Inner Rod Room*	(b)	
178	Flux Monitor Room	(b)	
180	Flow Monitor Valve Room--Right Side	(b)	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks	
181	Metal Prep. Room	AS		
182	Metal and Spacer Storage	AS		
183	Dryer No. 1	(b) AS		16
184	Dryer No. 2	(b) AS		
185	Refrigeration Equipment Room	(b)		
186	Gas Instrument Room	None		
187	Lunch Room	AS	Project H-595 installed sprinkler system	21
188	Non-irradiated Fuel Spacer Storage	AS		
189	Permanent Hot Storage Room	(b) EFW		
190	Hot Shop	(b) EFW		
191	Decontamination Facility	(b) EFW		
192	Tech. View. Seg. Pit Storage Basin	None		
193	Transfer Area	(e) AS	(e) Partial coverage	
194	Janitor Supplies	(q) AS	(q) Exposure to Rm 188	
195	Janitor Closet	(b) EFW	(b) Heat detectors	16
301	Electrical and Maintenance Storage Room	(b) EFW		
302	+14' Hot Tool Storage Rear Face Landing	(b) EFW		
303	Instrumentation and Electrical Storage Room	AS		

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
304	+14′ Change Room (off Rm 110)	AS	
305	+14′ Miscellaneous Storage	AS	
306	Acid Storage	AS	
307	+14′ Decontamination Room	AS	
308	Incore Flux Monitor Computer	(b)AS	Project H-639 installed sprinklers
309	+14′ Hot Shop	(b)AS	Project H-595 installed sprinklers
401	Hot Tool Storage and Remote Equipment Room	(b)AS	Project H-595 installed sprinklers
402	+28′ Lt. Rear Face Landing	(b)AS	Project H-595 installed sprinklers
403	Technical Contaminated Storage Room	AS	
404	+28′ Right Instrumentation Room	(b)AS	Project H-639 installed sprinklers
501	Contaminated Equipment Room and Work Area	AS	
502	+40′ Rear Ready Room	AS	
503	+40′ Change Room	AS	
504	Inlet Valve Operating Gallery	(b)EFW	
505	Zone I Exhaust Fan Room*	(b)EFW	
507	Outlet Valve Operating Gallery	(b)EFW	(b)Heat detectors
508-519	Not Listed	NA	
520	Ball System Relay Room*	(b)AS/EFW	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Room Number	Current Use or Contents	Protective Features	Remarks
521	+51' Hot Shop	(b) EFW	
601	60' Landing	(b) EFW	16
602	Left Pressurizer Transducer Room	(b) EFW	
603	Rear Emergency Access Room	(b) EFW	
604	Machinery Room	(b) EFW	
605	Zone I Exhaust Duct*	None Required	21
606	Right Pressurizer Transducer Room	(b) EFW	
607	Ball Return Room*	(b) EFW	
608	Machinery Room	(b) EFW	
609	Zone III Exhaust Fan Room	(b) EFW	
610	+60' Landing	(b) EFW	
611	Access Way to Room 604	(b) EFW	
612	Air Lock	None	16
613	Air Lock	None	
614	Air Lock	None	
615	Air Lock	None	
616	Air Lock	None	
617	Right 60' Landing	(b)	

*Category-1 facilities, systems and components. 21

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Corridor Number	Current Use or Contents	Protective Features	Remarks
1	Off Instrument and Equipment Room	AS	
2	Left Side Basement	AS	
3	Corridor to the Left Fuel Clad Failure Monitor Room	(b) AS	(b) Heat detectors
4	Right Side Basement	(b) AS	
5	Left Right Front Basement -16'	AS	
6	0' Corridor to Maintenance Shop Area	(b)	
7	Corridor to Discharge Align. Area	(b) AS	
8	Main Entrance	AS	
9	Corridor Straight on From Main Entrance	AS	
10	Converted to Offices (Elect. Prints Room)	AS	
11-A	Left Side Front 0' Level	(b) AS	
11-B	Right Side Front 0' Level	(b) AS	
12,12-A	Corridor in Front of RM Office	AS	
13	Corridor by Main Lunch Room	AS	
14-A, 14-B	Corridor to Left Rod RM and Ramp	(b) AS/EFW	
15	Corridor by Supply Fan Room	(b) EFW	

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Corridor Number	Current Use or Contents	Protective Features	Remarks
16	Corridor in Soiled Clothing Storage	(b)AS	
17	Corridor	(b)EFW	
18	Corridor by Left Side of Metal Prep.	(b)EFW	(b)Heat detectors
19	Corridor	(b)EFW	
20	Gas Wing Corridor 0' Level	None	
22	Corridor to Storage Basin, Right Side	(b)AS/EFW	
23-A	+14' Right Side Front	(b)AS/EFW	
23-B	Corridor 23-B	AS	Project H-639 installed sprinklers
26	+40' Right Rear	None	
27-A	+28' Left Front	(b)EFW	
27-B	+28' Right Front	(b)AS/EFW	
29-A	+40' Left Front	(b)	
29-B	+40' Right Front	None	
29-B	Corridor 29B	AS	
Stairwell Number			
1	Between Control Room and Room 6	None	
2	Stairs from Corridor 12 to Supply	None	

16

21

16

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

105-N Stairwell Number	Current Use or Contents	Protective Features	Remarks
3	Fan Room Basement	(b)	
4	Entire Front Left Stairwell	None	
6	Entire Rear Left Stairwell	None	
10	Entire Rear Right Stairwell	None	
13	Between Corridor #4 & Corridor 22	None	
15	Entire Front Right Stairwell	None	
105-N Elevators			
Pit "F"	Front Elevator Left Side	(d)$_{AS}$	(d)Automatic deluge system
Pit "R"	Rear Elevator Left Side	(d)$_{AS}$	
Pit "S"	Service Elevator to "W" Elevator	(d)$_{AS}$	
Rm 29 "W"	Charging Machine Elevator	(d)$_{AS}$	
109-N Room Number			
1	Cell #1 *	(b)(g)$_{EFW}$	(g)Rooms 1-6 have AS protection over reactor coolant system pumps. Supply for sprays changed from RWS-2 to Fire System by Project H-639.
2	Cell #2 *	(b)(g)$_{EFW}$	
3	Cell #3 *	(b)(g)$_{EFW}$	(b)Heat detectors
4	Cell #4 *	(b)(g)$_{EFW}$	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

109-N Room Number	Current Use or Contents	Protective Features	Remarks
5	Cell #5 *	$(b)(g)_{EFW}$	
6	Cell #6 *	$(b)(g)_{EFW}$	
7	Auxiliary Cell *	$(b)_{EFW}$	
8	Pipe Gallery *	None Required	
9	Air Lock	None	
10	Turbine Bay Basement	AS	
11	Cell Access Corridor	AS	
12	Air Locks to Access Corridor	AS	
13	Freight Elevator Machinery Room	AS	
14	Storage Room - 16'	AS	
14	Radiation Monitoring Office	None	
15	SWP Change Room	AS	
16	Shower Room-16'	None	
17-18	SWP Change Room	AS	
19	-16' Corridor No. 19	AS	
20	Soiled SWP Clothing Storage	AS	
21	Hot Tool Room	AS	
22	Decontamination Room	AS	
23	Hot Shop	AS	
24	Welding Shop	AS	

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

109-N Room Number	Current Use or Contents	Protective Features	Remarks
25	Hot Storage	AS	
31	Freight Elevator	None	
32	Solution Preparation Room	(b)EFW	(b)Heat detectors
33	Ion Exchange Room	(b)EFW	
34	Hydrogen Peroxide Storage	(b)EFW	
101	Stairwell #1	None	
102	Stairwell #2	None	
103	Radioactive Sampling and Monitoring Room	(b)EFW	
104	Change Room Lab. Entrance	(b)EFW	
105	Grab Sample Room	(b)EFW	
106	Radioactive Chemical Lab	(b)EFW	(b)Heat detectors
107	Turbine Drive Bay	AS,5	
108	Lunch Room	(b)EFW	
109	0' Level Corridor	(b)EFW	
110	Men's Lavatory	(b)EFW	
111	Closet	None	
112	Locker Room	(b)	
113	0' Level Shower Room	None	
114	Janitor's Closet	None	
115	Heat Ventilation Machinery Equipment Room	(b)	

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

109-N Room Number	Current Use or Contents	Protective Features	Remarks
116	Sheet Metal Shop and 0' Level Entry to Pipe Gallery	AS	
117	Stairwell #3	AS	
118	Stairwell #7	AS & None	
119	Lunch Room	(b) EFW	
120	Freight Elevator Landing	(b) EFW	
121	Control and Switchgear Room - 9'	(b) EFW	
122	Control Room	(f)	Halon installed by Project H-644
201	Gamma Spectrometer Room	(b) EFW	
202	Office	AS	
203	+12' Corridor	AS	
204	Office	AS	
205	Instrument Module Shop	AS	
206	Office	AS	
207	2nd Floor Water Quality Laboratory	(b)	(b) Heat detectors
209	Water Quality Laboratory	AS	
301	Hydraulic Shop	AS	
302	Stairwell Landing	AS	
303	Stairwell Landing +24' Level	AS	
304	Exhaust Ventilation Area and Clean SWP Storage	AS	

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

109-N Room Number	Current Use or Contents	Protective Features	Remarks
305	+24' Elevator Landing	AS	
306	Decontamination Facility Solid Materials Storage	(b)	
307	Stairwell Landing	(b)	
308	3rd Floor Water Quality Lab	(b) AS	
402	Cell 6 Heating and Ventillating Penthouse	AS	
Roof	Hydraulic units*	AS	
Bldg. or Area. Number			
107-N	Basin Recirc. Building	AS	
108-N	Acid and Caustic Loadout	None	
116-N	Stack*	None	
117-N	Stack Filter Building*	(f)	(f) Filter spray system. Project H-639 installed smoke detection in Instrument room.
151-N	Substation Building	(c) EFW	(c) Outside transformer protected by automatic fog spray water
153-N	Switchgear Building	EFW	
166-N	Fuel Oil Storage	Foam, Water	
181-N	River Water Pump House*	(d) AS	(d) Automatic deluge systems

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

Bldg. or Area Number	Current Use or Contents	Protective Features	Remarks	
182-N	High Lift Pump House*	(h)(i)$_{AS}$	(h)Protection available in basement, and partial on 0' level. Emergency Fire Pump and Emergency Diesel Generator sprinklers installed by Project H-639.	21
			(i)Automatic deluge in basement (partial). Control Room had Halon installed by Project H-595.	21
163-183N	Coagulate and Feed Pump House Water Treatment Building	(j)$_{EFW}$	(j)Control Room, Laboratory and electric control room	21
184-N	Power House and Annex	(k)(l)$_{AS}$	(k)Annex has a manual deluge system in addition to AS	16
			(l)Automatic deluge in power house 0' level (partial)	
1100-N	Administration Building	(b)$_{AS}$	(b)Heat detectors	
1101-N	Administration Building	AS	Sprinklers extended into storage vault by Project H-639.	21
1310-N	1310 & 1314-N Decontamination Facility	None		
1705-N	Electrical Shops	AS		
1714-N	Tool Crib	AS		16

*Category-1 facilities, systems and components.

TABLE 9.5-3

100-N AREA FIRE PROTECTIVE FEATURES

Bldg. or Area Number	Current Use or Contents	Protective Features	Remarks
1715-N	Diesel Oil Storage	(m)	(m) A manual foam system is provided for fuel covering within tanks
1723-N	Storage	AS	Project H-595 installed sprinklers
1734-N	Oil and Gas Storage	AS	
1900-N	Fuel Oil Unloading	None	
119-N	Stack Shack	None	
1301-N	Emergency Overflow Valve House	None	
1102-N	Drafting Office	(b)	(b) Heat detectors
1103-N	Offices	(n) AS	(n) Separation limits exposure to other structures
1104-N	Offices	(n) AS	
1129-N	Mobile Office	(f)	(f) Halon 1301
1120-N	Equipment Storage and Training Facility	AS/EFW	
1314-N	Liquid Waste Loadout	None	
100-N	Minor Construction Buildings	None	
1322-N	Waste Management Valve House	None	
1722-N	Hot Machine Shop	AS	
1706-N	Breaker Shop	(p) AS	(p) Present moderate exp to 1705-N and 1734-N

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	Page(s)	Date	Page(s)	Date
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11.2 Liquid Waste Management System

This section describes N Reactor's system for controlling, collecting, processing, handling, storing and disposing of radioactive liquid effluents and wastes. Section 11.5 describes the effluent monitoring and sampling systems.

11.2.1 Design Bases

11.2.1.1 Criterion

The liquid waste management system consists of a liquid waste disposal facility (1325-N), a retired facility (1301-N), and a liquid effluent retention facility (1312-N). The original design criterion for release of radionuclides in N Reactor liquid effluents is based on concentrations in the Columbia River at the plant boundary. This criterion required that the released radionuclides not exceed 0.03 percent of the maximum permissible concentration (MPC) for any critical organ at the project boundary, assuming a minimum river flow rate of 36,000 cfs (Reference 1). Since documentation of this criterion, the thyroid has been identified as the organ of interest since iodine-131 is the radionuclide of interest. The current standard for radioactive releases is 100 mrem/yr (Ref. 10). 21

Based on operating experience with the retired 1301-N facility, the design life of the new 1325-N liquid waste disposal facility is at least 10 operating years starting in September, 1985. Radioactive liquid effluents and wastes are routed to the 1325-N disposal facility during normal operation. In case of an event that would initiate the ECCS, liquid waste would be automatically diverted to the 1312-N Liquid Effluent Retention Facility (LERF). 21

11.2.1.2 Release Control Evaluation

Table 11.2-1 shows a sample radionuclide release rate from N Reactor to the Columbia River in 1975 when the 1301-N facility was in use. The table compares the average release concentrations with the DOE 5480.11 MPCs for the general population and shows the Columbia River releases in curies. 21

Table 11.2-2 shows the annual population dose rates resulting from radionuclides that are released in N Reactor liquid effluents (Table 11.2-1).

The new 1325-N Liquid Waste Disposal Facility (LWDF) was identified as one of the several alternatives (Reference 2) to reduce N Reactor releases to the environs. New release data acquired after the 1325-N LWDF installation are shown in Table 11.2-9. Since the release to the groundwater from the retired 1301-N facility will continue for some time, the long-lived radionuclides released (Sb, Ru, Sr) at N Springs are combined with releases from 1325-N.

11.2.1.3 System Capability and Capacity

N Reactor's radioactive liquid waste treatment relies on three major facilities: (1) the 1325-N LWDF (1301-N facility has been retired), (2) the 1310-N storage tank, and (3) the 1312-N LERF for emergency events. The 1325-N LWDF is an engineered soil column treatment facility designed to reduce the quantity of radionuclides released from N Reactor to the Columbia River. A full description of this facility is provided in Section 11.2.2. Design parameters used for construction of the 1325-N LWDF included:

1. A design percolation rate of 40 gal per day per square foot (0.028 gpm/ft^2) of plan view area
2. Sizing to accept a normal continuous flow rate of at least 4,000 gal per minute (910 m^3/h)
3. Sizing peak design flow rate of 12,000 gal per minute (2,730 m^3/h) for 5 minutes occurring every 20 minutes
4. Sizing to take annual decontamination design preheat and final rinse flow rate of 20,000 gal per minute (4,550 m^3/h) for 10 minutes
5. Diversion capability to the 1312-N LERF in case of overflow or emergency
6. Sufficient difference in elevation between the outlet piping from the 1301-N covered weir box and the bottom of the 1325-N facility, allowing effluent flow to the new 1325-N LWDF by gravity

7. Approximately 25 ft to 30 ft but not less than 20 ft of virgin soil between the ground surface of the 1325-N facility and the average ground water elevation at the site;
8. A slope for the sides of the 1325-N facility not steeper than 1.0/1.5 (vertical/horizontal);
9. A long, continuous cover in order to prevent (1) sunlight from penetrating inside the covered basin and promoting growth of plant life, (2) the uncontrolled transport of radionuclides to the environs, and (3) the contamination of the bottom ground surface with wind-blown sand and other debris which tend to reduce percolation rates into the ground;
10. Sloping away the bottom of the 1325-N facility from delivery points as required to spread effluent water uniformly over the bottom surface of the facility;
11. Performing designed functions during and following the Design Basis Earthquake (DBE) and wind gusts near ground level of up to 70 mph.

The above eleven design parameters were based upon experience with other projects and facilities. The time required for ground water to travel from the 1325-N facility to the river bank is much longer than that from the retired 1301-N facility, estimated to be minimum of ninety days.

The retired 1301-N facility measures 125 feet by 290 feet. The overflow portion of the facility is 1600 feet long by 50 feet wide. Table 11.2-3 lists the radionuclides discharge to the 1301-N facility and the radionuclides which reached the Columbia River in 1975. Excluding tritium, less than 0.06 percent of the radionuclides discharged to the 1301-N facility reached the Columbia River. Neither soil column nor modern treatment systems can remove tritium from a large effluent stream. The average annual release of tritium to the river results in a negligible dose to the offsite maximum individual of approximately 0.0002 mrem/yr (Ref. 13).

The 1325-N LWDF basin measuring 240 ft by 250 ft (total 60,000 ft^2), is 1,000 ft East and 200 ft North of the 1301-N facility (Fig. 11.2-9). The basin can accommodate a water depth greater than 2 ft. The 1325-N overflow trench is 3,000 ft long, 25 ft wide at the bottom, and has a working water depth of 5 ft. The cover installed over the facility is designed for a uniform live load of 30 lb/ft^2 (146 kg/m^2) plus its dead weight. The live load provides for a maximum loading caused by snow or by wet volcanic ash (but not both acting at the same time). Table 11.2-9 lists the radionuclides discharge to the 1325-N facility and the radionuclides which reached the Columbia River in 1986. Note that the effect of 1301-N still exists. The release to the N Springs is not entirely from the 1325-N facility.

A failure downstream of the 1301-N weir box or the 36-in. line to 1312-N and 1325-N during an accident (e.g., earthquake) would result in the possibility of flooding ground areas near these facilities but because of the ground contour would not result in a surface flow of the liquid effluent to the Columbia River (Refs. 9 and 10).

In case of ECCS initiation, automatic closure of 2 diversion valves will redirect reactor coolant flow away from the 1325-N LWDF to the 1312-N LERF. The storage capacity of the 1312-N facility is 11.2 million gallons. If the capacity of the facility is reached, passive initiation of flow to the 1312-N bypass will occur. This bypass provides an alternative path to the 1325-N facility.

The 1310-N radioactive chemical waste storage tank, is 900,000-gal spherical tank used for collection of reactor decontamination chemicals and rinse water. Approximately 600,000 gal of this capacity is used during reactor decontamination.

The specifications for the 1310-N facility are found in HWS-6653, "Specifications for Chemical Waste Storage Facility Building 1310-N," (Ref. 3). These specifications include civil, mechanical and electrical criteria.

The steel plates for the waste tank conform to the requirements of ASTM A36 or A283, Grade D, except for the section noted on drawing H-1-32227 which conforms to ASTM A441. See Figure 11.2-1.

An epoxy coating was applied to the tank interior in 1975 to provide additional protection against potential leakage and to prolong the useful life of the tank. A program of inspection prior to each decontamination is used on the 1310-N facility to verify tank wall integrity.

The 1310-N tank has a level gaging system that uses an electrical conductivity plummet-type sensor and transmitter. The indicator reads out in the 105-N control room and on the tank. System detection accuracy is ± 1200 gal when the tank is half full.

11.2.1.4 Equipment and Design Parameters

Section 9.3.3, Equipment and Drainage Design Parameters, describes and lists the components and design parameters for the:

- 105-N building space and equipment drain system
- 109-N building radioactive drain and waste disposal system (RDR)
- Reactor coolant system flush and waste disposal system 105-N building
- Reactor coolant system loop low pressure drain system - 109-N building
- Decontamination waste line - 109-N building.

Section 5.4.3, Reactor Coolant Piping, describes and lists the components and design parameters for the drain line from the reactor coolant system to the 1325-N facilities. This section describes the equipment and design parameters for the 1325-N facility, the 1310-N waste storage tank, the 1314 waste shipping facility, and the 1312-N LERF. 10 21

The 1325-N facility is a liquid radioactive waste treatment facility that uses a soil column as the treatment medium. The 1301-N facility is a retired soil column facility shown in Figure 11.2-2. The 1325-N Liquid Waste Disposal Facility (LWDF) is currently used for liquid waste treatment during N Plant 21

operation. The 1312-N Liquid Effluent Retention Facility (LERF) would be used for retaining liquid effluent after ECCS initiation.

The chemical waste collection system is subdivided into two facilities: (1) the 1310-N chemical waste storage facility and (2) the 1314-N decontamination waste railroad car loading facility. Table 11.2-4 shows the components and design parameters for the chemical waste collection system. The seismic design classification for the 1310-N tank foundation, footings and components is found in Section 3.2. Prior to each use of this system for collection and transfer of radioactive waste, the 1310-N system is physically inspected, where practicable, and tested hydraulically to insure leak-tightness.

11.2.1.5 System Waste Inventories

The chemical waste storage and loading facilities normally do not contain any radioactive liquids. About once each year, following a through-reactor decontamination, chemical wastes are routed to the 1310-N facility for temporary storage until loading through the 1314-N facility and the subsequent transfer, shipment and disposal activities are completed. Normally, the loading and shipment of the chemical wastes to the 200 Area for evaporation and solidification requires about 10 weeks.

Table 11.2-5 gives an inventory of major corrosion products contained in the 1310-N tank following the 1975 internal reactor decontamination. The original criteria for design of the 1310-N facility assumed that an aggregate activity of 2000 curies of these selected radionuclides would be contained.

Tables 11.2-10 and 11.2-11 give the radionuclide inventories as of 1986 for the 1325-N LWDF and the 1301-N LWDF respectively.

11.2.1.6 Control of Overflows

The potential exists for overflow of the 1310-N tank during decontamination operations.. There are safeguards, however, to prevent inadvertent tank overflow: first, tank level is monitored in the reactor control room during decontamination operations; second, only 600,000 gallons of the 900,000-gallo

tank capacity is used for chemical decontamination and rinse waters; third, detailed procedures for reactor piping decontamination with comprehensive check sheets are provided. Prior to initiation of each decontamination, the decontamination crew has several meetings to review the procedures, responsibilities and methods of communications required by the procedures. If overfilling should occur, the overflow would consist, primarily, of low activity demineralized water since the tank is filled from the top and the last solutions put into the tank are rinse solutions. See Chapter 15 for 1310-N tank failure analysis. 21

The 1314-N facility is used to load the decontamination solution (stored in the 1310-N facility) into a 20,000-gallon rail tank car for transportation to the 200 Area waste treatment facility. Potential for overflowing the transportation tank car exists. On September 24, 1973, due to operator error, 175 gallons of solution containing 0.026 curies of radionuclides overflowed from this tank, requiring soil removal and burial in the 200 Area waste management area. To prevent recurrence of this problem, several changes in design (Ref. 4) were implemented, including the following:

1. improved liquid level and dome pressure monitoring;
2. installation of alarms for high tank level and high dome pressure;
3. installation of a 750-gallon overflow catch tank and level monitoring equipment;
4. installation of an audible and visual alarm at the loading station to alert the operator and the reactor control room supervisor to high tank level, high dome pressure or high level in the new overflow tank;
5. provision for weather protection for new instrumentation and existing control equipment at the 1314-N loading station; and
6. the addition of new controls that will shut down pumping operations automatically should high level or high dome pressure occur in the rail tank car.

These physical safeguards should prevent future tank car overflows. 21

The 1325-N LWDF is not expected to overflow even under extreme conditions. If it does, the surrounding ground contour is such that directed flow to the river is prevented. If 1312-N is filled with ECCS flow during an emergency, the 1325-N facility becomes the overflow control.

11.2.2 Systems Description

Figure 11.2-3 provides a general schematic of 100-N reactor liquid effluent streams, waste facilities and sources. The primary liquid radwaste treatment system for N Reactor is the 1325-N facility with 1312-N emergency diversion. The 1301-N facility has been retired since September 1985. Two other facilities collect, store, and transfer liquid effluents containing significant quantities of radionuclides: the 1310-N and the 1314-N decontamination waste facilities. A minor amount of radioactivity is released directly to the Columbia River via the 102-inch line (see Table 11.2-1).

11.2.2.1 1325-N Liquid Waste Disposal Facility

The 1325-N facility, consisting of a 240 ft by 250 ft crib (Project H-616) and a 3,000-ft long trench (Project H-672), is a new liquid radioactive waste disposal system for N Reactor. Radioactive liquid effluents from the reactor coolant system, spent fuel storage basin, and various radioactive drain systems throughout the reactor facility are discharged to the 1325-N facility.

The 1325-N facility is an engineered ground disposal facility that makes use of the natural filtration properties of soil to remove radioactive material from effluent water. The soil layer lying between the facility and the river will function both chemically and physically to remove radioactivity. Chemically they act as an immense ion exchange column. The filtration of particles and the travel time for radionuclides to reach the river allows short half-lived isotopes such as Iodine-131 to decay significantly enroute.

The 1325-N facility has 1,300 ft of pipeline connecting to the header box. The 1325-N facility is approximately 1,000 ft East and 200 ft North of the retired 1301-N facility. (Figure 11.2-9). The 1325-N facility is located such that effluent from it avoids the contaminated soil beneath the retired

1301-N facility and increases the time required for ground water to travel from the new facility to the river bank (Reference 5). 21

The 1325-N LWDF is covered and has a percolation surface measuring 240 ft by 250 ft (total 60,000 ft^2) at the invert. The East and West sides of the facility are sloped 1.0 ft vertical to 1.5 ft horizontal. The facility crib is connected to an overflow trench through a weir box constructed at its northeast side. The sides and bottom of the weir box are constructed of reinforced concrete. The bottom of the weir box is located at an elevation below the bottom of the LWDF wall to allow water to flow under the wall from the LWDF into the weir box. The weir is designed so that water overflows into the new trench when the water rises to a depth of three feet in the 1325-N LWDF.

The disposal trench is shown on Figure 11.2-9. It is 3,000 ft long, 25 ft wide at the bottom, and has a working water depth of five feet. The sides of the trench are sloped at a rate of 1.0 ft vertical to 1.5 ft horizontal. The trench bottom elevation is 437 ft MSL over its entire length. 21

The trench is divided into four impoundment sections by reinforced concrete dams approximately 750 ft apart; these dams cause the water level to rise to its working depth of 5 ft before spilling over to the next impoundment section. Such an arrangement maximizes the percolation rate, which increases directly with water depth, and maintains the far reaches of the trench in a dry condition so that percolation is maximized in those areas during postulated upsets or surge conditions.

The entire length of the trench, including the weir box, has a solid, continuous cover with the following functions:

- Prevent the uncontrolled transport of radionuclides to the environs
- Provide protection of the contaminated area from intrusion by animals, birds and large insects
- Exclude sunlight from the trench so as to discourage the growth of plant life
- Prevent the intrusion of wind-blown sand and other debris into the trench.

Pre-cast concrete panels (approximately 182,000 ft^2) cover the trench at the same elevation as the cover for the 1325-N basin.

The facility is designed to infiltrate at a rate of 4,320,000 gal per day, a continuous average design flow rate. At peak design flow rate of 12,000 gpm for 5 minutes, occurring every 20 minutes for 24 hours; no overflow will occur.

Water discharged to the 1325-N facility percolates through the underlying soil to an unconfined aquifer in communication with the Columbia River. During the movement of water in the ground underlying the facility, radioactive materials are retained in the soil through precipitation and ion exchange reactions. Additionally, decay of the shorter-lived radionuclides in the discharge water occurs during transit to the river. After operating the 1325-N facility for some time, a detailed hydrological analysis was performed to evaluate related hydrological factors influencing the flow pathways from the 1325-N facility to the river (Ref. 12). Experience gained from operating 1301-N facility is documented in Section 11.2.2.2. The cumulative radionuclide inventory, including Co-60, Sr-90, Pu-239/240, etc., in 1325-N LWDF from 1983 through 1986 is listed in Table 11.2-10.

Control of the amount of liquid effluents routed to the 1325-N facility is accomplished via design and operating procedures. See section 13.5 for discussion of operating procedures.

Control of chemicals added to the reactor and other coolant systems is maintained via (1) Procurement Specifications and Quality Control Chemical Analysis (see Chapter 17), (2) Technical Specifications and (3) the Environmental Control Manual procedures. These specifications, analysis and procedures prevent the inadvertent introduction of noncompatible chemicals to both the reactor coolant and the 1325-N facility. 10 21

11.2.2.2 1301-N Liquid Waste Disposal Facility (Retired) 21

The retired 1301-N facility, similar to the 1325-N facility, is also an engineered ground disposal facility that makes use of the natural filtration properties of soil to remove radioactive material from effluent water. The facility was successfully operated for more than 20 years, and has been retired since September 1985. 10 21

The 1301-N facility is located on the river bank bluff, some 60 ft above and 860 ft back from the Columbia River. The facility plan is shown in Figure 11.2-4. The facility is 125 ft by 290 ft, created by excavating the existing soil and then surrounding the excavation with a soil and gravel embankment. The bottom of the basin is filled with a 3-ft layer of large stones. An extension trench is connected to the north side of the basin. 10 21 10 21

The extension trench is an excavated ditch approximately 1600 ft long, 50 ft wide and 12 ft deep, extending in a zigzag pattern. A pre-cast concrete cover is provided over the entire trench to prevent larger animals and birds from obtaining food, water or shelter within the trench. See Figure 11.2-5. 10 21

Hydrological analyses have indicated that flow pathways to the river are a complex system in that the water may enter the river from beneath the river bed or from springs on the river shoreline, and that the pathways may lead directly to the river, or may parallel the river for several miles before entry. Radionuclide sampling studies have shown that the most radioactive spring water enters the river via the shortest pathway. Figure 11.2-6 illustrates the results of an evaluation of hydrological factors influencing the flow pathways from the treatment facility to the river using a mathematical simulation technique. Figure 11.2-2 depicts a cross sectional view of the 1301-N facility.

Table 11.2-3 gives a summary of the sampling data for 1975 releases to the 1301-N facility. The effectiveness of the disposal system is indicated by the degree of decontamination provided by the facility. The variation in removal capability for different radionuclides is dependent on the half-life and the chemical properties of the radionuclide involved.

Table 11.2-6 gives results from a study performed to determine the physicochemical composition of radionuclides discharged to the facility. In the study, radionuclides were partitioned into their particulate, cationic, anionic and nonionic fractions. Since the 1301-N facility utilizes a natural soil column composed of sand (particulate filtration) and clay (cation exchange), the column is noticeably more effective for radionuclides predominantly in the particulate and cationic forms. As indicated in Table 11-2.3, the predominantly anionic and nonionic species such as iodine, ruthenium and antimony were removed by the soil column to a lesser extent than the particulate and cationic radionuclides such as iron, cesium, zirconium and manganese.

Some radionuclides exhibit characteristics indicating a mixture of the various physicochemical forms. Radionuclides of this type emerge at the river springs in essentially anionic and nonionic forms because their particulate and cationic fractions are retained by the soil column. Tritium oxide, which is chemically indistinguishable from water, is not removed during transit to the

river. Table 11.2-7 gives results of liquid effluent decontamination factors from 1971 to 1975.

The discharge of radioactive liquids to the 1301-N facility since N Reactor startup in 1963 has resulted in the accumulation of significant quantities of long-lived radionuclides in the soil on the bottom of the facility. Table 11.2-8 lists quantities of fission and activation products contained in this soil as of 1975. Table 11.2-11 lists the final inventory in the retired 1301-N LWDF as of 1986. 10 21

11.2.2.3 Decontamination Waste Facilities

Approximately 600,000 gallons of liquid effluents from reactor loop are collected in the 1310-N storage facility following a decontamination. The 1310-N facility is a 900,000-gallon capacity, spherically-shaped, steel storage tank located inside the northeast corner of the 100-N Area fence. See Figure 11.2-7. A 19-foot thick, earth-fill shielding wall surrounds the tank except the area adjacent to the 1301-N facility. The shielding wall reduces the dose rate from the radionuclides contained in the tank such that the dose rate at the nearest office building is less than 120 micro rem/hr. 10

The decontamination solution is routed to the 1310-N facility by closing the valve (FLV 802-1) on the 36-inch line to the 1325-N facility and opening the valve (FLV 801-1) to the 1310-N tank. Once the 250,000 gallons of 8-weight-percent phosphoric acid and 350,000 gallons of rinse water are in the tank, the valve sequence is reversed, and the 36-inch line returns flow to the 1325-N facility. Sodium hydroxide is then added to the 1310-N tank to neutralize the acid, and the recirculation pump is turned on to reduce the buildup of sludge in the tank bottom. 21 10

Shipment to the 200 Areas is accomplished by transferring the solution from the 1310-N tank into 20,000-gallon rail tank cars. The transfer is accomplished via the 1314-N tank car loading facility. See Figure 11.2-8 for the 1310-N and 1314-N facility flow schematic. The decontamination solution is transferred through a waste pump (S-1908) to a fill line and into the

20,000-gal tank. When the tank car is loaded, the lines are disconnected and the car transported to the 200 Area for solution evaporation and storage.

11.2.2.4 1312-N Liquid Effluent Retention Facility

This plant modification provides increased protection against potential contamination of the ground water from liquid radioactive effluent discharged from the reactor during postulated emergency events.

During normal reactor operation, excess low-level radioactive liquid is piped to the 1325-N LWDF through the weir box located in the retired 1301-N facility. Before the completion of Project H-797 in September 1987, the 1312-N facility was an unlined earthen basin whose purpose was to receive radioactive liquid overflow from the 1301-N and 1325-N LWDFs in the event of an Emergency Core Cooling System (ECCS) actuation.

The 1312-N LERF and portions of the Radioactive Drain System (including the 1301-N weir box) have been modified by Project H-797 as follows: the 1312-N LERF is double-lined and provided with a floating cover. See Figure 11.2-10. The new design of the basin is such that approximately 96% of the liquid-borne iodine-131 (I^{131}) anticipated to be released during the hypothetical accident will be contained (Ref. 11). Based on this requirement, the basin has been increased in capacity from 5.5 million gallons to 11.2 million gallons. The charcoal filtered vents were provided in the cover to minimize the airborne release of I^{131} to the atmosphere and also to help ensure liner integrity. A leak detection and collection system was installed to monitor liner integrity.

Flow to the 1325-N LWDF would be automatically diverted to the 1312-N LERF on activation of the ECCS. If the capacity of the 1312-N LERF is reached, passive initiation of flow through the 1312-N bypass will occur. This bypass was constructed between the 1312-N LERF and the 1325-N LWDF to provide an alternate path to the 1325-N facility. The flow path and location of the LERF is illustrated in Figure 11.2-11.

New valves were installed in the existing 1325-N LWDF line to divert the flow to the 1312-N LERF on ECCS actuation. Annunciation has been provided in the 105-N control room for valve operation, low-temperature warning for system components (for freeze protection), loss of power to the diversion system, low-pressure warning (valve actuating devices), and the 1312-N LERF leak collection and detection system. Surveillance requirements for liner leak integrity are given in Reference 8.

For relief of gas that might be trapped under the cover, six vents containing activated charcoal for radioiodine removal, are supplied. 21

The existing 1301-N LWDF weir box was modified to preclude overflow to the retired 1301-N facility during the predicted high ECCS flow rates. A standpipe downstream of the weir box ensures that passive pressure relief will occur at the 460-ft elevation to preclude possible damage to upstream components in the Radioactive Drain System. To increase system reliability, the inlet to the 1312-N facility has been designed to allow passive initiation of flow to the 1312-N facility. Flow into the 1312-N facility would initiate when the level in the 1301-N weir box exceeded 453.5 ft AMSL.

11.2.2.5 Stream Segregation and Routing

Figure 11.2-3 provides a simplified flow schematic of N Reactor liquid effluents. The 102-in. routes 300,000 gal per day of liquid into the Columbia River during reactor operation. Two minor sources of radionuclides drain into the 102-in. line, and four other sources have potential for leakage into the line as discussed below.

The two sources of radionuclides going into the Columbia River via the 102-in. line are (1) steam generator blowdown and (2) gas penetration coolant. Normal flow from the stream generator blowdown is to the 102-in. line; however, should the steam generator leak rate exceed 50 gal/h, the blowdown is routed to the 1325-N facility by way of a concrete dump basin and thence through the 36-in. radioactive drain line until the leak rate has been reduced to less than 50 gph. The gas penetration cooling drain has a measurable 10

concentration of manganese-56 (see Section 11.1). There is no diversion capability for this stream.

Four other sources that empty into the 102-in. river discharge line have potential for radionuclide leakage from the recirculating coolant to the heat dissipation coolant. Section 11.5.1.1 describes the monitoring program for leakage from these systems. In the event a leak is identified, the reactor is shut down and the failed components repaired.

11.2.2.6 Seismic Considerations

N Reactor site seismic considerations are covered in Section 2.5. Section 3.7.3 provides general information on seismicity for buried piping and piping penetrations at building walls which are applicable to waste disposal piping.

11.2.2.7 Operational Occurrences

Operational occurrences that could result in increased amount of radionuclides entering the Columbia River are fuel cladding failures, as discussed in Section 11.1.2, and leaks in the four heat-exchanger dissipation systems, as mentioned in Section 11.2.2.5. The course of action currently employed to correct a potential leak is to shut down N Reactor and fix the leak.

Operational occurrences could result in infrequent transfer and recirculating rod coolant or graphite and shield coolant to once through cooling and increase, in the short-term, the quantity of radionuclides going to the 1325-N facility. However, because of the hold up time, no measurable increased quantity of radionuclides to the river is expected.

11.2.3 Radioactive Releases

Table 11.2-1 shows the concentration and quantity of radionuclides released from N Reactor to the Columbia River in 1975 when the 1301-N facility was in use. The table compares the average release concentrations with DOE 5480.11 MPCs for the general population (Reference 6) and shows the Columbia River

releases in curies per year. Table 11.2-2 shows the associated annual dose rate received by the maximum and the average individual.

The new 1325-N Liquid Waste Disposal Facility (LWDF) was identified as one of several alternatives (Reference 2) to reduce N Reactor releases to the environs. Most of the radionuclides entering the 1325-N LWDF are retained in the soil. New release data acquired after the 1325-N LWDF installation are shown in Table 11.2-9. Since the release to the groundwater from the retired 1301-N facility will last for some time, the radioactive liquid releases of a few long-lived radionuclides (Sb, Ru, Sr) at N Springs are actually combined releases from both 1325-N and 1301-N facilities. 21

A series of wells, drilled for subsurface stratigraphy investigation are used to monitor the migration of radionuclides away from the soils directly underlying the 1301-N, 1325-N and 1312-N facilities at 100-N area, see Figure 11.2-9. These wells have been finished for use in radiation monitoring by placing a section of screen in each well, extending above and below the present water table to allow for changes in the hydrologic system resulting from discharges to the trench. The bottom of the screen was capped to prevent backfill material from entering the well. Also, a cement surface seal has been poured around each casing and a removal cap for each well provided. Each of the monitoring wells have been provided with a small submersible sampling pump. Power to these pumps have been provided as needed by portable generators.

The 1310-N facility could develop a leak after being filled with decontamination solution containing radionuclides. However, offsite population doses from such an occurrence would be insignificant because of the cleanup ability of the dry soil directly beneath the facility and increased efforts that would be made to ship the remaining solution from the facility to the 200 Area for treatment and disposal. See Chapter 15 for analysis of a postulated tank leak. 21

Increased frequency of fuel cladding failures could also increase releases of radionuclides to the 1325-N facility. This is unlikely because fuel irradiation experience and continuously improving fuel quality control have reduced the probability that increased frequency of fuel cladding failures will occur in future years. Also, N Reactor is rapidly shut down when a cladding failure is confirmed and the fuel with failed cladding is then removed. This minimizes the release or radionuclides to the 1325-N facility from the failed cladding and results in no measurable effect on the quantity of radionuclides released to the river.

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TABLE 11.2-9

RADIOACTIVE LIQUID RELEASES FROM THE 100-N AREA

21

(Discharges to the 1325-N LWDF and to the Columbia River via seepage from the N Springs in 1986)

Radionuclide	1325-N LWDF Release Ci	1325-N LWDF Average Concentration pCi/L	1325-N LWDF Peak Concentration pCi/L	N SPRING Release Ci	N SPRING Average Concentration pCi/L	N SPRING Peak Concentration pCi/L	DECONTAMINATION FACTOR
H-3	2.2E2	8.3E4	4.0E5	2.2E2+	4.7E4	7.9E4	1
P-32	1.4E1	5.3E3	2.5E4	5.2E-3+	4.0E0	4.1E1	2,700
Cr-51	6.9E1	2.6E4	1.2E5	*			
Mn-54	2.7E2	1.0E2	1.0E5	7.0E-3	5.3E0	8.0E0	40,000
Fe-59	1.7E2	6.2E4	5.7E5	*			
Co-58	6.2E3	6.2E3	5.1E4	*			
Co-60	3.9E2	1.5E5	1.5E6	1.2E-1	9.3E1	1.5E2	3,300
Zn-65	1.0E1	3.9E3	2.1E4	*			
Sr-89	2.0E2	7.6E4	3.5E5	1.7E0	1.2E3	6.8E3	NA
Sr-90	3.6E1	1.4E4	4.5E4	7.9E0	5.9E3	7.3E3	NA
ZrNb-95	2.1E2	7.8E2	5.4E5	*			
MoTc-99m	8.9E2	3.4E5	1.2E6	*			
Ru-103	4.3E1	1.6E4	7.0E4	4.9E-2	3.7E1	1.2E2	880
Ru-106	4.9E1	1.9E4	6.2E4	1.2E-1	8.9E1	1.8E2	410
Sb-12	4.7E0	1.8E3	6.9E3	*			
Sb-125	1.2E1	4.7E3	7.0E3	1.6E-1	1.2E2	1.6E2	75
I-131	4.3E2	1.6E5	6.4E5	3.0E-2	2.3E1	1.3E2	14,000
Xe-133	1.0E3	3.9E5	2.3E6	2.6E-2	2.0E1	2.1E2	38,000
Cs-134	7.4E0	2.8E3	5.3E3	*			
Cs-137	2.1E2	8.0E4	4.8E5	*			
Ba-140	1.5E2	5.5E4	1.5E5	*			

TABLE 11.2-9

RADIOACTIVE LIQUID RELEASES FROM THE 100-N AREA

21

(Discharges to the 1325-N LWDF and to the Columbia River via seepage from the N Springs in 1986)

	1325-N LWDF			N SPRING			DECONTAMINATION FACTOR
Radionuclide	Release Ci	Average Concentration pCi/L	Peak Concentration pCi/L	Release Ci	Average Concentration pCi/L	Peak Concentration pCi/L	
Ce-141	5.5E1	2.1E4	6.6E4	*			
CePr-144	9.4E1	3.5E4	1.5E5	*			
Pu-238	3.5E-2	1.3E1	5.0E1	2.8E-6	2.1E-3	1.2E-2	13,000
Pu-239/240	2.4E-1	9.0E1	3.5E2	1.2E-5	9.1E-3	8.9E-2	20,000
Np-239	3.1E3	1.2E6	1.8E7	*			
SLR	5.2E4	2.0E5	3.9E7	**			

\+ This is the same value as for H-3 (tritium) discharge to the 1325-N LWDF since all H-3 released there is assumed, because of its affinity for water, to eventually, though not necessarily within the time of one year, reach the Columbia River via the N Springs which are fled by the groundwater system underlying 1325-N. The average and peak concentrations represent the analyses of actual CY 1986 N Springs seepage samples.

* Indicates radionuclides in particulates form or with a high ionic-exchange potential which are trapped within the 1325-N LWDF soil column and removed beyond detection limits before reaching the seepage of the N Springs.

** Indicates short half-line radionuclides which have decayed beyond detection limits before reaching the seepage of the N Springs.

SLR= Short-lived radionuclides ($T^{1/2}$ 48 hours).

NA= The travel time of Sr-89 and Sr-90 from 1325-N to the N Springs is not yet determined, but estimated to be two years or more.

TABLE 11.2-10

CUMULATIVE RADIONUCLIDE INVENTORY IN 1325-N LWDF

1983 THROUGH 1986

(Curies)

21

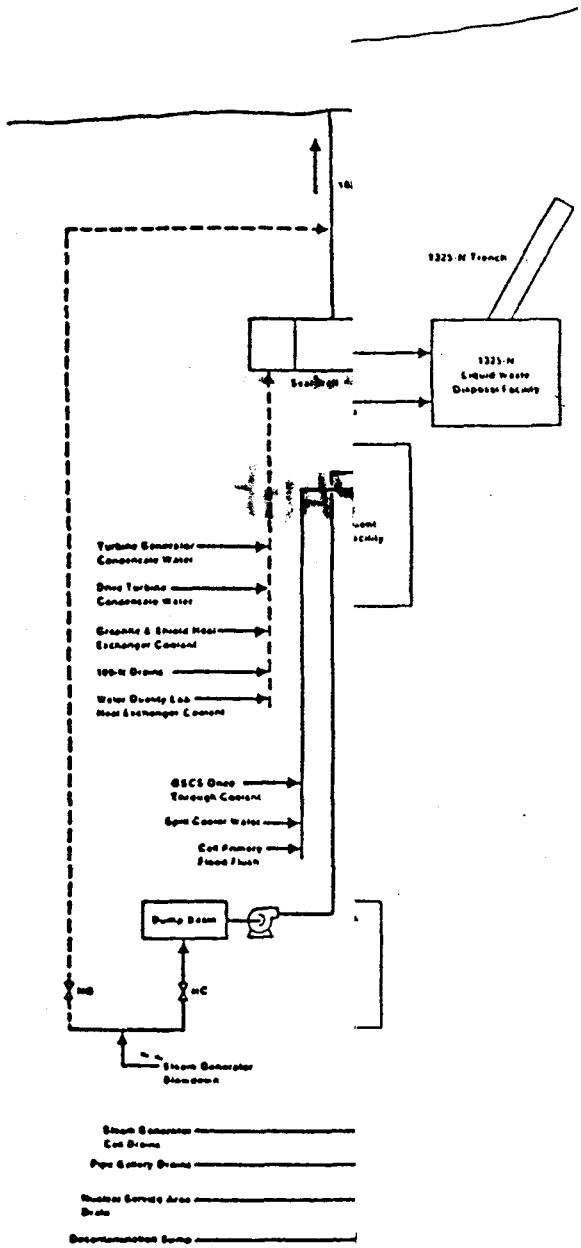
Radionuclide	Half-Life	1983	1984	1985	1986
Co-60	5.3y	160	400	510	810
Sr-90	29y	31	88	150	180
Ru-106	373d	10	23	28	49
Cs-134	2.1y	2.5	4.3	4.5	9.5
Cs-137	30y	44	82	110	310
Pu-239/240	2.4E4y	.44	1.2	2.1	2.4

TABLE 11.2-11

FINAL RADIONUCLIDE INVENTORY IN 1301-N LWDF (as of 1986)

(Curies)

Radionuclide	Half-Life	1986 Final Inventory
Co-60	5.3y	3900
Sr-90	29y	2100
Ru-106	373y	59
Cs-134	2.1y	44
Cs-137	30y	2800
Pu-239-240	2.4E4y	23



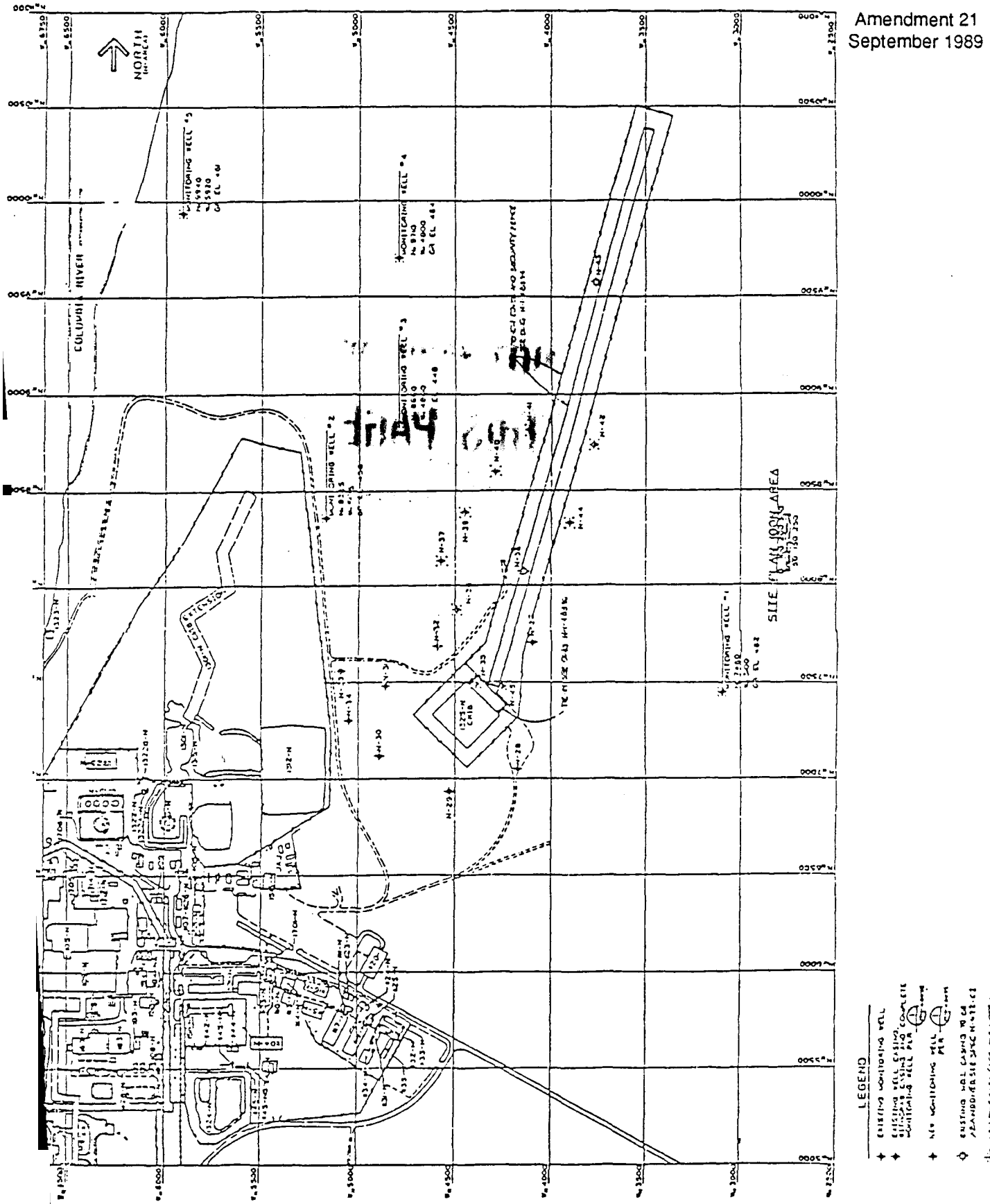
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Liquid Effluent Flow Schematic

Figure 11.2-3

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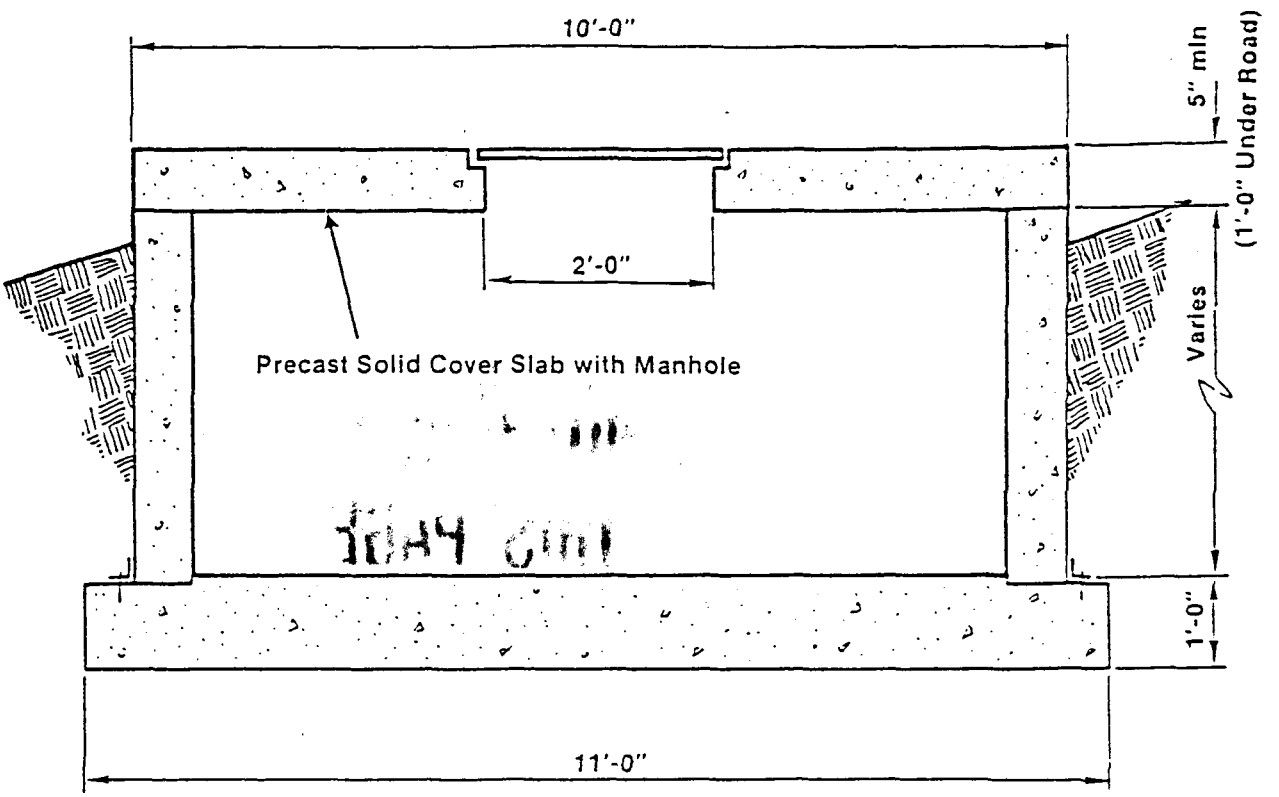
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1325-N Liquid Waste Disposal Facility

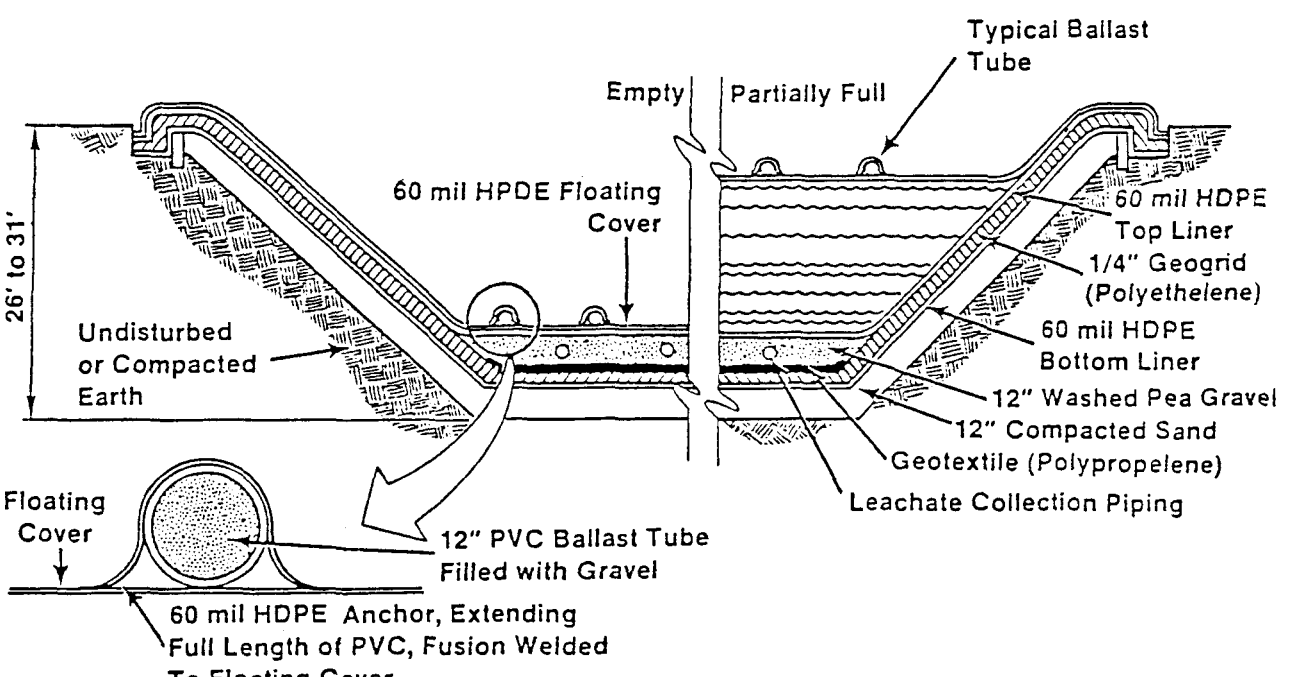
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Basin Elevation

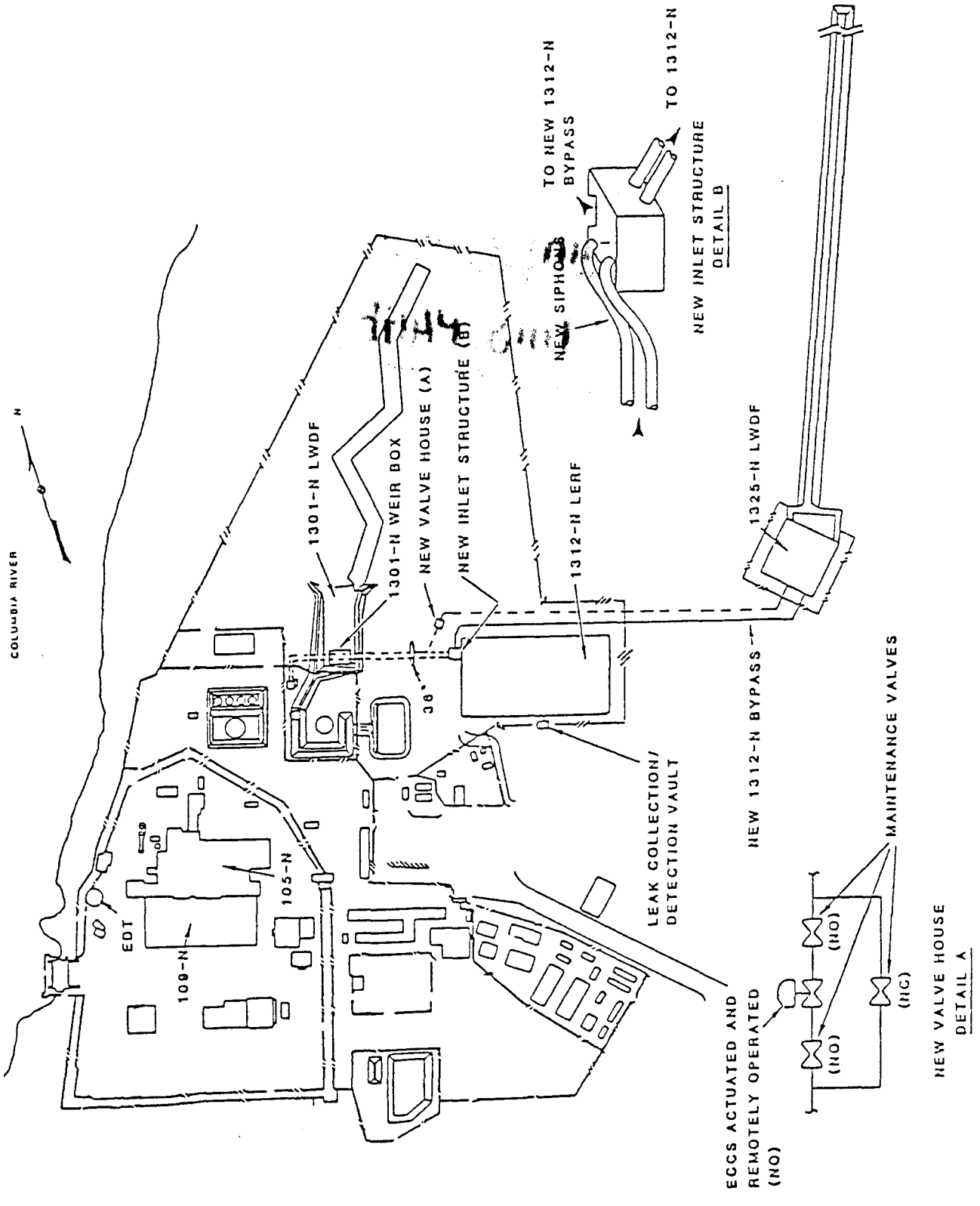


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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Liquid Effluent Retention Facility (LERF) Bypass and Basin Elevations Figure 11.2-10

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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	1312-N LERF Flow Path and Location Description
	Figure 11.2-11

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15.6 DECREASE IN REACTOR COOLANT INVENTORY

Section 15.6 provides an evaluation of events which could result in a decrease in the RCS inventory. The events include a spectrum of break sizes and locations in the major piping and components and single pressure tube events. Detection of sub-critical pressure tube cracks (leak before break) by monitoring gas dew point is discussed in Section 9.3.6. The single pressure tube events include a complete flow blockage of a pressure tube since it is a limiting event and has the potential for causing failure of the pressure tube. Section 15.6 also includes an evaluation of an hypothetical accident involving a LOCA coupled with an assumed nonmechanistic failure to provide direct ECCS Cooling to the reactor pressure tubes. 17

A variety of computer-based analysis tools have been used in the evaluation of the N Reactor response to the transient events postulated in this section. The major system computer analysis codes used in this section are RETRAN, RELAP4, and RELAP5. Each computer analysis code has been "bench marked," as described in each of the following sections.

The scale and consequences of the April 1986 accident at the Chernobyl Nuclear Power Station in the Soviet Union provided the impetus for reexamination of the U.S. Department of Energy defense production reactors. The reviews resulting from the reexamination of the N Reactor included recommendations for new accident analyses utilizing a combination of Best Estimate Analyses (RETRAN, HECTR, RELAP5 usually) to ensure that "conservative" calculations did not mask important system performance characteristics. Following the analyses of several transients with best estimate assumptions, conservative RELAP5 analyses of the limiting LOCA were performed utilizing 10CFR50, Appendix K as guidance. This conservative analysis, in conjunction with existing RELAP4 "conservative" analyses and best estimate analyses (RELAP5 and RETRAN) demonstrate that important system performance characteristics are not "masked," and provides the safety basis for the limiting LOCA. 21

In general, the new analyses performed utilizing RELAP5 have rendered the previous RELAP4 analyses, for the same events, superfluous. The original

RELAP4 analyses have, however, been retained in the NUSAR to provide the historic context for the N Reactor safety analysis. The RELAP5 analyses confirmed the conservative nature of the RELAP4 analyses for the blowdown phase of a loss of coolant transient. The RELAP5 analyses merely extended the analyses more thoroughly through the reflood phase of a loss of coolant transient.

The event which used RETRAN as the analysis tool is "Inadvertent Opening of an RCS Safety Valve," Section 15.6.1.

The events which used RELAP4 and are still considered sound, with no new RELAP5 analysis, are: Section 15.6.4.1, "Partial Failure of a Single Pressure Tube or Inlet Connector;" Section 15.6.4.2, "Complete Flow Blockage to a Pressure Tube;" Section 15.6.4.4, "Double-Ended Pressure Tube Break;" Section 15.6.5.3.2, "Cold-Leg Crossover--Four SG Cell Operation--55 Percent Split;" Section 15.6.5.4.1, "Hot-Leg Manifold Breaks, Double-Ended and 100 Percent Split;" Section 15.6.5.5, "Pump Suction Breaks, Double-Ended and 100 Percent Split;" Section 15.6.5.6, "Steam Generator Manifold Break;" Section 15.6.5.7, "Accidental V-4 Valve Actuation;" and Section 15.6.6, "Hypothetical Accident."

The events which used RELAP5 are: Section 15.6.4.5, "Minor Reactor Coolant System Pipe Breaks;" Section 15.6.5.3.4, "Determination of the Limiting LOCA;" Section 15.6.5.3.6, "Cold Leg Riser Break;" Section 15.6.5.3.8, "Cold Leg Manifold Break, Double-Ended Guillotine and Station Blackout;" and Section 15.6.5.4.3, "Hot Leg Riser Break, Double-Ended."

The RELAP4 analyzed events which have been rendered superfluous by the RELAP5 analyses, identified above, but which remain in the NUSAR to provide historic context are: Section 15.6.5.3.1, "Cold Leg Crossover Breaks, Double-Ended and 40 Percent Split;" Section 15.6.5.3.3, "Cold Leg Crossover Breaks, Double-Ended and 20 Percent Split;" Section 15.6.5.3.5, "Cold-Leg Riser Breaks, Double-Ended and 20 Percent Split;" Section 15.6.5.3.7, "Cold-Leg Manifold Breaks, Double-Ended and 100 Percent Split;" and Section 15.6.5.4.2, "Hot-Leg Riser Break, Double-Ended."

15.6.1 Inadvertent Opening of an RCS Safety Valve

RCS pressure relief is provided by the following valves in conjunction with an automatic reactor shutdown on system over-pressurization:

Valve	Number	Size	Function
RV-2	2	4 in. x 6 in.	RCS primary relief
RVTV-202	1	1 in.	Pressurizer relief when isolated during reactor shutdown
PCRV-210	6 (one/cell)	2 in.	Cell relief when isolated

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The ECCS V-4 dump valves also would provide pressure relief on over-pressure that is accompanied by abnormally high RCS bulk coolant temperatures. See Section 15.6.5 for analysis associated with ECCS actuation.

A study (Ref. 28) has investigated the effects on the primary system of small pipe breaks. This study concluded that pipe failures of a size that includes the smaller primary system PRVs will not cause the pressurizer to drain, and thus primary loop pressure control is maintained. From these results, it was concluded that further RETRAN analyses were unnecessary, since failures of these smaller PRVs are enveloped within the RV-2 failure results.

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The following sections evaluate the effects of inadvertent opening of one of the two RV-2 safety valves with or without loss of an electrical bus. The RV-2 valves would discharge into the ECCS dump line to the EDT tank. The valves are hydraulic pilot operated and are independent from each other. See Section 5.2.2.

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15.6.1.1 Identification of Cause and Frequency Classification

The most probable cause of spurious operation of an RV-2 valve is leakage through the pilot valve seat. This mode of failure occurred and opened an RV-2 valve in January 1965. The succeeding sections are both an account of that event and a description of results of primary loop overpressure relief studies. As a result of the January 1965 occurrence, the pilot valves were replaced with a more reliable and leak-free type, developed by the U.S. Navy

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for nuclear submarine coolant systems. Although the replacement pilot valves have not caused a malfunction, valve leakage is considered the most probable failure mode. The frequency classification for this accident is judged to be "Infrequent."

15.6.1.2 Sequence of Events and System Operation

15.6.1.2.1 Relief System Arrangement and Settings

The RV-2 safety valve settings are coordinated with the reactor scram high pressure trip and the ECCS extremely high pressure trip (required in conjunction with RCS high temperature or other out of limit condition to actuate ECCS). This allows a scram to occur before safety valve actuation. One safety valve opens at a lower pressure than the ECCS high pressure trip. These settings are as follows:

Reactor scram trip setting -	1655 psig
Safety valve RV-2-1	1700 psig (open)
	1200 psig (reset)
Safety valve RV-2-2	1775 psig (open)
	1200 psig (reset)
High pressure ECCS actuation -	1740 psig

An automatic ECCS actuation requires two independent ECCS trip signal inputs. Any two of the following trips will cause an actuation:

- o 2 of 3 high/low primary pressure
- o 2 of 4 primary coolant low flow
- o 4 of 6 low pump speed
- o 2 of 3 high primary temperature

In addition high confiner pressure plus low primary pressure will actuate the ECCS.

All of the RCS pressures are sensed on the hot-leg section of the Primary Coolant Loop piping. The RV-2 safety valves are connected to the hot-leg section, as shown in Section 5.2, Figure 5.2-2.

The RV-2 safety valve settings are deliberately spaced to preclude unnecessarily opening both of them simultaneously. One valve has enough capacity, approximately 1300 pounds/second, to depressurize the RCS for those types of events requiring pressure relief. See Section 5.2.2.1 for the design basis.

15.6.1.2.2 Effect of an RV-2 Valve Failure With no Loss of AC Bus

The effect of a single RV-2 valve failing open at full power conditions have been investigated with the RETRAN thermal-hydraulic analysis code to determine if a RV-2 valve failure would cause an ECCS actuation. To be conservative, a worst case scenario has been postulated whereby the RV-2 valve fails and sticks open. The loss of coolant causes a rapid pressure decrease, resulting in a low pressure scram about two seconds after valve failure. RETRAN calculations indicate that after the scram a maximum of 8900 gpm flow through the dump line. The pressurizer and surge line would drain in approximately 30 seconds. At 180 seconds the PCL begins to refill and repressurize.

The RV-2 opening would inevitably result in a low primary pressure trip signal. The second ECCS trip signal could occur on either low flow, or low pump speed if the primary coolant pumps have inadequate net positive suction head (NPSH) and experience cavitation. The RETRAN analysis indicates a minimum primary coolant pump suction pressure of 307 psia. Based on the pump performance data available, the minimum pump suction pressure necessary to avoid cavitation at that time (post-scram pump speed) would be 193 psia. Therefore, even allowing for pump-to-pump variations in coolant temperature, pressure, and pump rpm, there should be no danger of cavitation occurring.

Since the RETRAN analysis also shows that maximum coolant temperatures and minimum flows are far from ECCS trip values, it appears there is no possibility of an ECCS actuation in the event of a stuck open RV-2 valve, with no concurrent electrical bus loss.

15.6.1.2.3 Effect of a Stuck Open RV-2 Valve in Conjunction With Loss of an Electrical Bus

An additional RETRAN analysis was performed to determine if an ECCS actuation would occur if it were assumed that power from one electrical bus were lost at the same time the RV-2 valve failed open. The bus loss has the effect of reducing the number of high pressure injection pumps from four to two.

For that case, the peak dump line flow and the time the pressurizer drains is substantially the same as the stuck open RV-2 only case. However, with only two injection pumps on line, no repressurization will occur. In addition, there are short duration periods of two-phase flow in the outer annulus downstream end of an average power fuel column, and in the rear riser and outlet manifold, occurring about 100 seconds after the RV-2 valve failed. The amount of boiling was too low to perturb the coolant flow sufficiently to cause an ECCS low flow trip. Coolant pressures for this case are lower than the stuck open RV-2 valves only case, but it appears unlikely that main coolant pump cavitation will occur. Hence it is unlikely that an ECCS actuation will occur for this case. Recent studies have shown that there is no break size that will lead to fuel damage prior to ECCS actuation (See Section 15.6.4.5) and ECCS actuation, in conjuction with 2 HPI pumps, can mitigate the event should ECCS logic be completed by pump cavitation.

In addition to the above analysis, the possibility of an ECCS actuation as a result of a tube flow ECCS trip due to boiling induced flow perturbations for high power tubes was addressed. A series of RETRAN analyses were completed to quantify the probability of this occurrence including tube powers up to 5100 KW. These analyses showed coolant flow perturbations due to boiling in the flow channels and rear connector, but none severe enough to initiate an ECCS flow trip.

It can therefore be concluded that an ECCS actuation will not occur during a RV-2 failure with concurrent loss of one electrical bus.

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15.6.1.2.4 Description of Spurious Relief Valve Opening Event

On January 25, 1965, the right side RV-2 valve (now designated RV-2-2) spuriously opened at a time when the reactor was operating close to the design power level of 4000 MWt. The reactor was scrammed automatically upon reduction in RCS pressure or pressurizer water level.

Pressure in the system dropped from 1440 psig to 870 psig within 60 seconds and continued to decrease at a slower rate to the hot-leg saturation pressure of 430 psig at 240 seconds. It then rapidly increased to 760 psig as a result of the RV-2 reclosing and the pressurizer being refilled to approximately normal level.

Loss-of-coolant through the RV-2 valve caused a rapid reduction in pressurizer level. This rate of decrease was accelerated by coolant shrinkage that occurred as a result of the rapid reduction in power level caused by the scram. Within 19 seconds the pressurizer level dropped 21 feet (equivalent to 6150 gallons) to below the level instrumentation zero reference point of 5-1/2 feet of water in the pressurizer. The water level recovered to the zero reference point at 163 seconds into the event. The RV-2 valve was open approximately 240 seconds and released approximately 15,000 gallons of coolant.

The high pressure injection system responded by increasing makeup flow rate from an initial 650 gpm to 5000 gpm within 12 seconds and provided a maximum flow of approximately 6000 gpm at 30 seconds. Injection flow rate remained high until the pressurizer refilled.

The reactor coolant flow rate was reduced following the scram, by the programmed reduction of pump speeds, to about 30 percent of the normal operating flow rate.

Bulk coolant outlet temperature dropped rapidly after the scram from 500°F to 440°F and then slowed. It continued to decrease until it was within 20°F of the decreasing bulk inlet temperature at about 10 minutes following the valve opening.

Figure 15.6-1 provides examples of the data obtained from charts and evaluated in 1965. The following conclusions were reached:

- o The pressurizer drained completely, permitting the system to depressurize to the saturation pressure of the hot-leg coolant.
- o The pressure decrease was sufficient to trip some of the ECCS low pressure switches; however, no other ECCS trips occurred (requires trips from a second set of monitors on other RCS parameters, such as extremely low flow).
- o Pressure recovery and refill of the pressurizer was orderly.
- o Cooling of reactor fuel was adequate during the event. (Subsequent operation confirmed that fuel cladding integrity had not been adversely affected.)

15.6.1.3 Core and System Performance

The RETRAN thermal-hydraulic code was used to evaluate the capacity of the existing RV-2 valve. The model included piping from the reactor outlet header, through the V-43 valve, RV-2 valve, and V-23 valve out to the emergency dump tank. The RV-2 model has been compared with an actual test of the RV-2 valve with the following results:

	RETRAN MODEL	ACTUAL RV-2
PRESSURE	1650 psig	1650 psig
TEMPERATURE	300°F	300°F
FLOWRATE	12,000 gpm	10,922-13,653 gpm

These results verify the adequacy of the RETRAN Dump Line Model used in the analyses presented in Section 15.6.1.2.

Additional evaluation of the January 25, 1965 event was performed in 1967. The following conclusions were reached: 21

- For a maximum power tube with the reactor total power at 4800 MWt, the opening of an RV-2 would not cause boiling at the rear riser end of the pressure tube outlet connectors. Connector boiling suppression margins calculated for key points in the depressurization transient were:

Time After RV-2 Opening	Saturation Temperature	Actual Temperature	Margin
150 sec	486°F	439°F	47°F
240 sec	455°F	434°F	21°F

- Power would decrease more rapidly than pressure tube flow rate, and pressure would decrease. Therefore, even if boiling were to occur at the pressure tube outlet connector, the power decrease would result in promptly improved cooling conditions in the pressure tube.
- Flashing of the reactor outlet water prior to reaching the steam generators (lowest pressure point in hot-leg part of the recirculating coolant) maintains boiling suppression at the reactor outlet connectors.

On the basis of the experience and evaluations discussed above, it is concluded that inadvertent opening of an RV-2 safety valve or other RCS pressure relief valve would not result in damage to the fuel or result in actuation of the ECCS System even with concurrent partial loss of AC power. 21

15.6.1.4 Barrier Performance

Since the event does not result in damage to the fuel, the quantity of fission products in the reactor coolant is insignificant and no barriers, other than the fuel cladding, are needed to restrict the transport of radioactive material from the fuel to the public. See Section 6.5.

15.6.1.5 Radiological Consequences

The release of RCS inventory associated with an inadvertent opening of a RCS safety valve does not have significant radiological consequences. See

Section 6.5 on events involving partial release of the RCS inventory and the radiological consequences.

15.6.2 Break of a RCS Boundary that Penetrates Confinement

The sensing lines and reactor coolant piping that interface with the primary coolant pressure boundary and penetrate the confiner boundary are tabulated in Section 6.2.4. Of these penetrations, the 12-inch diameter high pressure injection (HPI) supply pipe is the largest in this category.

Failure of a sensing line would release reactor coolant outside of the confiner. The radiological release from the break of a sensing line would be significantly less than that discussed in Section 6.5.3 which discusses the consequence of a break in the largest RCS pipe. The 12" HPI pipe has a check valve located inside the confiner boundary that would prevent the loss of primary coolant outside of Zone I.

There are, however, further consequences of a break in the HPI piping. As this constitutes the only source of primary system make-up water, the pressure in the primary coolant loop would immediately begin to decay. The reactor would scram accompanied by a inventory shrinkage which in the absence of a make-up source would drain the pressurizer. At 645 psi, an ECCS trip signal would be made up. However, two are required for an automatic dump. None of the other conditions, i.e., low flow, high outlet temp., low pump speed, or high confiner pressure, would occur. The proper operator action would be to determine the course of the decay in primary loop pressure. This could be done by checking the HPI flow which would be very high combined with the maximum number of HPI pumps operating at maximum speed. Once it has been determined that the loss of PCL inventory is due to HPI system failure, the operator should open the V-3 valves and lock out the V-4 valves in the closed position. This will allow the use of the ECCS as a make-up source at 300 psig.

15.6.3 Failure of a Steam Generator Tube

The loss of reactor coolant from a failed steam generator tube is within the capacity of the high pressure injection (HPIS) system. Experience shows that there is no loss in RCS pressure following a steam generator tube failure. The increased injection flow rate would result in a manual shutdown of the reactor. A recent series of analyses of small RCS breaks concluded that there was no break size or location that would lead to fuel heat-up prior to ECCS actuation (Ref. 30). This was based on the availability of High Pressure Injection and covered break sizes to 4 in. nominal pipe diameter. The steam generator tubes are 0.625 in. outside diameter, so the recent analyses are clearly bounding. The analyses utilized state-of-the-art analytical tools recommended by the National Academy of Sciences and used best-estimate assumptions to accurately reflect plant performance. See Section 15.6.5 "Major Reactor Coolant System Pipe Breaks" for a more detailed discussion of the modeling codes and analysis techniques. There would be a detectable increase in radioactivity in the steam system but of no radiological consequences. 21

15.6.4 Single Pressure Tube, Connector and Fuel Ejection Events and Minor Reactor Coolant Pipe Breaks

The ECCS would provide adequate cooling to prevent fuel failure in the event of any postulated major LOCA or loss of both offsite and onsite ac power. However, postulated events that result in a reduction or loss of coolant to a single pressure tube may cause failure of a portion of the fuel in the affected pressure tube even though the ECCS were to perform its design function.

This section describes the consequences of a spectrum of single tube events, some of which may result in the release of fission and/or activation products.

The frequency of single tube events is classified as "Infrequent," not to be expected to occur during the life of the plant. None has occurred since initial operation.

The calculated doses to the public for those events with a potential for the release of fission and/or activation products are, in every case, a very smal fraction of the guidelines provided in 10 CFR 100. See Appendix 15.6B for discussion of calculational techniques and assumptions.

15.6.4.1 Partial Failure of a Single Pressure Tube or Inlet Connector

15.6.4.1.1 Partial Failure of a Pressure Tube

Cooling aspects of a complete circumferential failure of a reactor pressure tube are analyzed in Section 15.6.4.4. It has been determined that the resulting coolant flow toward the break would be adequate to prevent fuel heatup; however, a less than complete failure could produce a reduction in cooling and a subsequent fuel heatup.

Specifically, it is postulated that, at a given location along a fuel column, a leak in a pressure tube could develop to a size such that the downstream pressure would drop to the pressure at the outlet of the fuel column. In such a case, flow in the outer annulus of the downstream portion of the fuel column would stop and coolant in that portion would boil out rapidly. An analysis was made of the possible ramifications of such postulated leaks in terms of both cooling adequacy and capabilities for detection.

Several locations and leak sizes were chosen for study; the results are shown in Figure 15.6-2. The figure shows the supply and demand curves for the fuel column, outlet hardware and outlet connector, together with characteristic curves that would be representative of different locations and sizes of partial breaks of a pressure tube. The latter are termed "leak" curves because, as discussed later, they are not demand curves in the classical sense. For development of all of the curves, a power generation of 5000 kW was assumed (the design hot channel).

The broken line in Figure 15.6-2 represents the conditions that would exist immediately following occurrence of the leak while flow from the outer annul would still be carried into the outlet connector. All other leak curves (shown as solid lines) represent conditions that would exist after boilout

of the coolant in the annulus downstream from the leak. For such conditions, the flow rate in the outlet connector would consist only of that supplied by the inner annulus and center channel. In all cases, the abscissa of the graph represents the normalized total rate of inflow of coolant to the pressure tube (for the leak curves, the sum of flow rate from the leak and flow rate in the two inner subchannels). The methods used for development of the leak curves are described in the following paragraphs (See Section 4.4 for detailed discussions of the thermal-hydraulic methods used for treatment of an intact tube).

For purposes of analysis, the coolant passages of the fuel column were treated as discrete, continuous subchannels; these are the section of the outer annulus upstream from the break and the total lengths of the other two subchannels. This assumption enabled establishment of the hydraulic boundary conditions for the annulus; the pressure drop would be the same for all three subchannels, and the total demand pressure differential for a given break assumption (a point on one of the leak curves) would be the sum of that pressure drop and the pressure drop between the outlet of the fuel column and the outlet (hot leg) riser. The pressure drops and related flow distributions were calculated according to the following assumptions and principles:

- Inlet temperature, fuel column heat generation rate and outlet riser pressure would be constant.
- Chopped cosine heat generation for the fuel column.
- All heat generated in the outer fuel cylinder downstream from the leak would be transferred to the coolant in the inner annulus.
- The flow rate of coolant in the outer annulus would be equal to the flow rate out of the break. The assumption of a particular leak flow rate at a given location established the pressure drop for the other two subchannels. This pressure drop and the hydraulic characteristics of those subchannels enabled calculation of the subchannel flow distribution and total inflow to the pressure tube. Outflow from the inner (intact) subchannels was used to determine the pressure drop in the outlet connector.

- In subchannels for which net outlet quality was calculated to exist, the flow and pressure drop relationships across the boiling length (and/or outlet section) were determined by the method developed by Martinelli and Nelson (Ref. 1) for calculation of two-phase pressure drop. Single-phase pressure drop was calculated by conventional methods.

With reference to Figure 15.6-2, the linear part of a leak curve is the locus of assumed leak rates for which single-phase flow conditions were determined. The curved part is the locus of leak rate assumptions that resulted in calculated two-phase flow conditions in one or both inner subchannels and, at the extreme, the outlet section. It follows from the foregoing discussion that, as the assumed leak rate is increased, there will be a corresponding increase in total inflow to the tube; the converse is also the case. Additionally, an increase in the distance of the assumed break from the outlet of the fuel column produces a steeper pressure gradient in the outer annulus; this would correspond to a greater leak rate and, hence, an increase in tube inflow.

The intersection of the supply curve with any of the leak curves defines a specific leak rate that would satisfy the boundary conditions at the given location: e.g., loss-of-coolant flow in the outer annulus over the given distance to the fuel column outlet. Considering only the leak curves shown by solid lines, the assumed break locations have a definite bearing on the equilibrium flow rates established by intersection with supply curve. If the location of the leak is moved upstream from the outlet of the active zone, the leak curve is shifted upwards and to the right. As long as the supply curve intersects with the straight portion of the leak curve, the tube inflow increases with distance of the break from the outlet of the active zone. However, for a leak at a point between 10 and 12 feet from this location, the supply curve intersects the two-phase portion of the leak plot, and the resulting steady-state flow rate is significantly decreased. As an example, the steady-state flow rate for a critical leak 15 feet from the active zone outlet is approximately 90 percent of the initial value; while for a leak at 10 feet, the flow rate would be 114 percent of normal. At a distance of 12 feet, the total inflow would be unchanged.

Fuel temperature calculations were made for two cases of critical leaks as follows:

1. The first case was a leak 5 feet from the outlet of the fuel column. The flow rate increase would be short of that required to produce a flow monitor trip. Assuming no scram, the maximum fuel temperature that would be reached in the affected fuel column, if it were a hot channel tube, would be 1300°F. This temperature is above the alpha-beta transformation point, but below the fuel failure temperature of 1900°F. The temperature was calculated assuming steady state heat transfer from only the inner surface of the outer fuel cylinder.

2. The second case was a leak at a point just upstream from the point of maximum specific power. The resulting flow rate change would produce a reactor scram in this case, and the maximum fuel temperature in the affected column would increase to approximately 1420°F the beta-gamma transformation temperature. In this case, the maximum temperature was calculated by use of the conventional finite differences technique, assuming radiative and conductive heat transfer across the steam-blanketed surfaces.

While it appears from steady-state calculations that no net flow change would occur for the leak at 12 feet, there are indications that discharge fluctuations resulting from instability or changes in heat distribution would cause a reactor scram. In Figure 15.6-2, for the leak at 15 feet, the initial flow rate change (as determined from the broken line) would be an increase of about 17 percent; this would be followed shortly by a decrease of 27 percent to a final (equilibrium) inflow of about 90 percent of normal. In this case, the fluctuation in inflow would be caused by the redistribution of heat generated in the outer fuel cylinder.

Approximately equivalent conditions may exist for the 12 foot case, due to the effects discussed above. Moreover, the character of the leak curves is such that relatively long portions of the two-phase regions are nearly parallel to the supply curve. This condition, together with an apparently high degree

of sensitivity of the leak curves to both location and size of failure, indicates a significant potential for flow instability. Such fluctuations may be large enough to cause a flow monitor trip during or after leak development.

For any of the postulated critical-sized leak locations, the rate of discharge of coolant into the reactor graphite moderator would cause a rapid increase in gas moisture content. Since control room instrumentation would indicate the increase in moisture, a leak would be detected and the reactor would be scrammed manually.

For the special case of critical-size tube leaks from the center of the fueled region to approximately 8 feet from the end of the active zone, the measured change in the pressure tube flow rate may not be sufficient to initiate an automatic reactor scram by the pressure tube flow monitoring system (Section 7.2). The fuel and clad temperatures would increase to the point at which fuel failures would be expected because of accelerated clad corrosion or fuel melting.

The release of fission products would be significantly less than for the 100 percent pressure tube plug discussed in Section 15.6.4.2. See Section 15.6.4.4 for a discussion of the release of activation products associated with release of coolant inside the moderator.

15.6.4.1.2 Partial and Complete Failure of an Inlet Connector

Complete parting and partial failure of one of the couplings in an inlet connector have been studied to determine effects in terms of both RCS and individual reactor pressure tube performance. The affected reactor pressure tube was assumed to be operating normally prior to the failure. It was assumed the failure occurred upstream from the flow monitoring venturi; critical coolant flow in the constricted throat of the venturi would limit the reverse flow which constitutes the only source of cooling.

Results show that adequate reverse flow would occur with only a small, temporary rise in fuel temperatures during the 10 seconds required to

establish equilibrium. The coolant backflow is supplied under rear header pressure. The effects of temperature rise of the coolant in flowing backwards through the reactor pressure tube would be offset by a reactor scram initiated almost immediately after the coupling failure. Coolant loss in the RCS from both reactor pressure tube and associated inlet riser was found to cause the pressure in the RCS to decay to a minimum of about 900 psia. The capacity of the injection makeup system was found to be adequate to prevent draining the pressurizer. See Section 9.3.4.

Since a single coupling bolt failure would cause complete parting of the coupling, it is unlikely that a partial failure would take place. The high flow trip setting, protection against outlet connector boiling and subchannel DNB (high flow temperature limit), protects against the partial leak. Analysis of the less-than-complete failure indicates that a critical cooling situation could develop in the affected reactor pressure tube. Specifically, a critical size of leak was identified that would produce a leak rate equal to normal pressure tube flow rate; flow in the affected tube would diminish to zero. With such a leak in one of the couplings between the venturi and the inlet nozzle, the high flow trip would cause a reactor scram prior to fuel failure. Post scram actions would cause an immediate drop in the reactor coolant pressure in the rear header, resulting in resumption of partial flow past the break. Shutdown cooling would be sufficient to cool the fuel in the pressure tube fed by the leaking connector.

15.6.4.2 Complete Flow Blockage to Pressure Tube

The flow reduction, cooling and fuel temperature transients for partial pressure tube failures, with and without a reactor scram, are described in Section 15.6.4.1. Partial pressure tube plugging accidents are described in Section 15.3.4.

This section describes the effects of complete upstream flow blockage of a single pressure tube, which results in a reactor scram from an individual pressure tube low flow trip. The discussion utilizes conservative licensing basis assumptions to predict RCS and confinement response and radiological consequences. 21

Flow Blockage Analysis to Predict RCS and Confinement Response

A 0.5 second flow monitor instrumentation delay and a 0.2 second control rod circuit delay were assumed. Control rods were 75 percent inserted within 2.2 seconds after the assumed upstream tube plug occurred.

The initial thermal assumptions are shown in Table 15.6-1.

The fuel temperature transient was calculated using the MELT code with the assumptions described in Section 15.6.5.1 (Ref. 1a). The fuel temperature rise transient for the center element of a Mark IA fuel column is shown in Figure 15.6-3. The fuel temperature transient analysis was performed assuming a rapidly forming 100% upstream flow blockage. For the first 50 seconds of the transient following flow blockage, fuel temperatures rapidly increase to where both the inner and outer fuel assemblies reach the beta-gamma phase change at 1420 to 1430°F. The fuel temperature transient continues with only thermal radiation and conduction heat transfer modes until the inner fuel elements begin to melt at 245 seconds. The rate of temperature increase of the inner fuel temperature is reduced at this point because of the heat of fusion as the inner fuel assembly melts. At 250 seconds into the transient, the volume average inner and outer fuel element temperatures at the midpoint of the pressure tube reach the melting temperature of 1995 F. The fuel element is then assumed to slump, contacting the pressure tube and increasing the pressure tube temperature above 1300°F. The outlet riser pressure of approximately 1300 psi is assumed to cause a stress-rupture failure of the pressure tube at the point of the hot spot.

The fuel failure fraction, assuming a clad failure temperature of 1900°F, was determined using the MELT code. The fraction of fuel elements that fail is limited by heat transfer to the graphite cooling system; however, in this analysis, no credit was taken for decay power which would be transferred to the adjacent cooled pressure tubes, which adds conservatism to the analysis.

The fuel assemblies with clad failure would be limited to elements with initial linear heating rates of greater than 122 kW/ft. Between 122 kW/ft and 128 kW/ft, only the inner elements would fail, while above 128 kW/ft,

both the inner and outer elements would fail. For a hot channel case (5000 kW) with a 1.55 axial peaking factor, the fraction of the fuel column with clad failure would, therefore, be limited to the center 60 percent of the outer fuel and 63 percent of the inner fuel elements with a volume-weighted clad failure fraction of 61 percent. The fuel volume could release up to 80 percent of the total fission product inventory of the fuel column.

The stress-rupture failure of the pressure tube was modeled as a single ended tube break using the RELAP4 computer code (Ref. 2). Double-ended pressure tube rupture event is addressed in Section 15.6.4.4, and is a postulated fatigue flaw failure of the pressure tube with no fuel failure assumed. The enthalpy of the coolant and break flow rate into the graphite moderator were calculated for the conditions that exist at the time of failure, i.e., a system pressure of 1300 psi at 500 seconds after shutdown. The discharge flow rate and enthalpy were determined to be 75 lb/sec and 380 Btu/lb, respectively.

A portion of the coolant discharged from the break would be converted to superheated steam by the hot graphite. The steam would sweep the fission products released from the failed fuel into the graphite moderator.

Negligible amounts of CO and H_2 would be formed by the water gas and metal water reactions. See Section 15.6.4.4. Activation products in the moderator would add to the radioactive burden of the steam. See Section 15.6.4.4.

When the steam pressure within the moderator reaches 3 psi, the rupture discs would fail and release the contaminated steam into Zone I of 105 N.

The confinement sequence is shown in Table 15.6-2. The pressure transients in Zone I of 105-N and 109-N are plotted in Figure 15.6-4. The peak confiner atmosphere pressure of 0.61 psi is not sufficient to blow the weather caps on the steam vents. The confiner would remain sealed until 7000 seconds following the initial flow blockage. During this period the fog sprays would reduce the confiner atmosphere pressure to less than 3 inches w.g., permitting the establishment of a filtered release. The flow rate through

the filters and out the stack would be 1620 cfm. The Zone I building leakage would be less than 81 cfm.

As described in Section 6.5.3, greater than 99 percent of elemental iodine and 98 percent of the particulates would be removed from the confiner atmosphere by plateout on the internal confiner surfaces or by washout by the fog sprays.

The particulate and charcoal filters remove all but a few tenths of 1 percent of the elemental iodine and particulates that reach them. The organic iodine and the noble gases are held up in the confiner for significant time periods, as shown in Table 15.6-3.

The number of curies released from the stack and through building leakage are tabulated in Table 15.6-4.

The impact of these releases is shown in Table 15.6-5. Calculational techniques and assumptions used to calculate doses are described in Appendix 15.6B.

The dose rates are very small fractions of the guidelines published in 10 CFR 100.

Using Regulatory Guide 1.4 assumptions, the radioactive cloud would not reach the site boundary during the first 2 hours following the accident. The realistic calculation is based upon the wind speed that maximizes the dose at the site boundary.

The dose to operators in the control room is shown in Table 15.6-6. Operators are supplied with fresh air masks as well as a dedicated emergency ventilation system to control the thyroid dose. See Section 6.4. Rotation of operators would reduce the dose to any one individual to much lower values. See Section 6.5.3 for further discussion of accident doses.

15.6.4.3 Fuel Ejection Accident

If a reactor pressure tube nozzle or the end closure on a pressure tube should fail, the fuel and spacers would be expelled from the pressure tube. Upon failure, the reactor would be scrammed by any of the following trips:

- o High pressure tube flow rate,
- o RCS low pressure,
- o Low pressurizer water level.

The expelled fuel elements and spacers would impact against the thermo-barrier and would probably fall into a floodable pit at the front of the reactor or into the water filled fuel discharge pit, depending on the location of the postulated failure. Some fuel elements may become lodged in the thermo-barrier or possibly penetrate the barrier landing on the front-charging elevator. It is also possible that a fuel element could become lodged in the reactor piping.

The fuel temperature transient is dependent upon the final location of the expelled fuel assemblies.

- o Fuel which falls into the water-filled fuel discharge pit would be adequately cooled.
- o A fuel element which became lodged in the thermo-barrier would be adequately cooled from the steam-water mixture from the RCS blowdown through the affected nozzle.
- o Fuel which penetrated the thermo-barrier would be cooled by the fog spray stations which would be activated on confinement pressure.
- o The analysis of fuel elements which fall into the floodable pit at the bottom of the thermo-barrier and for a fuel element which is assumed to become lodged in the reactor piping is described below.

The analysis of a fuel element lodged in the reactor piping was performed for the maximum powered fuel element with an initial tube power of 5000 kW. After the fuel is ejected, the fission power drops to zero and the only heat source is decay power.

The first analysis was done with the following assumptions:

- o fuel element lodged in piping on face of reactor;
- o atmosphere is steam at 15 psia;
- o no water spray is impinging on the fuel element;
- o reactor face and protruding tubes are at 400°F;
- o remainder of walls are at 212°F;
- o emissivity of the fuel element is 0.80.

The natural convection caused by the hot reactor face in a closed space was calculated using the equation:

$$h = \frac{Ck_f}{X(L/X)^{1/9}} \left[\frac{L^3 \rho_f \, g\beta_f \, \Delta T \, N_{pr}}{\mu f^2} \right]^{1/3} \qquad \text{(Ref. 3)}$$

where:

C = 0.071
k_f = conductivity of steam at 306°F (Btu/hr-ft-°F)
L = length of reactor face (feet)
X = space width (feet)
ρ_f = density of steam (lb/ft^3)
ΔT = temperature difference between face and space (°F)
μf = viscosity of steam ($\frac{lb}{sec\text{-}F}$)
g = gravitational force (ft/sec^2)
β_f = buoyancy force (1/F)
(f corresponds to properties measured at average film temperature)
N_{pr} = Prandtl number

The convection was calculated to be 1.25 Btu/hr-ft^2-°F. The total decay heat at several time points was calculated, and the surface temperature of the outer fuel was calculated using the equation for combined convection & thermal radiation:

$$Q_{tot} = hA\ (T_s - T_a) + \sigma\ \epsilon_e A\ (T_s^4 - T_a^4)$$

where:

A = surface area 1.32 ft
T_s = surface temp. (°R)
T_a = ambient temp. (°R)
σ = Stephan Boltzman constant 0.1714 x 10$^-$ Btu/hr-ft^2-°R
ϵ_e = effective emissivity

Neglecting the temperature rise through clad, the outer fuel temperature was then calculated assuming a 20 mil gap between the uranium and zircaloy cladding.

$$Q_{tot} = \frac{A_g K_g}{\Delta X}\ (T_f - T_c)\ + \sigma \epsilon_e A_s\ (T_f^4 - T_c^4)$$

where:

A_g = heat transfer area of the gap (ft^2)
A_s = clad surface area (ft^2)
K_g = conductivity of the fission gas in the gap Btu/hr-ft-°R
X = gap width (ft)
T_f = fuel temp. (°R)
T_c = clad temp. (°R)

$$\epsilon_e = \frac{1}{\frac{1}{\epsilon u} + \frac{1}{\epsilon z} - 1}$$

where:

ϵ_e = effective emissivity
ϵ_u = uranium emissivity (0.54 for clean metal)
ϵ_z = zircaloy emissivity (0.6 for clean metal)

The results of this calculation show that the outer fuel steady-state temperature would exceed 1994°F for heating rates corresponding to a decay time of approximately 900 seconds. It is estimated that the fuel temperature transient to 1994°F will be approximately 600 seconds. It is therefore concluded that any near mid core fuel thus ejected which becomes lodged in the piping on the face of the reactor will melt and release fission products. Because of the spacing and configuration of the tubes, it is considered to be unlikely that any fuel elements would lodge in this position; however, the analysis assumes that a high-powered fuel element does become lodged in the reactor piping.

The second consideration was the fuel element which would be lying on the bottom thermo-barrier. If the end of the pressure tube fails, approximately 240 lb/sec of water and steam would be discharged into the thermo-barrier space. The quality would be 28.6% with a resulting water flow of 171 lb/sec or 2.8 ft^3/sec. Assuming that the evaporation of the water due to the hot reactor face equals the steam condensation on the thermal barriers, the head required to force this amount of water through the orificed 6" X 6" flood gates at either end of the barrier enclosure was calculated. The head was calculated to be 15 inches, using the correlation of a sharp edged pipe entrance.

$$H = 1 + K_L \frac{V^2}{2g} \quad \text{where } K_L = 0.5 \qquad \text{(Ref. 4)}$$

Assuming that 50% of the water exiting the broken tube falls on the panel with the remainder spraying beyond to the concrete floor, an analysis was made to determine the additional depth of water on the panel, using the equation for a frictionless broad faced weir:

$$q = C\ 2/3\ \sqrt{2g}\ H^{3/2}$$ (Ref. 5)

where:

q = quantity (ft^3/sec)
C = 0.577
g = gravitational force 32.2 ft/sec^2
H = feet of water

The head calculated was 0.5 inches. Since the panel height is 12 inches above the floodgates, a water depth of 3.5 inches would be sufficient to completely submerge the fuel elements. Using the conservative assumption of equal depth over the full area of the panel, the fuel element surface in contact with the water was found to be in nucleate boiling. The surface temperature of the outer fuel element was determined as a function of time. The inner fuel element temperature was then calculated as for the previous case. It was determined that the decay heat is not sufficient to maintain a steady-state fuel temperature of 1994°F after 5 seconds. Comparing this to the time required to reach 1994°F, it is concluded that no fuel failure would occur for this mode.

From the previous analysis and discussion, it is concluded that the ejected fuel elements would remain adequately cooled for all cases except for a fuel element that is postulated to become lodged in the reactor piping. It is possible that this element would remain cooled via the break flow; however, it was conservatively assumed that the highest powered fuel assembly would melt, releasing its fission products to the confinement system.

CONFINEMENT SEQUENCE

The confinement system transient was analyzed using the CONTEMPT computer code described in Section 6.2.1.1. The coolant flowing from the broken nozzle would flash to steam that would pressurize the confiner atmosphere. The reactor would be shut down on a high-flow trip. The high pressure injection system would maintain the RCS inventory, and ECCS would not be actuated. The sequence of events following the nozzle break is outlined in Table 15.6-9. The confiner atmosphere pressure transients are plotted as a function of time in Figure 15.6-8. The peak confiner atmosphere pressure of 0.54 psi is not sufficient to open the steam vents. The confiner would remain sealed until 1100 seconds after the accident.

During this period the fog sprays would reduce the confiner atmosphere pressure to less than 3 inches w.g., permitting the establishment of a filtered release. The flow rate through the filters and out the stack would be 1740 cfm. The Zone I building leak rate would be less than 87 cfm.

As described in Section 6.5.3, greater than 98 percent of the elemental iodine and 99 percent of the particulates would be removed from the confiner atmosphere by plateout or washout by the fog sprays. The particulate and charcoal filters remove all but a few tenths of 1 percent of the elemental iodine or particulates that reach them. The organic iodine and noble gases are released more rapidly during this accident than in the case of the complete flow blockage of a single pressure tube. See Table 15.6-3.

The number of curies of fission products released from the stack and through building leakage are tabulated in Table 15.6-10.

The impact of these releases is shown in Table 15.6-5. Techniques and assumptions used to calculate the doses are described in Appendix 15.6B.

Using Regulatory Guide 1.4 assumptions, the radioactive cloud would not reach the site boundary during the first 2 hours following the accident. The realistic calculation is based upon the wind speed that maximizes the 2-our dose at the site boundary.

The dose to operators in the control room is shown in Table 15.6-6. Operators are supplied with fresh air masks as well as a dedicated emergency ventilation system to control thyroid dose (see Section 6.4). Rotation of operators would reduce the dose to any one individual to much lower values. See Section 6.5.3 for further discussion of accident doses. 21

15.6.4.4 Double-Ended Pressure Tube Break

The N Reactor pressure tube design was established to be highly conservative to provide a large nuclear safety margin between operating and failure parameters and to provide protection against potential changes in pressure tube properties as a result of irradiation effects. The results of destructive examination of pressure tubes removed after varying amounts of reactor service have verified the adequacy of the tube design and tube properties for extended service. Section 5.3 discusses the tube design, fabrication controls and results of postirradiation pressure tube destructive examinations. Based on the pressure tube design and performance, the complete double-ended failure of a reactor pressure tube is a limiting fault not expected to occur during the operational life of the reactor.

The most likely failure mechanism for a pressure tube involves the fatigue propagation of a flaw through the wall to cause a small detectable tube leak. The pressure tubes received rigorous nondestructive examination prior to installation and an ongoing maintenance program monitors tube condition. It has been demonstrated that the pressure tubes possess leak-before-failure protection which affords a valid reactor shutdown indication to preclude further growth of a crack toward a critical crack geometry. In addition, operational stress levels including such factors as hydraulic loads, bending stresses and seismic loads, are restricted to values at which fatigue crack growth rates are extremely low, making the failure of a pressure tube by this mechanism during the reactor lifetime highly improbable. See Section 5.3 for additional information. 17

The complete double-ended failure would result in the discharge of high temperature, pressurized reactor coolant directly into the reactor graphite moderator. The reactor would be scrammed by a high flow monitor trip on the

affected pressure tube. HCRs would begin to enter the core at .7 seconds, which includes both the flow monitor delay (.5 seconds) and the safety system delay (.2 seconds). The HCRs would be 75 percent inserted at 1.95 seconds.

A portion of the coolant discharged from the break will immediately flash into steam. Heat transfer to the coolant from the moderator will vaporize the remaining coolant and superheat the resulting steam. The superheated steam will contain an insignificant amount of radioactivity from the reactor coolant but will not contain any additional fuel fission products since the fuel will be adequately cooled by the break flow. The superheated steam will become contaminated by activation products which result from the samarium oxide dust of previous ball system drops. Drops of the currently used boron carbide balls generate dust at a greater rate, but result in estimated lower activation levels by a factor of 1000.

The pressure within the moderator will reach 3 psi in less than 1 second, at which time the gas system rupture disks will breach, releasing the contaminated steam into the rear 105-N pipe space of Zone I of the confinement system.

Stress-rupture failure of a pressure tube due to flow blockage is addressed in Section 15.6.2. The flow blockage event does assume fuel damage and demonstrates adequate confinement response in mitigating radiological consequences.

15.6.4.4.1 Core And System Performance

The break discharge flow rates were determined using the RELAP4 computer code. The RELAP4 model shown in Figure 15.6-6 consists of an inlet riser to outlet riser representation with detailed nodalization in the connectors, fuel spacers and fueled region.

A break in the downstream spacer zones was not modeled since the break discharge flows would be less than for the upstream spacer case because the high 2-phase pressure drop in the fuel column would result in choked upstrea flow at the break.

The resulting mass and energy releases are presented in Table 15.6-12. The mass releases range from a maximum of 236 lb/sec for the upstream spacer break location to 157 lb/sec at the core center. The analyses of the pressure in the graphite moderator and the reactor shields, which are reported below, are based on these release values which were determined using a discharge coefficient of 1.0. 17

The moderator design provisions are discussed in detail prior to discussing the operation of the confiner to control the offsite release of the contaminated steam. The moderator design provisions include:

1. venting the steam from the reactor moderator without causing failure of other pressure tubes or the moderator atmosphere pressure boundary,
2. controlling the formation of flammable gases generated by the water-gas and metal-water reactions, and
3. controlling the contamination of the steam with activation products in the moderator.

15.6.4.4.2 Steam Venting From The Graphite Moderator

Provisions were included in the reactor moderator design to promote the venting of steam from the reactor core. These provisions are discussed in more detail in Section 4.5.3. Briefly, the provisions include:

1. an open graphite lattice structure to permit venting of steam from the region of the pressure tube failure;
2. plenums and large vent passages to conduct steam to the normal gas supply and exhaust line and special steam vent line;
3. rupture discs in the reactor gas line and vent line to permit exhausting the steam to a collection box; and
4. additional piping and valves to permit routing the steam either to the normal Zone I exhaust air tunnel, the exhaust air filter system and the ventilation stack, or to the 105-N pipe space within Zone I.

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A sketch of the system is included as Figure 15.6-9. Studies, both analytical and experimental, (Ref. 6, 7, 23 and 25) have been made to determine whether

these venting provisions are sufficient to prevent local and general pressure buildups which could lead to additional reactor damage. Special attention has been focused on the possibility of one pressure tube failure causing a lifting of the graphite stack sufficient to cause failure of additional tubes. It is concluded that the venting capacity is sufficient to prevent excessive general pressure buildup; and that the failure of one pressure tube would not propagate and lead to additional tube failures or impede operability of the safety systems.

15.6.4.4.3 General Pressurization Analyses

Calculations were made to determine the steam venting capacity of the system at the design pressure of 5 psig. Additional calculations were also made to determine venting capacity at 20 psig, the point where welds in the thermal shield may start to deform.

The enthalpy of the coolant discharged from the break would result in only a fraction of the coolant flashing to steam. The graphite moderator, however, represents a heat source of about 10^9 Btu and calculations were based on 100 percent vaporization of the flow. Additional calculations were made with an assumed steam temperature of 600°F which represents the upper limit steam temperature which could result from heat transfer from the moderator to the steam.

The moderator pressure calculations were made by summing up the resistance to flow of steam by the various parts of the system from the reactor moderator to the exhaust air ventilation stack and to the rear pipe space of Zone I, 105-N. Vent capacity was then determined using a compressible flow model. The results of the calculations are presented in Table 15.6-11b for vent flows to both the rear pipe space and the exhaust air ventilation stack and to the rear pipe space alone which is the current operating configuration.

Steam release rates from double-ended pressure tube failures are shown in Table 15.6-12.

Should the moderator pressure increase to beyond the design pressure but not exceed 20 psig, the expected effect would be steam blow-by past the gas seals into Zones I and II of the confinement system. At 20 psig, deformation of thermal shield welds might be expected and, at still higher pressures, more severe damage would be expected.

A comparison between Tables 15.6-11b and 15.6-12 show that the complete double-ended break of a single pressure tube would exceed the moderator design pressure of 5 psig but would not exceed the pressure at which deformation of the thermal shield welds could occur. The maximum moderator pressure of 15 psig was calculated to occur for the upstream spacer zone break with 600°F steam being discharged only to the rear pipe space of 105-N. For saturated steam, the maximum moderator pressure would be approximately 10 psig. The small amount of gas seal leakage which could occur by exceeding the design pressure will not increase the offsite consequences in the radiological analysis since the analysis conservatively assumes instantaneous release of the activation products into the confinement with 17 percent of those products being released through the steam vents.

15.6.4.4.4 Analysis of Local Effects

Scale models and analysis (Ref. 6 and 23) performed in the 1960's provided confidence that there was adequate protection against a single tube break resulting in a chain reaction of tube ruptures. However, since the initial work was done structural degradation of the reactor core has occurred. In 1985, dynamic analyses (Ref. 25) were performed to assess the effects of this degradation. 17

The response of the reactor core was evaluated to assess its ability to sustain a single pressure tube rupture, to maintain structural integrity, and to assess the operability of the safety systems. The dynamic response and integrity of the adjacent pressure tubes was the primary concern. However, effects on the graphite coolant tubes, HCR channels, vertical ball channels, and HCR T-blocks were also evaluated.

Acceptance criteria for the structural integrity of the reactor core and the operability of the safety systems which would permit the reactor to reach a safe shutdown condition were defined in Reference 24. The principal concern is that energy released from one tube rupture could induce a failure in an adjacent tube, in which case a cascading failure of all the pressure tubes could theoretically occur. This concern was addressed by assessing the structural response of adjacent tubes from thermal-hydraulic loads. Failure is defined when a second tube reaches its failure criteria of one percent strain. The integrity of the graphite blocks in the reflector and the moderator is not of direct concern. However, failure of the blocks must not impede operability of the safety systems.

The pressure tube rupture evaluation was performed in the following sequence:

- o blowdown data obtained (Ref. 15)
- o thermal-hydraulic load generation (Ref. 25)
- o dynamic structural analyses (Ref. 25)
- o evaluation of the reactor core safety systems (Ref. 25).

Three-dimensional thermal-hydraulic models (using RELAP5/Model) were used to generate the thermal-hydraulic loads. The thermal-hydraulic analyses used moderator block temperatures of 1250°F, a loss coefficient of 0.1 to account for pressure drop through the vent passages; applicable modes of the forced convention heat transfer in the two-phase and superheated regions of the vent passages, and a 40 percent blockage of vent paths due to graphite debris. Thermal-hydraulic analyses were performed for two postulated rupture locations. One location was the center of the core at mid-height of the fueled zone, and the other was at the edge of the fueled zone at mid-height. These loads were then applied to worst case structural models.

Dynamic time history analyses were performed to obtain the nonlinear respons of the pressure tubes and safety systems. Since the rupture causes a localized load, local analytical models of the reactor core were used to evaluate the dynamic response. Several local structural models were requir for these analyses. Since the pressure tubes and ball channels responded t

loads in the east-west direction and the graphite coolant tubes, HCR channels, and T-blocks responded to loads in the north-south direction, separate models were developed for loads in these directions. An example of a stress time history is shown in Figure 15.6-175 for a pressure tube at the location of maximum stress.

Displacements and tube strains obtained from the nonlinear analysis were used to evaluate the pressure tubes and safety systems. The results of the evaluations showed that all of the core components and safety systems met the acceptable criteria. Table 15.6-32 shows the comparison of allowable and predicted parameters.

The acceptance criteria used in the evaluation was based on incremental strain in excess of one percent for adjacent pressure tubes. Similarly, a technically conservative approach was adopted for the other components presented in Table 15.6-32. For instance, the Process Standards (Reference 26) criteria served as a primary basis for the vertical ball channel system, and horizontal control rod (HCR) system (Reference 24). 17

Referring to Table 15.6-32, no pressure tubes other than the assumed ruptured tube were predicted to fail. Also, no ball channels would exceed allowable displacement criteria. Two (HCR) channels and T-blocks adjacent to the failed tube were predicted to fail.

Graphite keys in the corner of the core may crack, but do not fail. The evaluation included two postulated conditions of key degradation: 50 percent key failure and 100 percent key failure. In either case core components and safety systems are predicted to remain operable.

In addition to the dynamic analysis of the graphite core performance under double-ended pressure tube rupture conditions, a recent best-estimate evaluation of the RCS response to a pressure tube rupture event has been performed (Ref. 31). Utilizing state-of-the-art analytical tools as recommended by the National Academy of Sciences, the best-estimate evaluation demonstrated that the High Pressure Injection system is capable of maintaining 21

adequate RCS inventory, that fuel temperatures remain below 1100°F, that the biological shield is not compromised, that the ECCS system will not automatically actuate and need not be actuated, and that the reactor will reach a safe shutdown condition. This best-estimate analysis does not postulate loss of any AC Buses, uses an initial reactor power of 3870 Mwt, and a realistic power vs time decay table developed by the Westinghouse reactor physics group.

The double-ended pressure tube break is postulated as a limiting fault. However, the critical crack length for the pressure tubes at operating conditions is such that pressure tube leaks would occur before failure. Several means of detecting coolant leaks are available, as discussed in Sections 5.2.5 and 9.3.6.3. These include the pressure tube flow monitoring system and the moderator gas system moisture monitoring system. Thus, for small tube leaks, which would be classified as infrequent events, the reactor core would be automatically scrammed via the flow monitoring system or manually shut down before a severe pressure tube failure could occur.

15.6.4.4.5 Potential For Production Of Flammable Gas Following A Reactor Tube Failure

The potential for producing hydrogen and carbon monoxide in the reactor following a pressure tube failure has been studied. Both hydrogen and carbon monoxide have flammable (explosive) concentration lower limits of about 4 percent and upper limits of 74 percent. A minimum oxygen concentration of about 6 percent is also required for a flammable mixture either as a mixture of air (or oxygen) with hydrogen and carbon monoxide, or as a diluted air mixture (diluted with steam vapor or helium) (Ref. 8).

Hydrogen and carbon monoxide are formed by the water-gas reaction between water vapor and carbon as shown by the equations:

$$C+H_2O \leftrightarrow H_2 + CO$$

Reactions also occur between graphite and the reaction products:

$$CO_2 + C \leftrightarrow 2CO$$
$$C + 2H_2 \leftrightarrow CH_4$$

Thermodynamic equilibria at one atmosphere total pressure between the various components are shown graphically in Figure 15.6-10 (Ref. 10). The equilibrium combined concentrations of hydrogen and carbon monoxide at maximum reactor graphite temperatures range up to 70 percent. Hence, the question of the potential for forming explosive quantities of flammable gases becomes one of kinetics, not equilibria.

Studies of the rate of reaction between steam and carbon have been reported in the literature (Ref. 11 and 12) and critically reviewed at Hanford (Ref. 13). The data are shown in Figure 15.6-11.

The data shown in Figure 15.6-11 were used in conjunction with graphite temperature calculations to determine the rate of hydrogen and carbon monoxide formation. Figure 15.6-12 shows the highest temperatures in layers of uncooled and cooled filler bars as a function of distance from the front of the reactor active zone. The temperatures in Figure 15.6-12 are representative of the graphite surrounding the highest powered fuel channel and are based on operating data obtained from thermocouples located in the uncooled graphite filler blocks. The maximum temperatures are 1200°F for the uncooled filler blocks and 1000°F for the cooled tube and filler blocks.

The total mass of graphite in the active core consists of approximately 640 tons equally divided between tube block graphite and filler block graphite. In order to determine a conservative upper bound on the water-gas reaction the core was divided into two temperature zones. Forty tons of graphite were assumed to be operating at 1200°F which represents the total mass of graphite in the uncooled filler blocks in the center third of the active core. The remaining 600 tons were assumed to be operating at 1000°F.

With these very conservative assumptions, the combined rate of hydrogen and carbon monoxide formation was calculated to be less than 1×10^{-4} ft^3/sec. Further, the rate calculated would be an initial rate, assuming that at the moment of the pressure tube failure the entire graphite mass (including the center of massive bars) would be bathed in steam without being cooled in the process.

The rate of gas formation would drop very rapidly. The formation rate calculated was based on the premise that steam would instantly bathe all graphite in the moderator. This would require a large flow out of the break, certainly sufficient to cause a scram.

Once the reactor scrammed, the graphite, particularly the hottest graphite, would cool rapidly by:

1. thermal conduction from higher temperature zones to lower temperature zones in a given bar;
2. normal cooling from pressure tubes and moderator cooling tubes;
3. cooling from the water injected into the moderator from the leak; or
4. cooling by the endothermic water-gas reaction.

No hydrogen is formed due to metal-water reactions since the fuel surface temperatures do not significantly increase during the double-ended pressure tube rupture event.

From the preceding, it is concluded that only one cubic foot of gas would be formed and that the formation of flammable gases following a pressure tube failure does not constitute a nuclear hazard.

15.6.4.4.6 Contamination of Steam With Activation Products

The samarium oxide balls formerly used in the ball safety system have deposited small quantities of samarium oxide dust throughout the moderator. Upon subsequent irradiation, the samarium atoms capture neutrons and are transmuted to other samarium isotopes and to other rare earths such as europium. During the repair of graphite ball channels during the 1977 summe

outage, tests were conducted to measure the quantity of activation products inside the ball channels and on the outside of the ball channels.

Based on these measurements, it was calculated that about 10,000 curies of activation products would be present on the inside of the ball channels and not available to contaminate the steam. About 5000 curies of activation products would be spread throughout the moderator and were assumed to be picked up by the steam released from the pressure tube break. Specific isotopes found to be present are tabulated in Appendix 15.6B.

15.6.4.4.7 Confinement Sequence

Following the failure of the rupture discs in the moderator gas system, the confiner atmospheric pressure would increase to 1.25 psi* as shown in Figure 15.6-13. The confiner sequence is outlined in Table 15.6-13. At 5.6 seconds into the accident, the special steam vent would open releasing contaminated steam to the outside atmosphere. Sixteen minutes later the fog sprays would have reduced the confiner atmosphere pressure to less than 3 inches w.g., permitting the special steam vent to close and the establishment of a filtered release. The activation products would be in the form of particulates and would be subject to plateout on the inner surfaces of the confiner and to washout by the fog sprays prior to escaping from the confiner.

As shown in Table 15.6-3, about 83 percent of the activated particulates would be removed from the steam by plateout and washout and about 16.6 percent are released to the outside atmosphere through the steam vent and building leakage.

Internal and external whole body doses from exposure to the activation products were calculated for the low population zone and site boundary. The results are tabulated in Table 15.6-14 and are only a small fraction of 10 CFR 100 guidelines. The control room dose for the duration of the accident

*Based on a discharge flow coefficient of 1.0.

would be 0.032 rem whole body and the 50 year dose commitment for internal whole body would be 0.7 rem.

15.6.4.5 Minor Reactor Coolant System Pipe Breaks

Although the double-ended pressure tube break is postulated as a limiting fault (see Section 15.6.4.4) and the High Pressure Injection System has sufficient capacity to mitigate that specific event, an evaluation of other small break loss of coolant accidents has recently been performed (Ref. 30). Utilizing state-of-the-art analytical tools as recommended by the National Academy of Sciences, the best-estimate evaluation included a series of small breaks in the pressurizer vent piping and primary coolant system. In the larger vent break case, a four-inch break, the ECCS was actuated on high confinement pressure and low primary system pressure. The smaller vent break, a two-inch break, did not cause ECCS actuation. Breaks in the primary coolant system were studied to determine if there was a break size or location that would cause the fuel to heat up prior to initiation of the ECCS. The study showed that there was no break size that would lead to fuel damage.

15.6.4.5.1 Justification for Availability of High Pressure Injection

The basis for reliance upon the High Pressure Injection system for small break LOCAs is that the N Reactor Restart Program for Seismic Issues (Ref. 47) adopted an interim NRC position for existing plants that (1) the Safe Shutdown Earthquake (SSE) seismic event does not cause a LOCA, nor does a LOCA occur simultaneously with an SSE, (2) offsite power may be lost due to an SSE, and (3) the plant must be capable of being brought to a safe shutdown condition following an SSE. Since an SSE is not postulated to occur simultaneously with a LOCA, the AC power supplies for the High Pressur Injection Pumps are available to mitigate small break LOCAs. Loss of offsit power alone, due to an SSE is mitigated by the diesel-driven ECCS pumps (Ref. 39) and not the High Pressure Injection pumps. Finally, the ECCS diesel-driven pumps provide the licensing basis safe shutdown capability without reliance upon the High Pressure Injection Pumps.

There are five (5) High Pressure Injection pumps, two of which are powered from the "A" AC bus, two of which are powered from the "B" AC bus and one which is a spare, which can be powered from either AC bus. This configuration provides a reliable means of ensuring High Pressure Injection for single failure of an HPI pump. The Section 3.1 description of N Reactor's compliance with General Design Criteria 33 indicates that the High Pressure Injection system provides substantial makeup flow (6,000 gpm with both AC buses and 3,000 gpm with one AC bus) to maintain reactor coolant system pressure and inventory in the event of small breaks.

Recognizing the interim nature of the NRC position to not require postulating a seismic event coincident with a LOCA, the Long-Term Service Program will demonstrate structural integrity for a 0.25 g earthquake for all remaining Category I safety-related structures, systems and components, consistent with the N Reactor Updated Safety Analysis Report (NUSAR) following plant restart (Ref. 47). 2.1

15.6.4.5.2 Single Failure Analysis

The worst case single failure for small break LOCAs is the loss of one AC bus. This loss renders 2 HPI pumps unavailable to mitigate the transient. As a result a qualitative assessment of the adequacy of two (2) High Pressure Injection pumps to mitigate a spectrum of small break LOCAs has been performed. The initial analysis was based on 4 HPI pumps being in operation in order to confirm that adequate core cooling is maintained until ECCS is actuated for moderate RCS line breaks (Ref. 30). However, as stated previously, recognizing the 3,000 gpm capacity of 2 HPI pumps, RCS line breaks with flow areas of approximately 2 in. diameter or less can be successfully mitigated without reliance on ECCS. Pipe breaks with flow areas from approximately 2 in. in diameter to 18 in. in diameter are mitigated by a combination of High Pressure Injection (2 pumps) and ECCS (2 HLDP). Breaks above 18 in. in diameter flow area are mitigated by ECCS alone (2 HLDP), which is both seismically qualified and independent of AC power supplies (see Section 15.6.5).

The basis for an 18 in. diameter break or greater being acceptable with no HPI flow is the limiting LOCA analysis described in Section 15.6.5.3.4. The limiting LOCA analysis does assume that the LOCA could be caused by a seismic event, due to portions of the non-"once-through cooling" RCS piping presumed to not yet be qualified to an SSE magnitude of 0.25 g. Assuming a seismic event coincident with the limiting large break LOCA requires the exclusion of all HPI since the AC power supplies (offsite and onsite) are not seismically qualified to 0.25 g. The remainder of the RCS piping is to be qualified to an SSE of 0.25 g, as part of the Long-Term Service (LTS) Program.

When the RCS piping has been entirely qualified by analysis or modification for an SSE of 0.25 g, a spectrum of deterministically induced partial line breaks at a spectrum of locations will be evaluated to ensure that ECCS flow in conjunction with 2 HPI pumps (allowing single failure on an AC power source) and partial RCS coolant pump flow can adequately mitigate the spectrum of moderate RCS line breaks. It should be noted that the N Reactor ECCS design which utilizes diesel-driven pumps is uniquely capable of coping with a Station Blackout Event. See Section 15.6.5.3.8 (no onsite or offsite AC power). The probability of losing both AC power sources (onsite boiler-generator, and multiple offsite transmission lines) coincident with a LOCA is considered sufficiently remote to permit reliance on at least one AC source being available in the event of a LOCA for both the Return to Service (RTS) and Long-Term Service (LTS) Programs.

A single failure of one HPI pump, being powered from the AC power source assumed to be available, can still be accommodated by use of the spare HPI pump, which can be powered from either AC bus. Should the spare pump already be aligned with the operating bus it could be started from the control room. Should the spare pump be aligned with the disabled bus, manual action to switch a breaker would be required to start the pump. This manual action would require at least a few minutes. It should also be recognized that the N Reactor LTS is for a limited operating lifetime of approximately 5 years. This further reduces the potential for coincident failures of AC power source with a postulated LOCA.

15.6.5 Major Reactor Coolant System Pipe Breaks

15.6.5.1 Introduction and Bases for Selection of Analysis Techniques

Loss-of-coolant accidents (LOCAs) are defined as postulated events involving reactor coolant system pipe breaks or other coolant pressure boundary component failures, up to the size of a double-ended break of the largest system pipe. The maximum-sized pipe break that can be compensated for by N Reactor's coolant makeup system is in excess of the release rate of the postulated double-ended pressure tube break or connector breaks. See Section 15.6.4.5. A discussion of single failures and available power supplies assumed for small and large break LOCAs is provided in Section 15.6.4.5.2. Nuclear piping systems are designed and constructed using exacting specifications to reduce the probability of catastrophic failure to the minimum practicality achievable. In addition, the systems are subject to "in-service inspections" designed to detect and permit the elimination of any service-related degradation. Nonetheless, fatigue and/or stress-assisted propagation of small flaws inherent in any system remain the most likely potential mechanism for a system failure. The presence of such flaws is, in fact, incorporated into the safety factors provided in the design of these systems. Although the stress corrosion assisted propagation mechanisms are much less important to the carbon steel components of the N Reactor RCS than for stainless steel piping systems, the remote possibility of fatigue mechanisms resulting in catastrophic failure of a component provides the mechanistic basis for analysis of N Reactor loss-of-coolant accidents.

The scale and consequences of the April 1986 accident at the Chernobyl Nuclear Power Station in the Soviet Union provided the impetus for reexamination of the safety of U.S. Department of Energy defense production reactors. One aspect of this review was the reexamination of N Reactor performance under major LOCA conditions. A number of independent reviews, assessments, and appraisals (Refs. 32 through 35) were conducted that, among many other issues, addressed core, RCS, and ECCS performance following a LOCA. The reviewer's recommendations and their resolutions are summarized below.

These reviews resulted in the following recommendations for improving capabilities for identifying the N Reactor limiting LOCA and updating the analyses using state-of-the-art methods and techniques.

1. Develop an expanded in-house capability to achieve an in-depth understanding of technical issues associated with major LOCAs:

   - o Methods should be comparable in technical sophistication to those used in the commercial nuclear industry
   - o State-of-the-art, best-estimate analytical tools should be used, as well as best-estimate assumptions
   - o Appropriate allowances should be made for uncertainties.

2. Develop credible, state-of-the-art analytical tools to assess N Reactor thermal-hydraulic behavior during LOCAs:

   - o Model nonhomogeneous and nonequilibrium two-fluid conditions during two-phase flow
   - o Model the horizontal-geometry core flow characteristics of the N Reactor
   - o Model the LOCA blowdown and reflood period
   - o Include detailed noding to account for local conditions in the core and hot channel, and to account for possible complex flow paths.

3. Complete confirmatory reanalyses of the ECCS showing satisfactory system performance:

   - o Meet the intent of the regulatory requirements currently met by commercial nuclear power plants
   - o Be comprehensive in scope (i.e., complete spectrum of break sizes and locations) and incorporate the reflooding period

- o Apply a rigorous quality assurance program to the computer codes selected
- o Account for condensation-driven flow and pressure dynamics in order to evaluate the potential for thermal shock and water hammer phenomena associated with LOCAs.

Sections 15.6.5.2, 15.6.5.3, 15.6.5.4, and 15.6.6 describe the LOCA reanalysis that was performed in response to the concerns expressed in these reviews. The discussion which follows addresses in detail how the above recommendations were implemented.

The analysis described in Sections 15.6.5.2, 15.6.5.3, 15.6.5.4, and 15.6.6 address the above stated concerns and recommendations as follows:

1. Capability to provide in-house, state-of-the-art, technically sophisticated analytical methods and tools was achieved by using the RELAP5/MOD2, Version 36.05 computer code (Ref. 37) to perform the LOCA reanalyses discussed in this section. This code has been assessed and judged acceptable for such analyses by the NRC, subject to demonstration of applicability to specific facilities (Ref. 36). A summary description of this light water reactor transient analysis code is provided in Section 15.6.5.1.2.

   RELAP5 is a best-estimate, state-of-the-art, advanced thermal-hydraulic code capable of modelling a two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic transient simulation of a two-phase system. Several modifications were also incorporated into the basic code to simulate reflood phenomena associated with filling long, horizontal tubes with liquid flowing in the vertical inlet and outlet risers. These modifications provided the recommended and necessary capability to account for the horizontal core geometry under reflood conditions. They are discussed in detail in Reference 38, Section 2.2. Additional validation and sensitivity studies are planned as part of a 10 CFR 50, Appendix K conformance program.

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2. RELAP5 capability to satisfactorily evaluate hot channel temperatures required modification of the system model (discussed in Section 15.6.5.3.1) to simulate gradual voiding of the fuel region during blowdown. This was accomplished by increasing the number of axial nodes in the fuel region model from 3 to 16 to reduce unrealistically rapid voiding and condensation of voids of coolant that are simulated when using the 3-node model.

   Limitations of computer core size did not permit this detailed noding scheme to be used in the entire system model, so it was implemented only for a single pressure tube. The more detailed model was then run separately, driven with boundary conditions that were derived from the coarser system model. A case was run to verify the acceptability of results using this approach. Details of the verification case and a comparison of its results with those of the single tube model are provided in Reference 45.

3. To account for uncertainties, and to meet the intent of regulatory requirements currently required of commercial nuclear power plants, a spectrum of postulated accident cases was analyzed with RELAP5 (Table 15.6-34). A series of calculations of postulated pipe breaks was performed (Refs. 38, 39, 40 and 44) using best-estimate assumptions; i.e., analysis that would predict a realistic, or most probable value of peak cladding temperature (Ref. 36). This spectrum of break analyses determined the system response to a range of large break LOCAs and from it, a worst-case break size and location, the limiting large break LOCA, was identified for further analysis.

   While the break spectrum studies were done on a best-estimate basis, the did include such conservative assumptions as loss of the HPI system and the Graphite and Shield Cooling System (GSCS). Additional conservatism was represented by the use of a decay heat rate in the outer fuel assembly (Table 15.6-38) approximately 3% higher than recentl updated rates, and an ECCS flow rate based on only 2 of 3 High Lift Diesel Pumps (HLDPs) (Refs. 40 and 45). The limiting large break LOCA

final analysis included additional conservatism in that it uses 1.2 ANS Decay Heat (1971) and loss of AC power.

The limiting large break LOCA was determined to be a longitudinal split in a cold leg crossover pipe, with area equal to 100% of the pipe cross sectional area. The limiting large break LOCA previously reported (see Section 15.6.5.3.1) was a 40% split in the cold leg crossover piping. Conservative estimates of peak clad temperatures were then calculated, using RELAP5 average channel and hot channel evaluation models that incorporated features and assumptions meeting the intent of 10 CFR 50.46 and 10 CFR 50, Appendix K. Some of the requirements of these regulations are not directly applicable to N Reactor, due to its horizontal geometry and its fuel elements being uranium metal, clad with Zircaloy-2, rather than unbonded, "...uranium oxide pellets within cylindrical Zircaloy cladding." The modifications to the computer models, and the assumptions and requirements that are incorporated in the LOCA analyses, are discussed in detail in WHC-SP-0285 (Ref. 45). 21

4. As noted in Item 3, analysis of a full spectrum of break types, sizes, and locations that includes the ECCS reflood period, and meet the intent of 10 CFR 50, Appendix K, fully addresses the recommendation for reanalysis that is comprehensive in scope. The improved techniques and re-evaluation of the assumptions used in previous analyses resulted in the current analysis identifying a different case (100% vs 40% split in the cold leg crossover piping) as the limiting large break LOCA.

5. RELAP5/MOD2, Version 36.05, is the culmination of approximately 12 years of development, testing, and use. This work was performed under the software quality assurance program of the Idaho National Engineering Laboratory (INEL) where the code was developed. Documentation of RELAP5/MOD2 and additional code modifications related to the limiting LOCA analysis models are provided in Reference 46.

6. An analysis methodology for the prediction of the occurrence and magnitude of condensation-induced water hammer in the N Reactor primary piping during ECCS water injection was developed and applied to the system. This analysis is discussed in Reference 42 and 29.

Evaluation of LOCAs is accomplished using the RELAP4, RELAP5/MOD2, NEWIGL, COBRA, HECTR, and MELT computer codes as described below.

A summary of RELAP5/MOD2 is provided above, with a more detailed description provided in Section 15.6.5.1.2.

The RELAP4 analyses were performed assuming that the reactor was initially at full power, flow rate and pressure. The transients were all initiated at time zero. It was assumed that all ac power was lost which rendered the high pressure injection system (HPIS), the pressurizer heaters and the secondary coolant system (SCS) feedwater pumps inoperable. Hence, there was no makeup to the RCS and no flow to the steam generators. The RELAP4 models used for the LOCA analysis included heat transfer between the primary and secondary sides of the steam generators. The SCS was modeled as a single control volume using a bubble rise model to account for phase separation. The normal SCS flow rate temperature and liquid level were used as initial conditions, and the heat transfer area adjusted to give the correct initial heat transfer rate. At the time of the break, the SCS flow rate was stopped simulating an ac power failure. Heat transfer after the break was assumed to be to a boiling pool of water in the SCS using the normal heat transfer correlations contained in RELAP4. The horizontal control rod (HCR) safety circuit would be tripped by low pressure in the hot-leg risers at less than 0.1 second for all cases. However, due largely to delays in the pressure monitoring instrumentation system, the control rods are assumed not to move into the core for 2 seconds following the trip. The HCR safety circuit trip signal also trips off the reactor coolant pumps, permitting them to coast down.

For the RELAP4 analyses, the ECCS trips, in all cases, on receipt of a combination of a low hot-leg riser pressure of 663 psia and a high confinemen system pressure of 25 in. w.g. This opens four V-3 and four V-4 valves.

The ECCS pumps start on the HCR safety circuit trip signal and, after the V-3 valves open, are deadheaded against the closed CV-2R check valves. The ECCS fluid cannot enter the cold-leg risers until the pressure in those risers decays to 306 psia which is the shutoff head of the ECCS high-lift diesel pumps (HLDP). The selected conservative value of 306 psia provides margin for pump shutoff performance degradation. For the RELAP4 analyses, the single failure criteria has been doubly applied to the V-4 valve system. Three valves are required to be in service on each side of the reactor; but all cases, excluding the accidental V-4 actuation, assume that only two valves are operable on each side, with an opening time of 25 seconds. 21

The RELAP4 loss of coolant analysis assumes multiple single failures to evaluate the adequacy of the ECCS system, i.e., loss of all onsite and offsite ac power, failure of one ECCS pump and failure of two V-3 and V-4 valves to open.

The initial conditions are given in Table 15.6-15.

Engineered safety features, described in Chapter 6, are provided to ensure that the consequences of this LOCA will not affect the public health and safety. The following sections describe the methods of analysis, results of analysis and environmental consequences of this postulated accident. See Appendix 15.6B for a discussion of the potential techniques and assumptions.

15.6.5.1.1 Safety Evaluation Criteria

The safety evaluation performance criteria for loss-of-coolant accidents are as follows:

1. The reactor protection system and the ECCS shall limit the course of the accident such that the essential heat transfer geometry of the fuel assemblies and reactor moderator are preserved, the maximum fuel cladding temperature does not exceed the fuel failure temperature, and the hydrogen generated by the metal-water reactions is limited as specified in 10 CFR 50.46, i.e. 17% local and 1% core average.

2. The confinement system pressure and temperatures shall remain within design values. See Section 6.2.

3. Resultant radiation doses shall remain within 10 CFR 100 guidelines.

15.6.5.1.2 Method of Analysis

RELAP5/MOD2 Analysis Methods

RELAP5/MOD2 is a light water reactor, thermal-hydraulic, transient analysis computer code, which was developed at the Idaho National Engineering Laboratory in support of need by the U.S. Nuclear Regulatory Commission (NRC) for an advanced, best-estimate prediction capability for applications pertaining to the regulatory process (Ref. 37). RELAP5 is an improved analysis tool when compared with RELAP4, which was based on a single fluid, homogeneous equilibrium model of the two-phase flow process. The principal new feature of RELAP5 is that it models a two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic transient simulation of a two-phase system. It uses a full, nonequilibrium, six equation, two-fluid model.

The RELAP5 code has been extensively applied in experimental comparisons and other analytical projects. Examples of NRC regulatory support applications include predictions for the LOFT and Semiscale test programs, simulation of design basis LOCAs, pressurized thermal shock investigations, and evaluation of commercial power reactor operational transients and operator guidelines.

A special version of RELAP5 (Version 36.05) was developed to support the N Reactor analyses. Modifications to the basic code were necessary to properly simulate the voiding and subsequent refilling of the horizontal N Reactor core during reflooding portions of a LOCA. The principal changes were made to correct the overprediction of interface condensation rates in the long, horizontal core channel containing the reflood front. Reference 38 provides the details of these modifications.

Several computer models are used to perform the limiting LOCA reanalyses reported in this section. The basic model is the system, average channel model, which is shown in Figure 15.6-176.

The system model is based on the N Reactor piping arrangement shown in a pictorial schematic in Figure 15.6-177. Major features of the system model, including the more detailed representations of the core noding arrangement, are summarized and shown in Appendix 15.6D. A detailed description of the model is provided in Reference 38.

The system model is used to analyze a series of "best-estimate accident" cases that include pipe breaks in the cold leg manifold, inlet riser, outlet riser, and station blackout sequences. A spectrum of cold leg crossover break sequences that also included station blackout conditions was performed using the basic system model, with some modifications, to establish the limiting LOCA. 21

Additional modifications of the RELAP5 system model were used to perform average and hot channel analyses of the limiting LOCA in accordance with the intent of requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K. Descriptions of specific key features of the average channel, hot channel, and hot channel verification evaluation models follow.

A RELAP5 Hot Channel Analysis verification model was used to verify the validity of using individual tube evaluation models, as discussed in the following three subsections.

Average Channel RELAP5 Evaluation Model

An average channel analysis of the limiting LOCA was performed using a RELAP5 evaluation model that incorporated appropriate conservatisms to meet the intent of Appendix K, and to ensure that applicable Technical Specification limits or requirements were observed. The method used for this conservative RELAP5 analysis was similar, up to the time of reflood, to that used in the RELAP4 analyses, whose results are presented later in this section.

The new evaluation model (EM) analysis was undertaken to establish the average channel peak clad temperature for the limiting LOCA, and to provide thermal-hydraulic boundary conditions for the hot channel analyses that calculated the LOCA peak clad temperature. These analyses were also performed using a RELAP5 code model, which was modified to provide more realistic estimates of the pressure tube thermal-hydraulic characteristics.

Several components of the system model were not required for the EM analysis and were removed from the EM to minimize computer memory requirements and running costs. These components were primarily those in the High Pressure Injection system, which is non-functional with loss of AC power, so does not participate in the transient.

Key modifications and pertinent boundary conditions for running the RELAP5 EM included:

1. Reactor Power -- 102% nominal, or 4080 MWt.

2. Decay Heat -- Decay heat is based on the 1971 ANS standard as required by Appendix K. The actinide decay power is based on end of cycle burnup, while infinite operating time is assumed for fission product decay. A multiplier of 1.2 is applied to the fission product isotopic decay.

3. Rod Worth/Insertion Time -- Rod worth available for insertion was assumed to be 40 mk with a time of 1.25 second to 75% insertion.

4. Reactor Scram Signal -- Reactor scram is conservatively assumed to occur only on the Tech Spec low system pressure of 1540 psig. Scram delay time is conservatively set at 1.15 second, which includes sensor, relay, and rod movement delays.

5. Primary System Pressure -- Primary system pressure is input to the model at 1825 psig (the Tech Spec limit) at the primary coolant pump discharge.

6. ECCS Actuation -- Assumed to occur only on coincident occurrence of the Tech Spec low pressure limit of 645 psig, and confiner pressure high limit of 25 in. water gage. A total delay of 6 seconds (1.0 second for sensor and 5.0 seconds for relay responses) is assumed until ECCS is actuated following the pressure trips.

7. V-3/V-4 Response Time -- Opening time for the V-3 and V-4 valves is assumed to be 15 seconds. The first 5.0 seconds is valve lift time with no flow; the remaining 10.0 seconds is linear opening time. If RCS pressure has not decayed to 306 psig in 15 seconds, ECCS flow does not begin until it does reach this value or is otherwise justified.

8. ECCS Supply -- The ECCS supply is based on a degraded high-lift diesel pump (HLDP) supply. Systems losses are maximized as determined from test data. Degradation is 4% head and 6% flow. ECCS coolant temperature is assumed to be 110°F for stored water (used for the first 20 minutes of the transient) and 80°F for river water. These temperatures represent conservative bounds. 21

9. Single Active Failure -- The single active failure is assumed to be a HLDP, so ECCS supply is based on only two of three pumps. This failure was determined to be more severe than failure of one V-3 or V-4 valve, based on test data, as discussed in Reference 45.

10. Loss of AC Power -- As required by Appendix K, a loss of AC power is assumed. However, no credit is taken for the SCRAM input that would normally occur from this event. A discussion of the basis for the loss of AC power assumption is provided in Section 15.6.4.5.2. As a result of the loss of AC power, the following events occur:

    - Primary coolant pumps coast down to zero rpm.
    - The High Pressure Injection (HPI) system is lost.
    - The steam generator secondary is isolated (HPV-6261 and HPV-201 valves close).

- o Steam generator bypass valves remain closed.
- o The Graphite and Shield Coolant System (GSCS) is conservatively assumed not to transfer to once-through cooling.

11. Confinement Pressure -- Confiner pressure is assumed to be at the design pressure of 5.0 psig to maximize backpressure for the LOCA analyses.

The above assumptions and boundary conditions are consistent with those listed in Table 15.6-33, and the rationale for their use is discussed in greater detail in Reference 45.

Hot Channel RELAP5 Evaluation Model

Hot channel analysis was performed using two single-tube RELAP5 evaluation models, modified to provide greater axial detail. The RELAP5 system model, used for calculating the average channel performance, includes only three axial nodes in the pressure tube region. This node arrangement is sufficiently detailed for establishing system responses. However, the coarse node structure leads to simulation of unrealistically rapid voiding and condensation rates during reflood in the fuel region during the LOCA transient, that results in non-physical condensate flows.

The models represent single Mark IV and Mark IA fueled pressure tubes in the peak power region. Both models are based on standard loadings. The Mark IV loading consists of 16 standard length (26.1 in. each) fuel elements. The Mark IA loading uses two standard length Mark IV fuel elements (26.1 in. each) at the front and rear of each fuel column, with 15 standard length Mark IA elements (20.88 in. each) between.

Both hot channel models consist of 3 radial and 16 axial nodes. Thus, for the Mark IV, the node-to-fuel element modeling is one-to-one. For the Mark IA model, the first and last two Mark IV elements are modelled explicitl (one-to-one), and the remaining 15 fuel elements are modelled by the 12 remaining nodes. Except for these differences, and the fact that the appropriate Mark IV and Mark IA fuel element radial dimensions are used, the

models are identical. Other details of the node arrangements are discussed and illustrated in Appendix 15.6D.

Two hot channel models were necessary to evaluate response of the peak power tube because calculations showed that peak LOCA temperatures in a Mark IA-fueled pressure tube are higher than in a Mark IV-fueled tube at the same nominal power input. This is attributed to the fact that the Mark IA outer fuel cylinder has a smaller volume of fuel in the outer cylinder than the Mark IV element. It, therefore, has less thermal capacitance and so is subject to a greater temperature increase for a given heating rate.

1. RELAP5 Evaluation Model Percentage Energy Deposition -- Relative percentage energy deposition in the hot channel model is shown in Table 15.6-35. These percentages are based on recently updated calculations for both a Mark IV and a Mark IA unit cell (Ref. 45). Energy deposition in the coolant channels and fuel cladding is conservatively assumed to occur in the fuel. After scram, the percentage energy deposition changes (as allowed by 10 CFR 50, Appendix K) because of a drop in neutron heating. After 60 seconds, decay heat is primarily gamma heating. 21

   For hot channel calculations, the percentages of heat deposition are conservatively held constant from 0 to 60 seconds of the transient. After 60 seconds, relative percentage values are modified as shown in Table 15.6-35 and held constant for the remainder of the transient (Ref. 45).

2. RELAP5 Evaluation Model Axial Power Profile -- The axial power profile for the hot channel calculations is based on an average Traveling Wire Flux Monitor (TWFM) profile 6, for a 6% operating mode, with a peak-to-average (P/A) ratio of 1.414. When averaged over the length of a 26.1-inch fuel element length, the resulting P/A is 1.396 (Ref. 45). The power profile is shown in Figure 15.6-178.

Based on an initialized tube power of 5000 kw for the Mark IV, and 4700 kw for the Mark IA models, the peak linear heating rates are 201 kw/ft and 189 kw/ft, respectively. The reduced tube power used for the Mark IA calculations was necessary to avoid excessive peak cladding temperatures that result from the lower thermal capacitance of the Mark IA fuel elements. The differences in dimensions and the relative volumes of the Mark IV and Mark IA fuel are shown in Table 15.6-36.

Power densities (kw/in^3) for Mark IV and Mark IA fuel, at their respective peak linear heating rates, are shown in Table 15.6-37, based on energy depositions in Table 15.6-35 and fuel dimensions in Table 15.6-36 (Ref. 45). Initial conditions for the hot channel models are shown in Table 15.6-50.

RELAP5 Hot Channel Analysis Verification Model

Because of computer size and cost limitations, it was not practical to provide the detailed, 16-node Mark IV/Mark IA pressure tube model in the full system model, which has only 3 axial nodes per pressure tube (see Appendix 15.6D). Therefore, the detailed, individual tube, hot channel models were run separately, driven with appropriate hydrodynamic boundary conditions derived from the system model. To verify the validity of this approach, a separate case was run using the coarser node scheme of the system model, again driven with hydrodynamic boundary conditions from the system model.

The verification model noding scheme was taken directly from the system model (see Appendix 15.6D, Figure 15.6-177) and represents the topmost channel of the broken crossover loop because this location was expected to yield the peak cladding temperatures. The model is sized to represent 6.25 pressure tubes at an average tube power of 4.1 MW and a peak linear heating rate of 175 kw/ft. Additional details of the model are discussed in Reference 45.

1. RELAP5 Verification Model Percentage Energy Deposition -- Relative percentage energy deposition in the fueled region of the system and hot channel verification models is shown in Table 15.6-38 (Ref. 45). These percentages are derived from earlier neutronic calculations for use in

the RELAP4 analyses reported in Sections 15.6.5.3 and are compared with current calculations of percentages, using more sophisticated techniques, in Table 15.6-38. The system and verification model calculations are based on the earlier neutronics percentages, which represent a difference of approximately 3% in heat deposition in the outer fuel region.

In addition, the energy depositions shown are pre-scram values. Early in the scram transient, decay heat is dominated by gamma heating, which increases energy deposition in the pressure tubes and graphite region as shown in Table 16.5-35. This phenomenon was not represented in the system and hot channel verification models; i.e., the pre-scram energy ratios were maintained throughout the transient, which constitutes an element of conservatism in the calculations.

RELAP4 Analysis Methods

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Description of Blowdown Model (RELAP4)

These calculations were performed with the RELAP4 code and provides a description of the thermal hydraulic response of the RCS. The times at which safety system trip points are reached are determined by the loop depressurization. The system depressurization rate determines the time of automatic actuation of the ECCS which opens the V-3 and V-4 valves and increases the rate of RCS depressurization. The average channel fuel-to-coolant heat transfer is determined throughout the blowdown transient. The time at which the inlet riser pressures decay to pressures at which ECCS coolant may enter is determined by RELAP4.

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The blowdown hydraulic analysis for large pipe breaks is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the core cooling analysis. This requires the ability to predict the flow rate, quality and pressure of the fluid passing through the core region. The RELAP4 code was developed by the Nuclear Regulatory Commission to provide the information.

A comprehensive program of code verification and checkout was performed prior to the analyses presented in order to verify the adequacy of the code and modeling technique to describe the N Reactor LOCA. This program included verification of the hydraulic model by comparison with experimental data, modification of the code to model the N Reactor fuel, assurance of conformance with the intent of 10CFR 50, Appendix K requirements and a series of sensitivity studies. This program of verification, including the code modifications, the DNB correlation choice, the sensitivity studies and the compliance with Appendix K, are described in detail in Reference 14.

The fundamental equations of conservation of mass, energy, and momentum are applied for single- and two-phase fluids in order to calculate the main dependent variables: density, internal energy, and mass flow rate. A set of auxiliary dependent variables is defined by the system governing algebraic equations.

The RELAP4 computer code performs a comprehensive space-time dependent analysis of a loss-of-coolant accident and is designed to treat all phases of the blowdown. The stages are:

1. A subcooled stage with the rapidly changing pressure gradients in the subcooled fluid.
2. A two-phase depressurization stage.

The code employs a one dimensional analysis in which the entire RCS is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coast-down and cavitation, core and steam generator heat transfer and the Bowring & Barnett DNB correlations are incorporated in the code (Ref. 14).

The RELAP4 model divides the RCS into discrete volumes called control volumes (CV) which are connected to each other by junctions (J). These refer to specified points in the model schematics and are placed in parentheses following the location description. The break areas are denoted as percentages (i.e., 50%) which is the percentage of the cross sectional area

of the pipe which is assumed to open as a result of the break. The reactor kinetics portion of the RELAP4 code is not incorporated into the blowdown analysis. Instead, the transient power decay is determined using the NEWIGL computer code. The NEWIGL code is described in Section 4.3.3.4.1.

The reactor power decay used in RELAP4 was determined by assuming that only 17 mk of control rods were available for insertion upon receipt of the reactor trip signal. This is very conservative in that, at full power operation, in excess of 55 mk of insertable control rods would be expected. The negative reactivity associated with coolant voiding was conservatively neglected. In addition, the effects of the automatic ball system drop on receipt of the signal which actuates the ECCS has been conservatively neglected.

A more realistic, but still very conservative, reactor power decay corresponding to 29 mk of scrammable control rods was applied to the hot channel and reflood analyses. In addition, the analysis will include a scrammable rod worth corresponding to initial attainment of full power after a long reactor shutdown for fuel recharging. Using more realistic, but still conservative, values of the core power decay would further reduce the peak fuel temperature presented in Section 15.6.5.3.1.

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The schematic of the RCS used in the blowdown analysis is shown in Figure 15.6-14. The specific models used in the analyses of the various postulated pipe breaks are provided in the specific results sections.

A more detailed description of the RELAP4 models used in the LOCA analysis is provided in Reference (2). The more detailed description of the manifold break RELAP4 models is provided in Reference (15).

The RELAP4 average fuel-to-coolant heat transfer rates are determined throughout the blowdown process. This heat transfer data is input into the MELT computer code which determines the hot channel fuel transient and the core volumetric transient. The description of the MELT computer code is presented in Appendix 15.6C.

15.6.5.2 Summary of Results

For the spectrum of postulated piping breaks within the reactor coolant system pressure boundary up to and including the double-ended break of the largest pipe, the emergency core cooling system will limit the fuel clad temperatures and assure that the essential heat transfer geometry is preserved. That is:

1. The maximum fuel element cladding temperature does not exceed 1890°F, which is below the clad failure temperature of 1900°F. The maximum local metal-water reaction is limited to less than 17 percent of the Zircaloy clad with average metal-water reaction less than one percent.
2. The "hot-leg breaks" result in peak fuel temperatures during the reflood phase of the transient.
3. The "cold-leg breaks" peak fuel temperatures occur prior to reflood for cases in which the CV-5 closes and remains closed during the transient.
4. The clad temperature transient is terminated at a time when the core geometry is still amenable to the cooling.
5. The core temperature is reduced and decay heat is removed for long periods of time.
6. The fuel heat-up transient is terminated before fuel clad failure could release fission products to the reactor coolant system. The maximum radiological consequence will be well below the 'realistic' case presented in Table 15.6-31 and discussed in Section 6.5.

The sequence of events for the limiting large break LOCA (RELAP5) is provided in Table 15.6-41. The maximum hot channel fuel temperature of 1890°F occurred for the 100 percent cold-leg crossover break (five cell operation) and is discussed in Section 15.6.5.3.4.

The maximum average channel fuel temperatures for the RELAP4 based analyses of cold-leg breaks and the hot-leg breaks are shown in Tables 15.6-16 and 15.6-17, respectively.

15.6.5.3. Cold-Leg Breaks

A spectrum of cold-leg breaks was selected for analysis.

The cold-leg breaks are generally characterized by a very short delay before the CV-5 valves close, stagnating the core and resulting in significant fuel temperature increases. Hot-leg breaks are characterized by a fairly long delay time before the CV-5 closure. This allows adequate cooling of the fuel during the initial high-power period, which delays the fuel heat-up. These phenomena cause the cold-leg breaks to result in higher fuel temperatures than the hot-leg breaks. The peak confiner atmosphere pressure, however, would be lower for cold-leg breaks than for hot-leg breaks of the same break size. The discussion of the results of the analyses is organized by break location and break size, with the RELAP4 based analyses listed first and RELAP5 based analyses listed last. The RELAP5 based analyses represent the design basis, with the RELAP4 based analyses for the same break size being included for its historical value. Section 15.6.5.3.4 describes the limiting large break LOCA. 21

15.6.5.3.1 Cold Leg Crossover - Five SG Cell Operation, 40 Percent Split (RELAP4)

This analysis discussion is provided for historical continuity, but has been superseded by RELAP5 based analyses of the same event, which are discussed in Section 15.6.5.3.4. 21

The schematic for the model used is shown in Figure 15.6-15. The transient is initiated at time t = 0 by opening a split equivalent to 40 percent of the pipe area (J31)* in 1 of the 16 cold-leg crossovers just upstream of the CV-5 valve (J48). This causes a local depressurization which immediately closes the CV-5 valve, stagnating flow in the pressure tubes fed by the affected riser. The reactor coolant system continues to depressurize through the break until the pressure in the hot-leg risers (CV-15) reaches 663 psia, at which time the ECCS is tripped. After a 6-second delay, four V-4 valves (J25), two on each side, are assumed to open on a 25-second ramp allowing

*J31 is Junction 31 and is shown as 31 on the model schematic.

some of the reactor coolant to blow down into the emergency dump tank (EDT) (CV-26). This increases the depressurization rate until the cold-leg riser (CV-14) pressure reaches 306 psia. At this time, ECCS water enters the RCS through the bottom of the cold-leg risers (J88, 89). This coolant flow closes the CV-5 valves on all 16 risers, forcing ECCS coolant through the core which terminates the fuel heat-up transient.

Figure 15.6-16 shows the pressure response of the cold-leg risers (CV-14). The increase in slope at 140 seconds is due to the opening of the V-4 valves (J25). See Figure 15.6-17. The pressure spike at 110 and 130 seconds is caused by the check valve model used which assumes only full-open or full-closed positions. When the CV-5 valves (J14) open at 110 seconds (see Figure 15.6-18), it predicts an unrealistic pressure rise in the cold-leg risers (CV-14). This pressure response has been corrected in Figure 15.6-16 as shown using the slope observed after the pressure spike. The break (J31) flow rate is shown in Figure 15.6-19, with the flow choking in less than 1 second. The fuel temperature responses for the portion of the core fed by the intact and broken risers are shown in Figures 15.6-20 and 15.6-21, respectively. The fuel surface temperatures are the outer surfaces of the outer elements in the axial center one-third of the core. The heat fluxes for these surfaces are shown in Figures 15.6-22 and 15.6-23. The temperature of the fuel fed by the broken riser shows an immediate and continual rise until reflood occurs since there is no flow through this portion of the core during this time. The heat flux for this surface exceeds the critical heat flux (CHF) within 2 seconds and falls off to the low values as shown in Figure 15.6-23. The plateaus in the fuel temperature rise are the α-β and β-γ phase changes, respectively, for metallic uranium. The temperature response for the fuel fed by the intact risers (CV-14) shows a delay caused by the CV-5 valves remaining open for 15 seconds (Figure 15.6-18) providing coolant to the fuel. Between 15 and 122 seconds, the heatup rate is continuous with a small amount of heat transfer to the coolant as shown by the heat flux in Figure 15.6-22. At 122 seconds, the CV-5 (J14) valves open due to the opening of the V-4 valves (J25) permitting some coolant to enter the core and cool the fuel surfaces. The sequence of events for the cold-leg crossover breaks are presented in Table 15.6-20.

Reflood would begin at 146 seconds with a peak average channel fuel temperature of 1425°F for this RELAP4 based analysis. The limiting large break LOCA is described in Section 15.6.5.3.4. 21

Hot Channel Analyses (RELAP4 Based Analyses)

A spectrum of cold-leg crossover break sizes were used to determine the specific break size which would close the CV-5 valve to the affected riser within the first second of the transient and remain closed for the duration of the transient. The smallest break size which meets this criterion will result in the longest system depressurization and, consequently, the longest time for the affected core to be essentially stagnated. This long stagnation period and the low fuel-to-coolant heat transfer results in the maximum fuel temperatures at the time when the ECCS system can provide cooling.

The transient which met the above criterion (see Figure 15.6-24) was the 40-percent cold-leg crossover break. This break results in the maximum fuel and cladding temperatures and was selected for the hot-channel and reflood analyses. Utilizing the RELAP4 based analyses all other cold-leg and hot-leg breaks would be less severe and would result in lower fuel temperatures. See Tables 15.6-16 and 15.6-17. Subsequent RELAP5 based analyses have identified the 100% cold leg crossover break as being the most limiting. 21
See Section 15.6.5.3.4.

The hot-channel fuel temperatures for RELAP4 based analyses were calculated using the MELT computer code described in Appendix 15.6C. The MELT code determines the fuel heat-up transient using the RELAP4 average channel fuel-to-coolant heat transfer rate throughout the blowdown transient to the point when the front riser pressure decays to 306 psia and the ECCS can enter the RCS and cool the core. MELT conservatively applies this heat transfer rate to the maximum core linear heating rate.

The maximum tube power during full power operation is 5000 kW. See Table 4.4-1. The axial peaking factor used in the hot channel analysis consists of a 1.55 center-peaked profile which corresponds to the maximum measured values shortly after reaching full power operation. These values

result in a maximum core linear heating rate of 223 kW/ft as compared to the maximum RELAP4 calculated heating rate of 150 kW/ft which is representative of a core average fuel channel. The heat-up analysis is applied to Mark IA fuel since it has the lowest capacitance and, therefore, the most rapid fuel temperature increase. In addition, the maximum tube powers normally occur in Mark IA fueled pressure tubes.

The RELAP4 blowdown analysis assumed 2 seconds of full power operation after the rear riser reached the low pressure trip point to account for instrument delay. The subsequent assumed rod scram corresponded to the insertion of 17 mk of control rods. The negative reactivity effects of core voiding or the automatic ball safety system trip were not included. A 20 percent uncertainty to the decay heat curve was applied as specified in 10 CFR 50 Appendix K.

A more realistic, but still very conservative, power decay curve corresponding to 29 mk of scrammable control rods was applied to the hot-channel analysis. This is far less than the 55 mk of scrammable rods which would be expected upon attaining full power operation after an extended shutdown. The 20 percent decay heat uncertainty was again applied as required by 10 CFR 50 Appendix K. The power decay factors for the Mark IA and the Mark IV fueled lattices are shown in Tables 15.6-18 and 15.6-19, respectively.

The fuel heat-up transient for the maximum powered element of the hot channel is shown in Figure 15.6-25.

15.6.5.3.2 Cold-Leg Crossover - Four SG Cell Operation - 55 Percent Split (RELAP4)

The model schematic for this case is similar to the five cell cold-leg crossover model shown in Figure 15.6-15 with the following exceptions: a) th riser break size was changed to 55 percent; b) the model volumes from the outlet manifold through the steam generator to the inlet manifold were adjusted to reflect four cell operation; and c) the volumes near the intact loop CV-5 valves (J14) were changed to approximate the broken loop arrangement. This last change was made to determine if the reopening of the intact loop CV-5 valves near the end of the transient was real or simply a

spurious event which occurs due to inadequacies in the valve model. The four cell model schematic is shown in Figure 15.6-164.

The initial core power for this case was 3535 Mwt (maximum four SG cell power plus one percent overpower). The critical break size of 55 percent was determined by analyzing several break sizes to find the worst case. The cold-leg crossover worst case break is the smallest break size that will immediately close the CV-5 valve with the valve remaining closed. This stagnates flow in that part of the core served by the affected riser while at the same time extending the riser blowdown longer than a larger size break.

For the five cell cold-leg crossover case, a spectrum of break sizes was investigated to determine the critical break size of 40 percent. The reduced flow rate associated with the four cell case results in a higher outlet enthalpy, which would make it appear that the critical break size of 55 percent for the four cell case is dependent on outlet enthalpy and/or core power. However, the critical break size was determined for the same four cell core power of 3535 Mwt but with a different flow (and outlet enthalpy) and found to be 55 percent also. This indicates that the critical break size is primarily a function of core power rather than outlet enthalpy.

The transient was started at time t = 0 by simulating a cold-leg crossover split (opening Junction 31). The reactor coolant system (RCS) depressurizes through the crossover break. When the intact hot leg risers (CV-15) reach 663 psia, ECCS is initiated (Figure 15.6-165). After a six second delay, the V-4 valves (J25) open on a 25 second ramp. This provides an additional outlet for RCS depressurization.

The opening of the V-4 valves (Figure 15.6-166) causes an increase in the depressurization rate of the intact and broken loop cold-leg risers (CV-14, Figure 15.6-167 and CV-45, Figure 15.6-168). The pressure spikes shown between 120 and 140 seconds in the intact cold-leg risers and to a lesser extent the broken loop cold-leg risers are caused by saturated liquid flashing into the core volumes. These are unrealistic spikes caused by desuperheating the fuel a section at a time, thus a spike as each core section is

encountered. A more realistic approach would be to analyze these events with a model incorporating a fine mesh quench front, which later generation safety analysis codes have. If a model of this type were employed, a repressurization would occur in the cold-leg riser when coolant first enters the core. This spike would be followed by a steady depressurization of the cold leg risers at the same rate as before the pressure increase.

When the cold-leg riser pressure (CV-14) drops to 306 psia, the CV-2R valves open, allowing ECCS coolant to enter the RCS. This allows ECCS injection into the core, closing any open CV-5 valves and preventing further fuel heatup. Initiation of ECCS injection terminates the transient blowdown analysis at 139 seconds.

The maximum broken loop fuel temperature is 1307°F (Figure 15.6-169) which occurs at 134 seconds. At this point, the CV-5 valve opens for a short time period and cools the fuel (Figure 15.6-170). It is not certain if this CV-5 opening (.2 seconds in length) is real or a valve model defect. Prior to the CV-5 valve opening, the broken loop fuel heatup rate was 4.39°F/second. If this heatup rate is carried out to reflood, the maximum broken loop fuel temperature would be 1327°F, well below that of the five cell 40 percent break case.

The intact loop maximum fuel temperature is 1154°F which occurs at 119 seconds (Figure 15.6-171). The intact loop CV-5 valves open from 115.7 to 120.5 seconds and from 133 seconds to reflood (Figure 15.6-172). This flow prevents any further heatup of the intact core. The amount of coolant flow through the intact loop CV-5 valves would indicate that these openings are real events, although slightly overstated due to the instant open-close valve model.

The intact and broken loop fuel surface heat fluxes (Figures 15.6-173 and 15.6-174) reflect the heatup and cooldown rates seen in the core.

Hot Channel Analysis (RELAP4) | 21

As discussed above, the worst case fuel temperature heatup transient during four steam generator cell operation is the 55 percent split cold-leg crossover break. Peak fuel temperatures in the center core of the broken loop were less than 1330°F at the time the ECCS terminated the fuel heatup transient.

As discussed in Section 15.6.5.3.1 the maximum hot channel fuel temperature for the worst case RELAP4 based analyses of five steam generator cell operation was 1570°F. The maximum RELAP4 core average fuel temperature was 1425°F which is in the β-γ phase transformation temperature plateau. The β-γ phase transformation is modeled as a high effective specific heat (a factor of 22 increase) in the temperature range of 1420 - 1430°F. The effective hot channel fuel temperature increase was approximately 250°F for the five steam generator cell case. The limiting large break LOCA is the RELAP5 100% cold leg crossover break which is described in Section 15.6.5.3.4. | 21 | 21

The hot channel fuel temperature increase for the four cell worst case will be slightly higher than the five cell case because of the lower fuel temperatures and consequently the lower heat transfer from thermal radiation.

The effective fuel temperature increase will result in hot channel fuel temperatures in the β-γ phase transformation temperature plateau of 1420 - 1430°F. The fuel temperatures are below that at which fuel-clad separation would occur.

The Reflood transient would be similar to the five cell case, in that the steam temperatures would rapidly drop to the saturation temperature of approximately 400°F. This would terminate the heatup transient. However, the hot channel fuel temperatures would remain in the β-γ temperature plateau until the fuel temperature is reduced to 1420°F. The cooldown rate would then be similar to the five cell case.

In summary, the worst case loss of coolant accident during four steam generator cell operation would be less severe than for the five cell case.

15.6.5.3.3 Cold Leg Crossover Breaks, Double-Ended and 20 Percent Split (RELAP4)

This analysis discussion is provided for historical continuity, but has been superseded by the RELAP5 based analyses described in Section 15.6.5.3.4.

These breaks use the model schematic shown in Figure 15.6-15 and are essentially the same as the 40-percent split case described above, with the primary exception of the break size. The double-ended break behaves similarly to the 40-percent split case with the exception that a slightly shorter time to reflood occurs due to the larger break area. The double-ended break has an area equivalent to 60-percent of the pipe area because a discharge coefficient of 0.6 was used for modifying the choked-flow tables.

The 20-percent split is not large enough to close the CV-5 valve (J48) for several seconds. This allows some flow through the core supplied by the broken riser which delays the heat-up transient. This flow also removes a significant amount of the initial core energy. The blowdown rate for this case is slower than for the larger breaks but the case is less severe because of the delayed temperature transient. The fuel temperature transient is interrupted again when the ECCS trips and the V-4 valves open. This causes the CV-5 valves to open momentarily, allowing subcooled fluid to enter the core. This transient is also ended when the cold-leg riser pressure drops to 306 psia.

The results of the double-ended break are shown in Figures 15.6-31 through 15.6-38, and the results of the 20-percent split are shown in Figures 15.6-39 through 15.6-47.

The similarity of the double-ended break to the 40-percent split is caused by the fact that the closing of the CV-5 valve (J48) in the broken riser in both cases causes the break to behave as a single-ended break. A discharge coefficient of 0.6 was used for the double-ended break making it equivalent to a 60-percent split. The plots for the double-ended breaks are presented, but not described, because they are similar to those presented for the

40-percent split case. The limiting large break LOCA is a RELAP5 analysis of the 100% cold leg crossover break. See Section 15.6.5.3.4. 21

The RELAP4 20-percent split behaves quite differently than the double-ended break due to the CV-5 valve (J48) response. Figures 15.6-39 and 15.6-40 show the CV-5 valve response in the broken and intact risers, respectively. Note that the CV-5 valve in the broken riser does not close for 8 seconds and the valves in the intact risers do not close for 30 seconds. This response allows a large portion of the initial core power to be removed by the coolant. This is evidenced by the fuel response shown in Figures 15.6-41 and 15.6-42. Note that the long delay in the fuel heat-up transient for both portions of the core and the temperature turn-around at 150 seconds is due to the CV-5 valves reopening. The pressure response of the cold-leg riser (CV-14) is shown in Figure 15.6-43. The long depressurization is a result of the small break area (0.27 ft^2). The transient was not extended past 170 seconds since it is clear that this is a less severe fuel temperature based on the RCS response to other breaks which show a linear decrease in the cold-leg riser pressure until the V-4 valves open, after which the depressurization rate increases. The transient is estimated to reach reflood at 200 seconds. Also, other analyses show that for small breaks such as this, the CV-5 valves open shortly after the V-4 valves. This is caused by the larger flow area of the V-4 valves versus the break. As shown in Figure 15.6-44, the V-4 valves will open at 165 seconds, further depressing the fuel temperatures due to the flow rate through the CV-5 valves.

The sequence of events for the cold-leg crossover breaks are presented in Table 15.6-20.

15.6.5.3.4 Determination of the Limiting LOCA (RELAP5)

The limiting LOCA is defined by the combination of worst single-failure type, break size, and location, that results in the highest peak cladding temperature (PCT). Discussion of the spectrum analysis of possible break sizes and locations performed to determine this limiting-case LOCA follows. In that discussion, the section of the core associated with the broken 21

crossover pipe is referred to as the "broken" or "affected" core. The other section is referred to as the "intact" or "unaffected" core.

Analysis to determine the limiting LOCA included consideration of a spectrum of simulated pipe breaks and station blackout sequences (see Table 15.6-34) is as follows.

Best-Estimate Inlet/Outlet Riser/Cold Leg Manifold Breaks and Station Blackout Analysis (RELAP5)

Best-Estimate analyses, reported in References 38, 39 and 40, evaluated results with:

- o Inlet and outlet riser break sequences at reduced (22,000 gpm) ECCS flow rate, including sensitivity studies of ECCS flow (Ref. 40)
- o Cold leg manifold breaks at full (27,000 gpm) and reduced (22,000 gpm) ECCS flow rates, including sensitivity studies of ECCS flow (Refs. 38 and 39)
- o Station blackout sequences at full and reduced ECCS flow, with ECCS flow sensitivity studies (Refs. 38, 39, 40).

Cold Leg Crossover Break Spectrum (RELAP5)

Based on results of the above analyses, and previous RELAP4 based limiting LOCA analyses reported earlier in Section 15.6.5.3, it was concluded that a split in one of the sixteen cold leg crossover pipes would again result in the worst case accident. This break location is also considered more probable due to it being not yet seismically qualified to 0.25 g, whereas the "once-through cooling" portion of the RCS is seismically qualified. See Section 15.6.4.5.2. Accordingly, a spectrum of six crossover pipe splits (no guillotine breaks), equivalent to 25%, 35%, 40%, 50%, 60%, and 100% of the pipe cross sectional area, was analyzed to identify the worst-case break (Ref. 44).

The initiating event for all of these cases was an assumed split break in a cold leg crossover pipe, upstream of the CV-5 check valve (see Figure 15.6-177). All crossover break sequences were analyzed assuming a

concurrent station blackout condition, as defined for the analyses reported in References 38 and 39, in which the following conditions existed.

- o Offsite and onsite AC power sources are not available
- o Onsite DC power sources (plant batteries) are availabl.
- o Reactor operating power was assumed to be 3780 MWt
- o Only two of three ECCS supply high-lift diesel pumps (HLDP) were assumed to start
- o ECCS capacity was based on two HLDPs operating at reduced capacity, in which flow was input by table as a function of system pressure. The ECCS flow rate used was as shown in Table 15.6-39.

For the cold leg crossover break spectrum analysis, loss of AC power was assumed to cause a reactor trip, a loss of power to the primary coolant pump motors, and primary coolant pump coastdown. Steam generator feedwater and steam flow paths were isolated, and the graphite cooling and high pressure injection systems were assumed not to function. Reactor trip for the limiting LOCA evaluation was assumed to occur on low rear riser pressure, with conservative delays. 21

From results of these calculations, it was determined that the worst case break is a 100% split of a cold leg crossover pipe, with the split located immediately upstream of the CV-5 check valve (Ref. 44). This case was then further analyzed using the Appendix K evaluation model and the Mark IV and Mark IA hot channel models (Ref. 45).

The sequence that causes the highest peak cladding temperature results from the smallest pipe break that closes the CV-5 valve and keeps it closed during the LOCA. The associated longer depressurization time increases the delay until actuation of the ECCS, which supplies coolant flow and subsequently refloods the core.

Closure of CV-5 results in almost instantaneous flow stagnation and immediate heatup of the fuel. Since coolant is not available from the normal flow

path when CV-5 is closed, core cooling is only provided by flashing to liquid upstream of the core, or by ECCS reflood.

Table 15.6-40 summarizes the results of calculations of the broken loop responses. It presents the peak temperatures and shows the response time of the CV-5 check valve and the V-4 dump valves, and the time to actuation of the ECCS. Time is in seconds after the initiation of the transient. The worst case, by a substantial margin, is seen to be the 100% split break.

The sequences for the 100% and the 40% break transients are discussed in detail in Reference 44. These sequences were chosen for detailed discussion because the 100% split is the limiting LOCA, and the 40% split calculation is representative of the 35%, 50%, and 60% splits, but was evaluated for the longest reflood time. It also provides comparison with the previously determined limiting LOCA case discussed in Section 15.6.5.3.1.

Figures 15.6-179 through 15.6-186 show key results of peak cladding temperature calculations for two of the spectrum of best-estimate crossover pipe cases analyzed. Figures 15.6-187 through 15.6-190 are plots of cold leg riser pressure for the 40% break, CV-5 check valve flow for the 40%, and 100% breaks, and broken loop ECCS flow for the 40% break case respectively.

From Table 15.6-40 it can be seen that, except for the 25% and 100% splits, the course of the transients was similar for all cases; i.e., for the 35%, 40%, 50%, and 60% splits, the break first caused CV-5 to close, followed by reopening after a delay between 63 and 84 seconds. Reopening of the valve resulted in some cooling of the fuel. Peak clad temperatures were 1320°F, 1325°F, 1410°F, and 1425°F for the 35% through 60% break sequences, respectively. Reference 44 provides comparison plots that illustrate in detail the similarities noted.

From Table 15.6-40 it can be seen that the 25% break caused CV-5 to close, but to quickly reopen at 100 seconds, and remain open for the remainder of the transient. The 100% break, on the other hand, closed CV-5 and kept it closed through the blowdown phase (Figure 15.6-189), resulting in a peak

cladding temperature of 1615°F. This was the highest PCT calculated from the spectrum study, and establishes this break as the limiting LOCA.

The 100% break transient was only analyzed to 300 seconds, but the fuel temperature had reached its peak value (Figure 15.6-179) and had started to decrease during this interval. Therefore, it was not necessary to simulate the transient beyond the early blowdown and reflood period for this analysis (Ref. 44). A more complete analysis of this limiting case is provided in the conservative Appendix K analysis reported in Reference 45 and described next.

Limiting LOCA Evaluation Model Analysis (Appendix K)

System performance and core behavior for the limiting LOCA case was calculated using RELAP5 Evaluation Model utilizing applicable requirements of 10 CFR 50, Appendix K. The model was appropriately modified to reflect the differences between the N Reactor's uranium metal fuel elements and the commercial reactor oxide fuel element designs which are specifically addressed by Appendix K. 21

N Reactor ECCS Acceptance Criteria (RELAP5)

10 CFR 50.46 specifies that each boiling and pressurized light-water nuclear power reactor, fueled with uranium oxide pellets within cylindrical Zircaloy cladding, shall be provided with an ECCS that is designed such that its calculated cooling performance subsequent to postulated LOCAs will conform to certain criteria.

10 CFR 50.46 also specifies that uranium oxide-fueled power reactor ECCS cooling performance be calculated in accordance with an acceptable evaluation model. Appendix K to 10 CFR 50, ECCS Evaluation Models, sets forth certain required and acceptable features for such evaluation models. The required features include: (1) use of the 1971 ANS standard for decay heat, with fission product decay power increased by twenty percent; (2) use of the Moody break flow model; and (3) the assumption of the worst single active failure. The remaining Appendix K features identify other analysis models and correlations that are satisfactory for use.

The use of these required and acceptable features, plus other conservative correlations and input data, has, in general, provided a substantial level of conservatism in large break LOCA ECCS performance analyses for these types of plants. Additionally, for economic reasons, certain phenomena have been conservatively modeled instead of using more realistic, but still acceptable models.

All of the 10 CFR 50.46(b) ECCS/LOCA acceptance criteria (peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling) are adopted for N Reactor analysis. To further assure that there is no cladding rupture during a LOCA, the peak cladding temperature limit for N Reactor is reduced from 2200°F to 1900°F (Ref. 41).

Results of the N Reactor fuel behavior experiments under LOCA conditions presented in Appendix 15.6A were reviewed in detail in Reference 41. The rupture temperature appropriate for current N Reactor operations is established as 1900°F. Thus, 1900°F can be conservatively considered as the minimum fuel element failure temperature, so below this temperature, no fission products should be released from an initially intact fuel element.

The PCT limit of 1900°F is also based on the need to prevent cladding failure that would expose the highly reactive metallic uranium to the LOCA steam/water environment. Such exposure would cause hydrogen generation that could result in hydrogen combustion and a loss of the confinement safety system function. Since no fuel element damage is expected at PCTs below 1900°F, detectable hydrogen generation, and distortion of core geometry are not expected. Furthermore, based on Baker-Just correlations (10 CFR 50, Appendix K), metal/water reactions below 1900°F are insignificant.

Conformance with Features of 10 CFR 50 Appendix K (RELAP5)

This section discusses the conformance of the analysis methods and assumptions used in the N Reactor LOCA reanalysis to the Appendix K required and acceptable features. In this regard, guidance was also taken from SECY-83-47

(Ref. 36) in which the NRC staff presents revised analysis methods that it finds acceptable for showing conformance to Appendix K.

Because the N Reactor fuel assembly is metallic uranium, with bonded Zircaloy-2 cladding, and the pressure tubes are horizontal rather than vertical, some of the Appendix K requirements are not directly applicable to these analyses. These exceptions, as well as the important realistic assumptions, are noted briefly in the Remarks section of Table 15.6-33.

Results of Average Channel Evaluation Model Analysis (RELAP5)

The RELAP5 Evaluation Model (EM) is derived from the two-loop system model, which represents a single broken cold leg crossover pipe, and fifteen intact crossover pipes.

1. Broken Loop -- Results of the system evaluation model analysis were not significantly different from the trends predicted by the break spectrum analysis for the 100% cold leg crossover break, in which rapid closing of the CV-5 check valve in the broken loop (Figure 15.6-189) caused immediate flow stagnation in the core and early heatup of the fuel (Figure 15.6-179). Similarly, in this analysis, the broken loop CV-5 valve closed rapidly (Figure 15.6-191), resulting in immediate flow stagnation and early heatup of the fuel. A sequence of events for the average and hot channel 100% crossover break analyses is provided in Table 15.6-41. 21

   Fuel temperature response in the upper and lower four core model regions of the broken loop are shown in Figures 15.6-192 and 15.6-193, respectively. ECCS flow began at 120 seconds, and immediately caused decreased temperatures in the third, fourth, and fifth levels. Temperature rise continued in the first and second (upper two) levels, peaking in the second level at 270 seconds and approximately 1425°F. At 280 seconds the peak cladding temperature for the transient reached 1720°F in the hottest, upper level (Ref. 45).

2. Intact Loop -- Fuel temperatures for the upper and lower four fuel regions in the intact loop are shown in Figures 15.6-194 and 15.6-195, respectively. Steam flow through the CV-5 valve began at approximately 280 seconds (Figure 15.6-196).

   Responses in the intact loop closely followed the response of the station blackout sequences discussed in detail in References 38 and 39. In those sequences, a temperature increase in the fuel occurs when steam flow through CV-5 begins. This steam flow depresses the ECCS level, but provides sufficient cooling to maintain fuel temperatures below 1200°F.

Results of Hot Channel Verification Model Analysis

Generally excellent agreement was observed between the verification and system models (Ref. 45). Results from the verification analysis, discussed in Section 15.6.5.1.2, were judged to confirm the acceptability of using the separate, more detailed pressure tube models, driven by hydrodynamic boundary conditions obtained from the system model, to perform the hot channel analyses. Additionally, an assessment of the applicability of RELAP5/MOD2 to actual plant data has been performed and has confirmed the suitability of using RELAP5/MOD2 for N Reactor Safety Analysis (Ref. 50).

Results of Hot Channel Evaluation Model Analyses

1. Hot Channel Analysis--Mark IV Fuel--Results of Mark IV hot channel analysis are shown in Figures 15.6-197 through 15.6-205. These figures illustrate results for the first and second peak power regions shown on the power profile of Figure 15.6-178.

   Plots of calculated temperatures for the first and second power peak, inner and outer fuel cylinders are shown in Figures 15.6-197 and 15.6-198, respectively. As shown in Figure 15.6-197, the peak cladding temperature for Mark IV fuel is 1835°F in the second peak power region, reached at 240 seconds into the transient. This temperature is associated with a specific tube power of 201.0 kw/ft.

Mass flow rates immediately upstream of the second peak power region are shown in Figures 15.6-199 through 15.6-201. Surface heat transfer coefficients for that region are shown in Figures 15.6-202 through 15.6-205. From the values of these coefficients it is concluded that a return to nucleate boiling did not occur (Ref. 45).

2. Hot Channel Analysis--Mark IA Fuel--Results of the Mark IA hot channel analysis are shown in Figures 15.6-206 through 15.6-211. These transients are carried past the time of temperature turnover, but are not continued as long as those for the Mark IV fuel because of the similar reflood characteristics.

   Plots of calculated temperatures for the first and second power peak, inner and outer fuel cylinders are shown in Figures 15.6-206 and 15.6-207, respectively. Temperature responses are similar to those for the Mark IV analysis, except that the peak cladding temperature was higher--1890°F. This temperature is associated with a specific tube power of 189.0 kw/ft, and is close to, but still below, the acceptable cladding limit of 1900°F. 21

   As discussed earlier, the higher peak cladding temperature in Mark IA fuel at 4700 MW vs 5000 MW in Mark IV fuel is attributed to the higher power density in the Mark IA fuel element.

   Surface heat transfer coefficients for the second peak power region are shown in Figures 15.6-208 through 15.6-211, as before. Again, from the values of these coefficients, it is evident that return to nucleate boiling did not occur.

15.6.5.3.5 Cold-Leg Riser Breaks, Double-Ended and 20 Percent Split (RELAP4)
This analysis discussion is provided for historical continuity, but has been supeseded by the RELAP5 based analyses described in Section 15.6.5.3.6.

The model schematic used for these breaks is shown in Figure 15.6-48. The transient is initiated by opening a break (J30 and 31) in the lower portion

of one of the sixteen cold-leg risers. There is reactor coolant flow into the confinement system from only one end of the break (J31) because the other end is supplied by the ECCS piping. This piping is designed to prevent the entire ECCS water from flowing out of the break of one riser. Due to the size (4") of the line between the CV-2R valve and the base of the cold-leg riser, it is possible for only 3600 gpm of the ECCS water to flow out of the break with 25,000 gpm available from the pumps. For the double-ended break, a discharge coefficient of 0.6 was used. Therefore, both breaks behave as splits of 60 percent and 20 percent, respectively. At the time of the break, the flow rate through the portion of the core supplied by the broken riser stops and then reverses due to the low pressure region at the break. The hot-leg risers are joined by flush piping at two locations (CV-46 and CV-52). This enables the coolant to flow from the cold-leg manifold (CV-11) through the intact risers (CV-12 and 13) and into the intact core. The coolant then flows into the hot-leg risers (CV-15, 16 and 17) where it is available for cooling through the flush piping (CV-46 and 52) into the affected hot-leg riser (CV-47, 48 and 49). The coolant then flows in the reverse direction through the affected core and into the broken cold-leg riser (CV-44 and 70) and then out of the break (J31). The flow reversal also causes a low pressure region in the hot-leg risers which causes the CV-3 valves (J20 and 56) to close. This forces the fluid downstream of the hot-leg manifold (CV-3 through CV-11) to expand through the cold-leg risers with some of the fluid forced through the intact core, providing cooling. When the hot-leg riser (CV-15) pressure reaches 663 psia, the ECCS is tripped and the V-4 valves (J25) open, allowing some of the RCS coolant to blowdown into the EDT.

When the intact cold-leg riser pressure (CV-14) drops to 306 psia, the CV-2R valves (J87) open, allowing ECCS coolant to enter the RCS and close the CV-5 valves (J14), forcing the ECCS coolant into the core and terminating the temperature transient. After the intact core has reflooded, the cold fluid enters the hot-leg risers and the flush piping. It then flows backwards through the portion of the core fed by the broken riser, reflooding the fuel, and then out the break.

The results for the RELAP4 based analysis of the double-ended break are shown in Figures 15.6-151 through 15.6-163, and those for the 20-percent split case are shown in Figures 15.6-49 through 15.6-56. These two cases behave quite similarly except for the slower depressurization of the 20-percent case. The pressure responses of the hot- and cold-leg risers (CV-15 and 14) are shown in Figures 15.6-151 and 15.6-49 for both cases. The flow through the intact CV-5 valves (J14) is shown in Figures 15.6-152 and 15.6-50. The long time before the valves stay closed, will delay the fuel temperature transient in the intact core. For these breaks, the flow in the core fed by the broken riser reverses due to the break location being downstream of the CV-5 valve. Figure 15.6-153 and 15.6-51 show the flow rate from the broken core into the hot-leg riser. The flow is initially positive but reverses to a large negative value and then decreases as the RCS depressurizes. 21

The reverse flow rate is adequate to cool the affected core. The fuel temperature increases do not begin until after the CV-5 valves close, stagnating the core. The temperature responses for the portion of the core fed by the broken riser are shown in Figures 15.6-154 and 15.6-52. The reverse flow is adequate to cool the fuel to below the initial temperatures. These cases are not as severe as the 40-percent cold-leg crossover split. The depressurization rates are very similar between the crossover breaks and the riser breaks of the same size, which means the times at which ECCS enters the RCS are similar. Therefore, since the fuel temperature transients are delayed in the riser breaks compared to the crossover breaks, the crossover breaks are more severe.

The sequence of events for the cold-leg riser breaks are presented in Table 15.6-21.

15.6.5.3.6 Cold Leg Riser Break (RELAP5)

General Discussion

The model schematic for this break is shown in Figures 15.6-212, 15.6-213, 15.6-214 and 15.6-215. The description of the system model and the details of the model modification are provided in Reference 40 and is virtually 21

identical to the limiting large break LOCA analysis discussed in Section 15.6.5.3.4.

In the discussion presented here, the section of the core associated with the broken inlet riser is referred to as the "affected" core, while the other section of the core is referred to as the "unaffected" core.

The inlet riser break sequence was initiated by a double-ended rupture of one inlet riser at the bottom. Most plant functions were assumed to function normally. As an exception, the high pressure injection (HPI) system and the graphite shield cooling system (GSCS) were assumed inoperable. Shortly after the rupture, the primary system depressurized rapidly. ECCS signals were generated, opening the V-3 (ECCS) and V-4 (emergency dump) valves. The core voided and fuel element heatups occurred. ECCS coolant refilled the unaffected core region, quenching the fuel elements; the peak fuel element temperature was 1240°F.

When the RCS pumps were tripped at 1000 seconds (16 minutes) the ECCS coolant flow was redistributed such that the fuel elements in the affected core quenched. The maximum fuel temperature was 1200°F in the unaffected core region and 900°F in the affected core region.

The inlet riser piping is part of the seismically qualified "once-through cooling" portion of the RCS. As a result, a break of this piping is considered less probable than the limiting large break LOCA (cold leg crossover break, currently not seismically qualified) described in Section 15.6.5.3.4.

Detailed Discussion

The postulated accident was initiated by a 200% double-ended rupture of one inlet riser pipe near the CV-2R check valve. The break included two flow paths from the primary coolant system to the containment. The first path was out the inlet riser through the break at the bottom. The second path was from the ECCS line through the broken stub via the CV-2R valve and into the containment. Because of the nature of the break, 1/16 of the core was

isolated from direct injection of the ECCS coolant. The amount of ECCS available to the remaining 15/16 of the core was also affected. The first 280 seconds of the calculations were performed with the production version of RELAP5/MOD2. After 280 seconds, the calculations were performed with the version of the code modified to simulate the phenomena of reflood in horizontal core channels. The reactor was assumed to be operating at 3870 MW, full power conditions when the rupture of the inlet riser occurred. It was assumed most systems operated as designed. As an exception, the HPI and GSCS systems were not available and the capacity of the ECCS was 22,000 gpm, a lower conservative bound on the delivery curve for two high lift diesel-driven ECCS pumps. The primary coolant pumps operated normally and were run back to 1240 rpm following the reactor trip signal. The steam generator secondary system also operated as normal by maintaining a level in the boiler region through regulation of the feedwater. At 1000 seconds a parallel sensitivity calculation was performed in which the primary coolant pumps were tripped and the ECCS capacity was increased to 24,000 gpm. 21

The timing of events for the base case inlet riser break is shown in Table 15.6-42. As stated above, the transient began by an instantaneous rupture of one inlet riser at the bottom. A reactor trip occurred at 0.5 second as a result of a low process tube flow. The reactor trip caused the steam generator bypass valves to open and a run back of the primary coolant pumps to begin. By 19 seconds the run back of the coolant pumps to 1240 rpm was complete. The first of two ECCS activation signals (low core flow) occurred at approximately 24 seconds. The second signal to activate the ECCS occurred 23 seconds later, at 47 seconds, due to low pressure in the hot leg manifold. At the activation of the second ECCS signal, the V-4 valves in the dump line and the V-3 valves in the ECCS system began to open. Because the break occurred in the inlet riser, ECCS coolant flow through the broken inlet riser CV-2R check valve began immediately, spilling limited ECCS coolant on the confinement floor.

Between 5 and 500 seconds several fuel heatups and quenches occurred as the reactor core and loops voided during the blowdown phase of the transient. At approximately 500 seconds a sustained heatup began in the upper half of

the unaffected core. At approximately 700 seconds the sustained heatup began in the upper half of the affected core. At about 1300 seconds the fuel temperature in the upper half of the unaffected core reached the first fuel phase change at 1240°F. However, at approximately 1350 seconds ECCS began to penetrate the upper half of the core and the fuel temperature turned over and declined. At 1544 seconds the base case calculation was terminated. The fuel temperature in the unaffected core was cooling down. However in the affected core the maximum fuel temperature was at 1300°F and increasing with no indication of it turning over. The "Inlet Riser Sensitivity Calculation" section below describes how tripping the primary coolant pumps rectifies this fuel heatup condition. However, before discussions of tripping the primary coolant pumps, a detailed discussion of the first 1000 seconds is presented.

Figures 15.6-216 and 15.6-217 show calculated fuel element cladding temperatures at each of the eight levels in the unaffected core. The temperatures shown are at the center of the core on the outer surface of the outer fuel element. The fuel elements in the lower half of the core experienced only small heatups at the beginning of the transient. The lower fuel element temperatures quickly declined as each of the core channels filled with the incoming ECCS coolant. The fuel elements in the top four channels experienced four heatups during the blowdown portion of the transient. The first heatup, 5 to 60 seconds, occurred due to voiding in the core as the primary system depressurized. At 47 seconds the V-4 valves opened and flashing commenced in the core inlet spacer region and the fuel temperatures declined. The next two heatups and quenches were a result of continued voiding and flashing upstream of the heated core and around the loop as the fluid approached saturation. Flashing in the loop resulted in bursts of two-phase fluid entering the core region through the CV-5 valves keeping the fuel elements cool.

The fourth heatup, which had a longer duration, began at approximately 280 seconds. Between 280 and 400 seconds steam generation in the core decreased, and condensation in the outlet riser declined, resulting in a very small core differential pressure as shown in Figure 15.6-218. As a

result, the incoming ECCS coolant filled the inlet riser in the unaffected core and began penetrating the inlet connectors and spacers as shown in Figures 15.6-219, 15.6-220, and 15.6-221. As the coolant penetrated the heated core section, the steam produced flowed into the outlet riser increasing the condensation. The outlet riser pressure decreased due to interphase condensation. As a result, the liquid in the inlet side of the core quickly flowed across the core, quenching the fuel elements as shown in Figure 15.6-216.

The final heatup in the unaffected core commenced at approximately 500 seconds. At about 500 seconds the fluid in the steam generator U-tubes approached saturation and began to flash. The steam generated by the flashing traversed through the loop and entered the unaffected core inlet riser through the CV-5 valve. The inflow of steam through the CV-5 valve kept the level in the inlet riser low as indicated by the high void fraction shown in Figure 15.6-219. At approximately 650 seconds the steam generator became a heat source to the primary system, and contributed to the persistent CV-5 flow. For approximately 800 seconds steam flowed through the upper half of the core while ECCS coolant flowed across the lower half of the core. The steam coming across the upper half of the core entrained the liquid coming up the outlet riser, carrying it over the outlet crossover and out into the outlet downcomer. At approximately 900 seconds, liquid began to pass from the unaffected core outlet downcomer to the RCS loop through the CV-3 valves. 21

At about 1000 seconds, large condensation spikes occurred in the outlet riser. The condensation was sufficient to increase the unaffected core differential pressure and draw fluid across the fourth channel. The fuel temperature in that channel declined as shown in Figure 15.6-216. As the fluid penetrated the core, steam was produced and the pressure increased. As the pressure increased, the frequency and duration of the CV-5 flow declined and the level in the inlet riser began to increase. At about 1300 seconds the CV-5 flow ceased. The liquid level in the inlet riser of the unaffected core quickly rose to the top and began to refill and quench the fuel elements as shown in Figure 15.6-216.

The affected core presents a unique situation for flooding the core during an inlet riser break. As stated above, the break occurred in the bottom of one inlet riser. The break flow came from three sources; through the CV-5 valve and into the inlet riser, back through the affected core and into the inlet riser, and through the ECCS CV-2R valve. The flow from the ECCS was limited due to the size (4 in.) of the CV-2R valve flow control leg.

The flow from the inlet riser initially was high, but declined due to inlet riser voiding as shown in Figures 15.6-222 and 15.6-223.

The ECCS coolant entering the affected core had to come through the unaffected core. Two paths were available for the coolant to enter the affected core. One path was through the flush line that connects the bottom of the outlet risers together. The other path was out the unaffected core outlet riser, into the outlet downcomer and into the V-4 valve inlet header. From the V-4 valve inlet header, the path was up the affected core outlet downcomer, into the outlet riser, across the affected core, down the inlet riser, and out the break.

Figure 15.6-224 shows the differential pressure between the unaffected and affected core outlet risers. For the first 700 seconds the differential pressure was positive. Fluid flowed from the unaffected core outlet riser to the affected core outlet riser through the flush line as shown in Figure 15.6-225. The flow through the flush line initially was high, but declined as the flush line voided. After 150 seconds the flush line became full of liquid from the ECCS coolant coming from the unaffected core. As the flush line filled the mass flow increased. However, condensation in the unaffected core outlet riser decreased the pressure locally. The differential pressure between the two core outlet risers declined (see Figure 15.6-224), and the mass flow to the affected core through the flush line also declined. By 700 seconds the affected core outlet riser was full of liquid. The head produced by the full outlet riser, in combination with the condensation in the unaffected core outlet riser, caused the differential pressure between the two cores to reverse. As a result, the flow through the flush line

reversed direction as shown in Figure 15.6-225. This condition persisted throughout the remainder of the transient.

In addition to the condensation in the unaffected core outlet riser region, the continued operation of the primary coolant pumps also affected the pressure and flow distribution in the core outlet region. After 900 seconds, the hot leg began to fill with liquid passing through the CV-3 check valves. As the hot leg filled, liquid reached the primary coolant pumps, increasing their efficiency. Between 1300 seconds and the termination of the calculation, the differential pressure across the pumps became increasingly positive as more liquid reached the pumps. As the primary coolant pumps became more efficient, the pressure in the hot leg lowered. In turn, more coolant exited the core outlet region through the CV-3 check valves. The lower hot leg pressure and the increased CV-3 valve flow affected the pressure and flow distribution in the core outlet region as mentioned above.

Figures 15.6-226 and 15.6-227 show calculated maximum fuel element cladding temperatures at each of the eight levels in the affected core. The temperatures shown are at the center of the core on the outer surface of the outer fuel element. When the break opened, the fuel temperatures increased due to the rapid depressurization and voiding in the core. The temperatures quickly turned over as flashing in the core outlet forced two-phase fluid back through the core and out the break at the bottom of the inlet riser. The primary system depressurization rate increased when the V-4 valves opened. Fluid in the affected core outlet downcomer reversed direction and began to flow out the dump line. The pressure in the affected core outlet riser decreased, resulting in a flow stagnation across the affected core due to a low core differential pressure. The flow stagnation resulted in the fuel temperature excursion between 100 and 170 seconds. As the depressurization rate slowed, the affected core differential pressure increased, the flow stagnation ceased, and the fuel temperatures turned over. 21

Between 200 and 500 seconds coolant coming from the unaffected core through the flush line kept the lower half of the affected core outlet riser liquid full. The fuel temperatures in the lower half of the core continued to

decline, following the coolant temperature. However, not enough coolant was entering the affected core outlet riser to keep the upper half of the outlet riser full during this time period. As a result, the fuel in the upper half of the affected core began to heat up as shown in Figure 15.6-226. At approximately 400 seconds the outlet downcomer in both the unaffected and affected core became liquid full. Liquid from the affected core outlet downcomer began to flow into the outlet riser, the riser became full, and liquid began to penetrate the core. The fuel temperatures in the upper half of the core declined.

At about 500 seconds in the lower half of the core, and 700 seconds in the upper half of the core, the fuel elements began a slow heatup. By 1000 seconds, the fuel temperature in the upper half of the core was increasing faster than in the lower half. The fuel temperature excursion was a result of flow stagnation due to the affected core differential pressure being nearly equal to zero.

The low affected core differential pressure, and thus the fuel element temperature excursion occurred for two reasons. First, condensation in the unaffected core outlet riser lowered the pressure in that region, which also lowered the pressure in the affected core outlet riser. As a result, coolant continued to flow into the affected core outlet riser from the outlet downcomer. However, the differential pressure between the affected and unaffected core outlet risers was greater than the differential pressure across the affected core. Therefore, a majority of the coolant flowed into the unaffected core outlet riser, rather than flow across the affected core. Thus, the fuel temperatures increased in the affected core.

Second, the continued operation of the primary coolant pumps tended to draw coolant out of the core region into the loop. After 900 seconds, the loop began to fill with liquid. The primary coolant pumps became more efficient and the pressure in the hot leg lowered. The differential pressure across the CV-3 valve increased and the flow through the valve increased. The increased CV-3 valve flow resulted in a flow and pressure redistribution in the core outlet region, and flow to the affected core was limited.

The calculation was terminated at 1525 seconds. At this time, the fuel element temperatures in the unaffected core had peaked at 1240°F and were declining. The fuel element temperatures in the affected core were at 1260°F and continuing to increase. ECCS coolant was entering the system through the unaffected core inlet riser, traversing the core to the outlet riser, and exiting through the outlet downcomer. From there, the coolant either entered the loop through the CV-3 valve, or exited through the dump line. The coolant entering the loop, flowed around the loop and exited through the affected core CV-5 valve, into the inlet riser and out the break. The coolant flowing through the dump line either continued through the dump line via the V-4 valves, or entered the affected core outlet downcomer. The pressure distribution was such that the coolant entered the outlet riser from the affected core outlet downcomer, flowed down the riser, exited through the flush line, and into the unaffected core outlet riser.

In summary, the base case inlet riser break calculation indicated that, with primary coolant pumps on and a 22,000 gpm ECCS capacity, fuel element heatups in the core were experienced. The heatups occurred mostly in the upper half of the core. ECCS coolant was sufficient to prevent fuel damage in the unaffected core. However, cooling in the affected core was insufficient to turn the fuel temperatures over; they continued to increase with no indication of fuel quench. The maximum calculated temperature in the unaffected core occurred in the second channel from the top and was 1240°F. The temperature in the affected core was at 1260°F, and continuing to increase.

Inlet Riser Break Sensitivity Calculation

At the termination of the base case inlet riser break calculation (1525 seconds), ECCS coolant was entering the primary coolant system at a rate of 2800 lbm/s (22,000 gpm nominal). The primary coolant loop was nearly liquid-full, and the primary coolant pumps were operating at 1240 rpm (reduced from 3600 rpm at the time of reactor trip). ECCS coolant was flowing throughout the unaffected core region and the fuel element temperatures were quenching. However, ECCS coolant was not sufficient to penetrate the affected core region. The fuel element temperatures in that region were still increasing, with no indication of quenching.

The sensitivity calculation was started from the base case calculation at 1000 seconds. At this time, the base case calculation was near the end of the blowdown phase of the transient. In this calculation, the ECCS capacity was increased from 22,000 to 24,000 gpm to be representative of two high-lift diesel-driven ECCS pumps with a degraded pump curve and maximum line losses. Also, the primary coolant pumps were tripped and allowed to coast down.

The RELAP5-calculated sequence of events for the sensitivity calculation is shown in Table 15.6-43. Events prior to 1000 seconds duplicate those of the base case transient previously discussed. Although the events up to 1000 seconds are duplicated in Table 15.6-43, a discussion of this time period is not repeated here. At 1000 seconds, the ECCS capacity was increased from 22,000 to 24,000 gpm, and the primary coolant pumps were tripped.

The ECCS mass flow rate to the unaffected core and out the break is shown in Figure 15.6-228. The step change in the ECCS flow is noted at 1000 seconds as the ECCS capacity was increased from 22,000 to 24,000 gpm. As noted in the base case calculation the ECCS responded to changes in the unaffected core inlet riser pressure.

The RCS pumps had been operating at 1240 rpm since the initial runback at 0.5 seconds. At 1000 seconds the operators tripped all the pumps. By 1120 seconds the pump impellers were calculated to stop rotating.

Figure 15.6-229 shows the calculated fuel temperatures for the upper four channels in the unaffected core. The temperatures shown are at the center of the core on the outer surface of the outer fuel element. As a result of the increased ECCS capacity, the fuel temperatures declined earlier than in the base case calculation (see Figure 15.6-216). Condensation in the outlet riser, coupled with the higher ECCS mass flow rate, allowed the fuel elements to cool faster than in the base case transient. From about 1250 seconds, the fuel temperature in the top channel slowly declined. At approximately 2260 seconds, a small fuel temperature excursion occurred. However, by

2580 seconds, ECCS coolant penetration in the top unaffected core channel resulted in a rapid decline of the fuel temperature.

Figures 15.6-230 and 15.6-231 show the calculated fuel temperatures for the eight channels in the affected core. The temperatures shown are at the center of the core on the outer surface of the outer fuel element. Shortly after 1000 seconds, flow stagnation across the affected core occurred and the fuel temperature increased. The cause of the flow stagnation was similar to that described in the base case discussion. Condensation occurring in the unaffected core outlet riser reduced the pressure. The lower pressure caused the affected core differential pressure to decline. At approximately 1170 seconds, the pressure increased due to steam generation in the core and the core differential pressure increased. The fuel temperatures decreased momentarily, then increased due to another decrease in the core differential pressure from condensation. Around 1300 seconds, the unaffected core outlet riser was nearly liquid-full reducing the condensation rate. As the condensation ceased, the pressure increased slightly resulting in a positive core differential pressure on the affected side. In addition to limiting the condensation, the effects of tripping the primary coolant pumps affected the pressure and coolant flow distribution in the core outlet region as discussed in the previous section. With the pumps tripped, the flow out the CV-3 valve was limited, thus redistributing the coolant elsewhere. Liquid traversed the affected core and the fuel temperatures began to decline. 21

In summary, it was shown in the sensitivity calculation that increasing the capacity of the ECCS from 22,000 to 24,000 gpm and tripping the primary coolant pumps was sufficient to completely quench the fuel elements in both the affected and unaffected cores. A maximum fuel element temperature of 1200°F was calculated to occur in the unaffected core. A maximum fuel element temperature of 900°F was calculated to occur in the affected core. Tripping the primary coolant pumps allowed a redistribution of the pressure and flow in the core outlet region. This redistribution aided in quenching the fuel elements in the affected core region.

Emergency procedures have been revised to require that plant operators trip the primary coolant pumps early into a large break LOCA to ensure appropriate margin to fuel peak cladding temperatures as demonstrated by this analysis.

15.6.5.3.7 Cold-Leg Manifold Breaks, Double-Ended and 100 Percent Split (RELAP4)

This analysis discussion is provided for historical continuity, but has been superseded by the RELAP5 based analyses described in Section 15.6.5.3.8.

The model schematic for these breaks is shown in Figure 15.6-57. The transient is initiated with a break (J30, 31) in the cold-leg manifold (CV-11). For both cases, the breaks are large enough to close the CV-5 valves (J14) in each riser, stagnating the core. The reactor scrams, and the RCS pumps begin their coastdown at 2 seconds. Hot-leg fluid flashes following the break, forcing first cold-leg and then hot-leg fluid out of the break. This blowdown continues until the ECCS trip signal is received and the V-4 valves (J25) open. The transient is ended when the cold-leg risers (CV-14) depressurize far enough to permit ECCS fluid to enter the RCS, commencing the reflood process.

Figures 15.6-58 through 15.6-63 show the results of the RELAP4 based analysis of the double-ended break with a discharge coefficient of 1.0, and Figures 15.6-64 through 15.6-69 show the results of the 100-percent split. The results for the double-ended case with a coefficient of 0.6 are shown in Section 15.6.6.2 for the hypothetical case of no ECCS being provided. Reflood would be reached at 43 seconds if the ECCS is available. The depressurization rates of the cold-leg risers (CV-14) for both breaks are shown in Figures 16.5-58 and 15.6-64. The double-ended case has a faster blowdown because of the large break area. For both cases, the CV-5 valves (J14) close immediately and remain closed throughout the transient, stagnating the core. The opening of the V-4 valves (Figure 15.6-59 and 15.6-65) does not have a large effect on the depressurization rate because of the large break sizes and the slow opening time (25 sec) of the valves. The fuel temperature transients are shown in Figure 15.6-60 and 15.6-66. The dips in the curves at 40 seconds are caused by code instabilities and have been

corrected as shown. These cases have the same fuel temperature transient as the affected core in the RELAP4 based analysis of the 40% cold-crossover break because the core is stagnant throughout the transient. However, because the depressurization rate is greater in the cold-leg manifold cases, the fuel temperatures are not as great at the end of blowdown. 21

The sequence of events for the RELAP4 based analysis of the cold-leg manifold breaks is presented in Table 15.6-22. 21

15.6.5.3.8 Cold Leg Manifold Break, Double-End Guillotine and Station Blackout (RELAP5)

The two accident sequences documented in this section are a double-end guillotine cold leg manifold break and a station blackout. The analyses reported here initially assumed a very conservative bound emergency core cooling system (ECCS) capacity (both sequences) and a conservative blowdown capacity (the station blackout sequence). Sensitivity calculations to the lower bound ECCS capacity cases were performed to determine the effect of increasing the ECCS capacity to a more realistic lower-bound value demonstrated by ECCS testing.

The calculations were performed using the RELAP5/MOD2 Version 36.06 computer code with modifications specifically designed to support modeling of the horizontal reflood phenomena. RELAP5 is a best-estimate, two-phase, two-component, nonhomogeneous, nonequilibrium, thermal-hydraulic, transient code. The specific modifications designed to support the horizontal core reflood include: horizontal water packing model, limiting the interphase heat transfer coefficient, and defining the interphase heat transfer coefficient as a function of the flow regime and the vapor velocity. 21

General Discussion - Cold Leg Manifold Break

The cold leg manifold break was initiated by a double-ended guillotine break of the cold leg manifold. The assumptions to the scenario follow. Power to the primary coolant pumps was interrupted at the time of the break and the graphite cooling system was inoperable. All other plant safety features functioned as designed. The emergency core cooling system (ECCS) was

functional, but with a lower bound on the ECCS capacity. After the rupture, the primary pressure rapidly decreased. As the pressure and the primary coolant mass flow decreased, the ECCS signal was activated and the emergency dump valves opened. During the blowdown phase of the transient, a fuel temperature excursion occurred; the maximum fuel temperature was 892°F. A second and prolonged fuel temperature excursion began early during the refill phase of the transient and persisted throughout the calculation. The maximum fuel temperature during this period was 1300°F. Near the end of the calculation the ECCS capacity was increased by about 10%, well within the demonstrated capacity of 2 ECCS pumps (Ref. 48). The increased ECCS capacity immediately cooled the fuel.

Detailed Discussion - Cold Leg Manifold Break

The model schematic for this break is shown in Figures 15.6-232, 15.6-233, 15.6-234, 15.6-235 and 15.6-236, with ECCS pump flow curves shown on Figure 15.6-237. The description of the system model and the details of the model modifications are provided in Reference 39 and the model is virtually identical to the limiting large break LOCA analysis described in Section 15.6.5.3.4.

The results presented are a RELAP5 best estimate simulation of plant response during the sequence. The initiating event for this transient was a 200% double-ended rupture of the cold leg manifold. The steady state conditions from which the transient was initiated are presented in Table 15.6-44. Figure 15.6-236 shows the modeling scheme used to represent the manifold break. The break was assumed to occur in the tie line connecting the two cold leg manifold halves. Each half was connected to the primary coolant pump discharge line and the core inlet piping, maintaining the communication between the two halves of the manifold. Both the tie line and the manifold are 24-in. diameter pipes.

Power to the primary coolant pumps was assumed to be interrupted at the initiation of the break, allowing the pumps to coast down completely. Also, the graphite cooling system was assumed inoperable. All other systems were assumed to operate as designed (i.e., the HPI system responds to the

pressurizer level response, ECCS availability is at capacity of 2 HLDP at reduced flow curve of 22,000 gpm, steam generator secondary responds to maintain secondary side liquid level, etc.).

The results of the revised RELAP5 calculation (Ref. 39) are compared to the results of the original RELAP5 cold leg manifold break calculation (Ref. 38) and to the results of two sensitivity calculations (Ref. 39). The first sensitivity calculation forced the check valve between the cold leg manifold and the core inlet riser, CV-5, to remain closed after the break was initiated. The second sensitivity calculation used an increased ECCS capability (24,000 gpm) (Figure 15.6-237). Because the overall response of the original cold leg manifold break calculation has been fully discussed in Reference 38, this discussion focuses on the analysis of the revised calculation, on the differences between the revised and original calculations, and in particular, the differences in the core responses.

21

In the original RELAP5 calculation (Ref. 38), two fuel temperature excursions were calculated. The first occurred during the blowdown phase of the transient while the second occurred early during the refill phase. The original calculation was stopped at 1500 seconds after the core was refilled.

The first 130 seconds of the revised calculation was essentially the same as the original RELAP5 calculation. The early events, the ECCS, dump valve, and scram signals, occurred virtually at the same time. A comparison of the sequence of events is shown in Table 15.6-45. One exception was the time that the ECCS injection actually began. In the original calculation, the ECCS injection started at 21 seconds. In the revised RELAP5 calculation, the ECCS injection started at 29 seconds. The 8 second difference was due in part to the control mechanism used in implementing the revised ECCS curve. Because of pressure instabilities encountered during the early injection of the ECCS, a 3 second delay was imposed on the pressure used to drive the ECCS injection. The remaining 5 seconds was due to the different pump shut off heads associated with the different ECCS capabilities. The pump shut off head representative of the three high lift ECCS pumps capability was 343 psi, while the 22,000 gpm capability pump shut off head was 288 psi. In actual

practice the control system will accelerate to full speed in 25 seconds. Technical Specifications require the pumps to accelerate approximately to full speed within 40 seconds, however, this does not significantly effect the reflood calculations.

Other exceptions between the original (Ref. 38) and revised RELAP5 calculations (Ref. 39) were events related to the change in the ECCS flow rate. In the model used for the original RELAP5 calculation, the ECCS capacity was modelled as representative of the three high lift pumps, while the revised calculation model used a lower conservative bound ECCS capacity of 22,000 gpm. As can be seen in Figure 15.6-238, which compares the two calculated ECCS flow rates, the assumed flow for the three ECCS high lift pumps was approximately 50% greater than the ECCS flow rate of the 22,000 gpm calculation. The terms 22,000 and 24,000 gpm are nominal rated values of the different ECCS capabilities and are used as labels for the different capabilities. In reality the delivery flow for the three high lift diesel pumps is approximately 27,000 gpm. As a result of the greater ECCS flow, the lower core channels filled earlier in the original RELAP5 calculation than in the revised calculation.

The initial fuel temperature excursions in both RELAP5 calculations were essentially the same. Comparisons of the outer cladding surface temperatures of the outer fuel element ("fuel temperature") of both calculations are shown in Figures 15.6-239 and 15.6-240. Figure 15.6-239 compares the center cell cladding temperatures in the seventh core channel (second lowest core channel); Figure 15.6-240 compares the center cell cladding temperatures in the second core channel (second highest core channel). As seen in these two figures, the fuel temperatures essentially overlaid, the only difference was that the fuel temperatures started to decrease slightly earlier in the original RELAP5 calculation. After the fuel quenched, at 60 seconds, the effect of the greater ECCS flow rate in the original RELAP5 calculation was seen in the lower core channels as the fuel temperatures decreased below the local saturation temperature while the revised RELAP5 calculation fuel temperatures remained at the local saturation temperature. The second fuel temperature excursion started at 95 seconds in both calculations. The fuel

temperature responses overlaid until 130 seconds after which the fuel temperatures in the original calculation decreased while the fuel temperatures in the revised RELAP5 calculation continued to increase. This difference in the fuel temperature responses was a direct result of the different ECCS flow rates.

The remaining discussion focuses on the revised RELAP5 calculation (Ref. 39).

By 70 seconds, ECCS flow was established in the lower two core channels, namely, channels 7 and 8. By 145 seconds, the core inlet riser was liquid full, and a quasi-steady core flow pattern was emerging. This flow pattern, seen throughout the remainder of the calculation, was characterized by liquid-full core inlet and outlet risers with the majority of the ECCS flow passing through the lower core channels. After 145 seconds, when the core inlet riser became liquid-full, liquid started penetrating the heated core volumes. The effect of the penetration was to change the heat transfer regimes in the core and cause the fuel temperature responses to change. As seen in Figure 15.6-241, at 145 seconds, the rate at which the fuel temperatures were increasing in the upper two core channels decreased, and the fuel temperatures in all of the remaining core channels stopped increasing and started decreasing. (The fuel in the lower channels later quenching.) ECCS flow was established in channel 6 by 210 seconds. By 250 seconds, ECCS flow was established in channel 5, the core outlet riser was nearly liquid full, and the quasi-steady core flow pattern was established. 21

Check valve CV-5 opened at 270 seconds allowing appreciable steam flow into the inlet riser until 480 seconds. The CV-5 valve opened because the pressure at the top of the inlet riser dropped below the pressure upstream of the valve in the piping between the break and the check valve. The pressure response was due to the ECCS flow reaching the hot leg downcomer. The cooler ECCS liquid condensed the vapor in the downcomer, and significantly reduced the pressure in the downcomer and surrounding volumes. The low, hot leg downcomer pressure caused three major events to occur. First, the CV-5 downstream pressure dropped below the upstream pressure, and the check valve opened. Second, the downcomer pressure dropped below the dump tank pressure,

and the dump line flow temporarily reversed. And third, the fluid in the lowest core outlet riser volume flashed, causing a sudden increase in the fluid mass flow rate up the outlet riser. The effect of these events was that the cooling in the pressure tubes increased with the increased flow through those tubes. Consequently, at approximately 280 seconds, the fuel temperatures in the upper two core channels stopped increasing and started decreasing. The fuel temperatures in the lower 5 core channels, however, were only slightly affected; those fuel temperatures decreased at a faster rate, but quickly stabilized. The fuel temperature in the third core channel was unaffected. After approximately 80 seconds, the upper two core channel fuel temperatures started increasing again.

Although the increased steam flow between 270 seconds and 430 seconds cooled the top two core channels, the increased flow had no significant effect on the other system responses. In addition, the flow through the CV-5 valve had little effect on the long term fuel temperature responses of the upper two core channels. The effect of the CV-5 valve steam flow was analyzed by performing a sensitivity calculation in which the CV-5 valve was forced closed after the initiation of the break. By 500 seconds, as shown in Figure 15.6-242, the fuel temperatures in the top two core channels from the revised calculation were approaching the fuel temperatures from the sensitivity calculation. Also as seen in Figure 15.6-243, the fuel temperatures in core channels 3 and 4 from the revised calculation virtually overlaid the fuel temperatures from the sensitivity calculation. The only other difference between the two calculations was the inlet riser voiding. The steam flow through the CV-5 valve caused the top of the inlet riser to void. The voiding in the top of the inlet riser, however, had no effect on the fuel temperature response in the core. And by 400 seconds, the core inlet riser was again liquid full. As illustrated by Figures 15.6-242 and 15.6-243, the CV-5 valve steam flow had no long term effect of the transient response.

At 310 seconds, the fuel temperatures in upper core channels 3 and 4 started increasing. The increase in the temperature was caused by the decrease in the heat transfer coefficient due to the increase in voiding in those core

channels. At 500 seconds, the fuel temperatures in the top two core channels stopped increasing and started decreasing. The fluid conditions in the top two core channels had not significantly changed. These fuel temperatures decreased because the core decay heat had decreased enough that the heat removal was now generally greater than the heat generated, and the fuel temperatures consequently decreased. Although the fluid conditions in the third and forth core channels were similar to those in the upper two channels, the fluid velocities in channels 3 and 4 were on the average lower. Consequently, the heat removal in those channels was less, and the fuel temperatures in those channels continued to increase. In addition, the fuel temperature in channel 4 was fluctuating. The fluctuations were caused by the void fraction fluctuating which in turn caused the heat transfer coefficient to fluctuate. At 580 seconds, the voiding in channel 4 on the average increased, causing the heat transfer coefficient to generally decrease which in turn caused the fuel temperature to generally increase. 21

By 600 seconds, channel 3 fuel temperatures increased above both channel 1 and 2 fuel temperatures. By 920 seconds, the channel 3 fuel temperature had reached the first phase change temperature (1242°F). During this time, channel 1 and 2 fuel temperatures were slowly fluctuating as the balance between decay heat and heat removed fluctuated, responding to the particular combination of fluid conditions and fluid velocities. By 990 seconds, the fourth channel fuel temperature increased above the fuel temperature in channels 1 and 2. The core flow, during this period, was in the quasi-steady flow pattern discussed earlier, i.e., both core inlet and outlet risers were liquid-full, and the ECCS flow was being carried in the bottom four core channels.

After approximately 1000 seconds, liquid was gradually penetrating the upper four core channels. The void fraction and the fuel temperatures in the first cells of the core volumes representing the heated sections of the process tubes were both decreasing. The effect of the increased liquid penetration was that the fuel in core channels 3 and 4 started to cool. At 1100 seconds, the fuel temperatures peaked in core channels 3 and 4, reaching at maximum fuel temperature of 1300°F in core channel 3. The void fraction

in both core channels decreased slightly and the average fluid velocities increased. The combined effect was to increase the heat transfer coefficients which caused the fuel temperatures to decrease. At 1250 seconds, the fourth core channel fuel temperature decreased below the fuel temperatures in core channels 1 and 2. At 1350 seconds, because of the decreasing void fraction, channel 4 fuel temperature started to significantly decrease nearly reaching the local saturation temperature by 1450 seconds. As can be seen in the Figure 15.6-241, however, the fuel in channel 4 did not quench. The ECCS flow had not been fully established yet in channel 4, and some voids were still present. At 1680 seconds, channel 3 fuel temperature had decreased to the fuel temperature in core channels 1 and 2. As shown in Figure 15.6-241, channel 3 fuel temperature asymptotically approached the fuel temperatures in the upper two channels.

At 1765 seconds, just prior to increasing the ECCS flow, the fuel temperatures and the pressure tube wall temperatures in the first cells of the heated core volumes were gradually decreasing, and liquid was slowly penetrating the upper core channels. Although fuel temperatures in the center cells were not decreasing, the top three core channel fuel temperatures were stabilized. The fuel temperature behavior in channel 4 was too erratic to make a judgement as to whether the fuel was heating or cooling.

At 1765 seconds, the ECCS flow was increased by about 10%. The fuel temperature in the top four channels decreased. By 1850 seconds, the fuel in core channel 4 quenched, and the fuel temperature in channel 3 was significantly decreasing. The fuel temperatures in core channels 1 and 2 also decreased until 1850 seconds, at which time however, both fuel temperatures stabilized. The fuel temperature in channel 3 continued to decrease until about 1900 seconds then stabilized about 50°F above the local saturation temperature.

At 1950 seconds, ECCS flow was established in core channel 4. The ECCS flow caused the fuel temperature to decrease below the local saturation temperatur to about 150°F. At 2100 seconds, both core channel 2 and 3 fuel temperatures started to decrease, and by 2150 seconds, core channel 3 fuel had quenched.

The calculation failed at 2150 seconds because of the high energy removal rates due to the ECCS flow penetrating the hot core channels. This problem has been circumvented in the other analyses (Ref. 45) by increasing the number of heated core volume cells from 3 to 15. The revised calculations were not restarted; the feeling being that the information to be gained by continuing the calculation would not be worth the cost of increasing the number of cells in the pertinent core channel.

In summary, the revised calculation with the lower conservative bound ECCS capacity (22,000 gpm) had a prolonged fuel temperature excursion during the refill phase of the calculation whereas the original calculation did not. The maximum fuel cladding temperature was 1300°F compared to 910°F in the original calculation and occurred during the refill phase of the calculation instead of the blowdown phase as in the original calculation. The flow through the CV-5 valve early in the calculation had only minor effects on the overall response, providing some cooling to the fuel in the upper two core channels. The increased ECCS flow (24,000 gpm), well within demonstrated 2 HLDP capacity (Ref. 48), had an immediate effect cooling the fuel in upper core channels and later quenching the fuel in core channels 3 and 4 with the fuel temperature in channel 2 rapidly decreasing at the end of the calculations. 21

General Discussion - Station Blackout Event (RELAP5)

The term "station blackout" refers to a situation where all offsite and onsite AC power sources are unavailable. During a station blackout, onsite DC power sources (plant batteries) are available. The reactor was assumed to be operating at 3870 MWt, full power conditions when a loss of offsite and onsite AC power event occurred. In the N Reactor, the emergency core coolant system (ECCS) pumps are mechanically driven by dedicated diesel engines. These engines were assumed to start as designed, and therefore ECCS is available. The loss of AC power resulted in a reactor trip. The primary coolant pumps were also tripped. Steam generator feedwater and steam flow paths were isolated. The graphite cooling and high pressure injection systems was assumed to cease functioning.

A base case N Reactor station blackout sequence has been previously analyzed (Ref. 38). The base case sequence assumed availability an ECCS capacity representative of the three high lift diesel-driven ECCS pumps and a full blowdown system (eight V-4 valves) capacity. This section presents an analysis of a revised station blackout sequence including an evaluation of the sensitivity of results to the ECCS flow rate.

The new sequence contains the same best-estimate assumptions used in the base case with two exceptions: reduced ECCS and blowdown system capacities. The new sequence assumed a very conservative lower bound (22,000 gpm) ECCS capacity and one half of the blowdown system capacity (4 V-4 valves). Approximately an hour of the new sequence was simulated with these assumptions. At that time, the sensitivity of the results to the ECCS flow rate was evaluated by increasing the ECCS capacity to 24,000 gpm, which is well within the demonstrated capacity of 2 ECCS High-Lift Diesel Pumps (Ref. 48).

Detailed Discussion - Station Blackout Event (RELAP5)

The model schematic for this event is the same as those used in the Cold Leg Manifold Break Event (Figures 15.6-232, 15.6-233, 15.6-234, 15.6-235, and 15.6-236, with ECCS pump flow curves shown in Figure 15.6-237).

The description of the system model and the details of the model modifications are provided in Reference 39 and the model is virtually identical to the limiting large break LOCA analysis described in Section 15.6.5.3.4.

A comparison of the RELAP5-calculated sequence of events for the original RELAP5 (base case) and revised station blackout transients is shown in Table 15.6-46. The ECCS signal was generated at 66 seconds, causing the V-4 valves to open. This signal resulted from coincident low flow (core inlet flow less than 3045 lbm/s with a 12 second delay) and low primary coolant pump speed (speed less than 650 rpm with a 1 second delay) conditions. The rotors of the primary coolant pumps coasted to a stop at 160 seconds.

When the V-4 valves opened, a general depressurization of the primary coolant system began. The calculated V-4 valve mass flow rate is shown in Figure 15.6-244 and the calculated inlet riser pressure response is shown in Figure 15.6-245. The early decline in pressure resulted from the reactor trip and loss of primary coolant pump head during pump coastdown. As would be expected with only 4 V-4 valves open, the total V-4 valve mass flow rate was less, and the primary system depressurization was slower than was calculated when all 8 V-4 valves opened in the base case (Ref. 38).

As a result of the depressurization of the primary coolant system, significant voiding occurred in the upper regions of the reactor core. Fuel elements in the upper four core channels heated up, but there was sufficient liquid in the lower four core channels to prevent fuel element heatup there. Figure 15.6-246 shows the upper core fuel temperatures and Figure 15.6-247 shows the lower core fuel temperatures. The temperatures shown in these figures are located on the outside surface of the cladding of the outer fuel elements at mid-channel. These fuel element temperatures are the highest calculated; the trends for other locations on the fuel elements are comparable. Inner fuel element temperatures were lower because of the lower fractions of power generated in the inner fuel elements. Axially, the power fraction is highest in the centers of the fuel channels, therefore the highest temperatures occurred there. 21

As shown in Figure 15.6-246, from 125 seconds to 257 seconds, minor fuel element temperature excursions occurred in the upper core regions. In the base case calculation, multiple blowdown fuel element heatups were calculated over a longer period (from 120 to 470 seconds). In this revised station blackout sequence, the primary system depressurization was slower, reducing the potential for blowdown fuel element heatup. Reference 38 provides a discussion regarding the periodic fuel element heatup and quenching during the blowdown portion of the station blackout transient sequence.

The ECCS flow began at 464 seconds when the core inlet riser pressure fell below 288 psia, the shutoff head of the ECC pumps at the assumed lower bound ECCS flow. The ECCS flow is shown in Figure 15.6-248. The flow increase at

3700 seconds was performed to evaluate the sensitivity of results to the ECCS flow rate and will be discussed later.

At 700 seconds the reflood fuel element heatup began in the upper five channels of the core. By this time the depressurization and flashing within the primary coolant system was nearly complete and the primary coolant loop was mostly voided.

At 860 seconds the computer code was changed from RELAP5/MOD2, Version 36.05 to the modified code version to better simulate the phenomena in the horizontal pressure tubes during reflooding.

When reflooding in the upper core began, the lower three core channels were liquid-filled by ECCS penetration during the blowdown period. The reflood process for the upper core was lengthy due to persistent steam flow through the CV-5 valve. An abbreviated discussion of these effects is presented here; a more complete discussion is found in Reference 38.

ECCS liquid is injected at the bottoms of the core inlet risers. ECCS tended to flow through the already-filled lower core channels rather than rise to the inlets of the upper core channels. In the outlet riser, liquid was entrained by steam from the upper core regions, and the resultant mixture was carried out of the core to the dump tank. The ECCS injection flow was thus effectively bypassed around the voided upper core regions, and was lost from the core.

Because the only pressure relief for the primary coolant system was through the V-4 valves to the dump tank, any overpressure in the primary coolant loop was relieved by steam flow through the CV-5 valves into the top of the core inlet risers. This steam flow: (1) was insufficient to quench fuel in the upper core regions, (2) tended to depress the inlet riser level and thus retard reflooding of upper core channels, (3) increased the carryout of liquid from the outlet riser as discussed in the previous paragraph, and, (4) was sufficient to provide adequate core cooling.

Steam flow through the CV-5 valve was persistent because the primary coolant loop was refilling. Liquid flow through the CV-3 valves (normal core outlet) into the hot legs was calculated as a result of core pressure fluctuations. As the coolant loops refilled, steam was produced by heat addition from the steam generator secondaries and from the piping walls. The steam produced flowed around the coolant loop and into the core through the CV-5 valves. The calculated CV-5 (normal core inlet) and CV-3 (normal core outlet) valve flow rates are shown in Figures 15.6-249 and 15.6-250, respectively. The calculated responses for these parameters were comparable to those observed in the base case analysis (Ref.38).

The calculated steam generator power is shown in Figure 15.6-251. Steam generator heat addition to the primary coolant system began at 520 seconds. As was true in the base case calculation, refilling the steam generator U-tubes was a lengthy process because the steam generator secondaries remained at pressure and temperature when feedwater and steam flows were isolated. The heat content of the steam generator secondary liquid provided a heat source to the primary coolant loop that caused vaporization of liquid entering the U-tubes. The steam generator secondary pressure, compared with the pressurizer pressure in Figure 15.6-252, initially increased as a result of the loss of steam and feedwater functions. After 520 seconds, the secondary pressure declined due to cooling by heat removal to the primary coolant system. Steam generator heat transfer phenomena in the revised station blackout calculation are comparable to those in the base case (Ref. 38). 21

The normalized core inlet riser level response is shown in Figure 15.6-253. The inlet riser was about 35% full when reflood commenced and the level increased to 40 to 50% full during the reflood period. Comparing Figures 15.6-249 and 15.6-253, periods of high CV-5 valve steam flow corresponded to periods of low inlet riser level. The inlet riser did not refill completely until liquid flow from the CV-5 valve began at about 2900 seconds. The inlet riser level response in the revised station blackout sequence (Ref. 39) was similar to that in the base case (Ref. 38). The lower ECCS rate in the revised sequence caused the level to be about 20% below that in the base case. This difference resulted in reflood fuel element

heatup in the top five core channels in the revised calculation, as compared to the top four channels in the base case.

The void fraction in the outlet riser at the elevation of the top core channel is shown in Figure 15.6-254. This figure shows the long period, from about 800 seconds to 2900 seconds, when liquid in the outlet riser was carried out the core exit. During this period the void fraction in the upper half of the outlet riser was 50 to 80%. The void fraction response in the revised (Ref. 39) station blackout calculation was similar to that in the base case calculation (Ref. 38) except that the outlet riser void fraction was higher than in the base case. This higher void fraction is attributed to the lower ECCS injection rate in the revised calculation.

During the reflood period, fuel in the upper three core channels was cooled by steam cooling for an extended period of time, as shown in Figure 15.6-246. This behavior compares with two channels experiencing extended heatup in the base case calculation. During the heatup period, the steam flowing through the CV-5 valve passed through the upper core channels, limiting the heatup of the fuel elements there. The peak fuel element cladding temperature, 1244°F, was reached at 1320 seconds on the outside surface of the outer fuel element in the centers of the upper two core channels. Fuel element temperature response for the revised station blackout calculation (Ref. 39) was generally comparable to that in the base case calculation (Ref. 38).

As the transient proceeded, the steam generator secondary fluid was cooled and the secondary pressure declined, as shown in Figure 15.6-252. The steam generator heat flow to the primary coolant diminished (as shown in Figure 15.6-251) and this reduced the CV-5 valve steam flow (Figure 15.6-248). With the CV-5 steam flow diminishing, the core inlet riser level recovered, as shown in Figure 15.6-253. By about 2900 seconds, some liquid flow around the coolant loop to the core was established. The combination of liquid flow through the CV-5 valve and a rising core inlet riser level quenched the fuel elements in the upper core channels.

Because of the coarseness of the axial nodalization within the fueled core region, the computer code encountered difficulties during the quenching of the third-from-top core channel (channel 3) at 2800 seconds. The energy released during rapid refilling of calculational cells containing high temperature fuel elements was so large that instabilities resulted and the calculation failed. To circumvent this problem, the model was renodalized such that the axial cells and heat structures representing the third core channel were expanded from 3 to 15 (Ref. 45). With this change, core channel 3 was successfully reflooded. Renodalization of core channels 1 and 2 was not required because quenching of fuel in these channels resulted from a gradual increase in liquid flow from the CV-5 valve and not by rapid introduction of liquid from a rising inlet riser level. A full discussion of core renodalization for reflooding was presented in Reference 38.

The pressure tube and graphite temperature responses for the revised station blackout calculation were comparable to those observed for the base case calculation (Ref. 38). Pressure tube temperatures generally followed the trends of outer fuel element temperatures shown in Figures 15.6-246 and 15.6-247. The pressure tubes in general were slightly cooler than the fuel elements, however because of the high stored energy in the graphite, were slower to quench. Graphite temperatures in the lower core channels declined continuously as a result of heat transfer to the fluid inside the pressure tubes. In the upper core channels the graphite temperatures declined until the channels voided, remained steady until the channels refilled, then resumed their decline.

To evaluate the sensitivity of station blackout reflooding phenomena to changes in ECCS injection rate, a sensitivity calculation was performed at the end of the revised station blackout calculation. At 3668 seconds, the ECCS injection capacity was increased from 22,000 gpm to 24,000 gpm and the calculation was continued to 4000 seconds. This change is evident in the ECCS mass flow rate responses shown in Figure 15.6-248. As shown in Figure 15.6-246, at the time the change was implemented the fuel elements in all but the top channel had been quenched. In the top channel the fuel element temperature was stable around 850°F at this time. During this period,

the core inlet riser was full as shown in Figure 15.6-253. When the ECCS injection rate was increased, within the demonstrated capacity (Ref. 48) of 2 ECCS high lift diesel pumps, the fuel elements in the upper core channel quenched immediately.

The findings of the ECCS sensitivity analysis are as follows. With the original 22,000 gpm ECCS injection rate, the inlet riser could be refilled. However, with the inlet risers full, the ability of the ECCS to quench fuel in the uppermost core channel was marginal. Because of the continually declining core decay heat, all fuel elements are expected eventually to be quenched. With the small increase in ECCS capacity to 24,000 gpm, well within the demonstrated capacity of 2 ECCS pumps (Ref. 48), quenching of fuel in the uppermost core channel was accomplished immediately. This finding indicates a 24,000 gpm ECCS capacity is sufficient to completely refill the inlet risers and subsequently quench heated fuel elements in all core channels.

In summary, a revised simulation of extended station blackout conditions in the N Reactor has been completed (Ref. 39). The simulation assumed a lower conservative bound ECCS capacity (22,000 gpm) and half blowdown capacity (4 V-4 valves). The results of the simulation were comparable in most respects to a previously reported (Ref. 38) station blackout base case simulation (ECCS capacity representative of three high lift ECCS pumps and eight V-4 valves). With the reduced ECCS capacity, fuel elements in the upper three core channels heated up during the reflood phase of the transient; this compares with two channels in the base case simulation. A peak fuel element temperature of 1244°F was calculated and this is slightly higher than the maximum temperature calculated in the base case (1208°F). In both the revised simulation and the base case, reflooding of the core proceeded slowly as a result of: (1) core inlet riser level depression caused by persistent steam flow through the CV-5 valve, and (2) a carryout of liquid from the core to the dump tank caused by entrainment in the outlet riser. While CV-5 valve steam flow depressed the core inlet level and thus retarded the reflood process, the steam flow provided significant cooling of the fuel elements in the upper core channels. The ability of 22,000 gpm ECCS injection

capacity to quench fuel in the uppermost core channel during a station blackout sequence was shown to be marginal. A sensitivity calculation was performed that indicated 24,000 gpm ECCS capacity, within the demonstrated capacity of 2 ECCS High Lift Diesel pumps (Ref. 48), is adequate for this purpose.

15.6.5.4 Hot-Leg Breaks

Hot-leg breaks would be characterized by very small fuel heat-up transients because the RCS blows down through the core to reach the break. The hot-leg manifold break would result in the maximum confinement system pressure. See Section 6.2.

15.6.5.4.1 Hot-Leg Manifold Breaks, Double-Ended and 100-Percent Split (RELAP4) 21

The model schematic is shown in Figure 15.6-70. Three cases were analyzed with this model two double-ended cases with discharge coefficients of 1.0 and 0.6, and a 100-percent split case. The only significant difference among the cases was the blowdown rate. The transient is initiated by opening a break (J30 and 31) in the hot-leg manifold (CV-3). This causes a reactor scram and pump trip. The pumps coast down over a period of several seconds, providing adequate coolant to the core until the pressure in the pump suction lines is low enough to close the pump discharge check valves (J10). At this point, the core is stagnant with the break being supplied by flow, due to flashing of the hot-leg coolant. This proceeds until the ECCS is tripped, opening the V-4 valves (J25) and eventually allowing cold ECCS coolant to enter the cold-leg risers (CV-14), closing the CV-5 valves (J14) and initiating the reflood process.

The hot-leg manifold breaks are in the largest single pipe in the RCS. All of these breaks are characterized by continued flow through the CV-5 valves, which removes a large portion of the initial core power and delays the fuel heat-up transient.

The results of the double-ended breaks are shown in Figures 15.6-71 through 15.6-75 for a discharge coefficient of 1.0 and in Figures 15.6-76 through

15.6-80 for a discharge coefficient of 0.6. The results are very similar, with the largest difference being the depressurization rate. Figures 15.6-71 and 15.6-76 show the pressure response of the hot- and cold-leg risers, and Figures 15.6-72 and 15.6-77 show the flow rate through the CV-5 valves. The slow depressurization rate following 30 seconds is a result of reaching the saturation pressure of the fluid in the cold-leg piping between the pumps and the manifold (CV-8-CV-11). The V-4 valves opened at 20 seconds, but because of their slow opening time and the size and location of the break, had little effect on the depressurization rate. The fuel temperature transients shown in Figure 15.6-73 and 15.7-78 are smooth and decreasing because the flow rate through the CV-5 valves is adequate to cool the fuel (See Figures 15.6-75 and 15.6-80).

The results of the 100-percent-split case are presented in Figures 15.6-81 through 15.6-86. This case is similar to the double-ended breaks except for a slower depressurization rate. The cold-leg flow rate through the CV-5 valves is shown in Figure 15.6-82. The valves do not close in this case, providing flow to the core throughout the transient. This results in adequate cooling of the fuel as shown in Figure 15.6-85.

The sequence of events for the hot-leg manifold breaks are presented in Table 15.6-23.

A series of analyses have recently been performed, utilizing state-of-the-art analytical tools, to confirm the peak confiner pressure (Ref. 49). These analyses utilized RELAP5 and the HECTR code and determined that the calculated peak confiner pressure occurs from an outlet manifold break (200%, 2 HLDPs, no AC power) and that the peak confiner pressure is not appreciably different than the RELAP4 based analyses. This serves to confirm the adequacy of the existing RELAP4 based analysis presented in this section. See Section 6.2.1.3.

15.6.5.4.2 Hot-Leg-Riser Break, Double-Ended (RELAP4)

This analysis discussion is provided for historical continuity, but has been superseded by the RELAP5 bases analyses described in Section 15.6.5.4.3.

The model schematic is shown in Figure 15.6-87. The transient is initiated by opening a break (J83, 84) in one of the 16 hot-leg risers (CV-49). The break areas shown in the cold-leg risers (J30, 31) are not opened for this case. The blowdown causes a reactor scram and a pump trip but the pump does provide coolant flow through the core for a period of several seconds. The break causes the CV-3 valve (J56) in the broken riser to close immediately. The RCS continues to depressurize with some cooling of the fuel caused by flashing in the hot-leg piping between the CV-3 valves and the core. The V-4 valves (J25) open on receipt of the ECCS trip, providing another path for RCS coolant to exit. The transient is terminated when cold ECCS water enters the RCS through the CV-2R valves (J90 and 91).

The results of the hot-leg-riser, double-ended break are presented in Figures 15.6-88 through 15.6-95. A break flow coefficient of 1.0 was used, with the break flow rate presented in Figure 15.6-93. This break is typical of all hot-leg breaks in that the CV-5 valves remain open long enough to provide sufficient coolant to the core to minimize any fuel temperature transient. The analysis was done only for the first 20 seconds because the peak confiner pressure was reached at less than 5 seconds, and it is apparent that the fuel temperature transient is less severe than for the cold-leg breaks. The cold-leg-riser pressure response is shown in Figure 15.6-88. The CV-5 valve flow for the intact and affected risers is shown in Figures 15.6-89 and 15.6-90, respectively. The fuel temperature response in both the intact and affected cores is typical of hot-leg breaks with the fuel temperature decreasing after an initial increase as shown in Figures 15.6-91, 92, 94 and 95. The fuel temperature in the affected core (Figure 15.6-91) shows a decrease at 2 seconds due to an increase in the CV-5 valve flow rate (Figure 15.6-90) and the reactor scram. The sequence of events for the hot-leg-riser break is presented in Table 15.6-24.

15.6.5.4.3 Hot Leg Riser Break, Double-End (RELAP5)

21

General Discussion

The model schematic for this break is shown in Figures 15.6-212, 15.6-213, 15.6-214 and 15.6-215. The description of the system model and the details

of the model modification are provided in Reference 40 and are virtually identical to the limiting large break LOCA analysis described in Section 15.6.5.3.4.

In the discussion presented here the section of the core associated with the broken outlet riser is referred to as the "affected" core, while the other section of the core is referred to as the "unaffected" core.

The outlet riser break sequence was initiated by a double-ended rupture of one core outlet riser at the bottom. Most plant functions were assumed to function normally. As an exception, the HPI and GSCS systems were assumed inoperable. RELAP5-calculated plant response follows. Shortly after the assumed pipe break, the primary system depressurized rapidly. ECCS signals were generated, opening the V-3 and V-4 valves. The core voided and fuel element heatups occurred. ECCS coolant quenched the fuel elements in the lower half of the core. The ECCS capacity was insufficient to completely fill the core, however, steam cooling was adequate to prevent damage to the upper core fuel elements. The peak fuel temperature was 1244°F and 1000°F in the unaffected and affected core regions, respectively.

A sensitivity calculation of the base case outlet riser break scenario was performed. The ECCS capacity was increased from 22,000 to 24,000 gpm, well within the demonstrated capacity of 2 ECCS pumps (Ref. 48), and the primary coolant pumps were tripped. The sensitivity calculation was initiated at 1000 seconds into the base case calculation. The effect of the increased ECCS capacity and tripping the pumps was a complete quench of the unaffected core region. A maximum fuel element temperature of 1221°F was calculated in the unaffected core region. The higher ECCS flow was capable of quenching the lower half of the affected core region. Steam and two-phase cooling of the upper affected core region was adequate to prevent fuel element temperature excursions above 950°F.

Detailed Discussion - Hot Leg Riser Break (RELAP5)

This section presents analysis of two outlet riser break sequences. The first 1000 seconds of the sequences is identical. The reactor was assumed

to be operating at 3870 MW, full power conditions, when the rupture occurred at time zero. The full blowdown valve capacity was assumed to be available (total of 8 V-4 valves operated). High pressure injection and graphite cooling systems were assumed to be unavailable. ECCS capacity was assumed to be 22,000 gpm maximum. After 1000 seconds, two calculations were performed. In the first calculation, the primary coolant pumps continued at the degraded speed of 1240 rpm, and the ECCS capacity remained at 22,000 gpm. In the second calculation, the primary coolant pumps were assumed to be tripped and ECCS capacity was increased to 24,000 gpm at 1000 seconds. In the following discussion, the first calculation is termed the "base case" and the second calculation is termed the "sensitivity" calculation.

The first 650 seconds of the calculations were performed using the production version of RELAP5/MOD2. After 650 seconds, the calculations were performed using the version of RELAP5/MOD2 modified to simulate the phenomena of reflooding in horizontal core channels. Both calculations were performed using the double, 8-channel core modeling. 21

Outlet Riser Break Base Case Calculation (RELAP5)

The base case outlet riser break calculation assumed a 22,000 gpm ECS capacity and continuous operation of the primary coolant pumps (the pumps were allowed to run during the transient, but at a reduced rate after reactor trip). The RELAP5-calculated sequence of events for the base case outlet riser break calculation is shown in Table 15.6-47. A reactor trip was calculated at 1.6 seconds as a result of the outlet downcomer pressure falling below 1565 psia. Representative calculated primary and secondary coolant system pressures are compared in Figure 15.6-255. The initial decline in the pressurizer pressure lagged that in the remainder of the primary coolant system because the fluid in the pressurizer was initially saturated at the operating pressure. Pressurizer flashing therefore occurred immediately; early depressurization of the pressurizer was limited by choked flow in the pressurizer surge line. The steam generator secondary pressure initially decreased in response to the decreased secondary heat load following reactor trip (secondary pressure was a controlled parameter in this calculation).

The reactor trip also caused the steam generator primary-side bypass valves to open and the initiated runback of the primary coolant pumps. At 19 seconds the runback of the primary coolant pumps to 1240 rpm was complete.

At 36 seconds the hot leg manifold pressure fell below 660 psia, producing the first of two conditions required for ECCS actuation. At 187 seconds the second ECCS condition was met when, due to increasing voids within the core coolant, the core flow fell below the setpoint flow rate. The ECCS actuation signal was therefore generated causing the V-4 blowdown valves and V-3 ECCS injection valves to open. Because the core inlet riser pressure was already below the ECCS pump shutoff head, ECCS flow to the core began immediately at 187 seconds. It should be noted that the high confinement trip, which would occur within 10 seconds, was conservatively ignored. Actual ECCS actuation signal would occur at 36 seconds.

The mass flow rates through the V-4 valves are shown in Figure 15.6-256 and the ECCS mass flow rates are shown in Figure 15.6-257. The V-4 valves are divided into two groups, each one serving half of the core. Because of the proximity of the break to the affected V-4 valve, flow through this valve was slightly less than for the unaffected V-4 valve flow. The ECCS injection inlets for the 16 inlet risers are not interconnected. Therefore the affected and unaffected ECCS mass flow rates were generally in a ratio of 1 to 15.

Representative calculated fuel element cladding temperatures are shown in Figures 15.6-258 through 15.6-261. The temperatures shown are at the centers of the core channels on the outer surfaces of the outer fuel elements. During the blowdown period, fuel elements in the top 5 channels of the unaffected core experienced two periods of heatup; 190 to 240, and 450 to 530 seconds. In the affected core, the fuel in Channel 2 experienced a fuel element heatup from 530 to 580 seconds. Heatup of the fuel in the affected core was less than that in the unaffected core because affected core fluid conditions were more dynamic, due to the proximity to the break, than in the unaffected core. Fuel element heatup during the blowdown phase of the transient was moderate; the maximum calculated fuel element temperature was 670°F.

The break flow rates are shown in Figure 15.6-262. The affected side break represents the rupture of the lower stub of the 16-in. diameter outlet riser. The unaffected side break represents the 5.5-in. diameter flush line from the bottoms of the 15 intact outlet risers to the broken outlet riser. The affected side break flow rapidly declined in response to the increasing void fraction within the broken outlet riser, as shown in Figure 15.6-263. The unaffected side break flow also declined in response to the more slowly increasing void fraction in the bottoms of the unaffected outlet riser, as also shown in Figure 15.6-263. Shortly after ECCS injection flow began, ECCS fluid was drawn across the lower channels of the unaffected core, filling the lower unaffected outlet riser. Thereafter, the unaffected-side break passed liquid. The affected side break continued to pass mostly steam, as primary coolant flashed on its way to the broken outlet riser.

The computer code was changed from the production version of RELAP5/MOD2 to the version modified for reflooding of horizontal core channels at 650 seconds. Reflood fuel element heatup was calculated beginning at 700 seconds in the unaffected core and at 830 seconds in the affected core. By 650 seconds the unaffected core pressure had declined to about 100 psia; a gradual continuation of the primary coolant system depressurization continued until about 1100 seconds, as shown in Figure 15.6-255.

The events at about 1100 seconds, surrounding the time of minimum core pressure, are of some interest. Prior to that time, coolant flowing across the lower affected core channels was saturated by the heat removed from fuel and graphite. At about 1100 seconds, the heat removed was insufficient to saturate the coolant; this effect is especially noticeable in the fuel element temperature response for Channel 6 in Figure 15.6-261. As a result, subcooled coolant flowing across the lower affected core channels interacted with steam flowing in the outlet riser. This interaction resulted in high condensation rates within the lower affected core outlet connectors and outlet risers. The high condensation rates resulted in a momentary, local depressurization within this region. The effects of this depressurization are observed in most calculated parameters. The most dramatic effects included rapid decreases in fuel element temperatures (Figures 15.6-258

through 15.6-261) as fluid was drawn into the core channels toward the condensation site, and reversals of flows through the V-4 blowdown valves (Figure 15.6-256). The condensation effects ceased after a short time because steam production increased significantly as reflooding of the core continued.

After 1100 seconds, the primary coolant system pressures stabilized. The unaffected core pressure was slightly higher than that of the coolant loop. In turn, the coolant loop pressure was slightly higher than that of the affected core. Primary system coolant from the unaffected core therefore flowed through the unaffected CV-3 valves into the hot legs, and around the coolant loop. At the steam generators, a portion of the liquid was boiled to steam. At the cold leg manifold, the flow generally separated, with steam rising through the cold legs to the CV-5 valves on both the unaffected and affected core inlet risers. Because of this circulation pattern, the voided primary coolant system refilled slowly. The CV-3 check valve flow rates are shown in Figure 15.6-264. Note the CV-3 valve flow from the affected outlet downcomer stopped when the break was opened and passed little fluid during the transient. Unaffected CV-3 valve flow continued until the blowdown valves opened at 190 seconds, was stopped until about 980 seconds, then resumed through the remainder of the calculation.

Flow through the unaffected CV-3 valve was responsible for refilling the coolant loop. Liquid entering the steam generator primary side was mostly bypassed around the steam generators; the bypass valves were opened as a result of reactor trip. The liquid that did reenter the steam generator tubes was boiled by heat addition from the steam generator secondary. The power reversed at about 500 seconds, when the primary side pressure fell below the secondary side pressure, as shown in Figure 15.6-255. Representative loop void fractions are shown in Figures 15.6-265 and 15.6-266, for the primary coolant pump and cold manifold, respectively. Figure 15.6-267 shows the mass flow rates through the CV-5 valves on the unaffected and affected core inlet risers. After 1100 seconds, these flows were generally steam with a small liquid content.

Figures 15.6-268 and 15.6-269, respectively, show the void fractions in the unaffected core inlet and outlet risers. Figures 15.6-270 and 15.6-271 show the same information for the affected core inlet and outlet risers. For clarity, only information at elevations adjacent to core channels 1, 3, 5, and 7 is presented. Significant voiding of all four risers at each core channel level was calculated during the blowdown portion of the transient.

Both inlet risers momentarily refilled when the minimum primary system pressures were reached at about 1100 seconds. After that time, the unaffected inlet riser level was stratified, however, at a lower elevation. Both inlet riser levels were depressed by the steam flow through the CV-5 check valves, as shown in Figure 15.6-267. The affected inlet riser level generally was below the unaffected inlet riser level because the break at the bottom of the affected outlet riser was larger, thus providing a lower-resistance path to the confinement. 21

After 1100 seconds, the lower half of the unaffected outlet riser was liquid-filled. A small part of the coolant flowing across the lower core channels was diverted through the flush line to the break at the bottom of the affected outlet riser (this break mass flow is shown in Figure 15.6-262). The major portion of this coolant, however, rose into the unaffected outlet riser, establishing its liquid level. Steam flowing through the upper core channels entrained liquid at the mid-riser elevation, producing two phase fluid in the upper half of the unaffected outlet riser. The steam flow carried out most of the ECCS liquid over the unaffected outlet downcomer. From the outlet downcomer a portion of the flow was diverted through the unaffected CV-3 check valves into the coolant loop. Another portion of the flow was diverted to the affected outlet downcomer through the blowdown valve inlet headers. The remaining flow passed through the blowdown valves to the dump tank.

Throughout the calculation, a mist flow was calculated in the affected outlet downcomer. ECCS injection into the affected core channels was heated by the fuel elements and then flashed as the local pressure fell rapidly along the path to the break at the bottom of the affected outlet riser. Another source

of flow through the outlet riser was the reverse flow through the affected outlet downcomer as discussed in the previous paragraph.

With the unaffected inlet riser level at the mid-core elevation, the fuel elements in the upper three unaffected core channels heated up, as shown in Figure 15.6-258. Fuel elements in the lower core channels remained cool because of liquid flow from the inlet riser. The upper core channels were steam cooled, primarily from steam flowing through the CV-5 check valves. Steam cooling limited the unaffected core temperatures to a maximum of 1244°F. This temperature was calculated on the outer surface of the outer fuel element at the center of the top core channel. Note this temperature is within the temperature range of the alpha fuel phase change.

Because the inlet riser level in the affected core was lower than in the unaffected core, the top five channels of the affected core experienced fuel element heatup during the reflood period. As shown in Figures 15.6-260 and 15.6-258, the fuel element temperatures in the affected core were lower than those in the unaffected core. Conditions in the affected core were more dynamic than in the unaffected core because fluid there was accelerated to the break at the bottom of the outlet riser. The fluid velocities were therefore higher, and the resulting steam cooling more effective in the affected core than in the unaffected core.

The base case outlet riser break calculation was terminated at 1777 seconds. At that time, refilling of the coolant loop was proceeding, aided by the continued operation of the primary coolant pumps. The inlet riser levels of both the affected and unaffected cores were rising and steam cooling had stabilized the fuel element temperatures. A more satisfactory conclusion for the calculation would have been to continue until the coolant loops had refilled. This would allow the inlet risers and core channels of both cores to fill and quench all fuel elements. The calculation was terminated because of the findings from similar analyses performed for inlet riser break sequences described in Section 15.6.5.3.6.

For the inlet riser break sequence, continued operation of the primary coolant pumps was determined to result in unacceptable heatup of the fuel elements in the affected core (see Section 15.6.5.3.6). Tripping the primary coolant pumps was demonstrated as an effective remedial action that prevented fuel damage following an inlet riser break (see Section 15.6.5.3.6). Since the ability of the operators to diagnose the break location (i.e., is the break in the inlet riser, outlet riser, both, or neither) is questionable, a common remedial action for all break locations is desirable. The base case outlet riser break calculation was therefore terminated and a sensitivity outlet riser break calculation performed to evaluate plant response using the remedial action assumptions. An analysis of the sensitivity outlet riser break follows.

In summary, the base case outlet riser break calculation indicated that, with primary coolant pumps powered and a 22,000 gpm ECCS capacity, fuel element heatups were experienced in the upper four channels of the unaffected core and upper five channels of the affected core. Steam cooling of the upper core channels was adequate to prevent fuel damage. The maximum calculated fuel element temperature was 1244°F in the top channel of the unaffected core. Steam cooling in the affected core prevented fuel element temperatures from exceeding 1000°F. The calculation was terminated prior to quenching of the core because the response of the plant to an outlet riser break with primary coolant pumps tripped and 24,000 gpm ECCS was of greater interest. This scenario is investigated in the sensitivity calculation presented next.

21

Outlet Riser Break Sensitivity Calculation (RELAP5)

As described above, the base case outlet riser break calculation was terminated and the sensitivity calculation initiated so the inlet and outlet riser break sequences could be evaluated using a common set of assumptions regarding primary coolant pump operation and ECCS capacity. The assumptions in the sensitivity outlet riser break calculation were identical to those in the inlet riser break sensitivity study as presented in Section 15.6.5.3.6. The assumptions in the sensitivity outlet riser break were also identical to those in the base case outlet riser break calculation, with two exceptions:

(1) the ECCS capacity was increased from 22,000 to 24,000 gpm at 1000 seconds, and (2) all primary coolant pumps were tripped at 1000 seconds.

The RELAP5-calculated sequence of events for the sensitivity outlet riser break sequence is shown in Table 15.6-48. Events prior to 1000 seconds duplicate those shown in Table 15.6-47 for the base case outlet riser break; discussion of events during that period is not repeated here. At 1000 seconds the ECCS capacity was increased from 22,000 to 24,000 gpm and it was assumed the operators tripped the primary coolant pumps.

The step change in ECCS flow rate to both the affected and unaffected cores is shown in Figure 15.6-272. The primary coolant pumps had been operating at a constant 1240 rpm since the runback from the normal operating speed was completed at 19 seconds. When all pumps were tripped at 1000 seconds, a coastdown of the primary coolant pumps began. The pumps rotors were calculated to stop turning at 1106 seconds.

Representative primary and secondary system calculated pressures are shown in Figure 15.6-273. Responses of these pressures during the sensitivity calculation were comparable to those from the base case calculation shown in Figure 15.6-255. Minimum primary system pressures were reached at about 1100 seconds.

At about 1800 seconds, the lower affected core channels had been cooled and ECCS liquid reached the bottom of the broken outlet riser. The break void fractions are shown in Figure 15.6-274. Prior to 1800 seconds, flow was calculated from the coolant loop through the CV-5 check valves into both the affected and unaffected cores, as shown in Figure 15.6-275. The change in break flow at 1800 seconds changed the pressure distribution so that coolant flowed from the unaffected core outlet downcomer, through the CV-3 check valves into the hot legs, and around the coolant loop. The reactor loop began refilling at 1800 seconds.

Void fractions in the unaffected core inlet and outlet risers are shown in Figures 15.6-276 and 15.6-277. Void fractions in the affected core inlet

and outlet risers are shown in Figures 15.6-278 and 15.6-279. The decrease in CV-5 valve flow to the unaffected core after 1800 seconds allowed the unaffected inlet riser level to increase to the top of the core. Because of the proximity of the break, the affected inlet riser level also increased, but only to about 75% full. Further increase in the affected inlet riser level was very gradual. The unaffected outlet riser completely filled.

The fuel element temperature responses are shown in Figures 15.6-280 through 15.6-283. The temperatures shown are located on the outer surface of the outer fuel element at mid-channel. After 1800 seconds, fuel temperatures in the lower halves of both the affected and unaffected cores remained at or below saturation.

In the upper unaffected core, fuel elements in the top 3 channels experienced heatups. As was discussed for the base case, steam cooling of fuel elements in these channels limited their heatup. When the unaffected core inlet riser level increased to the top at 1800 seconds, fuel in Channel 3 was quenched immediately. Quenching of fuel in Channel 2 was delayed for about 200 seconds and quenching in Channel 1 for about 800 seconds. These delays represent the time required for the head of ECCS liquid at the inlets of these core channels to reflood the process tubes. The maximum calculated fuel element temperature, 1221°F, was reached at 1709 seconds in the outer fuel element in the middle of the top core channel. 21

In the upper affected core, the top 4 core channels had fuel element temperatures above saturation temperature. Because the increase in the affected inlet riser level was less than in the unaffected core, fuel elements in the upper core region were not quenched by single-phase liquid flow from the inlet risers. Cooling from the steam flow through the affected core CV-5 check valve was an important source of cooling. Because the hydraulics within the affected core were so dynamic, the affected inlet riser level was oscillatory. Liquid was transported by these oscillations to the inlets of the upper affected core channels. This intermittent liquid flow provided an additional source of cooling in the upper core. As a result of this cooling, after 1800 seconds the fuel element temperatures in the upper affected core

remained below 750°F. At about 2800 seconds, refilling of the primary coolant loop was completed. When this happened, liquid flow through the affected loop CV-5 check valve into the top of the affected inlet riser provided additional cooling for upper affected core channels. Because the affected core pressure was low, the liquid flowing through the CV-5 valve flashed, providing two-phase cooling of the upper core.

In summary, the results of the outlet riser break sensitivity calculation indicate that, with the primary coolant pumps tripped, 24,000 gpm ECCS capacity, which is within the capacity of 2 ECCS pumps (Ref. 48), was sufficient to completely quench the unaffected core. A maximum fuel element temperature of 1221°F was calculated. In the affected core, ECCS was capable of quenching the lower half of the core. Steam and two-phase cooling of the upper affected core was adequate to prevent fuel element temperature excursions above 950°F. For long-term behavior, the core energy decays and the overall system pressure and temperature approaches atmospheric conditions. The steam production stops, and the affected core refills completely.

15.6.5.5 Pump Suction Breaks, Double-Ended and 100-Percent Split (RELAP4)
No RELAP5 based analysis of this transient has been performed because it is not the limiting event and the RELAP4 analysis is bounding.

The model schematic is shown in Figure 15.6-96. The transient is initiated by opening the break (J30 and 31) in the suction line of one of the five operating pumps. Because of the pump discharge check valve (J40), the two cases differ only by the area of the break 60 percent for the double-ended break because of the 0.6 discharge coefficient, and 100 percent for the split. The pump discharge check valve in the broken loop closes immediately. The reactor scrams, and the pumps begin to coast down. Flow is provided through the core by the intact loops for several seconds. Eventually, the low pressure in the broken loop causes the flow in the four intact loops to reverse and flow through the hot-leg manifold (CV-3), into the broken loop and to the break. At this time, the pump discharge check valves in the intact loops close, reducing the core flow rate to that provided by expansion of the cold-leg fluid. The V-4 valves (J25) open on the ECCS trip signal,

and the transient is terminated when cold ECCS fluid enters the RCS through the CV-2R valves (J72).

The results for these cases are shown in Figures 15.6-97 through 15.6-108. These breaks behave as typical hot-leg breaks because of the location of the check valves (J10 and 64) just downstream of the pump discharge. The valve in the broken loop closes immediately, allowing the pumps in the intact loops to continue supplying coolant to the core for most of the transient, resulting in low fuel temperatures. The depressurization rates for the two cases are shown in Figures 15.6-97 and 15.6-103. The cases were not run to the end of blowdown because it is apparent that these cases have less severe fuel responses than the cold-leg breaks. The flow rates through the CV-5 valves (J14) are shown in Figures 15.6-98 and 15.6-104. The flow rate through the CV-5 valves provides adequate cooling to the fuel to delay any heat-up until 40 seconds after the break. See Figures 15.6-99 and 15.6-105.

The sequence of events for the pump suction breaks are presented in Table 15.6-25.

15.6.5.6 Steam Generator Manifold Break (RELAP4)

No RELAP5 based analysis of this transient has been performed due to this event not being limiting and the RELAP4 analysis being bounding. 21

Figures 15.6-109 and 15.6-110 show the model schematic used and a cross section of a steam generator showing the manifold and tube arrangement. The transient is initiated by breaking the top of the vertical manifold inside of one of ten operating steam generators (CV-55 through 58). The break (J30) allows the RCS coolant to blow down into the secondary side of the steam generator and then into the SCS piping. Except for the fluid discharging into the SCS, the break behaves essentially as the pump suction breaks. See Section 15.6.5.5. The SCS is modeled as two volumes (CV-61 and 60) which represent the single affected steam generator and the rest of the SCS, respectively. The pressure relief valves in the SCS (J72) open when the pressure reaches 600 psig and reclose at approximately 575 psig. 21

The results of this analysis are given in Figures 15.6-111 through 15.6-119. This case was analyzed to determine if a break in the steam generator manifold, with the coolant blowing down into the SCS, had the potential of overpressurizing the SCS. The initial conditions assumed for this case are those that would result in the highest SCS pressure. The initial SCS pressure was assumed to be at the Technical Specification limit of 465 psia although the normal operating limit is 165 psia. Also, the system was modeled with the minimum volume possible, by assuming that the plant was not exporting steam to WPPSS Hanford Generating Project and only the SCS volume to the dump condensers was available. The break behavior is similar to the pump suction break. The break flow rate is shown in Figure 15.6-117. The cold-leg riser pressure response is shown in Figure 15.6-111, and the CV-5 valve flow rate is shown in Figure15.6-112. Again, the flow rate through the CV-5 is adequate to cool the core throughout the first 50 seconds of the transient as shown in Figures 15.6-118 and 119. The ECCS trip signal has opened the V-4 valves (J25) at 70 seconds which will increase the depressurization rate. The pressure response in the secondary side of the affected steam generator is shown in Figure 15.6-114, and the response in the rest of the SCS is shown in Figure 15.6-115. The pressure relief valves in the SCS open at 35 seconds (Figure 15.6-116), terminating the pressure rise in the SCS below the design pressure of 675 psia.

The sequence of events for the steam generator manifold break is presented in Table 15.6-26.

15.6.5.7 Accidental V-4 Valve Actuation (RELAP4)

No RELAP5 based analysis of this transient has been performed because it is not a limiting event and the RELAP4 analysis is bounding.

The model schematic for these cases is shown in Figure 15.6-120. The transient is initiated by opening some, or all, of the V-4 valves (J25) allowing the RCS coolant to blow down into the emergency dump tank (EDT, CV-26). The reactor scrams and the RCS pumps trip off as in the previous cases. This transient is assumed to be an infrequent incident and, as such, it has been assumed that the high pressure injection system (HPIS) is

functional. The local depressurization in the hot-leg risers forces the CV-3 valves (J20) to close but the HPIS is providing adequate cooling for the core. Two cases were analyzed: the first case is the opening of all eight V-4 valves on a 7-second ramp which is normally expected; the second case is the opening of four V-4 valves on a 25-second ramp, these being the Technical Specification minimum requirements. The blowdown would continue until the ECCS coolant enters the cold leg risers.

The results of the two cases are shown in Figures 15.6-121 through 15.6-132. The first case provides the fastest depressurization rate possible for this accident, and the second case assumes the minimum required V-4 valves and the maximum allowable opening times. Both cases assume that the HPIS is operable, with a maximum capacity of 5000 gpm at 120°F. Figures 15.6-125 and 131 still show the flow rate using 6000 gpm. However, there is no effect on the end results, such as fuel temperatures. This water is injected at the pump suction line (J60). The cold-leg riser pressure response is shown in Figures 15.6-121 and 15.6-127. The accident behaves similarly to a hot-leg-manifold break due to the location. See Section 15.6.5.4.1. The flow rate through the V-4 valves is presented in Figures 15-6-124 and 130. The CV-5 valves (J14) remain open throughout the transient due to the injection of water at the pump suction, allowing for adequate cooling of the core. See Figures 15.6-122, 15.6-123, 15.6-126, 15.6-128, 15.6-129 and 15.6-132. It should be recognized that ECCS would actuate on low RCS pressure and high RCS flow, therefore total reliance on HPI is not required to mitigate these transients. 21

The sequence of events for accidental V-4 valve actuation is presented in Table 15.6-27.

15.6.6 Hypothetical Accident

15.6.6.1 Introduction

The results of the analysis of a spectrum of postulated loss of coolant accidents were presented in Section 15.6.5. In these cases, reflood was initiated when the front riser pressure decayed to the high lift diesel pump 21

shutoff head, at which time the ECCS provides direct cooling to the reactor pressure tubes, thus, terminating the fuel heatup transient. Fuel failure was prevented since the maximum fuel temperature was less than the 1900°F clad failure temperature for reflood conditions (see Appendix 15.6A). Fission products were not released to the coolant, and there were no radiological consequences except for the RCS coolant released through the steam vents (See Section 6.5.3.).

This section assumes a nonmechanistic failure to provide direct cooling to the reactor pressure tubes following a LOCA. The case considered represents the hypothetical accident for N Reactor with respect to fuel failure hydrogen generation and the release of fission products. It is a limiting occurrence which is not expected to occur. The case is postulated, however, in response to Regulatory requirements to assume a release of a large fraction of fission products from the core to demonstrate the capability of fission product removal systems in preventing offsite radiological consequences in excess of 10 CFR 100 guidelines. The case is also postulated to serve as a limiting design basis for the severe accident hydrogen mitigation system, consistent with 10 CFR 50.44. Section 15.6.6.5 provides the limiting Hydrogen Mitigation System Source Term (HMSST) Analysis and Section 15.6.6.2 and 15.6.6.3 provides the Hydrogen Mitigation Design Basis Accident (HMDBA) Analysis. The HMSST is a conservative, bounding assessment of the hydrogen mitigation issues, with the HMDBA forming a more practival but still conservative, prediction of the Hydrogen Source term for the non mechanistic hypothetical accident.

The high thermal conductivity and large mass of the N Reactor fuel and graphite result in a relatively slow rate of fuel temperature increase following loss of direct cooling of the fuel, and fuel failure would not occur until several minutes after the initiation of the event. This slow heatup rate permits N Reactor to use a confinement concept, rather than a containment concept, which allows for separation of the steam release phase from the fission product release phase in a hypothetical accident initiated by a major pipe break. The Regulatory Guide 1.4 assumptions of instantaneous release of fission products to containment, therefore, must be modified so

that the behavior of the N Reactor fission product removal systems may be adequately described and evaluated. In addition, the graphite and shield cooling system (GSCS) provides an indirect means of cooling the fuel and limiting the extent of fuel failures in the event of loss of all direct cooling of the fuel. The GSCS and fission product removal systems are assumed to function in the hypothetical accidents described below.

The sequence of events during the postulated LOCA is as described in Section 15.6.5.3.7 except that, when the front riser pressure decays to the high lift diesel pump head shutoff, no direct cooling of the fuel is assumed. The initial conditions are shown in Table 15.6.15. 21

The case considered is a double-ended break of the 24-in. cold-leg manifold in 109-N. A break at this location closes the CV-5 check valves and stagnates the coolant in the core with very little fuel-to-coolant heat transfer. This results in the maximum rate of fuel temperature rise, and, therefore, the maximum fraction of the core with failed fuel.

Sensitivity studies on the break discharge coefficient (RELAP4) were performed to determine the coefficient which would maximize the fuel heat-up rate. A discharge coefficient of 0.6 was found to result in the maximum fuel temperatures during the RCS blowdown. The only means of cooling the fuel after the RCS was void of coolant would be thermal conduction and radiation from the fuel elements to the pressure tube and ultimately to the coolant in the graphite and shield cooling system (GSCS). 21

For the Hydrogen-Mitigation Design Basis Accident Analysis (HMDBA) transient (Section 15.6.6.2 and 3), a RELAP5/MOD2 analysis was performed (See Section 15.6.5.3.8) to determine the rate of system depressurization, stored energy removal, etc., to establish the initial boundary conditions for the core thermal analyses as performed by the TRUMP-BD (Ref. 52) computer code. Fuel heatup was determined to begin following core dry-out 150 seconds after the initiation of the accident. The ECCS, including the HPI system, was assumed to be ineffective. The GSCS was assumed to be functional throughout the HMBDA with a heat-removal capacity within its nominal operating range of 21

between 40 to 60 MW thermal. The GSCS serves as the ultimate heat sink for the nuclear decay and exothermic metal/water chemical reaction heat sources.

The TRUMP analyses were performed for each of the three core-loading zones (i.e., fringe, spike and central) for four power conditions. The analyses were performed using a half-length model of one fueled pressure tube and its surrounding graphite moderator. The half-length model included the non-GSCS-cooled regions of the core. A total of 24 TRUMP-BD simulations were therefore necessary to completely characterize the core during an HMDBA, given the two halves of each pressure tube, three enrichment zones, and four power bins in each enrichment zone. A core power/volume weighing approach allowed for the construction of a whole-core hydrogen source term and energy balance from the individual TRUMP-BD results.

Fuel melting is calculated to occur within 10 minutes followed by decreasing temperatures as the GSCS becomes effective in removing the decay heat and chemical energy. The maximum fuel temperatures are predicted to occur in the core end region (non-GSCS cooled). Fuel melting occurs outside the zone where GSCS cooling occurs in this core end region after 2 hours, and temperatures are stabilized after about 10 hours as energy is transferred axially to the reflector graphite and GSCS-cooled regions. The GSCS is effective in mitigating the progression of the HMDBA and limiting whole-core fuel failure to less than 26.2%. Of the 26.2% predicted, 5.8% was in the end region (non-GSCS cooled) and 20.4% was in the GSCS region. The calculated hydrogen HMDBA source term is approximately 30,000 ft^3 at 2 hours and near 70,000 ft^3 at 10 hours. A discussion of hydrogen consentrations for the limiting Hydrogen Mitigation System Source Term (HMSST) is provided in Section 15.6.6.5.

These analyses provided a basis for the design and operation of the hydrogen mitigation system. This confirms the previous safety assessments that the GSCS is effective in mitigating the course of a severe degraded-core accident and in limiting the extent of core damage and hydrogen production to predictable and manageable levels (Ref. 51 and 52).

15.6.6.2 Sequence of Events and System Operation

The hydrogen mitigation design basis accident (HMDBA) scenario is the N Reactor's hypothetical loss-of-coolant accident described in Section 15.6.6.1.

For the HMDBA transient a RELAP5/MOD2 analysis (See Section 15.6.5.3.8) was performed to determine the rate of system depressurization, and stored energy removal, to establish the initial boundary conditions for the core thermal analyses as performed by TRUMP-BD(N). Fuel heatup was determined to begin following core dryout 150 seconds after the initiation of the accident. The ECCS, including the high-pressure injection (HPI) system, was assumed to be ineffective. The primary system vents its steam inventory into the confinement and, in turn, to the atmosphere through vents provided to avoid overpressurization of the confinement. However, without primary coolant the reactor core slowly overheats due to continued decay heating. The GSCS was assumed to be functional throughout the HMDBA with a heat-removal capacity within the nominal operating range of 40 to 60 MW thermal. The GSCS also serves as the ultimate heat sink for the decay and exothermic metal/water chemical reaction heat sources. Z.1

Table 15.6-28 provides the RELAP4 hypothetical accident confinement sequence. This table is included because the RELAP4 cold-leg manifold break blowdown data, which is virtually the same as the RELAP5 HMDBA blowdown data, was used in the Containment of Radionuclides Released After LOCA (CORRAL) code analysis of this event for radiological dose consequences. (See Section 15.6.6.6). This table provides the typical confinement system response sequence.

The effect of a different initiating event (inlet riser break vs. assumed cold-leg manifold break) is primarily to change the rates of system depressurization and initial core heatup. Different initiating events are not particularly important with respect to long-term core behavior and hydrogen generation. Consequently, the assumption of a cold leg manifold break with the loss of ECCS to the entire core serves as the boundary conditions for the development of a hydrogen source term and postulated

hypothetical radiological releases. In addition, this event appropriately bounds, from a PRA-frequency and accident-progression perspective. Therefore, the HMDBA accident analyzed herein and HMSST analyzed in Section 15.6.6.5 do not represent one hypothesized scenario, but rather a bounding plant end state that envelopes all severe-accident states having a probability of 10^{-8} or greater per reactor year.

For purposes of dose projections, the confinement system is assumed to function, consistent with its description in NUSAR Section 6.2. This assumption permits the evaluation of dose due to the time dependent nature of confinement pressure response and fog spray plate-out of fission products. A discussion of the actual confinement system response and its affect on the offsite dose consequences of this hypothetical accident are provided in Section 15.6.6.4 and 15.6.6.6 respectively.

15.6.6.3 Core and System Performance

15.6.6.3.1 Mathematical Model

The RELAP5/MOD2 analysis code has been previously described in the limiting LOCA analysis, Section 15.6.5.3.4. The RELAP5/MOD2 code was used to establish the initial boundary conditions for the core thermal analysis performed by TRUMP-BD(N) (Reference 52).

The HMDBA thermal-response and hydrogen source-term analyses were performed utilizing the TRUMP-BD(N) computer program. This code has been used for many years as justification for the safety of the U.S. Nuclear Regulatory Commission (NRC)-sponsored National Research Universal (NRU) Coolant Boilaway and Damage Progression Program.

Twenty-four HMDBA simulations were required to completely characterize the whole-core performance. The HMDBA simulations were performed for four power conditions for the inlet and outlet halves of each pressure tube for each of the three core-loading zones (i.e., fringe, spike, and central).

15.6.6.3.2 Input Parameters and Initial Conditions

The N Reactor TRUMP-BD analyses used half-length axial model and unit cell geometries. The N Reactor concentric fuel elements, associated cladding, pressure-tube block, cooled and uncooled filler blocks, helium-filled gaps, transition-tube block, and the axial reflector blocks were explicitly modeled. A temperature-dependent heat-transfer coefficient for each of the helium-filled gaps was derived based on a $K(t)/\delta$ treatment of the temperature-dependent thermal conductivity of helium over the given as-built gap size. (Reduced radial gap sizes are possible given the swelling and distortion effects associated with 20 years of operation. However, the use of as-built nominal dimensions was conservative.) Radiative heat transfer from the inner to outer fuel elements, the outer fuel cladding to the pressure tube, pressure tube outer surface to the pressure tube graphite block, the graphite block to the outer surface of the GSCS tube, filler block to filler block, and filler block to the reflector blocks was explicitly modeled. Material properties are discussed in Section 4 of Reference 52. 21

The half-length model is adiabatic at the outside reflector face. Heat transfer within the end regions (non-GSCS cooled) was modeled via axial conduction along the pressure-tube blocks and fuel elements and between segments of the uncooled filler blocks, and by radiation between the uncooled filler blocks, and the transition/reflector regions.

Appropriate GSCS region 4-in. unit cells were modeled assuming a thermal connection between the inner diameter of the GSCS tube and an external heat sink representing the GSCS cooling water bulk temperature. The GSCS cooling water was assumed to be at a constant 150°F during the HMDBA transient. A constant heat transfer coefficient of 1870 $Btu/h \cdot ft^2 \cdot °F$ was assumed at the inner wall of the GSCS tube. This resulted in an initial GSCS heat-removal capability of approximately 44 MW thermal. In addition, radiative heat transfer between the cooled and uncooled filler-block pairs within the GSCS region was modeled.

Following system depressurization, the core was assumed to have a continuous and unlimited supply of steam for Zircaloy and uranium oxidation and, hence,

hydrogen generation. Unlimited steam availability provides a conservative upper bound for the analysis of the HMDBA and confirmation of the design basis for the hydrogen mitigation system (HMS). It should be noted that RELAP5 analyses predict that steam availability would end near 1200 seconds after the initiation of a hypothetical loss-of-coolant accident.

The Baker-Just (Ref. 18) and Wilson (Ref. 53) parabolic rate equations (temperature- and oxide layer thickness-dependent) were assumed as the bases for hydrogen generation from Zircaloy and uranium metal, respectively. Hydrogen generation from the following unit cell components was allowed:

- Cladding (Zircaloy-2)
- Fuel (uranium metal)
- Pressure tube (Zircaloy-2)
- GSCS tube (Zircaloy-2).

Oxidation of the Zircaloy-2 cladding and pressure tube was continuously modeled, whereas oxidation of the uranium fuel was assumed only after either the inner or outer cladding of each element exceeded the assumed cladding-failure threshold of 1900°F. Once the cladding-failure threshold was exceeded, the exposed uranium area was assumed to remain constant, even if temperatures decreased below the failure threshold of 1900°F.

Finally, a best-estimate decay-heat as contrasted to the assumption of 120% decay rate for the Appendix K analyses was assumed during the analyses.

In summary, the important HMDBA analysis assumptions are as follows:

- Cold-leg manifold break with subsequent core dryout and heatup
- Unlimited steam availability
- Radiative cooling/heat transfer allowed for fuel. No material relocation was modeled
- Axial conduction allowed. No axial convective losses/steam cooling were allowed

- o Zirconium and uranium oxidation allowed (Baker-Just and Wilson parabolic rate equations) (Ref. 18). Oxidation of the Zircaloy-2 components was continuously modeled. Oxidation of the uranium fuel was modeled only after the cladding-failure threshold of 1900°F was exceeded
- o Best-estimate decay heat
- o No ECCS. The GSCS was assumed to be functional throughout the HMDBA within the nominal operating range between 40 to 60 MW thermal.

The TRUMP-BD analyses were performed for a core power level corresponding to 4000 MW thermal. The analyses were performed for four power conditions (i.e., four power ranges or "bins"); the maximum-powered (i.e., peak power tube), average-plus-one-sigma, average, and the average-minus-one-sigma power pressure-tube for each of the three core-loading zones. This is referred to as the 4-bin approach. The maximum power bin can be defined as those pressure-tube powers greater than the average-plus-one-sigma tube powers. The average-plus-one-sigma powers can be defined as those pressure-tube powers greater than the average tube power but less than the maximum power bin. The average and average-minus-one-sigma powers can be defined in a similar manner. The information necessary to produce the power bins and obtain the average linear powers utilized by these analyses was supplied in Reference 54. 21

The relative axial power profile of the maximum power tube from the maximum power bin was assumed to establish the worst-case axial power profile within the GSCS region for each of the remaining three power bins in each region.

The limiting axial power profiles for the non-GSCS end regions were derived from power histograms. The histograms provide the predicted specific linear power frequencies for each of the three core zones for five axial slab thicknesses (front to rear). The first and last two slabs, corresponding to 48 in. each, provide an estimate of the maximum linear power within each of the end regions. The middle slab of 326 in. represents the entire GSCS region. By scaling (assuming the existing slope of) the initial relative axial power profiles for each core-loading zone to the maximum linear powers within each of the 24-in. slab sections, a limiting axial power profile

within each of the end regions for each of the core-loading zones was established.

The maximum linear power of approximately 200 kW/ft represents the maximum predicted linear power from Reference 54. The maximum linear power within the GSCS region (200.2 kW/ft) occurs for the spiked loading zone. The maximum linear power within the end region (non-GSCS) (152.5 kW/ft) occurs for the central zone. Power profiles shown in Figure 15.6-284 provide the bounding specific linear powers for both the GSCS and end regions.

The four power bins were derived from a statistical treatment (Ref. 54) of the known pressure-tube powers for each of the 1003 pressure tubes. By characterizing the whole core in terms of four power bins (i.e., maximum, average plus one sigma, average, and average minus one sigma) for each of the three core-loading zones, and by analyzing a bounding power condition for each bin, each of the four bins could be weighted by the number of tubes within each given core-loading zone and power bin. Core-wide weighing factors were derived. These factors were determined by dividing the number of pressure tubes within each power bin by 2006 to account for the two halves of each pressure tube.

By weighing each of the 24 HMDBA simulation results by the appropriate weighing factor, a whole-core hydrogen source term, a total core fuel-failure fraction, and a whole-core energy balance can be obtained.

As a check on the weighing factors, the total core power was back-calculated from the pressure-tube powers and the number of tubes within each power bin. A total core power of 4171 MW thermal was obtained. This is in close (+4.3%) agreement with the actual design basis of 4000 MW thermal for full-power conditions. A total core power greater than the design basis was to be expected, given the conservative approach of utilizing the appropriate "maximum" tube within each bin for each of the HMDBA simulations.

By dividing the number of power bins even further (i.e., plus and minus two sigma, plus and minus one-half sigma, etc.), 4000 MW thermal can be

approached. However, this quickly becomes computationally prohibitive. The 4-bin approach was considered adequate and provides for additional conservatism.

15.6.6.3.3 Results of TRUMP-BD Analysis

After system depressurization (time >150 seconds) the fuel begins to heat up until it reaches thermal equilibrium with the surrounding pressure tube and graphite mass.

The inner fuel temperatures, as shown in Figures 15.6-285 and 15.6-286, can be considered as the bounding results.

The peak inner-fuel temperature for most of the GSCS cooled axial locations reach the 1900°F cladding-failure threshold at approximately 750 seconds. The subsequent exothermic uranium metal/water and continuing Zircaloy/water reactions drive the inner-fuel temperature to a peak value of 2155°F at approximately 1090 seconds. The outer-fuel response is somewhat delayed and attains a peak temperature of 1970°F at 1150 seconds. After 1200 seconds, the fuel temperatures begin to decrease as the energy removal capacity of the GSCS exceeds the energy input to the GSCS cooled region. (The GSCS effectively limits the extent of fuel failure and hydrogen production within the GSCS cooled region.) By 10 hours all of the fuel temperatures within the GSCS region are approaching 1500°F.

The inner-fuel temperatures for the inlet half of the fringe zone and the outlet halves of the fringe, spike, and central zones are considerably lower. This is expected, given the low initial powers for the fringe zone and outlet halves. In most cases, the maximum inner-fuel temperatures for the GSCS cooled regions do not exceed 2000°F. The one exception is the inner-fuel temperatures for the spike-zone outlet half, which obtained a maximum value of 2080°F.

Most of the fuel failures occurred during the first 3 hours and were located within the GSCS cooled region. Although cladding failure occurs within the GSCS region, the GSCS has stabilized fuel temperatures by 1200 seconds and

eventually provides a very effective means of reducing fuel temperatures. The GSCS limits fuel failures and hydrogen production from the GSCS region to times that are less than 3 hours.

In the uncooled end regions, the local decay powers are much lower. Cooling is limited to radiative heat transfer and axial conduction into the GSCS-cooled region and reflector graphite. A very slow and sustained temperature rise occurs for most of the end-region axial locations. The upward trend of the fuel, cladding, and graphite temperatures for three of the end region axial levels (5, 10, 14) is strongly evident in Figures 15.6-285 and 15.6-286. At approximately 3 hours (10,800 seconds) fuel temperatures in the inlet end regions reach the cladding failure threshold of 1900°F. The end-region within the central zone, eventually attained the maximum predicted temperature for the entire reactor core of 2441°F after 8.9 hours (see Figure 15.6-286). (Axial level 14 corresponds to a position approximately 62 in. from the outside edge of the axial reflector and is located near the center of the end region.) Peak temperatures are eventually stabilized via axial conduction and radiative heat transfer and begin to decrease.

It should be emphasized that fuel failures do occur for the end regions, but only for the inlet half. This is due to the effects of increased water density and control-rod insertions on the neutron; hence, power distribution within the reactor. This results in a power distribution that is skewed toward the inlet end of the reactor. As a result, the linear powers in the fuel within the outlet end regions are lower than the inlet half. The majority of fuel failures occurred before 3 hours and were located within the GSCS region. The small number of fuel failures that do occur after 3 hours are associated with the inlet-half end regions.

Small temperature increases were noted in Figures 15.6-285 and 15.6-286 for several of the end-region axial levels. The first small temperature increase is associated with the inner fuel element reaching the cladding-failure threshold of 1900°F, thereby allowing the uranium metal to oxidize. The resulting exothermic uranium metal/water reaction increased the adjacent radial and axial node temperatures and is the cause of the temperature

increase. The remaining fuel temperature increases were determined to be associated with either the failure of the outer fuel element or of axially adjacent nodes.

The calculated whole-core fuel-failure percentage was 26.2%. The majority of fuel failures were determined to occur before 3 hours, occurred within the high-power GSCS regions, and were concentrated within the inlet half. Of the 26.2% predicted, 5.8% was in the end region and 20.4% was in the GSCS region.

Of the whole-core fuel failures, 69% occurred within the inlet half with most of these in the central zone. Less than 1% of the fringe zone failed. No fuel failures were predicted within the outlet-half end regions.

The TRUMP-BD analyses clearly demonstrate that autocatalytic oxidation behavior is not characteristic of N Reactor fuel under severe accident conditions. The principal reasons for this result are as follows: 21

1. The thermal mass provided by the graphite surrounding the pressure tubes and core peripheries and the large graphite heat capacity limit the extent of heatup.
2. The unique concentric fuel design results in a very large surface-area-to-volume ratio of approximately 9.0. This provides an efficient means of heat transfer either by convective cooling or by radiative transfer to the surrounding surfaces.
3. The bonded design of the fuel elements and increased thermal conductivity associated with metal-versus-oxide fuels result in a much greater ability to transfer heat. Of secondary importance are the much larger uranium fuel inventory and lower steady-state operating fuel temperatures (approximately 900°F), which influence oxidation behavior. The combined decay heat and chemical reaction energy is eventually removed by heat transfer through the pressure tube to the graphite and subsequently to the GSCS. (A system not available in commercial LWRS). The fuel temperatures, as presented by Figures 15.6-285 and 15.6-286 are eventually stabilized and begin to decrease. These combinations provide

for a dramatically different response under severe accident conditions than calculated for LWR systems, thus avoiding autocatalytic behavior.

The maximum hydrogen-generation rate of 1080 ft^3/min corresponds to the timing of the maximum rate of fuel failures. After 2 to 3 hours, the inlet end regions become the primary source of hydrogen and hydrogen generation has decreased to a relatively constant 80 ft^3/min. The minor hydrogen-rate spikes predicted to occur for times greater than 3600 seconds are associated with individual fuel node failures within the end regions.

The quantity of both Zircaloy and uranium metal consumed versus time to produce the 4-bin whole-core hydrogen source term corresponds to a maximum of 2.3% of the total Zircaloy-2 and 0.5% of the total uranium fuel inventories.

The 4-bin hydrogen production is 32,960 ft^3 at 2 hours, 37,570 ft^3 at 3 hours, and 71,950 ft^3 at 10 hours. The hydrogen production was determined to split nearly evenly between the GSCS and end regions given the 3-hour GSCS contribution limitation.

These analyses assume the availability of unlimited steam for the metal reactions. This is a conservative assumption imposed to ensure that the HMDBA represents a conservative bound in terms of hydrogen generation for the given initial conditions and event frequencies. The RELAP5 calculations indicate that steam would no longer be available after 1200 seconds. With steam-limiting arguments, the total hydrogen produced would be about 15,000 ft^3.

A whole-core energy balance was performed. The total energy gain consists of the sum of the decay power and the exothermic Zircaloy and uranium metal/water chemical reactions. The single energy-loss term is the energy transferred out of the core by the GSCS system. The net energy is the difference between the total energy gains and energy losses (i.e., removal via the GSCS).

At time zero (referenced 150 seconds into the HMDBA simulation) the GSCS was determined to be removing approximately 44 MW thermal of energy from the reactor core. This is slightly greater than the minimum nominal GSCS capability of 40 MW thermal, but well within the nominal operating range of 40 to 60 MW thermal. By approximately 4500 seconds the GSCS removes more energy than is being input into the whole core. The stored energy of the core is decreasing. From a whole-core perspective, temperatures should begin to decrease. However, local hot spots within the end regions persist for as long as 10 hours before temperatures begin to decrease.

15.6.6.4 Confinement Performance

The confinement system, performance description includes the typical confinement system steam vent/fog spray/filtered release mode functions (described in greater detail in Section 6.2) and a description of the affects of the use of various options of the fog spray and hydrogen mitigation systems. 21

The input data for the analysis is derived from the RELAP4 cold leg manifold break analysis (Section 15.6.5.3.7) for steam/water mass flow rates and the TRUMP-BD code analysis of core performance with no ECCS and only GSCS cooling available (Section 15.6.6.3). The TRUMP-BD code is used in providing the HMDBA hydrogen source term.

The confinement performance analysis is documented in Reference 55.

15.6.6.4.1 Mathematical Model

CONTAIN is a state-of-the-art containment analysis code developed as part of the United States Nuclear Regulatory Commission's ongoing reactor accident research program. CONTAIN is designed to analyze the physical, chemical, and radiological conditions inside a containment building and the resulting fission product release to the environment following the severe reactor accident.

Many accident analysis codes are limited in the physical phenomena they consider. They typically analyze either thermal-hydraulic behavior, fission

product decay, or aerosol behavior, but they do not consider these phenomena simultaneously. CONTAIN is unique because it simultaneously treats thermal-hydraulic, aerosol, and fission product behavior. This approach can provide a more accurate analysis because of the important closed loop feedbacks which exist among these phenomena. CONTAIN has been used extensively for reactor containment analysis and to the extent possible CONTAIN models have been validated by comparison with available experimental results.

The current release of the CONTAIN computer code is version 1.1. Options provided by CONTAIN 1.1 for modeling flow paths between cells include the ability to specify opening and closing characteristics for those flow paths based on time and pressure. These options are adequate for modeling containment systems which can withstand elevated internal pressures and are designed to remain closed during an accident.
N Reactor utilizes a confinement system which has a low structural design pressure rating. It is designed to vent during the steam release period following a large pipe break and then re-close after the steam release is over. The N Reactor confinement includes a system of vents and dampers for accomplishing this that are controlled by signals based on multiple conditions. The simple time and pressure dependent flow control specifications provided by CONTAIN 1.1 are not adequate to accurately model this type of confinement system.

A special version of CONTAIN, referred to as CONTAIN 1.1N, was developed specifically for application to N Reactor. CONTAIN 1.1N includes options for modeling the flow path logic associated with a confinement system. The calculations herein were performed with the N Reactor version of CONTAIN.

Figure 15.6-287 is a simplified sketch of the confinement building showing the CONTAIN model cell locations. Nine cells are included in the model to represent the confinement building and four additional cells, represent the environment. Details of the model are provided in Reference 55.

The hydrogen mitigation system comprises two features which have not been discussed. The first mitigation system feature is a recirculation duct and

fan between the reactor building front and the pressurizer penthouse to help mix the atmosphere throughout confinement. A constant volume flow rate of 25,000 cfm is specified beginning at 360 seconds after accident initiation. The gas flow direction in the recirculation duct is from the reactor building to the penthouse; the open cross vent between the pipe gallery and the reactor building provides a path for return gas flow.

The second mitigation system feature is a nitrogen purge in the pipe gallery extension to inert confinement and prevent the long-term buildup of a combustible gas mixture. The source begins at 3 hours after accident initiation and maintains a constant flow rate of 3000 cfm at 1 atmosphere and 77°F.

The details of the hydrogen mitigation system effect on hydrogen concentrations is provided in section 15.6.6.5. 21

15.6.6.4.2 Input Parameters and Initial Conditions

The accident assumes a loss of coolant from a large pipe break in the primary cooling system cold-leg manifold. Subsequent failure of the emergency core cooling system (ECCS) is also assumed, but the graphite shield cooling system (GSCS) remains operational. High temperatures in the core following the postulated pipe break result in metal-water reactions and the production of hydrogen. Fuel failure also results in the release of aerosols and radionuclides into confinement.

The steam/water mass flow rate from the pipe break described in Section 15.6.5.3.7 gives the corresponding moisture enthalpy. The break flow peaks 0.6 seconds after accident initiation at a rate of 40,800 lb/sec. Although the flow rate is decreasing thereafter, it remains above 33,000 lb/sec. for several seconds. Steam flow terminates 156 seconds after the break occurs.

The hydrogen source rate (HMDBA) is based on calculations made with the TRUMP-BD computer code (Ref. 52).

Hydrogen generation begins at 135 seconds and reaches a maximum rate of 1080 cfm at 1 atmosphere and 68°F at 720 seconds. Thereafter, hydrogen production decreases rapidly and by 36,000 seconds, which is as far as the TRUMP-BD calculations were carried out, the rate has fallen to about 50 cfm at 1 atmosphere and 68°F. The CONTAIN model assumes that the hydrogen generation rate remains constant from 36,000 seconds on. The limiting analysis for hydrogen mitigation (HMSST) is provided in Section 15.6.6.5 and is based on HECTR analyses.

The fission product and aerosol sources for this analysis are based on a siting source term developed for an analysis performed with the CORRAL computer code (Ref. 56). This source term meets the intent of applicable regulatory guideline requirements for performing siting evaluations of nuclear facilities.

Consistent with Regulatory Guide 1.4, the siting source term for this analysis assumes that 100% of the total fission product inventory is available in the failed portion of the fuel and that 100% of the noble gases and 50% of the iodine are released from the fuel. It is assumed that 50% of the elemental iodine that leaves the failed fuel remains in the primary system, which is a conservative assumption considering the length of piping available for fission product plateout in N Reactor. Therefore, the masses of fission products entering confinement represent 100% of the total noble gas inventory and 25% of the total iodine inventory.

The iodine inventory entering confinement is partitioned according to Regulatory Guide 1.4, which assumes 91% is in the form of elemental iodine (I_2), 5% is in the form of particulate iodine as cesium iodide (CsI), and 4% is in the form of organic iodine as methyl iodide (CH_3I).

Radionuclides, other than the noble gases and the iodine species, are not included in the fission product input for this CONTAIN model, although they are included as aerosol masses.

CONTAIN requires that fission product source rates be specified, so a normalized rate curve was used to generate rate versus time tables that would release the desired total fission product masses. The rate curve for this analysis releases the fission products to confinement between 465 and 1550 seconds with the peak release rate occurring at 700 seconds. The pipe break location (cold leg manifold) assumed for this hypothetical accident is in the center of the pipe gallery. For the CONTAIN N Reactor model, this is cell 5. All of the sources discussed above (i.e., steam/water, hydrogen, and fission product) were released into cell 5 of the model.

The limiting analysis for hydrogen mitigation (HMSST) is provided in Section 15.6.6.5, which assumes the release steam/water and hydrogen at the more conservative locations of the reactor front-pipe barrier space, the pressurizer penthouse, and the pipe gallery extensions.

21

15.6.6.4.3 Results of CONTAIN Analysis

This section presents results from two similar CONTAIN cases for N Reactor with the hydrogen mitigation system. Both CONTAIN cases use the same model and accident conditions; the only differences are spray operation times and run duration. The first case, which was run to a problem time of 30 days, turns the sprays off at 100,000 seconds. The second case, which was run to about 3.5 days problem time, allows the sprays to operate continuously.

The early part of the hypothetical accident sequence is dominated by confinement venting during the steam blow-down followed by a period of inward airflow to replace condensing steam. Initial Release rates for hydrogen and fission products to confinement occur during the period of inward airflow and there is no early release from confinement. Table 15.6-29 shows the timing of significant events for the first 3 hours of the accident. Actions of the vents and dampers that control confinement venting during the steam blow-down can be observed. During the steam condensation period, inflow of air through the stack is insufficient to maintain adequate building pressure, so the vacuum breakers open for a short time. The times shown in Table 15.6-29 are only approximate; their accuracy is limited by the print intervals that were chosen for this calculation.

Pressures are quite uniform throughout confinement except for the filter building, which is isolated during the steam blow-down period by closure of the filter vent damper. The peak pressure occurs in the pipe gallery center at about 10 seconds into the accident. Thereafter, the pressure decreases rapidly and actually falls below atmospheric as steam condensation progresses. The minimum pressure of -0.5 psig is observed at 200 seconds. After that time, the filter vent damper reopens and the building pressure starts rising again slowly.

Pressures stabilized by about 10,000 seconds. Gas flows after that time are very low and small pressure differences between various cells and atmospheric pressure was due primarily to the gravitational head resulting from differences in elevation.

Confinement peak gas temperatures occur at about 10 seconds into the transient. The steam release rate decreases after that time and temperatures begin to fall in most cells due to the cooling effect of the fog sprays. A rapid decrease in reactor building temperatures occur after the initial steam blow down due to the high fog spray flow rates there. Pipe gallery temperatures also decrease, but much more slowly since fog spray flow rates in the pipe gallery are lower.

The general cooling trend continues until the fog sprays are turned off at 100,000 seconds. At that time, all cell temperatures rise rapidly to a higher level and then begin to fall again, although very slowly, as heat losses from the confinement to the atmosphere continue.

Hydrogen concentrations throughout confinement show an early peak in the source volume (cell 5) at about 1000 seconds, which follows shortly after the peak in the hydrogen release rate. Although some peak in the source cell could be expected, the magnitude predicted by this CONTAIN model is exaggerated as comparisons with HECTR have demonstrated. This is due to nodalization in the source region that is too coarse. The other cells show a more gradual increase in hydrogen concentration that continues for about 10,000 seconds. At that time the nitrogen purge is activated, causing

hydrogen concentrations to stabilize and then decrease. Termination of the confinement fog sprays at 100,000 seconds produces no visible effect on the hydrogen concentration curves. All cells in confinement remain below the 4 mole percent lower flammability limit, so a hydrogen burn is not predicted. This is consistent with calculations performed with HECTR and COBRA-NC for similar cases. The limiting analysis for hydrogen mitigation, however, is provided in Section 15.6.6.5, which assumes a larger hydrogen source term (HMSST) at more limiting locations.

Filter building flow is a small positive flow at time zero, which immediately terminates as the filter vent damper is shut to protect vent duct-work and filter elements from high pressures during the steam blow-down. Shortly after 205 seconds, when the confinement pressure has fallen below 3 in. w.g. and the sequential timers have timed out, control logic allows the vent damper to reopen. At this point the confinement pressure becomes negative, which draws air from the outside into confinement. As the outside air replaces condensed steam, the building pressure rises and the flow rate gradually approaches zero. By 10,000 seconds the flow rate returned nearly to zero. At 10,800 seconds the mitigation system nitrogen purge is activated and the filter flow immediately goes positive as expected. 21

Filter building flow reflects downward trend in the flow-rate from the 20,000 seconds to 100,000 seconds. This is caused by a steadily decreasing stack effect as the sprays continue to cool the confinement atmosphere. At 100,000 seconds the sprays are turned off and the confinement atmosphere temperature rises again as it is heated by building structures. This causes buoyancy forces to increase again and the stack flow rises rapidly to a new stable value.

All of the leak flow areas are very small, so the flow rates are low when compared with the previous filter building flows.

There is a high positive leakage flow during the steam blow-down, followed by a period of negative flow as steam condenses and outside air is drawn in. Although the leakage flows approach zero as steam condensation rates decrease,

they stabilize at some small negative value. The exception is one leak near the top of the volume representing the steam generator and auxiliary cells. There is a short period of outward flow through that leak after the nitrogen purge is activated.

Before the sprays are turned off at 100,000 seconds, the negative leakage flows are approaching zero. The leak in the pressurizer penthouse at the top of the pipe gallery and one leak at the top of the steam generator cell volume actually go positive during that time. These trends are consistent with the fact that gas temperatures are decreasing while the sprays are on, resulting in a reduction in the confinement stack effect.

The stack effect causes warm air inside confinement to flow out through the stack, which in turn draws outside air in through the leaks at lower elevations. The stack effect is strong enough to maintain negative flow through the majority of the leaks, even though the nitrogen purge is in operation. At 100,000 seconds when the sprays are turned off, the air inside confinement begins to heat up and the magnitude of the stack effect increases. This reverses the earlier leakage flow trends until a stable condition is established.

There is good agreement between the CONTAIN code analysis results for radioactive material transport and release and the CORRAL code (Ref. 56 and 55). As a result the radioactive dose projections from the hypothetical accident will rely upon the CORRAL code results (Ref. 56) (see also Section 15.6.6.6).

In regard to confinement performance there has been a good fit between the CONTAIN code results and the HECTR code results (described in Section 6.2 and 15.6.6.5) and the CORRAL code. Therefore, the foregoing description is considered meaningful in determining realistic confinement performance, and realistic hydrogen mitigation system success in precluding hydrogen deflagration. The limiting analysis for hydrogen mitigation, however, is provided in Section 15.6.6.5, which assumes a larger hydrogen source term (HMSST) at more limiting (conservative) locations.

Analysis with CORRAL concluded that beyond about 10 to 30 hours, the fog sprays have no benefit in removing fission products from confinement. A similar conclusion can be drawn from the CONTAIN results. For each fission product species, nearly all of the fission product mass that was released from confinement escaped by 100,000 seconds when the sprays were turned off. Therefore, although turning off the sprays at that time has a significant effect on the temperatures and flows, it has a negligible effect on the fission product release. For the case with the mitigation system in operation, running the fog sprays beyond 100,000 seconds (27.8 hours) will produce no benefit in reducing fission product release from the N Reactor confinement.

Summary

There are several conclusions that can be drawn from the results presented in the preceding section. 21

The fission product release predictions for hydrogen mitigation system operation show that most of the release to the environment occurs within 30 hours after accident initiation. Therefore, with the mitigation system in service, operating the sprays beyond about 30 hours will produce no benefit in reducing fission product release from the N Reactor confinement.

Comparisons of predictions with and without the mitigation system in operation show that the non-mitigation case results in a much slower radiological release. Without the mitigation system in operation, significant fission product release is still taking place beyond 30 hours.

Comparisons between two CONTAIN predictions with and without the hydrogen mitigation system in operation show significant differences in fission product release to the environment. The mitigation system causes all fission product species to be release more rapidly than they would be without the mitigation system in operation. This results in a significantly larger total release of molecular and particulate iodine, since there is less residence time in confinement during which the sprays and other settling process can remove

these species from the atmosphere. Although this larger release caused by the mitigation system is undesirable, it is necessary that the mitigation system be operational to insure against the possibility of a hydrogen deflagration or detonation that could breach confinement.

15.6.6.5 Limiting Hydrogen Mitigation Analysis

Following the Chernobyl accident in the Soviet Union in April 1986, several independent reviews were conducted to consider safety issues raised concerning that accident and their relationship to N Reactor. In response to these reviews, a N Reactor safety enhancement program was developed to resolve issues raised by the reviewers. A major part of this program has been to evaluate the effects of hydrogen generated from a postulated degraded-core accident in N Reactor and to resolve any related concerns.

In response to concerns over the hydrogen issue, a hydrogen-mitigation and control system design effort was initiated. A conservative limiting design-basis hydrogen source term, HMSST, was developed, results from hydrogen distribution and mixing studies were evaluated, and numerous alternatives for hydrogen control were considered. On the basis of those efforts, a system was designed and analyses were performed to verify the adequacy of the design.

This section presents results of analyses performed with HECTR to confirm that the hydrogen-mitigation system will prevent conditions that could support a major burn and challenge confinement integrity.

The analyses covered here focused on hydrogen releases in three locations: (1) the reactor front-pipe-barrier space, (2) the pressurized penthouse, and (3) the pipe gallery extension.

15.6.6.5.1 Mathematical Models

Following the accident at Three Mile Island, the Nuclear Regulatory Commission sponsored a program to investigate hydrogen behavior in light water reactor containments. A major part of this program was the development of HECTR (Hydrogen Event Containment Transient Response), which is a computer code to model the transport and combustion of combustible gases during postulated

accidents in Light Water Reactors (LWRs). In order to perform the necessary calculations for N Reactor in a realistic manner, some modifications to the most recent version of HECTR (Version 1.5) were required.

HECTR is a lumped-parameter containment analysis code developed for calculating the containment atmosphere pressure temperature response to combustion. Six gases--steam, nitrogen, oxygen, hydrogen, carbon monoxide, and carbon dioxide--are modeled in HECTR along with sumps containing liquid water. To calculate the pressure, temperature, and composition of gases in a containment or confinement system, the containment is divided into "compartments" with flow between compartments occurring through "junctions." Flows between compartments are pressure and buoyancy driven with inertial and resistance terms included. Steam is treated as a real gas, and the other gases are treated as ideal. Gases in each compartment are instantaneously mixed, and source terms are user-specified. Simplified conservation equations are solved to determine compartment and junction conditions during the transient. The thermal response of surfaces and equipment in the confinement can also be calculated in HECTR, using either one-dimensional finite difference slabs or lumped masses. Models are included to calculate combustion, radiative and convection heat transfer, steam condensation or evaporation, and leakage. The engineered safety features (ESFs) modeled in HECTR are sprays, fans, and sumps. 21

The basic physical models in HECTR remain unchanged for version 1.5N. Most of the changes are in the logic models dealing with system actuation and junction opening and closing.

The spray model has been altered to allow multiple spray trains with separate actuation criteria for each train. Each train can now be actuated based on combinations of pressure or temperature signals from different locations. The spray trains can be individually turned off and restarted at later times.

Two additional types of leaks are added to treat the steam vents and vacuum breakers at the N Reactor. The steam vent option allows the vents to open based on exceeding a specified pressure and to later close based on a timer.

The closure can be specified to occur over an additional time interval. Different criteria can be used for individual steam vents, if desired. The vacuum breaker option allows a leak to open as a function of differential pressure between the compartment and the outside atmosphere. Tabular input is used to define the flow area versus differential pressure relationship.

An additional junction option was added for the cross vents between the 105 and 109 buildings at N Reactor and for the filter building isolation valves. This option allows a junction area to be controlled by trips and tables. Different trips and tables can be specified for opening or closing the junction. For example, a junction could be specified to open according to a flow area versus differential pressure table until a compartment pressure exceeds a setpoint, then reclose following a flow area versus time table. Finally, this option allows a door to "blow out" if a third trip (e.g., the differential pressure in a particular direction exceeding a specified value) is satisfied. After blowing out, the junction remains fully open for the remainder of the calculation.

Three separate HECTR models were used for the analyses. These models were developed to provide additional detail in the regions near the source release point. The models are described separately below. A more detailed description of the confinement system and the various accident-mitigation functions is given in Reference 58, 61 and 62.

Front Pipe Barrier Space Release

A pipe break and subsequent hydrogen release in the reactor front-face pipe-barrier space was analyzed using the 49-vol HECTR model shown in Figure 15.6-289. Volumes 1 to 18 represent the reactor building. The front and rear pipe-barrier spaces are represented by volumes 16 and 17, respectively, while the graphite gas space is represented by volumes 18. The break location was in volume 16 of the model. For this analysis, volume 18 does not interact with the other confinement volumes because the blowout panel that isolates it remains in place. The pipe gallery is modeled by volumes 19 to 28 and volumes 42 to 44. The six steam generator cells and the auxiliary system cells are represented by volumes 30 to 36. Volume 37

represents the filter building and volume 38 represents room 605, which connects the reactor building to the filter building. Volumes 29, 39, 40, 41, and 45 through 48 represent the pressurized penthouse. Volume 49 represents the hydrogen mitigation system recirculation duct between the reactor building front and the pressurizer penthouse.

Appropriate flow junctions are included to connect adjacent volumes and to represent the vents, ducts, doors, and sumps between the various regions of the confinement space. This model includes new junctions between the front-pipe-barrier space and the reactor building volumes. These new junctions model the lower blowout panels located at the top of the pipe-barrier spaces, which were included in previous calculations. The new blowout panels at the bottom have been added by replacing personnel access door latches which shear pins. In the event of a break in the pipe-barrier spaces, openings at both the bottom and the top will improve circulation and decrease the maximum hydrogen concentration there. To remain within HECTR's current blowout panel limit, this model does not include blowout panels in the rear-pipe-barrier space. 21

Special types of confinement leaks are used to model the steam vents and vacuum breakers. Sprays are included in the reactor building and the pipe gallery. The pipe gallery spray actuation logic includes a 44 second delay between confinement trip and spray activation to account for the time required to attain full flow conditions. The reactor building spray delay time was increased to 700 seconds in this particular model to avoid early code convergence problems induced by the spray model. This adds some conservatism to the calculation because it represents an additional time during which the mixing effects of the sprays are not included.

The design features of the hydrogen mitigation system are represented as follows. A fan is modeled at the junction between the reactor building front (volume 23) and the mitigation system recirculation duct (volume 49) to provide forced recirculation from the reactor building to the pressurizer penthouse. The nitrogen-purge system is modeled by specifying an appropriate nitrogen source rate in the volume corresponding to the nitrogen-purge

location. The nitrogen purge in this model is located in volume 28 in the pipe gallery; the specified purge rate is 3000 ft ^{3}/min beginning 3 hours after the start of the transient.

The steam source used in this model represents a front riser break and the hydrogen source is the HMSST. Plots of these source rates are included in Figures 15.6-290 and 15.6-291 respectively.

Pressurizer Penthouse Release

A pipe break and subsequent hydrogen release in the pressurizer penthouse was analyzed using the 65-volume HECTR model shown in Figure 15.6-292. The volume numbering scheme for this model is identical to that already described for the 49-volume model except for the additional detail within the pressurizer penthouse. This model divides the pressurizer penthouse into 24 volumes to provide better characterization of the flow patterns induced by the recirculation fan and the hydrogen source.

As Figure 15.6-292 indicates, the pressurizer penthouse is represented by volumes 29, 39, 40, 41, 45 to 48, and 50 to 65. The mitigation-system recirculation duct (volume 49) connects volume 2 of the reactor building front to volume 46 of Figure 15.6-292.

The modeling of vents, ducts, doors, and sumps between various compartments is identical to the 49-volume model except for the blowout panel modeling in the pipe-barrier spaces. Only the upper blowout panels for the front- and rear-pipe-barrier spaces were included in this model, to avoid exceeding HECTR's current blowout panel limit. This is not considered an important deficiency since the main interest in this calculation is with the behavior in the pressurizer penthouse. The modeling of steam vents, vacuum breakers, and fog sprays is identical to that used in the 49-volume model. The hydrogen-mitigation system includes a recirculation fan between volumes 2 and 49, and a nitrogen source in volume 28. The fan actuation logic includes a 120-second delay between confinement trip and fan activation. The 3000 ft^3/min nitrogen purge is set to start 3 hours after the confinement trip.

The steam source used in this model represents a cold-leg manifold break and the hydrogen source is the HMSST. Flow rates for these sources are shown in Figure 15.6-293 and 15.6-291 respectively.

The break flow was located in volume 24 of the pipe gallery directly below the pressurizer penthouse.

Emergency Dump-Tank Vent Release

For transients such as a complete loss of AC power, the emergency core-cooling system (ECCS) will be activated and the primary system will be depressurized through the V4 valves into the emergency dump tank. If the ECCS flow fails to enter the core, the fuel will heat up and hydrogen will be generated and released to the emergency dump tank. The hydrogen release through the emergency dump-tank vent line was analyzed using the 51-volume HECTR model shown in Figure 15.6-294. The volume numbering scheme for this model is identical to the 49-volumes model except in the pipe gallery extension. Two volumes were added there to provide three vertical levels within that region so that the sensitivity of results to hydrogen source elevation could be evaluated. 21

The 51-volume model includes some modifications to the trip logic controlling the action of vents and doors between internal compartments and between confinement and the environment. These changes represent an improvement to the modeling of these components. Confinement sprays are modeled, but were not activated in the cases reported here. This is because the confinement pressure remained below the spray actuation limit since there was no steam source into confinement for this postulated accident.

The mitigation-system recirculation and exhaust fans are modeled, but a nitrogen purge is not included. The fan-actuation logic included a 240- to 360-second delay between confinement trip and fan actuation. Only one of the cases evaluated with this model extended beyond 3 hours, when the nitrogen purge is scheduled to start; that case was terminated at 3 hours and 20 minutes. The effect of the nitrogen purge would be negligible in this time frame.

A steam source was not included in the model. As described earlier, this accident sequence represents an event such as loss of all AC power, which causes the dump valves to open and the reactor cooling-system inventory to be discharged to the emergency dump tank. Water in the dump tank condenses the steam, so there is no steam source to confinement. Noncondensible gases, such as hydrogen, enter confinement through the emergency dump-tank vent line.

The hydrogen source used with this model was derived from the HMSST by accounting for mixing of hydrogen with air in the dump tank prior to being released through the vent line into the pipe gallery extension. The flow rate for the hydrogen source is shown in Figure 15.6-295.

The hydrogen source was released into volume 19 near the bottom of the pipe gallery extension.

15.6.6.5.2 Input Parameters and Initial Conditions

Table 15.6-49 summarizes the HECTR analyses performed to assess the consequences of hydrogen releases in three major locations: (1) the reactor front-face pipe-barrier space, (2) the pressurizer penthouse, and (3) the pipe gallery extension. The HMSST was used as the hydrogen source for all calculations except the final one. That case used an emergency dump-tank vent-line hydrogen source, which was derived by accounting for mixing of the HMSST source with air in the dump tank prior to release into confinement. The steam source rates varied depending on the break location.

15.6.6.5.3 Analysis Results

The HECTR analyses described in this section assessed the consequences of hydrogen releases in three major locations within N Reactor confinement. The locations evaluated were the reactor front-face pipe-barrier space, the pressurizer penthouse, and the pipe gallery extension.

In the front-pipe-barrier space, HECTR predicted a peak hydrogen concentration of 7 mol% (Figure 15.6-297). This is above the lower flammability limit but below the hydrogen-detonation limit. The hydrogen concentration confirms

the effectiveness of new blowout panels that were added to the pipe-barrier spaces to improve natural circulation and reduce the peak hydrogen concentration in the event of a pipe break in that location. HECTR analyses have confirmed that a deflagration in the pipe-barrier space would not challenge confinement integrity (Ref. 59). Figure 15.6-296 demonstrates that the pressure peak produced by the burn in the front-pipe-barrier space, approximately 0.25 psig, was well below the pressure produced by the initial steam blowdown. Therefore, confinement integrity would not be challenged. In addition, a comfortable margin exists between the predicted peak hydrogen concentration and the hydrogen-detonation limits.

In the pressurizer penthouse, HECTR analyses performed with a detailed model predicted peak hydrogen concentrations of 8 mol% (Figure 15.6-298). The HECTR results do not compare well with predictions from COBRA-NC/NAI (Ref. 60). The detailed distributed-parameter approach provided by COBRA-NC/NAI appears necessary to adequately model the complex flow patterns within the pressurizer penthouse. 21

Consequences of a hydrogen deflagration within the pressurizer penthouse have been evaluated. A deflagration does not pose a threat to confinement integrity because of the limited volume within the penthouse that is above the flammability limit. However, the COBRA-NC/NAI calculations indicate that a significant volume near the top of the penthouse could also be above the detonation limit.

These results are extremely conservative because the HMSST assumes the instantaneous release of hydrogen in the core at the break and because of certain other conservative modeling and input assumptions. Additional analyses of the pressurizer penthouse are ongoing as part of the SEP to verify the effectiveness of the hydrogen mitigation system to maintain hydrogen concentration below 10 vol%. Analyses have shown hydrogen burns at this level to be acceptable (Ref. 59).

An analysis of possible hydrogen generated from a worst case, adiabatic core heat-up accident was performed (Ref. 51). This accident is identical to the HMDBA except that the GSCS is also assumed to fail, leaving no means of

reactor core heat removal. Using this hydrogen source term, an analysis showed that with full flow from both redundant nitrogen purge system trains (40,000 ft^3/min starting 15 minutes after the accident), the confinement will be inerted (i.e., oxygen concentrations below 5 vol%) before the general confinement hydrogen concentration reaches the lower flammability limit of 4 vol% (Ref. 51). This analysis demonstrated the margin available at N Reactor to safely accommodate accidents more severe than the HMDBA.

In the pipe gallery extension, HECTR was used to evaluate an accident sequence involving a blowdown to the emergency dump tank and subsequent release of hydrogen through the dump-tank vent line. Those analyses indicated that with the vent-line exit located near the bottom of the pipe gallery extension peak hydrogen concentrations are below the lower flammability limit (Figure 15.6-299). The HECTR predictions were confirmed with a detailed COBRA-NC/NAI model.

In summary, the following conclusions have been demonstrated by the analyses:

- The Hydrogen Mitigation System recirculation fans are adequate to control the hydrogen mitigation design basis accident (HMDBA) (See Section 15.6.6.4.3).

- The Hydrogen Mitigation System Source Term (HMSST), which is larger than the HMDBA (see Section 15.6.6.3), results in the potential for limited hydrogen burns in two localized areas (pressurizer penthouse and front-pipe barrier space) but which do not challenge confiner integrity. Analyses are continuing for the pressurizer penthouse using a more realistic hydrogen source location.

- For accidents beyond the HMSST, the nitrogen inerting portion of the Hydrogen Mitigation System has sufficient capacity to mitigate the accident. Nitrogen inerting of the confiner will increase accident doses, so emergency procedures will be developed from implementation of manual control of the nitrogen purge system.

- Additional details are provided in References 58 through 62.

15.6.6.6 Radiological Consequences

The radiological consequence analysis is based on Reference 56 which utilizes the CORRAL computer code, that, when compared to the CONTAIN computer code overestimates radioactive releases. This is due to the CONTAIN results demonstrating that fission products are retained in the N Reactor confinement for a longer time than predicted by the CORRAL code. This longer retention time allows for greater natural deposition, retention due to the fog spray system, and radioactive decay. Therefore the use of the CORRAL computer code to project the radiological consequences of the hypothetical accident is conservative.

The CORRAL code analysis tool was used to characterize both a siting source term and best estimate source term release. The site boundary and low-population dose estimates for both source terms, with and without hydrogen mitigation system operation have been developed and are presented in the following sections. The control room dose estimates are provided in NUSAR Section 6.4 for the performance of the dedicated control room habitability system. 21

The siting source term used in the CORRAL code analysis described herein is the same as the source term used in the CONTAIN analysis described in the previous section (15.6.6.4.2).

An additional conservatism introduced in the CORRAL analysis is the postulation of the fission product release occurring at the reactor front face (105-N) closer to the filter building and stack than where the cold-leg manifold is actually located (109-N pipe gallery). The cold-leg manifold steam blowdown rates, hydrogen generation rates, and fission product release rates were simply transposed to this more limiting location for conservatism. The cold-leg manifold break input data used are consistent with the CONTAIN analyses described in the previous section, 15.6.6.4.

15.6.6.6.1 Mathematical Models

The CORRAL code is a multicompartment computer model which evaluates transport, deposition, and leakage of airborne fission products during postulated accidents. The CORRAL code was originally developed to support the Rasmussen Reactor Safety Study. An N Reactor model of CORRAL (Ref. 20) was developed to support the N Reactor Updated Safety Analysis Report in 1978. The N Reactor version of CORRAL (CORRAL-N) was developed to properly describe the complex fog spray systems used in the N Reactor confinement building.

The CORRAL models variable fission-product source terms, leakage from any compartment as a function of time, inter- and intra-compartment filtering, natural deposition in all compartments, water spray washing, and filtration prior to stack release. Natural fission product deposition and spray washout employ mass transfer models as well as empirical data generated as part of the Containment Systems Experiment program.

The major consideration in the use of CORRAL is that time-dependent thermal-hydraulic behavior of confinement must be provided as input. Of specific importance is accurate time-dependent confinement leak rates. The CORRAL code does not correct for radioactive decay of isotopes during transit through the confinement system. Although spray washout for particulates is the dominant early-time removal mechanism, CORRAL conservatively assumes that when the airborne particulate fraction is reduced to 1%, then the sprays become ineffective and no more particulate removal is allowed.

Based on a unit fission product source to confinement, CORRAL tracks the transport and retention of inert gases (noble gases and organic iodide), molecular iodine (I_2), and all remaining radionuclides (e.g., Cs, Rb, Te, Ba, Sr, Ru, La) as particulates. Each of these groups exhibit different physical and chemical behavior in natural deposition and removal by fog sprays. The code then provides the fraction of unfiltered fission product inventory leaked from confinement at ground level, the fraction of filtered fission product inventory leaked through the stack, the fraction of fission

product inventory present in the filters, and the fraction of fission product inventory washed and plated out as functions of time.
The CORRAL code does not consider radioactive decay of fission products during transit through the confinement system. Since radioactive decay during holdup in the confinement system is allowed in Regulatory Guide 1.4, radioactive decay for each radionuclide is treated outside of the CORRAL code. Radioactive decay is calculated for each radionuclide from the time of release from the fuel until the time of release from confinement. However, transit time in the primary system is not considered (i.e., the only transit time considered is the transit time through confinement as determined by the CORRAL code).

15.6.6.6.2 Input Parameters and Initial Conditions

The accident scenario used for evaluation of the N Reactor confinement operation and associated radiological consequences with the hydrogen mitigation system was the Hydrogen Mitigation Design Basis Accident (HMDBA). (Ref. 52). The postulated HMDBA was assumed to be a loss of coolant accident (LOCA) initiated by a double ended break of the cold-leg manifold in the N Reactor pipe gallery (109-N building) coupled with a non-mechanistic failure of the emergency core cooling system (ECCS). The graphite shield cooling system (GSCS) was assumed to be operational during the course of the accident. However, to evaluate the radiological consequences resulting from accident situations where there is a different fission product release location than the pipe gallery (109-N building), the HMDBA hydrogen source term, steam release, and fission product release for the same time sequence were assumed to be released from a front riser in the reactor front face (105-N building). These fission product and thermodynamic sources are not necessarily characteristic of a front end riser break; however, they will represent conservative input assumptions and do provide the ability to evaluate radiological consequences for at least two fission product release locations.

21

Siting Source Term Assumptions

Regulatory Guide 1.4 states that 100% of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core and

25% of the radioactive iodine inventory should be assumed to be immediately available for leakage from the primary reactor containment.
For N Reactor the highest fuel burn-up occurs in the central portion of the fuel column. During operation and during design basis postulated accidents, the outer ends of the fuel column are at lower temperatures than the central portion (Reference 52). Therefore, if fuel failure were initiated by a loss of coolant accident, only part of the N Reactor fuel would reach failure temperature. The fission products in the unfailed portion of fuel would not be involved in the fission product release. However, to be consistent with Regulatory Guide 1.4 recommendations for siting calculations, it will be non-mechanistically assumed for the siting source term that 100% of the total fission product inventory is available in the failed portion of the fuel. Also as specified in the regulatory guides, it was assumed that 100% of the noble gases are released and 50% of the iodine is released from the failed fuel.

It was assumed that 50% of the elemental iodine that leaves the failed fuel would be plated out in the primary system by inherent mechanisms. This assumption is conservative considering the length of primary piping available for plateout removal in N Reactor. Thus, for the HMDBA siting source term, it was assumed, consistent with Regulatory Guide 1.4 assumptions, that 100% of the total noble gas inventory and 25% of the total iodine inventory are released to the N Reactor confinement.

Of the iodine inventory released to confinement, Regulatory Guide 1.4 assumptions are used; namely, 91% is assumed in the form of elemental iodine, 5% is assumed in the form of particulate iodine (CsI), and 4% is assumed in the form or organic iodides (methyl iodine).

The radionuclide inventory available from N Reactor fuel is provided from a RIBD calculation (see Appendix 15.6B) for N Reactor fuel with an exposure of 2.81×10^5 MWd. The important fission products, in terms of radiological consequences, together with the end of irradiation cycle activity (Ci) are provided from the RIBD calculation.

The source term (Ci) entering confinement for each radionuclide is obtained by multiplying the activity of each radionuclide derived from the RIBD calculation by the appropriate fraction of fission products entering confinement assumed by Regulatory Guide 1.4 as discussed above. For the siting source term, only the noble gases and iodine species of the RIBD calculation were used since they are limiting contributors to dose.

Best Estimate Source Term Assumptions

For N Reactor, the highest fuel power occurs in the central portion of the fuel, and during operation the outer ends of the fuel are initially at lower powers (Reference 52). Consequently fuel failure for high power fuel elements would not mean fuel failure for all elements in a pressure tube when the graphite and shield cooling system is functioning. Thus, it is conservatively assumed for the best-estimate HMDBA source term that 50% of the total fission product inventory is available in the failed portion of the fuel. Of the 50% of the total fission product inventory available in the failed portion of the fuel, it is further assumed that 100% of the noble gases are released and 50% of the iodine is released. In addition, it is assumed that 30% of the cesium in the failed portion of the fuel is released along with 100% of the tellurium, 5% of the ruthenium and 1% of all other volatile fission products. These assumed release rates are based on an evaluation of available data and limited fuel melt experiments (Reference 41). 21

An evaluation of the chemical and physical forms of fission products available from N Reactor fuel under severe accident conditions suggest that the predominant form of iodine will be cesium iodide. Therefore, it has been assumed for the best-estimate fission-product source term that 95% of the iodine released from the failed fuel will be in the form of particulate cesium iodide (CsI) and 5% will be in the form of elemental iodine. Of the elemental iodine that reaches confinement, it is assumed that 1.6% reacts with organic material to form organic iodide (methyl iodide). Because of the length of primary system piping that is at lower temperatures than the fuel and because of the lower transit flow velocities through these larger diameter pipe systems, there will be substantial retention of particulates due to chemi-absorption and internal deposition. Experimental data from the

Marviken Large Scale Aerosol Transport Tests program and the LWR Aerosol Containment Experiments program, demonstrated that the characteristics of aerosol material passing through representative interface piping are changed significantly. Deposition and/or resuspension of aerosols in the primary systems piping tend to produce large particles, that are then easily deposited in lower flow velocity regions, such as the inlet or outlet downcomers in the N Reactor primary piping system. Based on the experimental data, it is assumed that 95% of the particulates released from the failed fuel will be retained in the N Reactor primary piping system. As before, it is assumed that 50% of the elemental iodine that leaves the failed fuel will be plated out in the primary system by inherent mechanisms.

The fraction of total fission product inventory that enters confinement for the best-estimate fission-product source term is provided in Table 15.6-30a. The source term (Ci) entering confinement for each radionuclide is obtained by multiplying the activity of each radionuclide derived from the RIBD calculation by the appropriate fraction of fission products entering confinement from Table 15.6-30a.

Both the Siting Source Term and the Best Estimate Source Terms utilize the following assumptions relative to confinement performance, atmospheric diffusion and dose conversion.

Radioactive decay during transit and holdup in confinement is considered in these analyses as allowed in Regulatory Guide 1.4. Similarly, reduction in fission product inventory due to engineered safety features is also considered. Specifically, credit has been taken for the reduction in fission product inventory due to confinement sprays and the filter system. The N Reactor filter system efficiencies have been thoroughly evaluated. Consistent with this evaluation, filter efficiencies of 99.95% for particulates, 99% for elemental iodine, and 70% for methyl iodine have been used. In addition, inherent retention mechanisms in confinement (such as deposition, agglomeration, plate-out, etc.) have been considered. The aerosol particle size assumed for these analyses was an aerodynamic mass medium diameter of 1.5 μ with a standard deviation of 2.

Regulatory Guide 1.4 allows an assumed leak rate of containment consistent with the technical specification requirement at peak accident pressure. For a filtered vented confinement with large volume, such as N Reactor, a containment code must be used to describe the time dependent transit, retention, and filtered and unfiltered leak rate of fission products. For these analyses the CORRAL code (Reference 20) was used.

The N Reactor technical specifications leak rate of 200 ft^3/min at 0.1-in. w.g. pressure was used to define the leak rate area input for each of the computational volumes in the code. A recent confinement pressure test demonstrated that the leak rate of N Reactor confinement is better than the technical specification requirement by approximately a factor of 3 (65 ft^3/min at 0.1-in. w.g.).

Regulatory Guide 1.4 also provides acceptable assumptions for atmospheric diffusion and dose conversion. Table 15.6-30b, provides recommended dispersion factors, wind speeds, and receptor breathing rates for various periods following the accident. The 0 to 2 hour parameters correspond to a low wind speed with very stable wind direction and mixing conditions leading to a worst-case concentration at the receptor. Subsequent time periods assume shifting winds with successively higher wind speeds and increasing mixing leading to lower concentrations at the receptor. The 0 to 8-hour period is assumed to start at the beginning of the active part of the day, with a correspondingly high breathing rate. The following 16-hour period assumes a resting breathing rate followed, after the first 24 hours, by a time-averaged breathing rate. 21

Other assumptions provided by Regulatory Guide 1.4 are utilized with the exception that Regulatory Guide 1.4 suggests that no correction should be made for radiological decay of iodine in transit to the receptor. This assumption was originally provided in Regulatory Guide 1.4 in order not to neglect radioactivity associated with the decay of mother-daughter chains. However, modern radiological consequences computer codes are able to adequately keep track of mother-daughter decay chains with their associated

radioactivity. Thus, decay of all radionuclides in transit to the receptor is considered, and all decay chains are properly accounted for.

Regulatory Guide 1.4 also requires that site specific meteorology be used where available. For these calculations, a new set of site-specific dispersion factors, X/Q, were generated in accordance with the methods recommended in Regulatory Guide 1.145 using the NRC PAVAN code. These site-specific dispersion factors were calculated based on meteorological data for April 1, 1982 through March 31, 1986.

Based on the topography in the vicinity of N Reactor, site-boundary radiological consequences were evaluated using the site-boundary diffusion estimates computed from wind data from the N Reactor telemetry station. The low-population zone radiological consequences were evaluated using the LPZ diffusion estimates computed from wind data from the Hanford Meteorology Station tower. The ground level release dispersion factors in Table 15.6-30c are somewhat lower than those derived from Regulatory Guide 1.4 at short times, but tend to be higher for longer times.

The site boundary receptor for N Reactor was assumed to be located 5.5 mi (8,850 m) from the source and was assumed to be evacuated 2 hour after the accident. The LPZ receptor was assumed to be located 20 mi (32,000 m) from the release point and was assumed to be present for 30 days following the accident. The 20 mi LPZ was located 3/4 of the distant to the nearest population center. The actual distance to the outskirts of the nearest town (Richland) is approximately 25 mi with the intervening space being mainly an unpopulated semi-arid region.

Submersion (whole-body) doses were calculated using the SUBDOSA code that assumes a semi-infinite cloud (thermodynamic equilibrium) model. This model tends to be very conservative, as real plumes are finite in extent and the radionuclide concentrations decrease away from the plume centerline. The calculated external gamma dose is the gamma dose in tissue at a depth of 5 cm. This is also a good approximation of the external gamma dose to the internal organs commonly of interest (including the thyroid). The inhalation

doses were calculated using the DACRIN computer code. DACRIN incorporates a physiological transport model that tracks each radionuclide within the body according to solubility type and calculates doses based on location within the body, activity, and radiation produced. Both codes account for decay in transit to the receptor. DACRIN and SUBDOSA computer codes include any buildup of daughter radionuclides during transit.

Both the iodines and the noble gases are included in the calculation of submersion (whole body) doses. Only the iodines are included in the inhalation dose calculations because nuclides generated by noble gas decay do not contribute significantly to the thyroid dose. No credit was taken for plume transit time with regard to determining the exposure time, X/Q, wind speed, or breathing rate at either the site boundary or the LPZ. The site boundary receptor was therefore exposed to the full first 2 hours of the plume. The conditions at the LPZ were determined without taking credit for plume travel time; i.e., the clock starts at the time of plume arrival. This is common practice for commercial power reactors, as the site boundaries are usually close enough that plume transit time is not significant compared to the two hour receptor residence required by 10 CFR 100. 21

15.6.6.6.3. Radiological Consequences Results

The results of the radiological consequence evaluation are presented in Tabular format in Table 15.6-31. This table includes both the siting source term and best-estimate source term results for the site boundary and low population zones, with and without the hydrogen mitigation system in operation. The difference between the whole body dose for the two source terms is strictly the assumption of 100% verses 50% availability of the total fission product inventory in the failed portion of the fuel. For the thyroid dose, there are two items responsible for the differences. First, is the assumption of 100% verses 50% availability of the total fission product inventory in the failed portion of the fuel. The second is the assumption in the best-estimate source term that the dominant iodine form is cesium iodine particulate, because there are several very efficient particulate retention mechanisms (i.e., primary system, fog sprays, plateout, and filters).

An evaluation of the effects of various hydrogen mitigation systems operating modes during and following the maximum hypothetical accident on offsite dose have been performed (Ref. 56).

Operational parameters which were studied included varying nitrogen purge supply and operation or non-operation of the hydrogen mitigation exhaust fan. The recirculation flow was consistently held at 25,000 ft^3/min.

As expected, large nitrogen purge rates cause fission products to be released from confinement early in time. Hence, there is a significant increase in the site boundary (2 hour) whole body and thyroid dose from early stack release. However, for large nitrogen purge flow rates, operation of the recirculation fan drawing flow back into confinement results in smaller site boundary dose. The operation of the exhaust fan eliminates the contribution from unfiltered building leakage. The increase in the LPZ doses due to a large nitrogen purge flow rate are not as evident due to the large LPZ (20 mi) and the fact that the LPZ doses are averaged over 30 days. Even with large nitrogen purge rates, the low population doses are still well within the 10 CFR 100 guidelines. These results demonstrate that if a large nitrogen purge rate were required to control the presence of hydrogen, there would be no dramatic increase in the radiological consequences to the general public. Appropriate emergency procedures and controls are implemented for operation of the nitrogen purge system.

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TABLE 15.6-6

DOSE TO CONTROL ROOM PERSONNEL
FOLLOWING POSTULATED SINGLE
PRESSURE TUBE EVENTS*

	DOSE FOR DURATION OF ACCIDENT - REMS	
	COMPLETE FLOW BLOCKAGE	FUEL EJECTION
Whole Body Dose		
Exposure to Plume	0.134	0.006
Immersion From Inleakage	0.789	0.004
TOTAL	0.923	0.010
Thyroid Dose		
Immersion From Inleakage (Operators supplied with fresh air)	10.7	0.031
Skin Dose	10.1	0.059

*10 CFR 50 Appendix A, Criterion 19 dose requirements:
whole body - 5 rem and thyroid - 30 rem.
Does not include control room habitability project modification effect on Operator Dose. See Section 6.4.

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TABLE 15.6-16

RELAP4

COLD-LEG BREAKS RESULTS SUMMARY

21

Case	Time[a] (seconds)	Cold-Leg Riser[b] Pressure	Maximum[c] Fuel Temp.	Time[d] (seconds)
Double-ended cold-leg manifold D.C. = 1.0	43 sec	306 psia (REFLOOD)	1078°F	43 sec
Double-ended cold-leg manifold D.C. = 0.6	48 sec	(REFLOOD)	1106°F	48 sec
100% Cold-leg manifold	52 sec	(REFLOOD)	1106°F	52 sec
Double-ended cold-leg crossover	125 sec	(REFLOOD)	1423°F	125 sec
40% cold-leg crossover	146 sec	(REFLOOD)	1427°F	146 sec
20% cold-leg crossover	172 sec	(646 psi)	1050°F	168 sec
Double-ended cold-leg riser	140 sec	(REFLOOD)	1025°F	110 sec
20% cold-leg riser	150 sec	(650 psi)	860°F	150 sec

(a) Time problem was terminated.
(b) Pressure when problem terminated.
(c) Maximum fuel temperature in average power channel.
(d) Time at which maximum fuel temperature was observed.

TABLE 15.6-17

RELAP4

HOT-LEG BREAKS RESULTS SUMMARY

Case	Time[a] (seconds)	Cold-leg Riser[b] Pressure	Maximum[c] Fuel Temp.	Time[d] (seconds)
Double-ended hot-leg manifold D.C. = 1.0	50 sec	306 psia (REFLOOD)	750°F	2 sec
Double-ended hot-leg manifold D.C. = .6	73 sec	(REFLOOD)	750°F	2 sec
100% hot-leg manifold	75 sec	(322 psi)	750°F	2 sec
Double-ended pump suction	75 sec	(630 psi)	753°F	2 sec
100% pump suction	75 sec	(617 psi)	753°F	2 sec
Accidental dump	100 sec	(422 psi)	750°F	2 sec
Double-ended hot-leg riser	20 sec	(1100 psi)	780°F	2 sec
Steam generator double-ended D.C. = 1.0	75 sec	(660 psi)	750°F	3 sec

(a) Time problem was terminated.
(b) Pressure when problem was terminated.
(c) Maximum fuel temperature in average power channel.
(d) Time at which maximum fuel temperature was observed.

TABLE 15.6-18

POWER DECAY FACTORS FOR MARK-IA FUEL 29 MK SCRAMMABLE RODS

RELAP4 INPUT

21

Time (Seconds) From Accident	Inner Fuel	Outer Fuel	Pressure Tube	Graphite
0.0	0.2345	0.7233	0.0072	0.035
2.0	0.2345	0.7233	0.0072	0.035
2.1	0.2023	0.6144	0.0081	0.0357
2.2	0.1550	0.4688	0.0066	0.0268
2.3	0.1162	0.3489	0.0054	0.0227
2.4	0.0902	0.2691	0.0046	0.0188
2.5	0.0737	0.2180	0.0041	0.0163
3.0	0.0459	0.1327	0.0031	0.0120
4.0	0.0402	0.1156	0.0029	0.0109
5.0	0.0365	0.1044	0.0027	0.0101
6.0	0.0337	0.0961	0.0026	0.0095
7.0	0.0315	0.0893	0.0025	0.0091
12.0	0.0247	0.0694	0.0021	0.0076
22.0	0.0195	0.0543	0.0018	0.0064
32.0	0.0168	0.0463	0.0016	0.0057
42.0	0.0150	0.0412	0.0015	0.0052
52.0	0.0137	0.0374	0.0014	0.0048
102.0	0.0103	0.0277	0.0011	0.0038
202.0	0.0085	0.0227	0.0010	0.0033
302.0	0.0076	0.0203	0.0009	0.0030
402.0	0.0070	0.0186	0.0008	0.0027
502.0	0.0066	0.0176	0.0007	0.0026
1002.0	0.0055	0.0146	0.0006	0.0022

TABLE 15.6-19

POWER DECAY FACTORS FOR MARK-IV FUEL 29 MK SCRAMMABLE RODS

RELAP4 INPUT

Time (Seconds) From Accident	Inner Fuel	Outer Fuel	Pressure Tube	Graphite
0.0	0.2366	0.7295	0.0056	0.0283
2.0	0.2366	0.7295	0.0056	0.0283
2.1	0.2047	0.6250	0.0063	0.0284
2.2	0.1572	0.4764	0.0052	0.0227
2.3	0.1181	0.3559	0.0042	0.0180
2.4	0.0921	0.2756	0.0036	0.0148
2.5	0.0754	0.2242	0.0032	0.0128
3.0	0.0474	0.1382	0.0025	0.0093
4.0	0.0415	0.1204	0.0023	0.0084
5.0	0.0376	0.1087	0.0021	0.0079
6.0	0.0347	0.1001	0.0020	0.0074
7.0	0.0324	0.0931	0.0019	0.0070
12.0	0.0254	0.0723	0.0016	0.0058
22.0	0.0201	0.0566	0.0014	0.0049
32.0	0.0173	0.0482	0.0013	0.0043
42.0	0.0154	0.0429	0.0012	0.0040
52.0	0.0140	0.0389	0.0011	0.0037
102.0	0.0105	0.0288	0.0009	0.0029
202.0	0.0087	0.0236	0.0008	0.0025
302.0	0.0078	0.0211	0.0007	0.0023
402.0	0.0071	0.0193	0.0006	0.0021
502.0	0.0068	0.0183	0.0006	0.0020
1002.0	0.0056	0.0151	0.0005	0.0016

TABLE 15.6-20

SEQUENCE OF EVENTS

COLD-LEG CROSSOVER BREAKS (RELAP4)

21

Event	Double-Ended Time (sec)	40% Split Time (sec)	20% Split Time (sec)	Four Cell 55% Split Time (sec)
Pipe Break	0	0	0	0
Low Pressure Scram Trip	<.1	<.1	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1	.1	.1
Control Rods Start In and Pumps Start coastdown	~2	~2	~2	~2
Control Rods 75% Inserted	3.5	3.5	3.5	3.5
Pressurizer Vessel Empties	58	59	82	40
CV-5 Valve Initial Closure	13 (Intact) 0 (Affected)	15 (Intact) <1 (Affected)	30 (Intact) 8 (Affected)	10 (Intact) 1 (Affected)
CV-3 Valve Initial Closure	DNC	DNC	DNC	DNC
low Pressure ECCS Trip	106	114	159	103
V-4 Valves Begin Opening	112	120	165	109
CV-2R Valves Open	124	146	~200	139
Peak Fuel Temp Occurs	124	146	~150	134

DNC - Did not close

TABLE 15.6-20

SEQUENCE OF EVENTS

COLD-LEG CROSSOVER BREAKS (RELAP4)

Event	Double-Ended Time (sec)	40% Split Time (sec)	20% Split Time (sec)
Pipe Break	0	0	0
Low Pressure Scram Trip	<.1	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1	.1
Control Rods Start In and Pumps Start Coastdown	~2	~2	~2
Control Rods 75% Inserted	3.5	3.5	3.5
Pressurizer Vessel Empties	58	59	82
CV-5 Valve Initial Closure	13 (Intact) 0 (Affected)	15 (Intact) <1 (Affected)	30 (Intact) 8 (Affected)
CV-3 Valve Initial Closure	DNC	DNC	DNC
low Pressure ECCS Trip	106	114	159
V-4 Valves Begin Opening	112	120	165
CV-2R Valves Open	124	146	~200
Peak Fuel Temp Occurs	124	146	~150

DNC - Did not close

TABLE 15.6-21

SEQUENCE OF EVENTS

COLD-LEG-RISER BREAKS (RELAP4)

21

Event	Double-Ended Time (sec)	100% Split Time (sec)
Pipe Break	0	0
Low Pressure Scram Trip	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2	~2
Control Rods 75% Inserted	3.5	3.5
Pressurizer Vessel Empties	40	60
CV-5 Valve Initial Closure	15 (Intact Riser) DNC (Affected Riser)	35 (Intact Riser) DNC (Affected Riser)
CV-3 Valve Initial Closure	~70	~145
low Pressure ECCS Trip	104	146
V-4 Valve Begin Opening	110	152
CV-2R Valves Open	145	~200
Peak Fuel Temperature	110	~200

DNC - Did not close

TABLE 15.6-22

SEQUENCE OF EVENTS

COLD-LEG-MANIFOLD BREAKS (RELAP4)

Event	Double-Ended DC = 1. Time (sec)	Double-Ended DC = .6 Time (sec)	100% Split Time (sec)
Pipe Break	0	0	0
Low Pressure Scram Trip	<.1	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2	~2	~2
Control Rods 75% Inserted	3.5	3.5	3.5
Pressurizer Vessel Empties	~18	~18	~18
CV-5 Valve Initial Closure	.1	.1	.1
CV-3 Valve Initial Closure	DNC	DNC	DNC
low Pressure ECCS Trip	30	36	37.8
V-4 Valves Begin Opening	36	42	43.8
CV-2R Valves Open	42	48	52
Peak Fuel Temperature Occurs	42	48	52

DNC - Did not close

TABLE 15.6-23

SEQUENCE OF EVENTS

HOT-LEG-MANIFOLD BREAKS (RELAP4)

21

Event	Double-Ended Discharge Coefficient = 1.0 Time (sec)	Double-Ended Discharge Coefficient = .6 Time (sec)	100% Split Time (sec)
Pipe Break	0	0	0
Low Pressure Scram Trip	<.1	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2	~2	~2
Control Rods 75% Inserted	3.5	3.5	3.5
Pressurizer Vessel Empties	16	16	16
CV-5 Valve Initial Closure	DNC	DNC	DNC
CV-3 Valve Initial Closure	DNC	DNC	DNC
Low Pressure ECCS Trip	11	17	20
V-4 Valves Begin Opening	17	23	26
CV-2R Valves Open	50	73	90
Peak Fuel Temperature Occurs	2	2	2

DNC - Did not close.

TABLE 15.6-24

SEQUENCE OF EVENTS

HOT-LEG-RISER BREAK (0-20 sec) (RELAP4)

Event	Double-Ended Time (sec)
Pipe Break	0
Low Pressure Scram Trip	<.1
Steam Generator SCS Flow Stops	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2
Control Rods 75% Inserted	3.5
Pressurizer Vessel Empties	~25
CV-5 Valve Initial Closure	DNC
CV-3 Valve Initial Closure	DNC
Peak Fuel Temperature Occurs	2

DNC - Did not close

TABLE 15.6-25

SEQUENCE OF EVENTS

PUMP SUCTION BREAKS (RELAP4)

Event	Double-Ended Time (sec)	100% Split Time (sec)
Pipe Break	0	0
Low Pressure Scram Trip	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2	~2
Control Rods 75% Inserted	3.5	3.5
Pressurizer Vessel Empties	17	17
CV-5 Valve Initial Closure	55	50
CV-3 Valve Initial Closure	DNC	DNC
Low Pressure ECCS Trip	64	62
V-4 Valve Begin Opening	70	68
CV-2R Valves Open	~100	~100
Peak Fuel Temperature	2	~2

DNC - Did not close

TABLE 15.6-26

SEQUENCE OF EVENTS

STEAM GENERATOR MANIFOLD BREAK (RELAP4)

Event	Time (sec)
Pipe Break	0
Low Pressure Scram Trip	<.1
Steam Generator SCS Flow Stops	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2
Control Rods 75% Inserted	3.5
Pressurizer Vessel Empties	~18
CV-5 Valve Initial Closure	55
CV-3 Valve Initial Closure	DNC
Low Pressure ECCS Trip	64
V-4 Valves Begin Opening	70
CV-2R Valves Open	~100
Peak Fuel Temperature Occurs	2

DNC - Did not close

TABLE 15.6-27

SEQUENCE OF EVENTS

ACCIDENTAL V-4 ACTUATION (RELAP4)

Event	Eight V-4 Valves Open Time (sec)	Four V-4 Valves Open Time (sec)
V-4 Valves Begin Opening	0	0
Low Pressure Scram Trip	<.1	<.1
Steam Generator SCS Flow Stops	.1	.1
Control Rods Start In and RCS Pumps Start Coastdown	~2	~2
Control Rods 75% Inserted	3.5	3.5
Pressurizer Vessel Empties	18	25
CV-5 Valve Initial	DNC	DNC
CV-3 Valve Initial Closure	DNC	DNC
Low Pressure ECCS Trip	37	52
Peak Fuel Temperature Occurs	2	2

DNC - Did not close

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TABLE 15.6-28

HYPOTHETICAL ACCIDENT CONFINEMENT SEQUENCE (RELAP4)

Time (sec)	Event
0.0	Break of cold-leg manifold.
0.1	Reactor trip on low pressure.
0.2	Trip signal set at 2 in. w.g. for confinement valves closure in 109-N building.
0.23	Fog sprays in 109-N building turn on.
0.3	Confinement trip signal in 105-N bldg. set at 2 in. w.g. confinement valves in 105-N bldg. start to close, start confinement timer.
0.52	Fog sprays in 105-N bldg. turn on.
1.3	105-N confinement exhaust valves closed to 80 degrees.
2.0	109-N steam vents weatherhead open at 2 psig. Peak confinement pressure.
2.0	Special steam vent weatherhead in 105-N bldg. opens at 1.25 psig.
2.0	Insertion of horizontal control rods begins.
2.3	105-N confinement supply valves closed to 80 degrees.
3.5	Control rods 75% inserted.
5.2	109-N confinement valves closed.
10.0	Peak pressure of 2.7 psig 105-N bldg.
10.0	Peak pressure of 3.0 psig in 109-N bldg.
92.0	Confinement atmosphere pressure becomes negative
150.3	Steam vents in 109-N bldg. being to close (only special steam vent in 105-N bldg. open).
205.3	Filtered flow exhaust valve opens, special steam vent closes. (Confinement is at a negative pressure with reverse flow through filter).
420	The highest power fuel elements reach 1830°F.

TABLE 15.6-28

HYPOTHETICAL ACCIDENT CONFINEMENT SEQUENCE (RELAP4)

Time (sec)	Event
1500	No additional fuel would reach 1830°F.
2820	Confinement atmosphere pressure becomes positive and filtered release is established.

TABLE 15.6-29

SIGNIFICANT EVENTS FOR CONTAIN ANALYSIS OF N REACTOR

Event Description	Time (sec)
Steam release begins; filter vent closes	0.0
Confinement fog sprays are activated	0.2
Regular steam vents in pipe gallery open; special steam vent opens	2.3
Regular steam vents in reactor building open	4.3
Peak confinement pressure is reached	10.3
Hydrogen release begins	135
Pipe gallery steam vents close	150
Steam release terminates	156
Reactor building steam vents close; vacuum breakers open	200
Special steam vent closes and filter vent opens	215
Vacuum breaker in reactor building closes	315
Vacuum breaker in pipe gallery closes	415
Fission product release begins	465
Hydrogen source rate peaks	720
Fission product release to confinement terminates	1550
Nitrogen purge activated; filter vent flow becomes positive	10,800

TABLE 15.6-30a

FISSION PRODUCT FRACTIONS RELEASED TO CONFINEMENT BEST-ESTIMATE

FISSION PRODUCT SOURCE TERM MAXIMUM HYPOTHETICAL ACCIDENT

Fission Product Group	Fraction FPs in Failed Fuel (%)	Fraction FPs Released From Fuel (%)	Partition Coefficients (%)	Primary System Retention Coefficients (%)	Fraction FPs Entering Confinement
Noble gases (Kr-Xe)	50	100		0.00	5.00 E-01
Iodine	50	50			
Elemental (I_2)			5	50.00	6.25 E-03
Particulate (CsI)			95	95.00	1.19 E-02
Organic (CH_3I)			1.6[a]	0.00	1.00 E-04
Other Species					
Cesium	50	30		95.00	7.50 E-03
Tellurium	50	100		95.00	2.50 E-02
Ruthenium	50	5		95.00	1.25 E-03
Other	50	1		95.00	2.50 E-04

[a]It is assumed 1.6% of the elemental iodine entering confinement (6.25×10^{-3} times total) reacts with organic material to form methyl iodide.

FP = Fission Product

TABLE 15.6-30b

RECOMMENDED LOW-POPULATION ZONE DOSE CALCULATION PARAMETERS

Time After Accident (hours)	Dispersion Factor at Low-Population Zone (m^3/sec)	Wind Speed (m/sec)	Breathing Rate (m^3/sec)
0 - 8	5.4 E-06	1.0	3.47 E-04
8 - 24	9.6 E-07	1.0	1.75 E-04
24 - 96	3.0 E-07	2.4	2.32 E-04
96 - 720	6.3 E-08	2.67	2.32 E-04

TABLE 15.6-30c

DISPERSION FACTORS BASED ON N REACTOR

SITE-SPECIFIC METEOROLOGY

Location	Time Interval (hours)	Regulatory Guide 1.145 X/Q (m^3/sec)	Old X/Q (m^3/sec)
Site boundary	0 - 2	2.0 E-05	2.6 E-05
Low-population zone	0 - 2	3.6 E-06	5.4 E-06
	2 - 8	1.6 E-06	5.4 E-06
	8 - 24	1.1 E-06	9.6 E-07
	24 - 96	4.6 E-07	3.0 E-07
	96 - 720	1.3 E-07	6.3 E-08

TABLE 15.6-31

COMPARISIONS OF RADIOLOGICAL CONSEQUENCES USING

THE SITING AND BEST-ESTIMATE SOURCE TERMS

FOR MAXIMUM HYPOTHETICAL ACCIDENT

	Without Hydrogen Mitigation System		With Hydrogen Mitigation System[a]		
Location	Siting Source Term	Best-Estimate Source Term	Siting Source Term	Best-Estimate Source Term	10 CFR 100 Guideline
Site boundary					
Whole body	3.3	1.4	20	9.8	25
Thyroid	212	15.4	201	16.9	300
Low-population Zone					
Whole body	1.9	0.8	6.6	3.5	25
Thyroid	149	10.4	212	8.9	300

[a]Recirculation flow 25,000 ft^3/min; nitrogen purge 3,000 ft^3/min for 3 to 30 hours, 1000 ft^3/min thereafter.

TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level		Remarks
	Conservative	Realistic	
1. Initial reactor coolant system conditions:			
o Coolant pressure = 1825 psig at primary coolant pump discharge		X	
o Coolant temperature = 537°F in hot leg and 395°F in cold leg		X	
o Coolant flow rate = 85.8 x 10^6 lbm/hr		X	
2. Initial power conditions:			
o Initial reactor power 102% of related power level = 4080 MWt	X		
o Fraction of thermal power generated in fuel = 96.36%	X		Assumed for 60 sec. following break; 5 sec more likely value
o Fraction of thermal power generated in process tubes and graphite = 3.64%	X		
o Steady-state heat removal by graphite and shield cooling system = 1% (40 MWt)	X		
3. Initial stored energy in the fuel:			
o Axial peak-to-average power factor = 1.5 (average channel model)	X		
o Axial peak-to-average power factor = 1.4 (hot channel model)		X	
o Hot Channel Factor = 1.25		X	
4. Cold leg crossover line 100% split break (1.353 ft) occurs instantaneously	X		

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TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level		Remarks
	Conservative	Realistic	
5. loss of AC power concurrent with break at 0 seconds	X		Seismic event assumed coincident with LOCA due to non-"once-through cooling" RCS piping not being qualified to SSE of 0.25 g.
6. Primary coolant pump inertial coast down with closure of Trip and Throttle valves at 0 seconds		X	
7. Primary coolant pump rotors running free during blowdown and reflood periods		X	
8. Dump condenser condensate pumps trip on loss of AC power at 0 seconds	X		
o Feedwater flow coasts down linearly to 0 lbm/sec by 12 seconds		X	
9. Secondary side steam lines isolated by closure of Export Valves (HPV-6261) and Dump Condenser Valves (HPV-201) on loss of AC power	X		
o Steam flow decreases as a function of linear change in valve opening area from full open at 0 seconds to full closed at 10 seconds		X	

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TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level: Conservative	Assumption Level: Realistic	Remarks
10. Reactor shutdown conditions:	X		
o Reactor trip signal on low RCS pressure setpoint of 1540 psig			
o Delay time before control rod movement:			
- Instrument line response	X		
- Pressure monitor response = 0.95 seconds	X		
- Relay response	X		
- Trip breaker closure	X		
- Rod movement delay = 0.20 seconds / 1.15 seconds			21
o Control rod insertion time to 75% inserted = 1.25 seconds		X	
o Control rods available for scram = 40 mk	X		Single-sided linear insertion of rods assumed
o Negative reactivity feedback effects ignored (e.g., coolant voiding, Doppler)	X		
11. Decay heat conditions:			
o Contribution from actinides (U239, Np239 and others) at peak end of cycle burnup		X	
o N Reactor-specific calculated fission product decay heat multiplied by 1.2	X		little difference with ANS standard for LOCA period of interest
o Decay heat distribution:			
- Fraction deposited in fuel = 88%	X		Assumed to take place 60 sec. following break
- Fraction deposited in process tubes = 3%X			
- Fraction deposited in graphite = 9%	X		

TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level: Conservative	Assumption Level: Realistic	Remarks
12. Metal-water reactions:			
o Zirconium-water reaction not included in average channel model or hot channel model		X	Reaction rate assumed not significant at average channel temperatures
o Zirconium-water reaction rates not steam limited	X		
13. Reactor internal heat transfer:			
o RCS piping passive heat structures included		X	
o Heat transfer between fuel, process tubes, and graphite included		X	
14. Primary-to-secondary heat transfer from secondary side of steam generation to RCS included	X		large amount of energy assumed stored in secondary side
15. One high lift diesel pump (ECCS) fails to start on demand (worst single active failure)	X		
16. Two high lift diesel pumps establish ECCS flow conditions on demand:			
o Pumps at full speed at 41 seconds after scram signal	X		Earliest RCS pressure allowing ECCS injection expected 100 sec after break
o Pumps dead head at 330 psia at V-3 manifold		X	
o ECCS water temperature = 110°F from Demineralizer Water Tank and 80°F from Columbia River	X		Bounding high temperature; switch to river assumed after 20 min.
o Degraded flow on ECCS injection:			
- Pump head degradation = 4%	X		
- Pump flow degradation = 6%	X		
- Diesel driver speed = 745 rpm	X		

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TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level		Remarks
	Conservative	Realistic	
17. One RCS Dump Valve (V-4) out of service and seven V-4 valves open on demand:		X	
o Actuation signal on RCS pressure of 645 psig coincident with confinement pressure of 25 in. wg.	X		
o Delay time before V-4 valves start opening:			
- Timer delay = 5.0 seconds		X	
- Instrument line response = 1.0 seconds		X	
6.0 seconds			
o V-4 valve opening time (cone lift and rotation) = 15 seconds		X	
o V-4 valve flow discharge rate during opening:			
- No flow from 0 to 5 seconds	X		
- Increases as a function of linear change in valve opening area from full closed at 5 seconds to full open at 15 seconds	X		
18. No credit for HPI system operation	X		Siesmic event assumed coincident with large break LOCA

TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level		Remarks
	Conservative	Realistic	
19. Eight ECCS Inlet Valves (V-3) open on demand:		X	
o V-3 valve opening time (cone lift and rotation) = 15 seconds		X	
o V-3 valve ECCS injection flow rate during opening:			
- No flow from 0 to 5 seconds	X		
- Increases as a function of linear change in valve opening area from full closed at 5 seconds to full open at 15 seconds	X		
20. Sixteen CV-2R check valves open when RCS depressurizes to less than 330 psia and high lift diesel pumps are operating at full speed		X	
21. RELAP5/MOD2 choked flow model for subcooled and two-phase critical flow used at break		X	For a given break size, slower depressurization yields more severe impact
22. Frictional pressure losses calculated using Colebrook correlation for variation of friction factor with Reynolds number, and modified Baroczy correlation for the two-phase friction multiplier		X	
23. Critical heat flux (CHF):			
o Combination of Biase and modified Zuber correlations used for CHF during blowdown		X	
o Return to nucleate boiling during blowdown not precluded by code logic	X		No return to nucleate boiling during blowdown expected
24. Post-CHF heat transfer:			

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TABLE 15.6-33

SUMMARY OF N REACTOR LIMITING LOCA ASSUMPTIONS

RELAP5 Analysis Features/Assumptions	Assumption Level		Remarks
	Conservative	Realistic	
o Combination of Dougall-Rohsenow and Bromley-Pomeranze correlations used for film boiling heat transfer		X	
o CHEN correlation used for transition boiling heat transfer		X	
25. Confinement pressure not calculated as part of LOCA ECCS performance analysis	X		Assume confiner peak pressure of 5 psig during period of transient. Increases backpressure and RCS depressurization time.
o Low confinement pressure signal for actuation of ECCS occurs at 10 seconds after break	X		Low pressure signal should occur within 1-2 sec. following break

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TABLE 15.6-34

BEST ESTIMATE ANALYSIS MATRIX

Break location	Break Size	Comments
Cold leg manifold	200% guillotine	3 HLDP(a)
Cold leg manifold	200% guillotine	minimum ECCS (2 HLDP)
Station blackout	8 V-4 valves	3 HLDP
Station blackout	4 V-4 valves	minimum ECCS (2 HLDP)
Outlet riser	200% guillotine	Primary coolant pumps off (2 HLDP)
Inlet riser	200% guillotine	Primary coolant pumps off (2 HLDP)
Cold leg crossover	35% split	2 HLDP
Cold leg crossover	40% split	2 HLDP
Cold leg crossover	50% split	2 HLDP
Cold leg crossover	60% split	2 HLDP
Cold leg crossover	100% split	limiting LOCA, 2 HLDP

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(a) High-lift diesel pump

TABLE 15.6-35

PERCENT ENERGY DEPOSITION -- HOT CHANNEL MODEL

	Percent Deposition at T = 0 s (% x 100)		Percent Deposition at T > 60 s (% x 100)	
	MKIV	MKIA	MKIA	MKIA
Inner Fuel	26	23	26	26
Outer Fuel	70	73	62	62
Process Tube	0.85	0.84	3	3
Graphite	3.4	3.2	9	9

TABLE 15.6-36

FUEL DIMENSIONS

	MKIV	MKIA
Overall diameters, nominal		
Outer cylinder O.D., in.	2.425	2.404
Outer cylinder I.D., in.	1.701	1.767
Inner cylinder O.D., in.	1.279	1.246
Inner cylinder I.D., in.	0.479	0.440
Cladding thickness, mils		
Outer cylinder, outer cladding	25	25
Outer cylinder, inner cladding	20	25
Inner cylinder, outer cladding	30	40
Inner cylinder, inner cladding	20	25
Relative total volume		
Outer cylinder $in^3/in.$	2.346	2.087
Inner cylinder, $in^3/in.$	1.105	1.067

TABLE 15.6-37

INITIAL POWER DENSITIES (3)

	KW/ft^3 at T = 0 sec		KW/ft^3 at T > 60 sec	
	MKIV(1)	MKIA(2)	MKIV(1)	MKIA(2)
Inner fuel	47.56	41.16	47.20	45.93
Outer fuel	59.51	65.64	53.01	56.0

(1)MKIV power density at 201 kw/ft
(2)MKIA power density at 189 kw/ft
(3)Decay heat is kw/ft^3 times decay heat factor calculated by ORIGEN.

TABLE 15.6-38

PERCENT ENERGY DEPOSITION--SYSTEM AND HOT CHANNEL

VERIFICATION MODELS

	Verification Model Percent Deposition	Updated Percent Deposition (For Comparison)
Inner Fuel	24	26
Outer Fuel	73	70
Pressure Tube	0.61	0.85
Graphite	3.0	3.3

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TABLE 15.6-39

DEGRADED ECCS FLOW RATE

Pressure Differential (psid)*	Flowrate (lb/sec)
0.00	4494.
13.69	4400.
27.89	4300.
41.78	4200.
55.35	4100.
68.60	4000.
81.53	3900.
94.15	3800.
106.45	3700.
118.43	3600.
130.09	3500.
141.44	3400.
152.46	3300.
163.17	3200.
173.57	3100.
183.64	3000.
193.40	2900.
202.84	2800.
211.96	2700.
220.76	2600.
237.42	2400.
252.80	2200.
266.92	2000.
279.76	1800.
291.34	1600.
301.64	1400.
310.68	1200.
318.44	1000.
324.93	800.
330.16	600.
334.11	400.
336.79	200.
338.20	0

*Pressure differential between V-3 inlet manifold and inlet riser--two high lift diesel pumps

TABLE 15.6-40

BREAK SPECTRUM RESULTS

Parameter	Break Size--Percent					
	25	35	40	50	60	100
CV-5 closes (initially) (sec)	1	1	1	1	1	1
CV-5 reopens broken loop (sec)	100	84	78	77	63	(a)
CV-5 recloses broken loop (sec)	(a)	130	121	115	67	(a)
V-4 opens (sec)	40	40	40	40	40	40
ECCS initial flow (sec)	128	136	125	121	113	111
Peak fuel temperature (°F)	1310	1320	1325	1410	1425	1615

(a) not predicted to occur in analysis

TABLE 15.6-41

SEQUENCE OF EVENTS--100% CROSSOVER BREAK

Time (sec)	Event
0.005	Break initiated.
0.1	Broken loop check valve CV-5 closes.
0.14	Scram low pressure limit reached 1540 psig.
1.1	Rod scram trip breaker closes.
1.3	Rod movement initiated.
2.6	Rods 75% inserted.
10.0	ECCS high confiner pressure locked in (25 in. w.g.).
11.	Intact loop check valve CV-5 closes.
44.	ECCS low pressure limit reached (645 psig).
49.	ECCS actuated.
50.	Secondary loop peak pressure reached (420 psig).
54.	Dump valve V-4 flow initiated.
65.	Intact loop check valve CV-5 reopens.
90.	High lift diesel pumps start upon scram--reach full speed in less than 90 sec.
130.	ECCS flow initiated, broken loop.
140.	ECCS flow initiated, intact loop.
200.	Cooling stabilized in bottom half of core, broken and intact loop.
280.	Mark IA hot channel reaches peak cladding temperature (1890°F at 4700 kw).
325.	Mark IV hot channel reaches peak cladding temperature (1835°F at 5000 kw).
430.	Peak cladding temperature reduced to 1430°F.
440.	ECCS flow reaches 95% of steady state.
790.	Intact loop average channel peak cladding temperature reached (1150°F)
800.	Bottom half of core quenched. Upper half not yet quenched, but stable cooling established. ECCS providing 23,600 gpm "steady state" flow.
1000.	Maximum temperature below 1400°F.
1100.	Average channel tube temperatures for the upper broken loop are below 1000°F, and decreasing. Average channel tube temperatures for the intact loop are less than 1200°F and stabilized.

TABLE 15.6-42

RELAP5-CALCULATED SEQUENCE OF EVENTS,

BASE CASE INLET RISER BREAK TRANSIENT

Time (sec)	Event
0.0	200% double-ended guillotine break at the bottom of one inlet riser stand pipe. First fuel heatup begins in the affected core.
0.5	Reactor trip on low core flow. Steam generator bypass valve begins to open. Primary coolant pump begins to run back to 1240 rpm.
5 - 60	First fuel heatup in unaffected core.
18.8	Primary coolant pump at 1240 rpm.
23.7	First of two ECCS signals activated--low core flow.
47.1	Second of two ECCS signals activated--low hot leg manifold pressure. V-4 (dump line) and V-3 (ECCS) valves begin to open.
63.0	Pressurizer empty (level below 0.5 ft).
70 - 170	First and second fuel heatup in affected core.
75.0	Unaffected core pressure decreased below the pressure upstream of the CV-2R valve and ECCS coolant begins to flow into the unaffected core.
130 - 200	Second fuel heatup in unaffected core.
200 - 400	Third fuel heatup in affected core.
280	Computer code changed to the modified version for simulation of the reflood process.
280 - 500	Fourth fuel heatup in unaffected core.
500	Reflood heatup begins in the unaffected core and in the lower half of the affected core.
650	Steam generator becomes a heat source to the primary.
700	Reflood heatup begins in the upper half of the affected core.

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TABLE 15.6-42

RELAP5-CALCULATED SEQUENCE OF EVENTS,

BASE CASE INLET RISER BREAK TRANSIENT

Time (sec)	Event
1350	Maximum fuel temperature of 1240°F attained in the unaffected core.
1544	Base case calculation terminated. Fuel temperature in the affected core at 1300°F and decreasing.

TABLE 15.6-43

RELAP5-CALCULATED SEQUENCE OF EVENTS,

SENSITIVITY INLET RISER BREAK TRANSIENT

Time (sec)	Event
0.0	200% double-ended guillotine break at the bottom of one inlet riser stand pipe. First fuel heatup begins in the affected core.
0.5	Reactor trip on low core flow. Steam generator bypass valve begins to open. Primary coolant pump begins to run back to 1240 rpm.
5 - 60	First fuel heatup in unaffected core.
18.8	Primary coolant pump at 1240 rpm.
23.7	First of two ECCS signals activated--low core flow.
47.1	Second of two ECCS signals activated--low hot leg manifold pressure. V-4 (dump line) and V-3 (ECCS) valves begin to open.
63.0	Pressurizer empty (level below 0.5 ft).
75.0	Unaffected core pressure decreased below the pressure upstream of the CV-2R valve and ECCS coolant begins to flow into the unaffected core.
100 - 170	Second fuel heatup in affected core.
130 - 200	Second fuel heatup in unaffected core.
200 - 400	Third fuel heatup in affected core.
280	Computer code changed to the modified version for simulation of the reflood process.
280 - 500	Third fuel heatup in unaffected core.
500	Reflood heatup begins in the unaffected core and in the lower half of the affected core.
650	Steam generator becomes a heat source to the primary.
700	Reflood heatup begins in the upper half of the affected core.

TABLE 15.6-43

RELAP5-CALCULATED SEQUENCE OF EVENTS,

SENSITIVITY INLET RISER BREAK TRANSIENT

Time (sec)	Event
1000	ECCS capacity increased from 22,000 to 24,000 gpm and all primary coolant pumps were tripped.
1120	Primary coolant pump impellers stop rotating.
1240	Maximum fuel temperature of 1200°F attained in the unaffected core (top channel).
1330	Maximum fuel temperature of 900°F attained in the affected core.
2700	Sensitivity calculation terminated. Fuel elements in the unaffected core quenched.

TABLE 15.6-44

COMPARISON OF RELAP5- AND RETRAN-CALCULATED STEADY STATE CONDITIONS WITH PLANT DATA, USED FOR COLD LEG MANIFOLD BREAK AND STATION BLACKOUT EVENTS

Parameter (units)	Plant Data	RELAP5	RETRAN
Core power (MW_t)	3870	3870	3870
Pressurizer pressure (psia)	1595	1595	1595
Pressurizer level (ft)	24.5	24.4	24.5
Hot leg temperature (°F)	NA	532	531
Cold leg temperature (°F)	NA	394	394
Loop flow rate (lbm/sec)	NA	23,745	23,750
Steam line pressure (psia)	152	152	152
Feedwater flow rate (lbm/sec)	NA	3729	3763
HPI flow rate (lbm/sec)	800	979	1126
Core differential pressure (psid)	NA	111	111
SG primary differential Pressure (psid)	NA	27	30

NA = Not Available

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TABLE 15.6-45

COMPARISON OF RELAP5-CALCULATED SEQUENCE OF EVENTS, REVISED (Ref. 39) AND ORIGINAL (Ref. 40) COLD LEG MANIFOLD BREAK TRANSIENTS

Time (sec)		
Revised	Original	Event
0.0	0.0	Cold leg manifold ruptured. Power to primary coolant pumps interrupted.
0.46	0.46	Reactor power tripped on low core flow. Steam generator bypass valves opened.
12.11	12.1	ECCS actuation low flow signal set. One of two signals required to actuate emergency dump and ECCS injections valves.
15.13	15.2	ECCS actuation low pressure signal set. Second of two signals required to actuate emergency dump and ECCS injection valves.
23.0	24.0	Maximum fuel temperature reached during blowdown phase temperature excursion (892°F, revised; 910°F, original).
29.0	21.0	Primary pressure dropped below ECCS shutoff head, and ECCS injection started.
64.0	60.0	Fuel elements quenched.
70.0	52.0	Core channels 7 and 8 were liquid full.
95.0	95.0	Second temperature excursion began.
210.0	60.0	ECCS flow established in core channel 6.
250.0	160.0	ECCS flow established in core channel 5. Quasi-steady core flow pattern established in revised calculation.
1100.0	130.0	Maximum fuel temperature reached in second fuel temperature excursion (1300°F) in third core channel revised; 656°F in top core channel, original).
1764.7	NA	ECCS capability increased 10%.
1850.0	145.0	Core channel 4 fuel quenched.

NA = Not Applicable.

TABLE 15.6-45

COMPARISON OF RELAP5-CALCULATED SEQUENCE OF EVENTS, REVISED (Ref. 39) AND ORIGINAL (Ref. 40) COLD LEG MANIFOLD BREAK TRANSIENTS

Time (sec)		
Revised	Original	Event
1950.0	177.0	ECCS flow established in core channel 4.
2150.0	145.0	Core channel 3 fuel quenched.
2150.0	1500.0	Calculation stopped.

TABLE 15.6-46

COMPARISON OF RELAP5-CALCULATED SEQUENCE OF EVENTS, REVISED (Ref. 39) AND ORIGINAL (Ref. 38) STATION BLACKOUT TRANSIENTS

Time (sec)		
Revised	Original	Event
0	0	Loss of offsite and onsite power occurs, feedwater and steam functions isolated, graphite cooling isolated, primary coolant pumps trip.
66	67	V-4 valves open and ECC pumps are started on ECCS signal (coincident low flow and low pump speed).
125	124	Blowdown period fuel element heatup begins.
161	161	Primary coolant pump coast down complete.
257	466	Fuel elements quenched as a result of coolant loop flashing.
464	213	ECC flow to core begins (core pressure below ECCS pump shutoff head).
523	308	Steam generators become heat source to the primary system.
698	495	Reflood period fuel element heatup begins.
859	500	Computer code changed to the modified version for simulation of the reflood process.
1304	NA	Alpha fuel phase change region entered by rising fuel temperatures in top two core channels.
1324	977	Peak fuel element temperature reached, 1244°F (1208°F in original calculation) in outer fuel elements at centers of top two core channels.
1336	NA	Alpha fuel phase change region exited by falling fuel element temperatures in top two core channels.
3668	NA	Beginning of sensitivity calculation. ECCS capacity increased from 22,000 gpm to 24,000 gpm. At this time top core channel fuel element temperatures were stable at about 850°F; all other fuel elements had been quenched to saturation temperature.
4000	NA	End of sensitivity calculation, all fuel elements quenched to saturation temperature.

NA = Not Applicable

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TABLE 15.6-47

RELAP5-CALCULATED SEQUENCE OF EVENTS, BASE CASE

OUTLET RISER BREAK TRANSIENT

Time (sec)	Event
0	Double-ended rupture occurs at bottom of one outlet riser.
1.6	Reactor trip caused by low outlet downcomer pressure.
19	Primary coolant pump runback to 1240 rpm complete.
36	First ECCS requirement attained (low hot leg manifold* pressure).
95	Pressurizer empty (level below 0.5 ft).
154	Feedwater throttled when setpoint level attained.
187	ECCS actuation signal. Low core flow condition attained as second ECCS requirement. Blowdown and ECCS injection valves opened. ECCS flow begins immediately.
190 - 240	First blowdown fuel element heatup in unaffected core.
447 - 553	Second blowdown fuel element heatup in unaffected core.
534 - 576	Blowdown fuel element heatup in affected core.
650	Computer code changed to the modified version for simulation of the reflood process.
706	Reflood fuel element heatup begins in unaffected core.
827	Reflood fuel element heatup begins in affected core.
1684	Maximum fuel element temperature reach 1244°F in center of top core channel, outside surface of outer fuel element.
1777	Base case calculation terminated.

TABLE 15.6-48

RELAP5-CALCULATED SEQUENCE OF EVENTS, SENSITIVITY

OUTLET RISER BREAK TRANSIENT

Time (sec)	Event
0	Double-ended rupture occurs at bottom of one outlet riser.
1.6	Reactor trip caused by low outlet downcomer pressure.
19	Primary coolant pump runback to 1240 rpm complete.
36	First ECCS requirement attained (low hot leg manifold pressure).
95	Pressurizer empty (level below 0.5 ft.)
154	Feedwater throttled when setpoint level attained.
187	ECCS actuation signal. Low core flow condition attained as second ECCS requirement. Blowdown and ECCS injection valves opened. ECCS flow begins immediately.
190 - 240	First blowdown fuel element heatup in unaffected core.
447 - 553	Second blowdown fuel element heatup in unaffected core.
534 - 576	Blowdown fuel element heatup in affected core.
650	Computer code changed to the modified version for simulation of the reflood process.
706	Reflood fuel element heatup begins in unaffected core.
827	Reflood fuel element heatup begins in affected core.
1000	ECCS capacity increased from 22,000 to 24,000 gpm and all power to primary coolant pumps tripped.
1106	Primary coolant pump rotors stop turning.
1709	Maximum fuel element temperature reached, 1221°F in center or top core channel, outer surface of outer fuel element.
2873	Quenching of fuel elements in unaffected core complete.
3395	Sensitivity calculation terminated.

Table 15.6-49

Summary of HECTR Calculations

Hydrogen release location	Hydrogen source	Steam source	Simulation time(s)	Comments
Front-pipe-barrier	HMSST	Riser break	18,000	Included lower blowout panels in pipe-barrier spaces
Pressurizer penthouse	HMSST	Cold-leg manifold break	2,000	Steam source in pipe gallery
Pressurizer penthouse	HMSST	Cold-leg manifold break	860	Steam source in pipe gallery; hydrogen burn
Pipe gallery extension top	HMSST	None	10,000	Recirculation fan on
Pipe gallery extension bottom	HMSST	None	10,000	Recirculation fan on
Pipe gallery extension bottom	Dump-tank-vent-line source	None	12,000	Recirculation fan on

Figures 15.6-133

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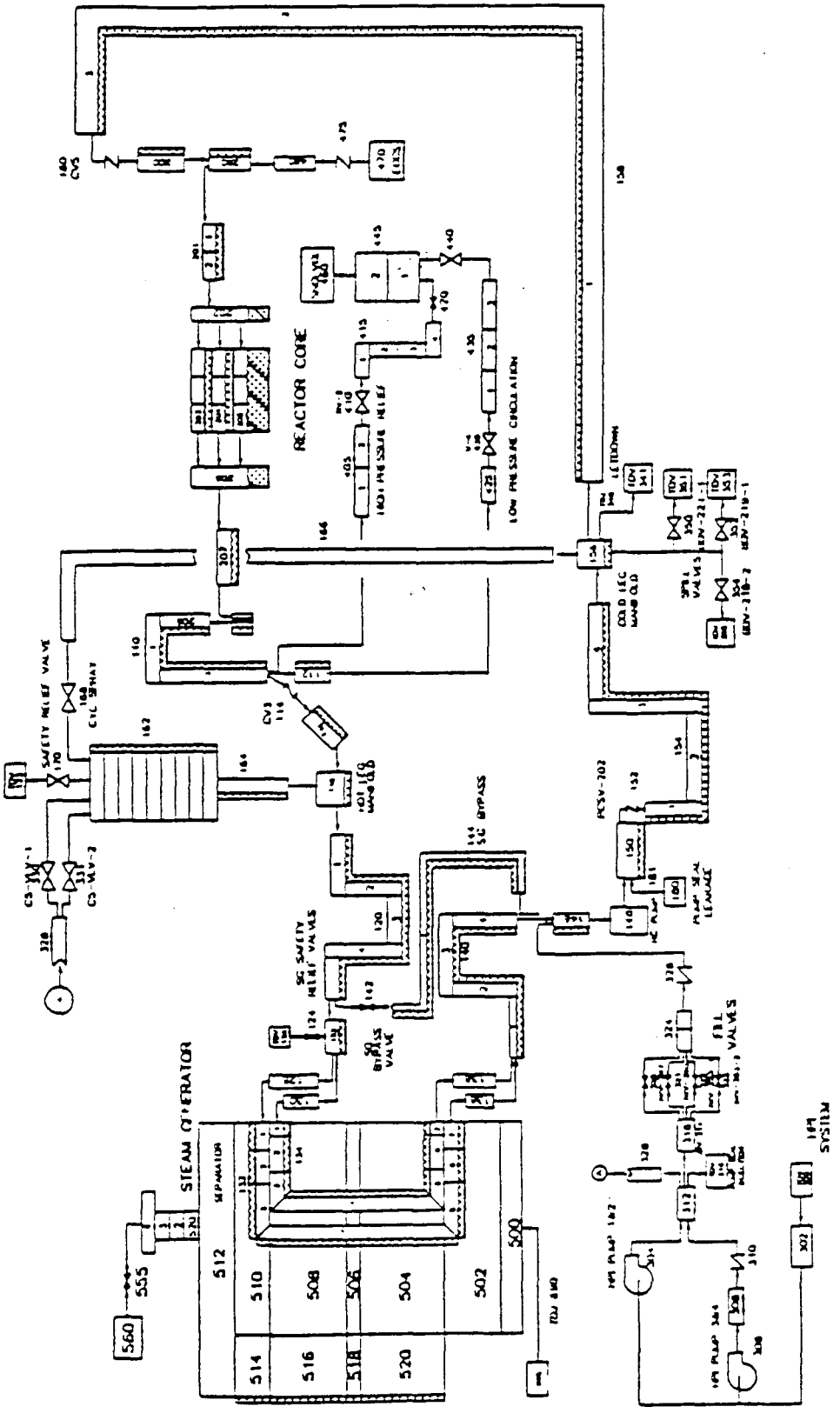
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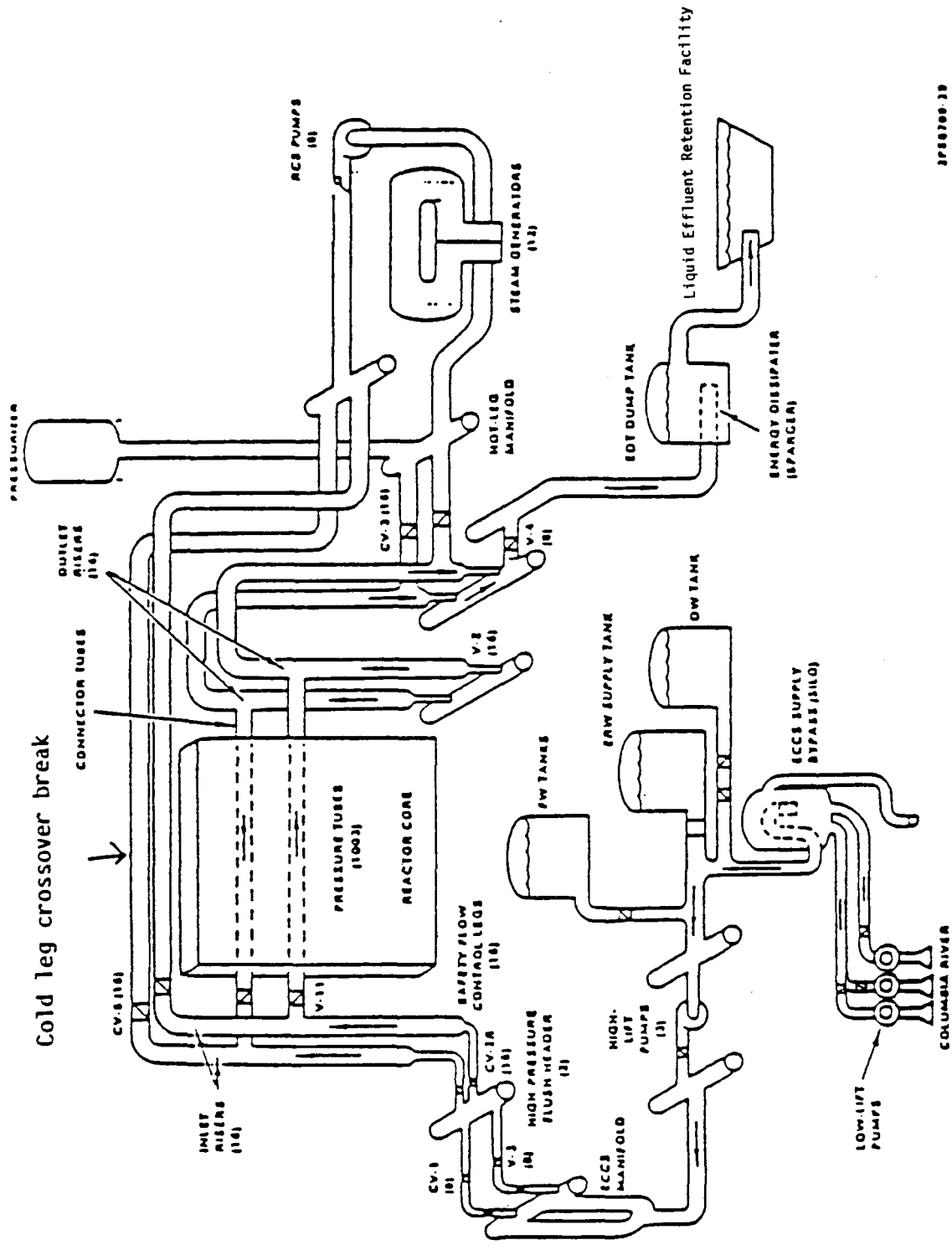
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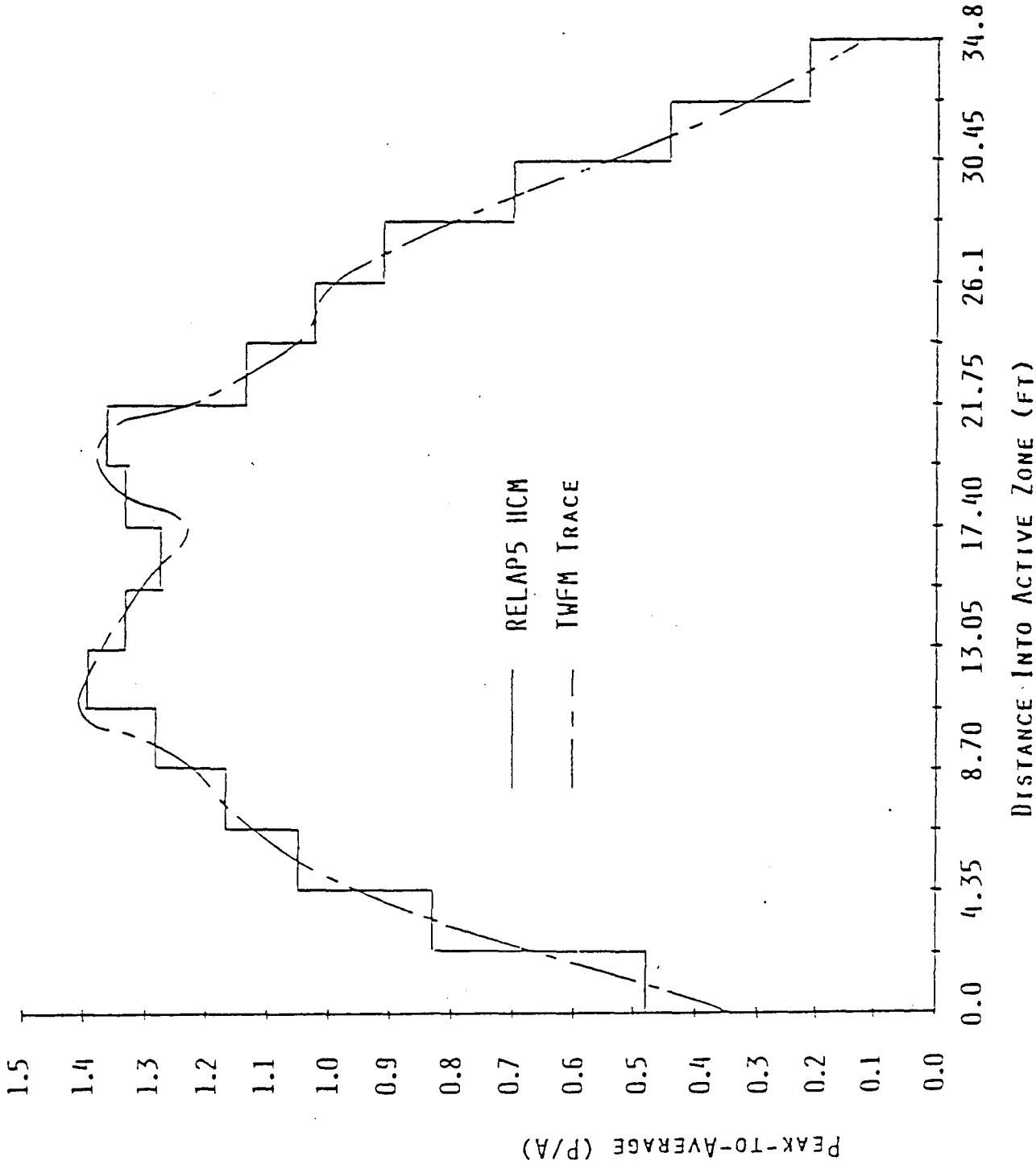
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	RELAP5 Nodalization of N Reactor Coolant Loop Piping Figure 15.6-176



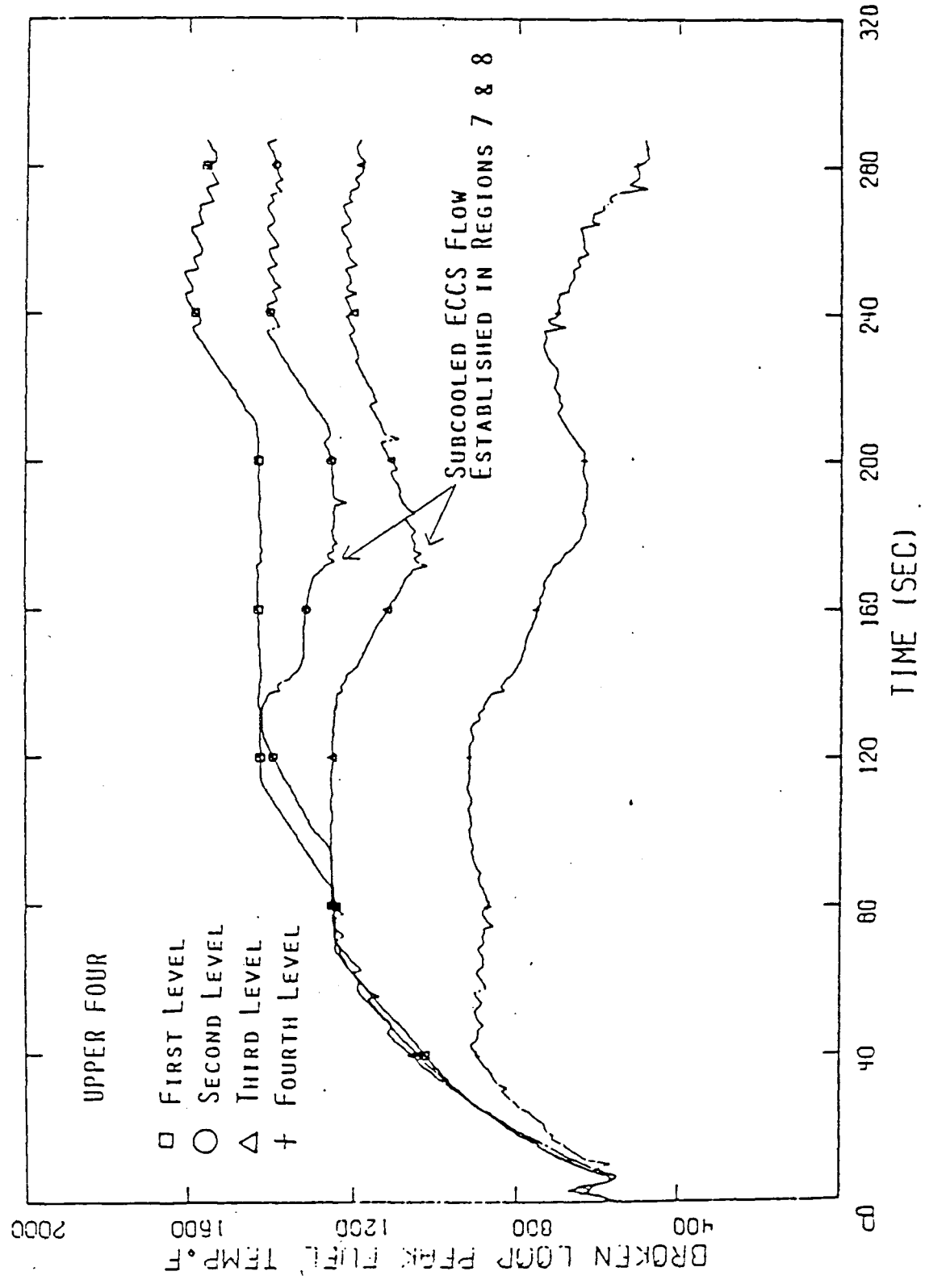
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Layout of the N Reactor
	Figure 15.6-177



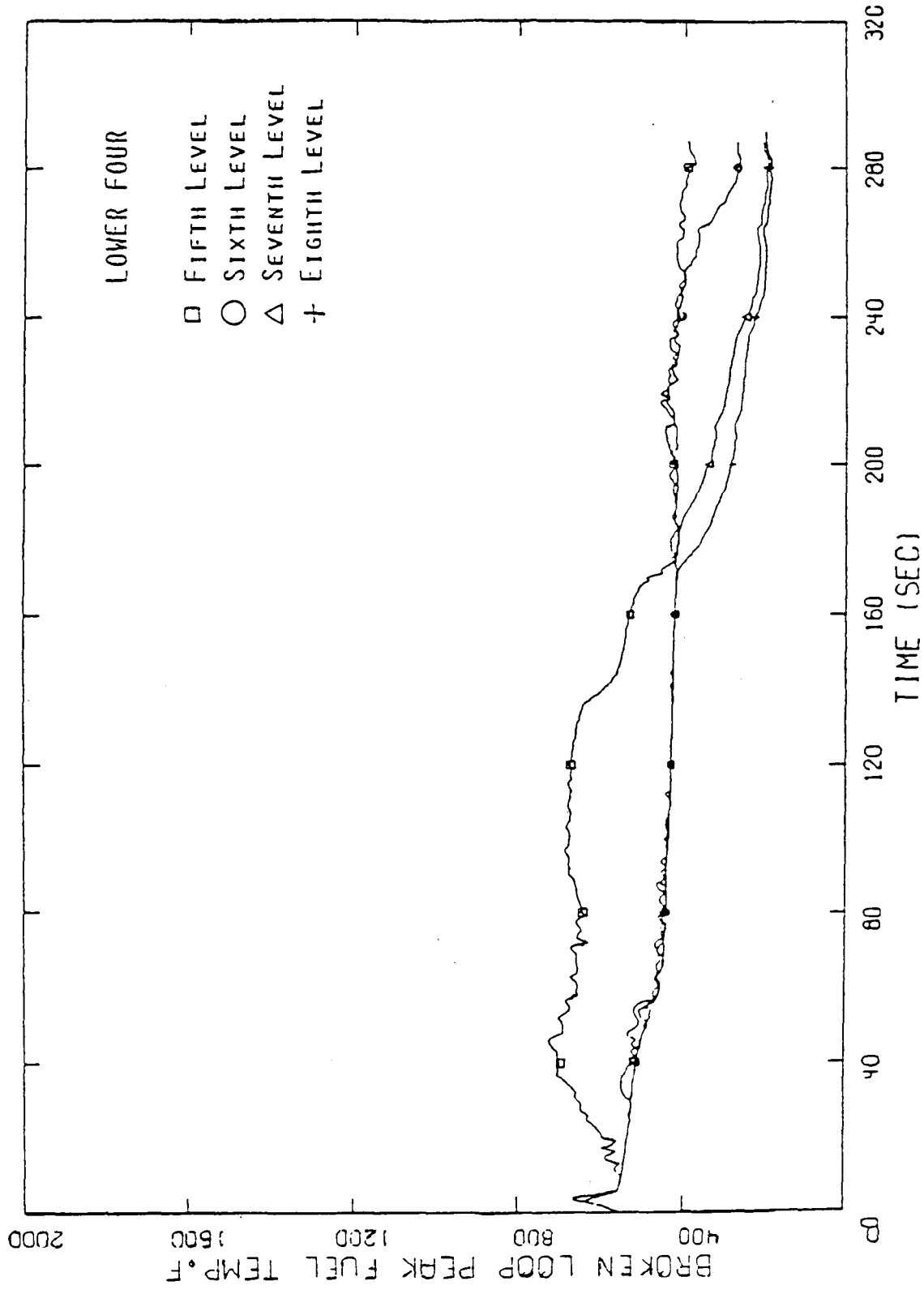
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	HCM Axial Power Profile
	Figure 15.6-178



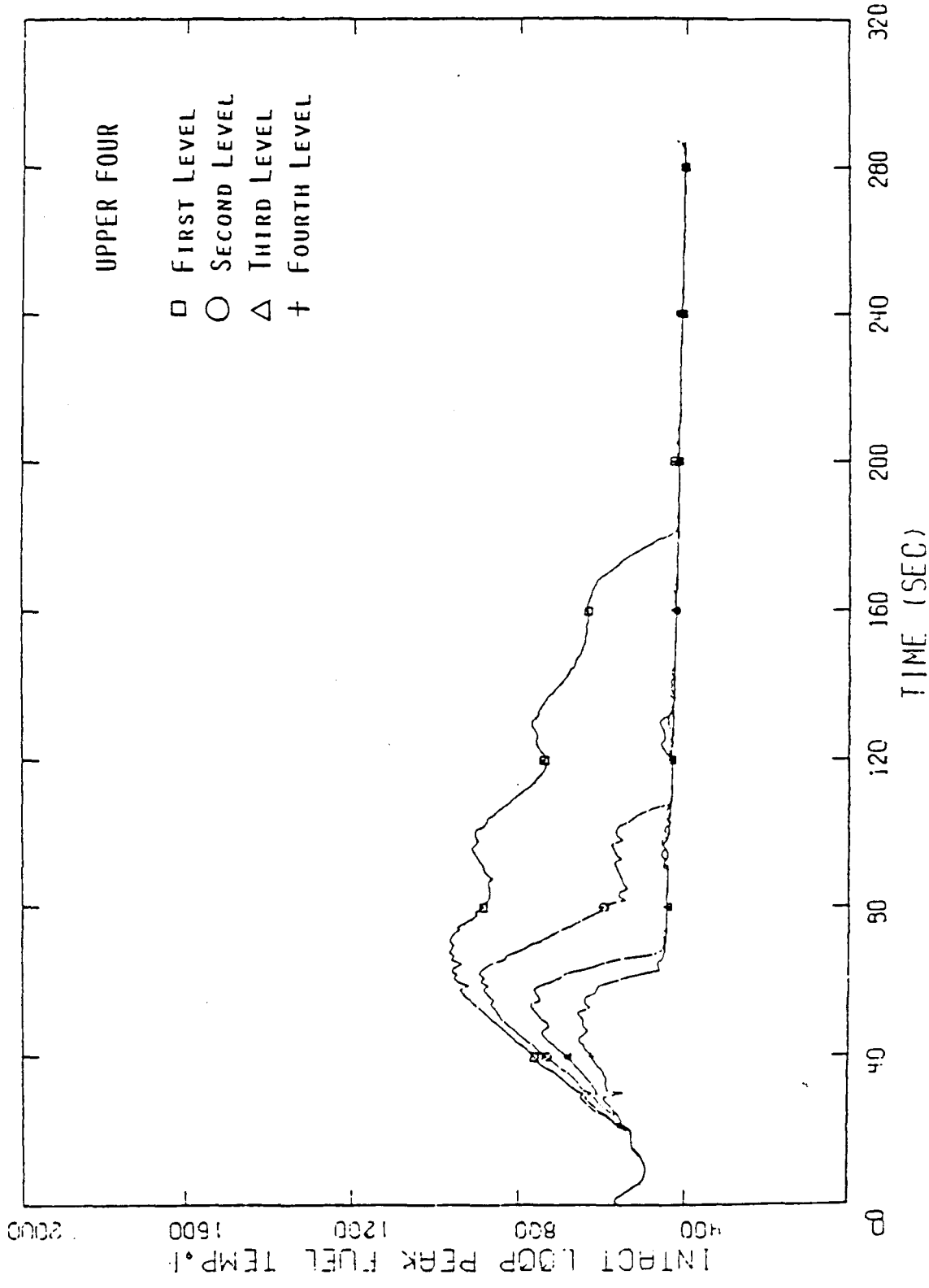
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures for the Upper Four Core Levels for the 100% Break Case Figure 15.6-179



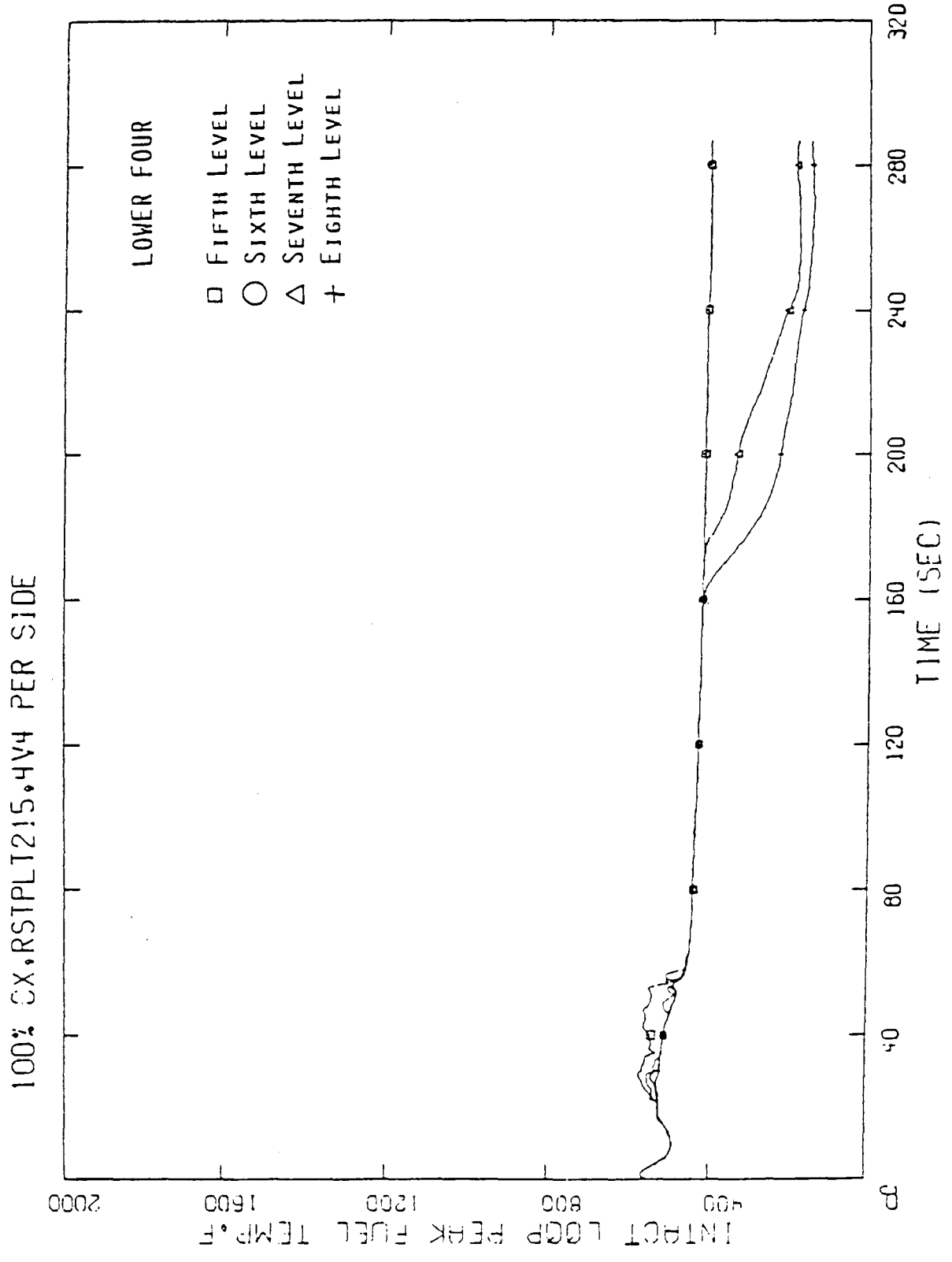
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures for the Lower Four Core Levels for the 100% Break Case Figure 15.6-180



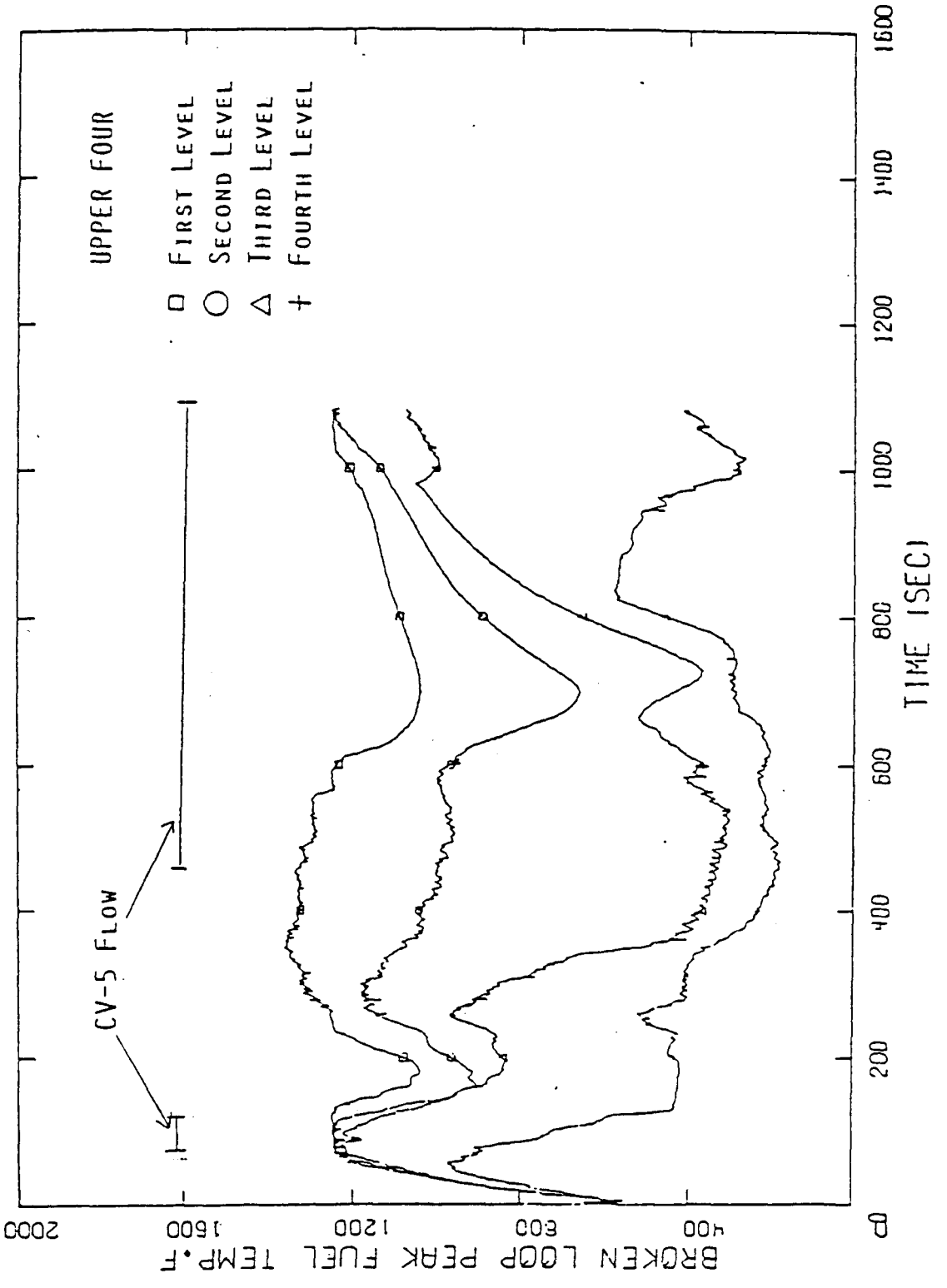
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Intact Loop Fuel Temperature Response for the Upper Four Core Levels for the 100% Break Case Figure 15.6-181



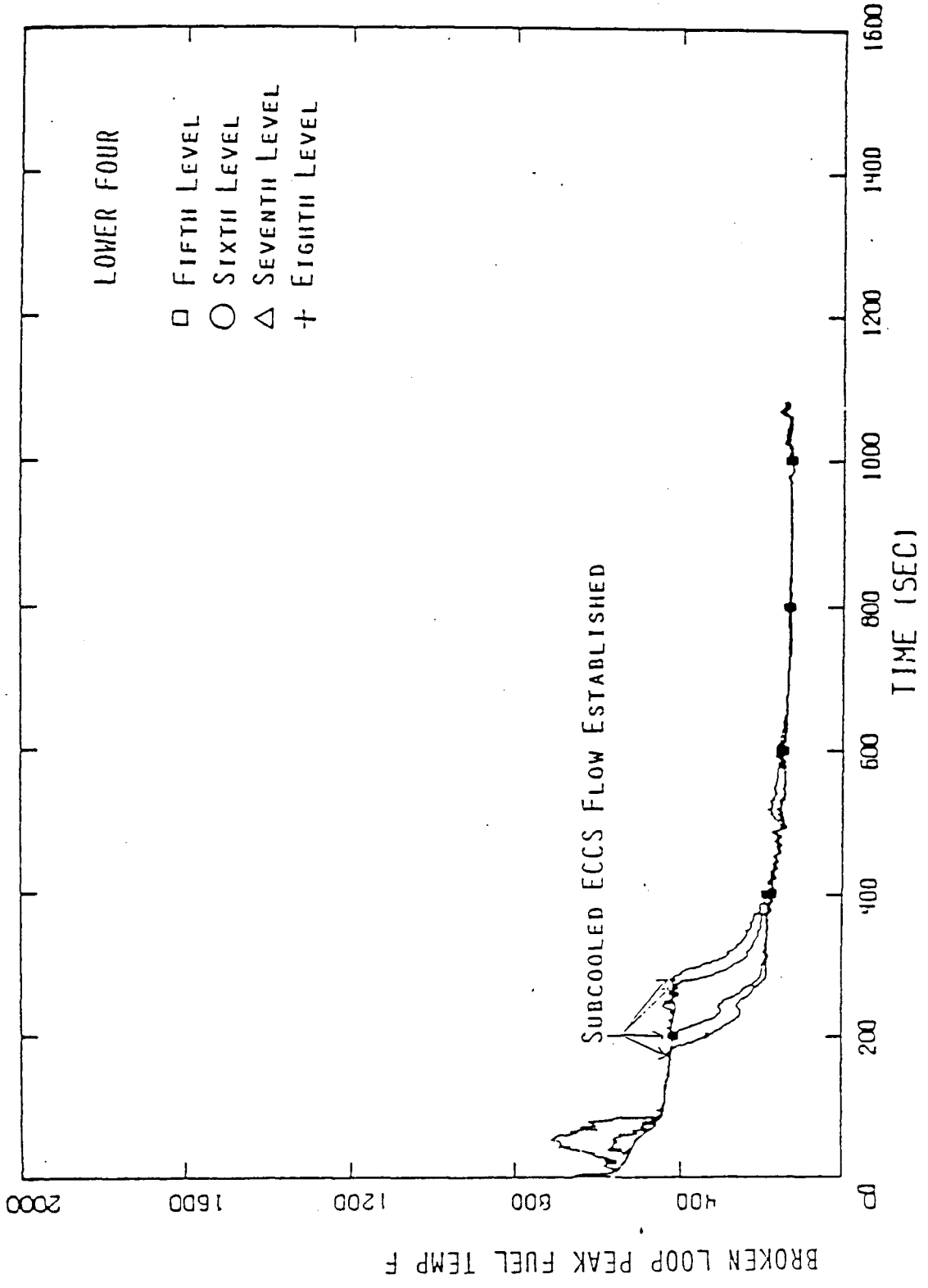
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Intact Loop Fuel Temperature Response for the Lower Four Core Levels for the 100% Break Case Figure 15.6-182



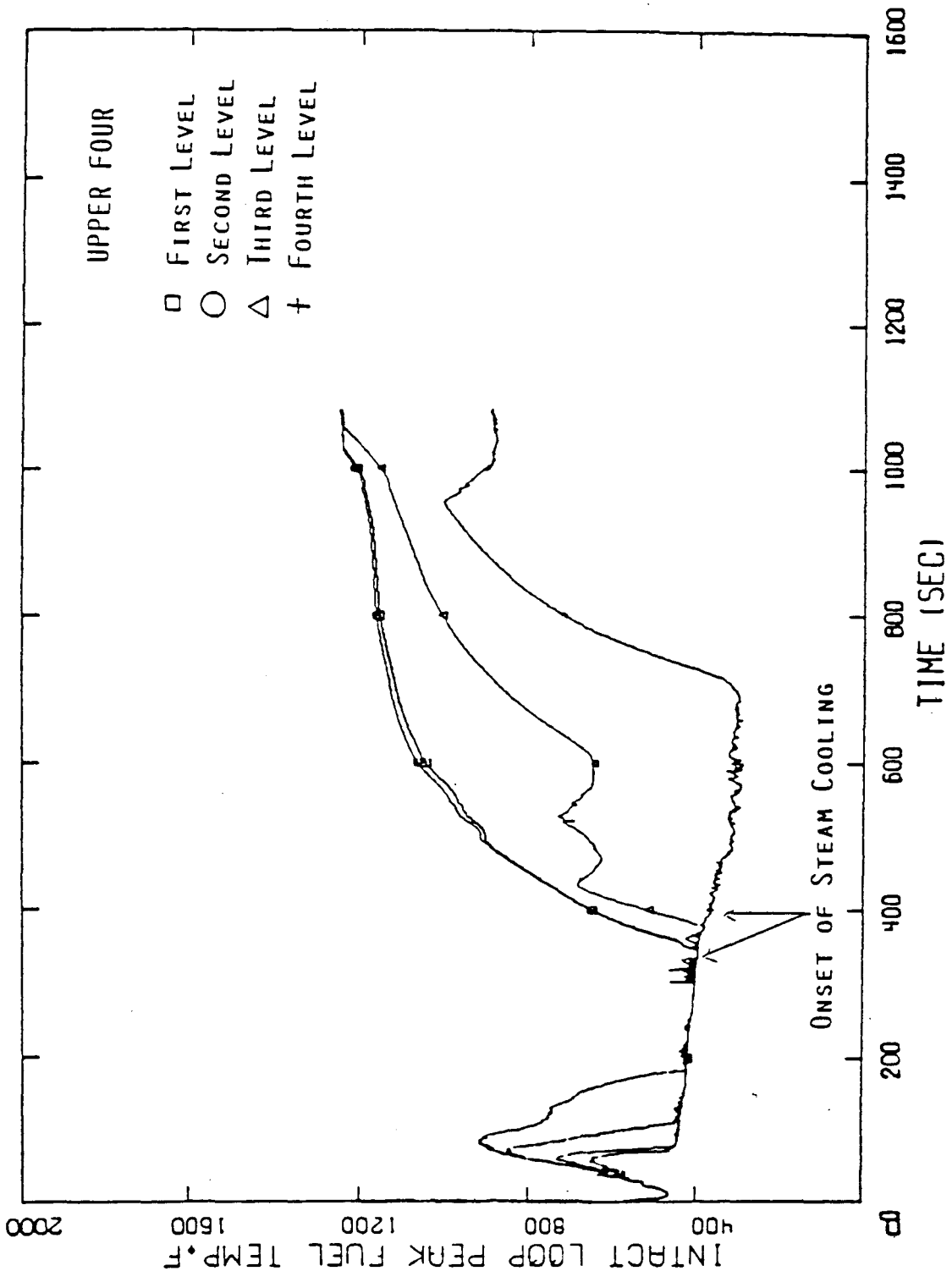
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures for the Upper Four Core Levels 40% Case Figure 15.6-183



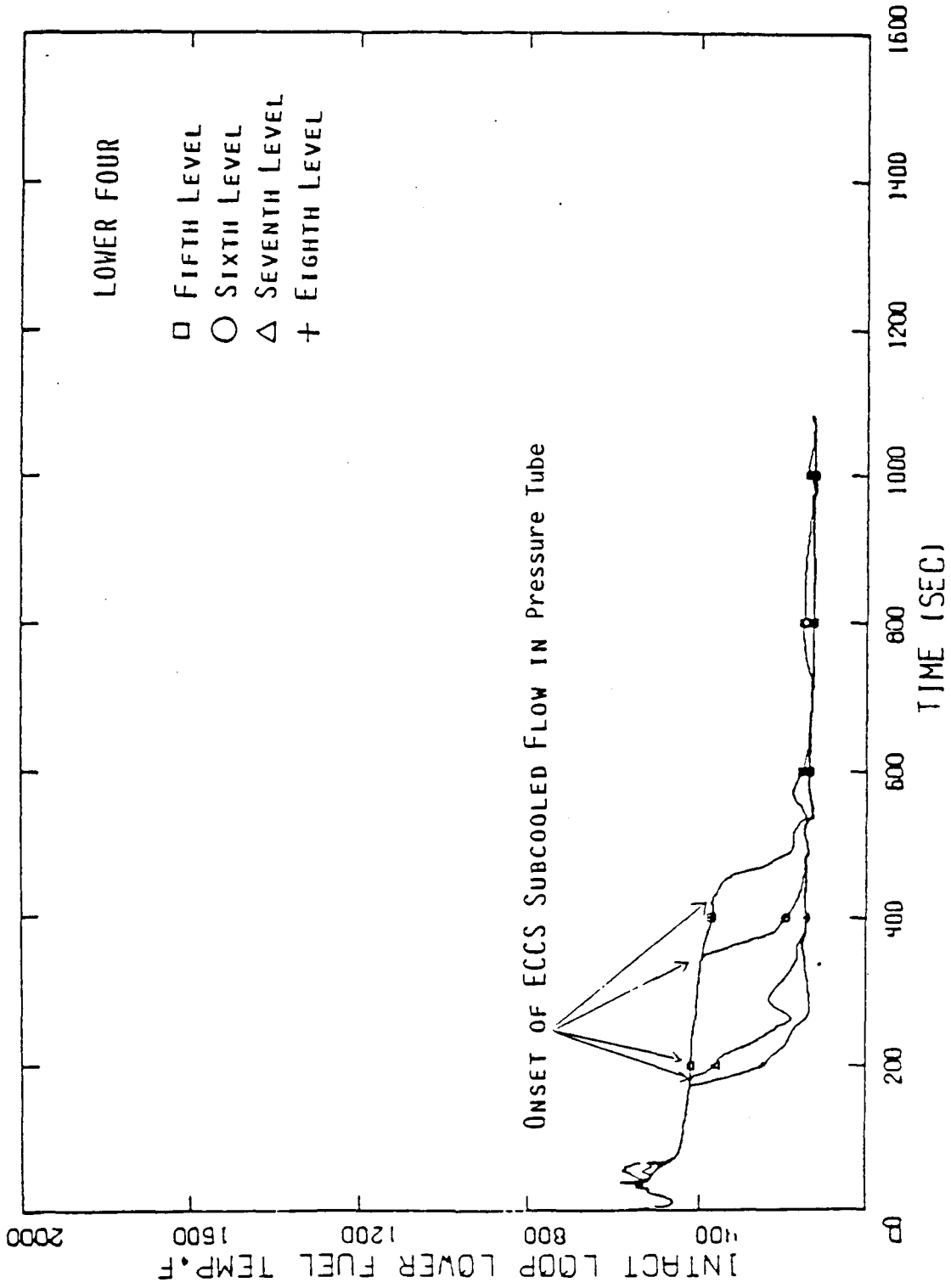
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures for the Lower Four Core Levels 40% Case
	Figure 15.6-184



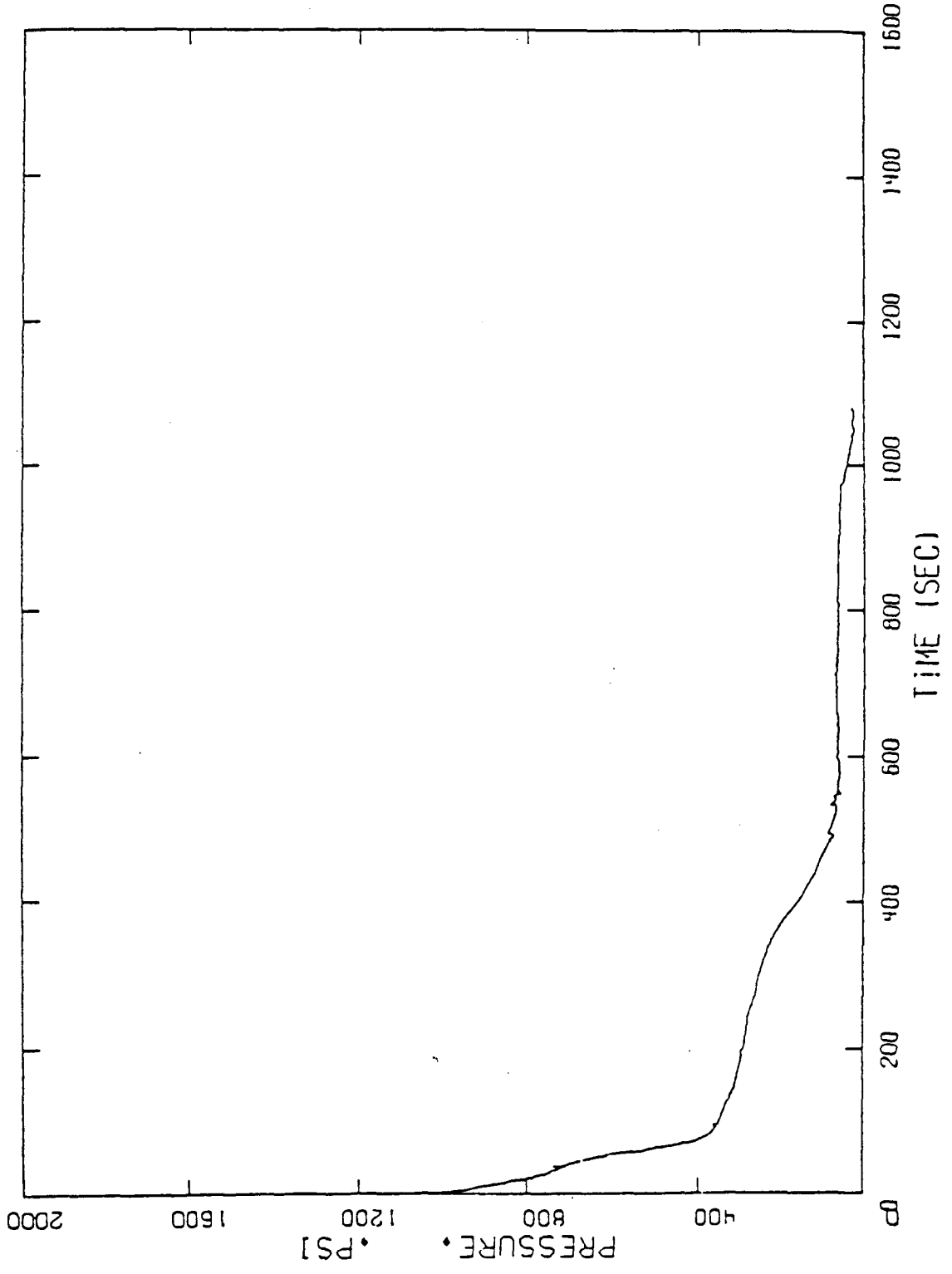
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Four Core Temperatures, 40% Break Case Figure 15.6-185



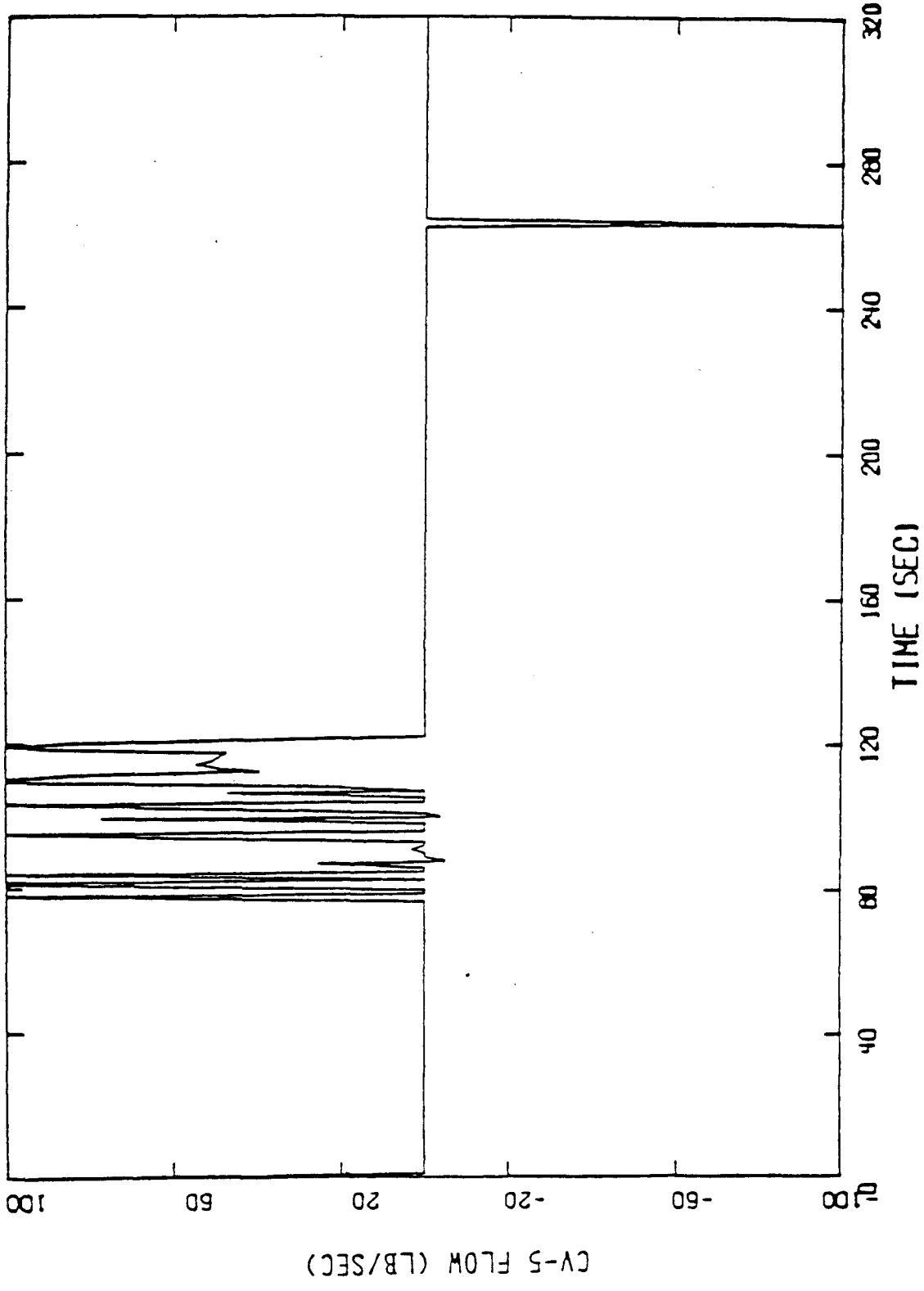
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Four Levels Peak Fuel Temperatures 40% Break Case Figure 15.6-186



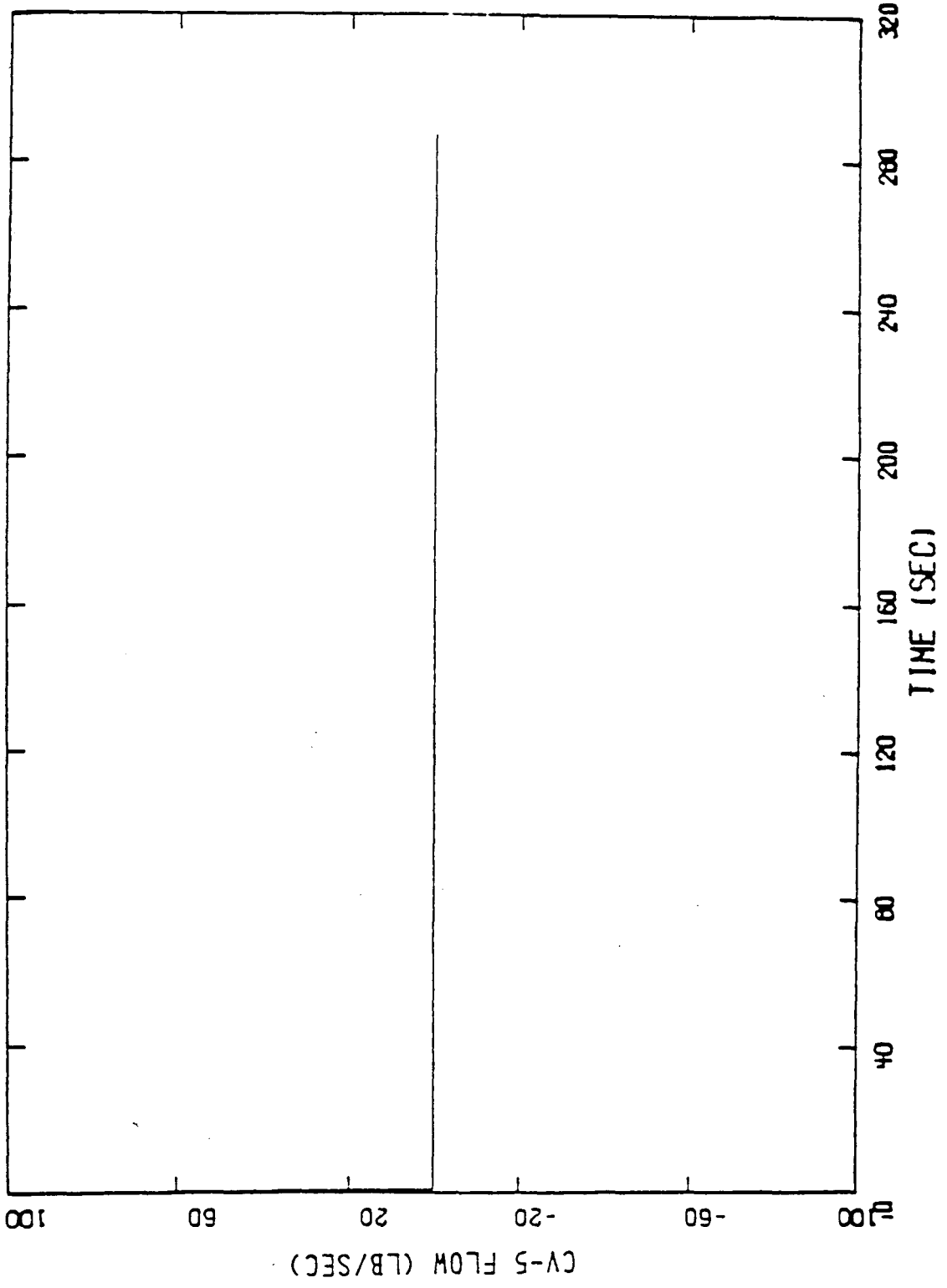
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Riser Inlet Intact Loop Pressure 40% Break Figure 15.6-187



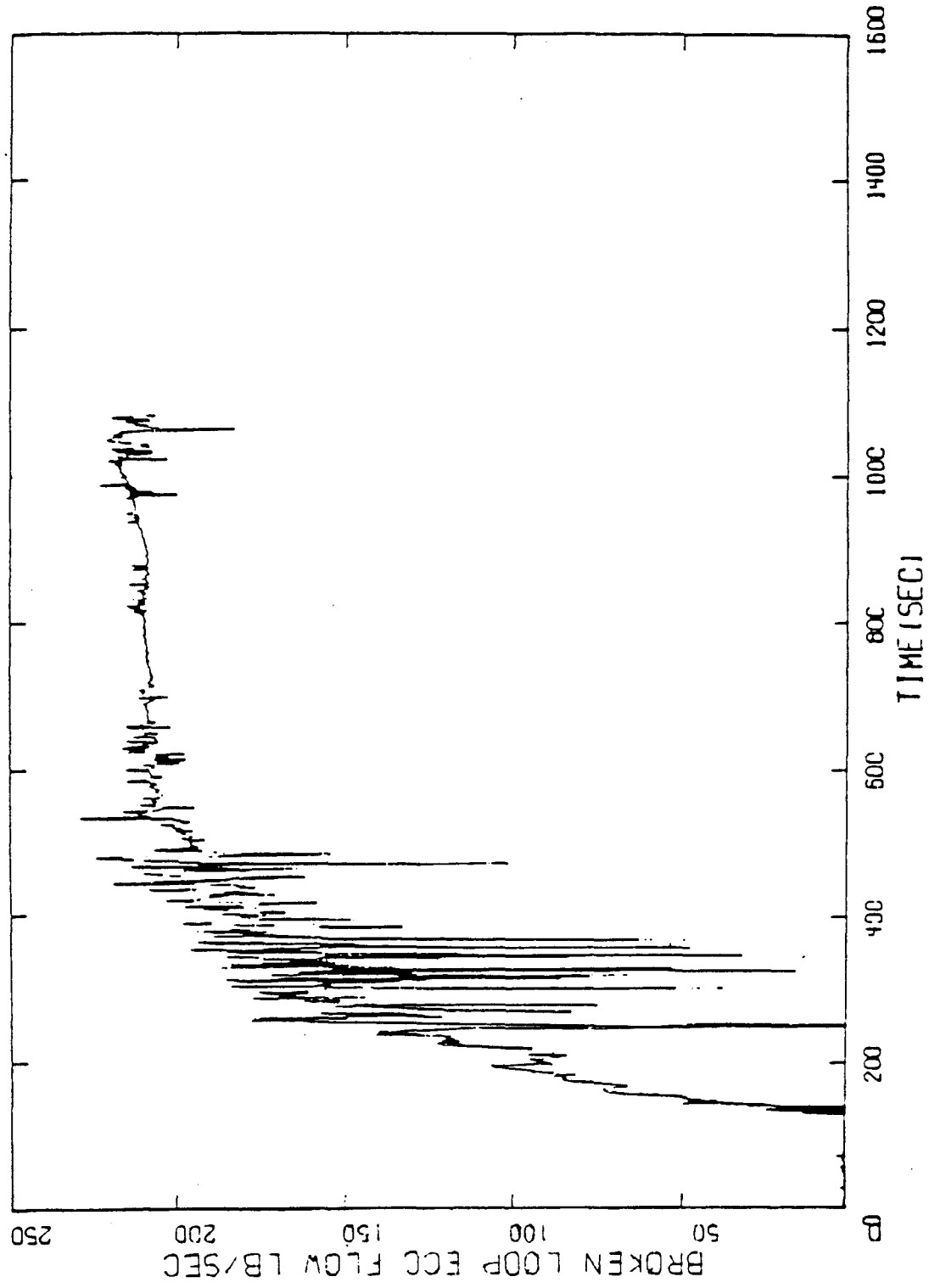
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	40% Break Case, Broken Loop, CV-5 Mass Flow Rate
	Figure 15.6-188



28909072.9

Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	100% Break, Broken Loop, CV-5 Mass Flow Rate
	Figure 15.6-189

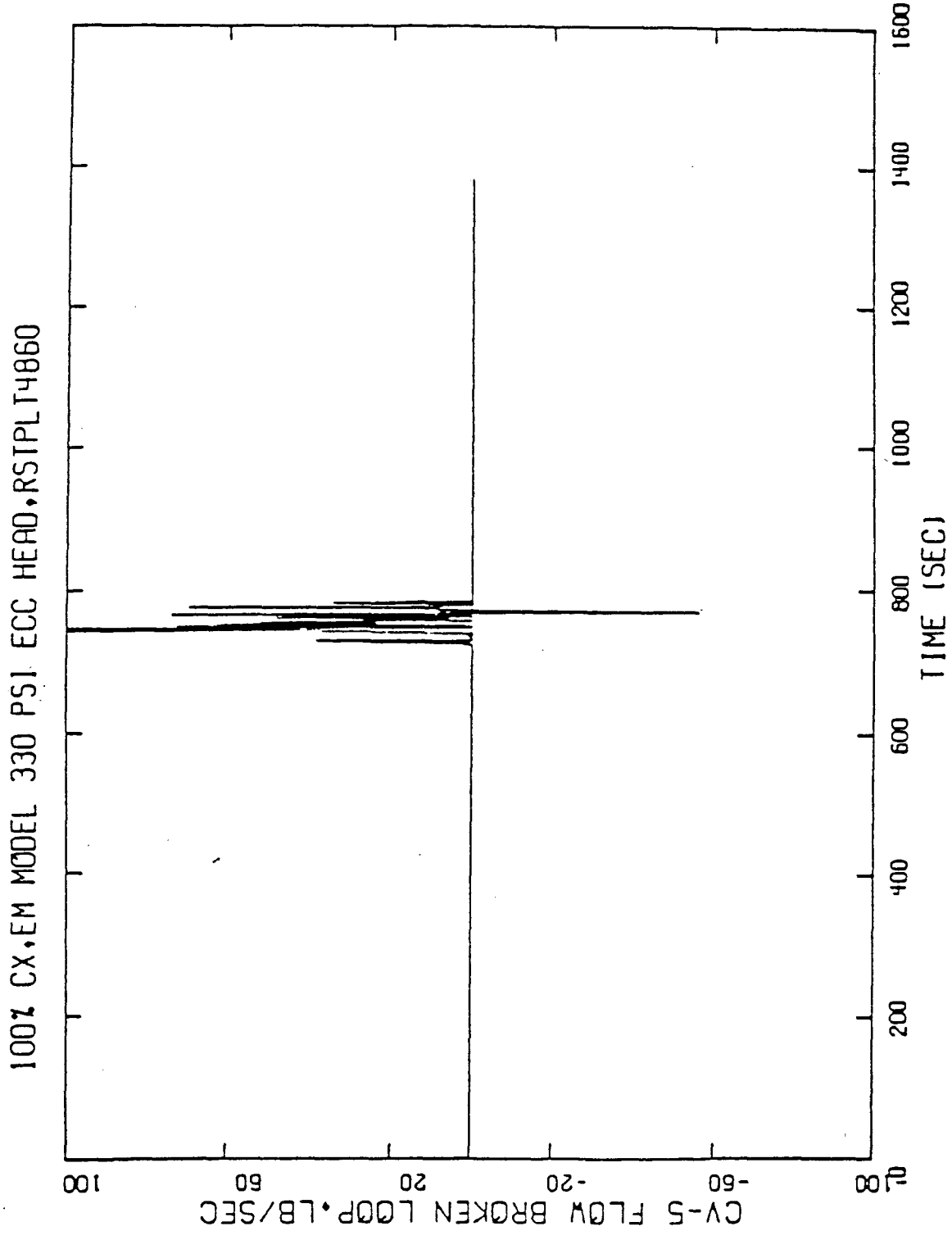


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Westinghouse Hanford Company
N REACTOR
SAFETY ANALYSIS REPORT

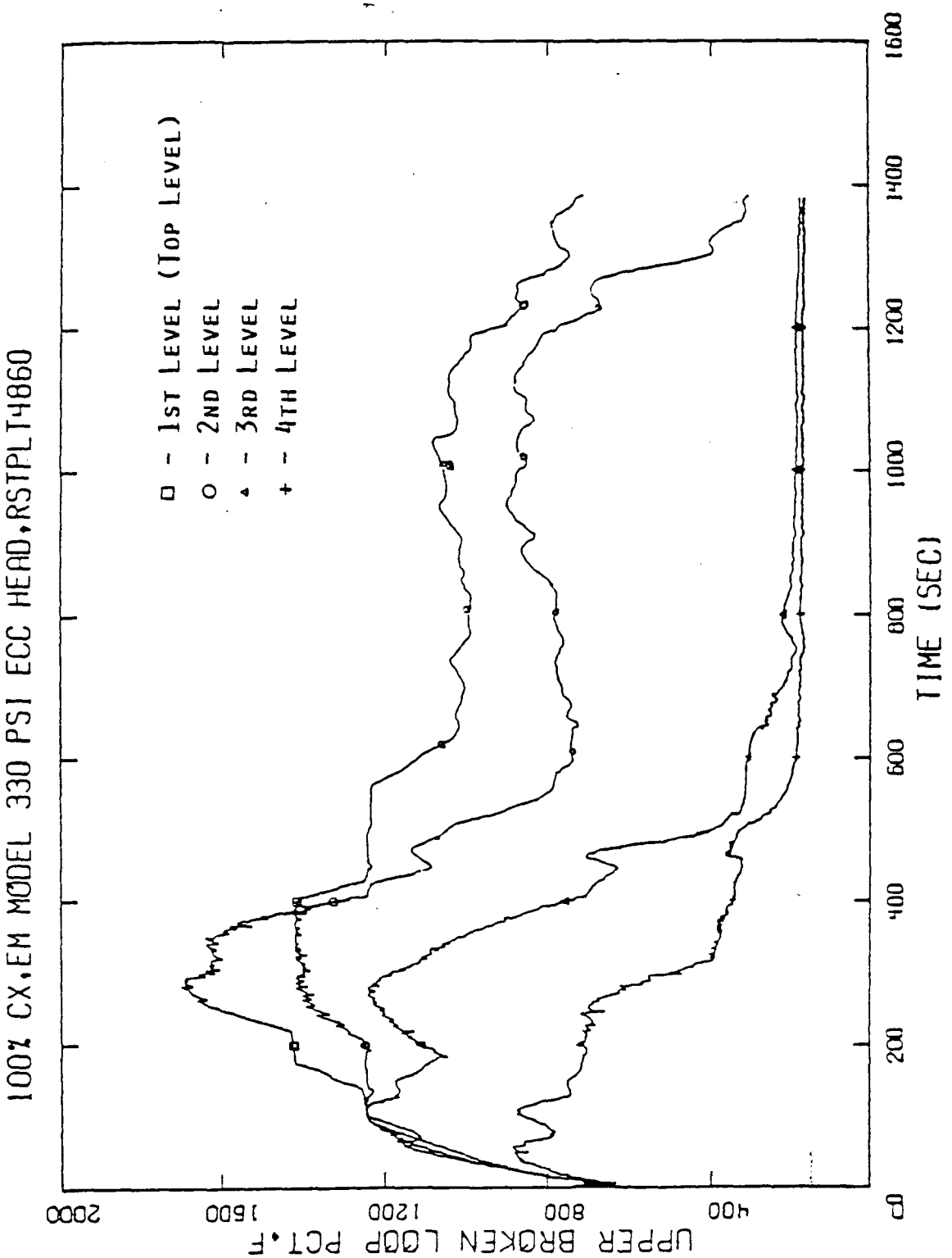
Broken Loop ECCS Flow
40% Break Case

Figure 15.6-190



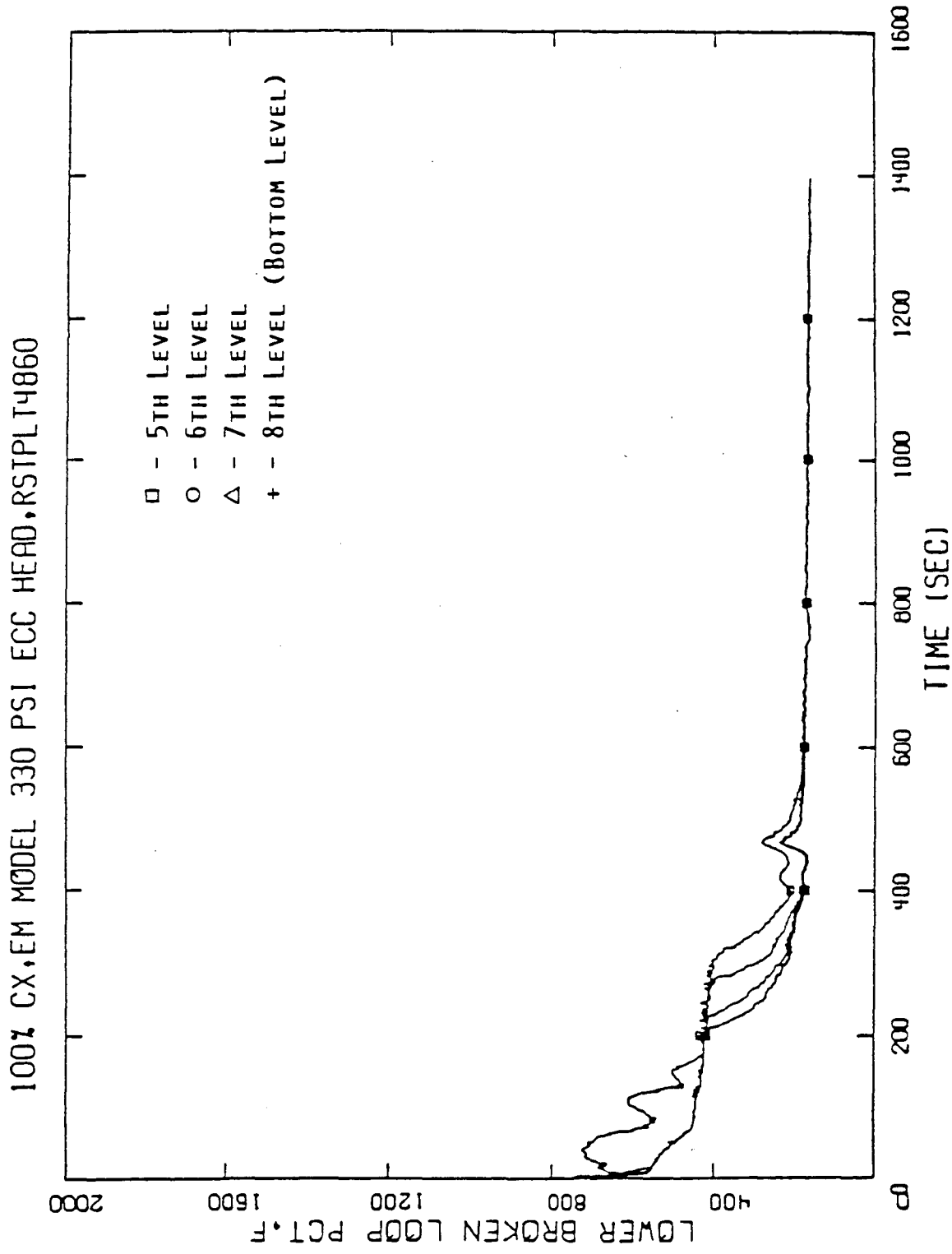
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Loop CV-5 Flow 100% EM Figure 15.6-191



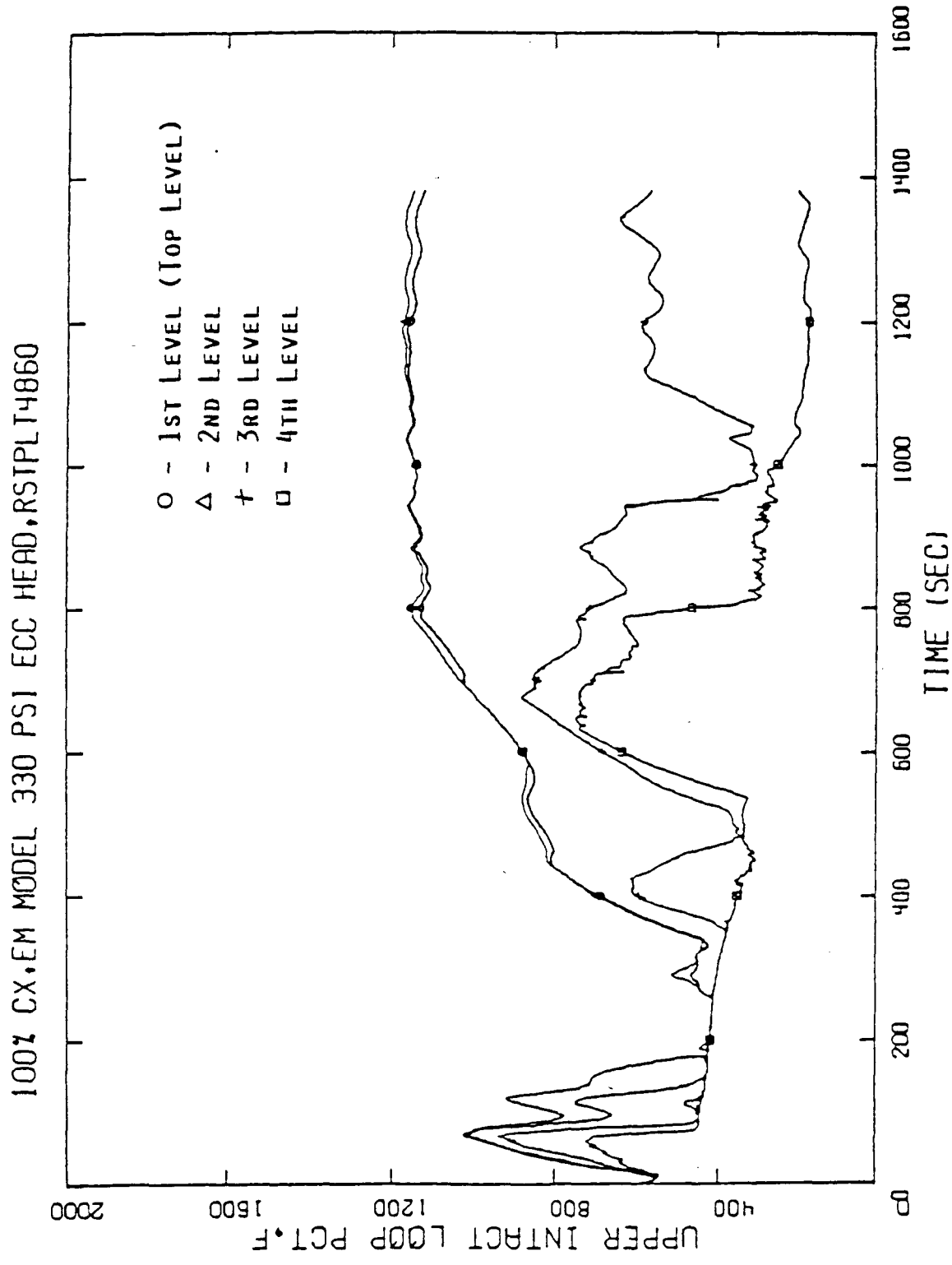
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Broken Loop PCT 100% EM Figure 15.6-192



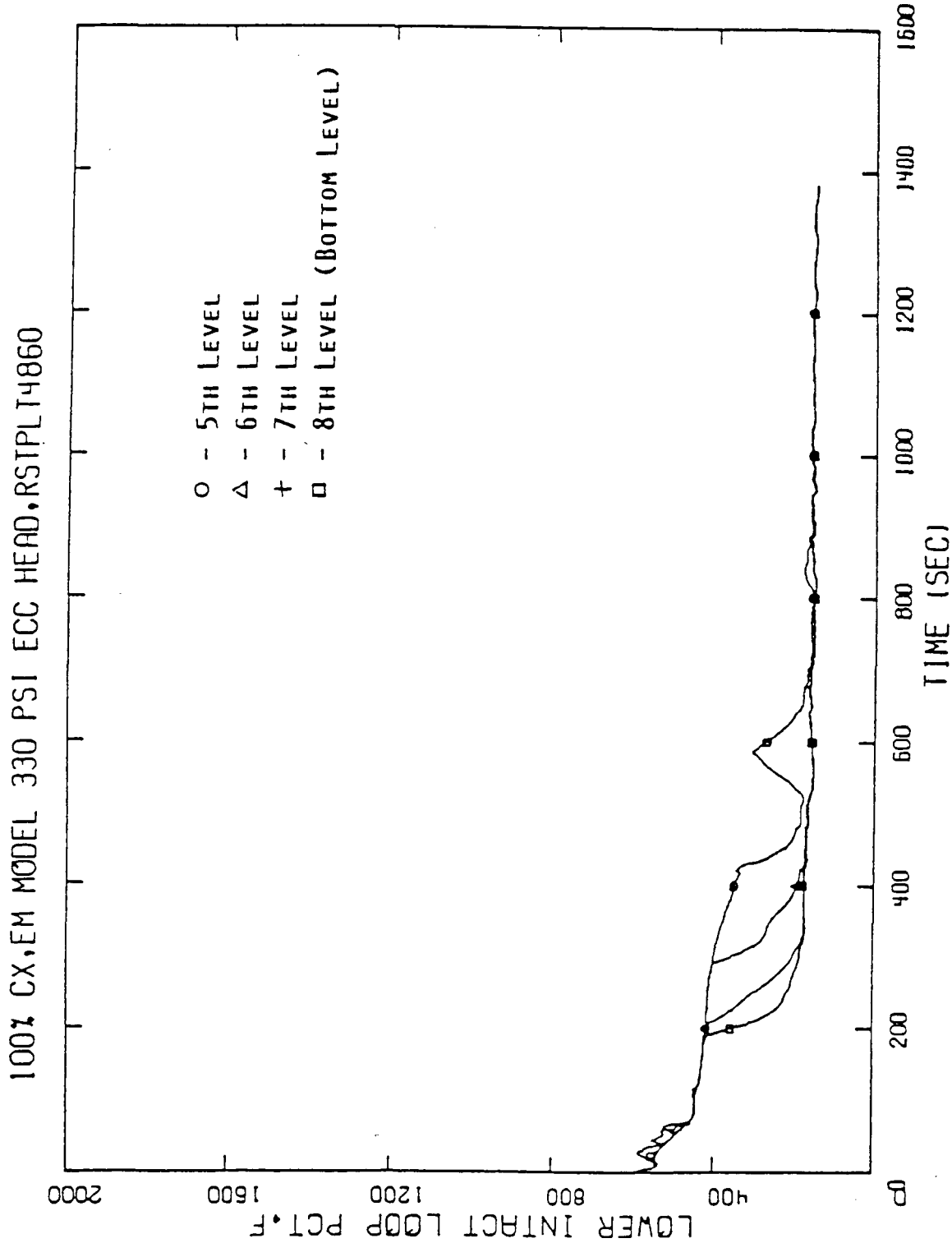
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Broken Loop PCT 100% EM
	Figure 15.6-193



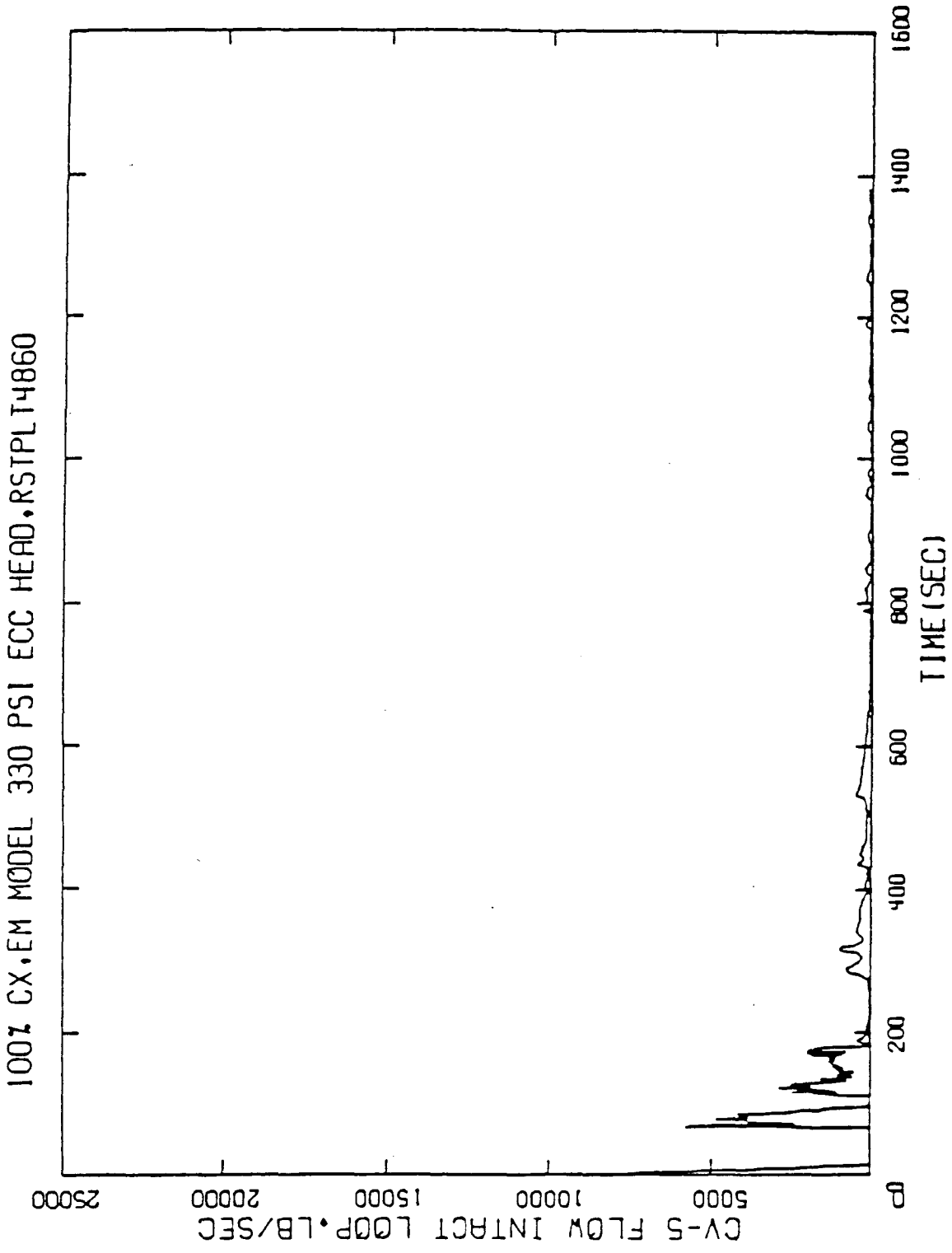
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Intact Loop PCT 100% EM
	Figure 15.6-194



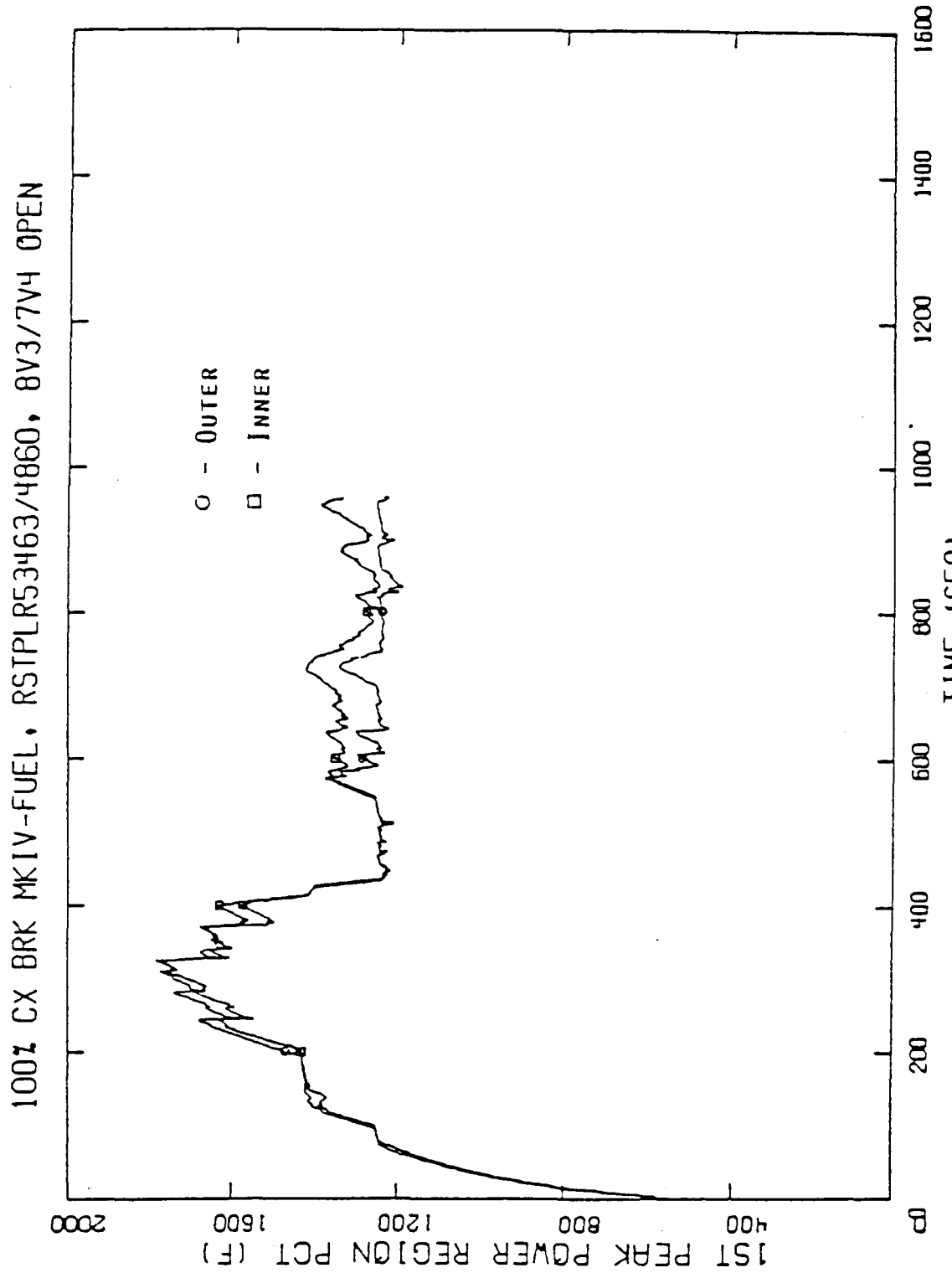
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Intact Loop PCT 100% EM Figure 15.6-195



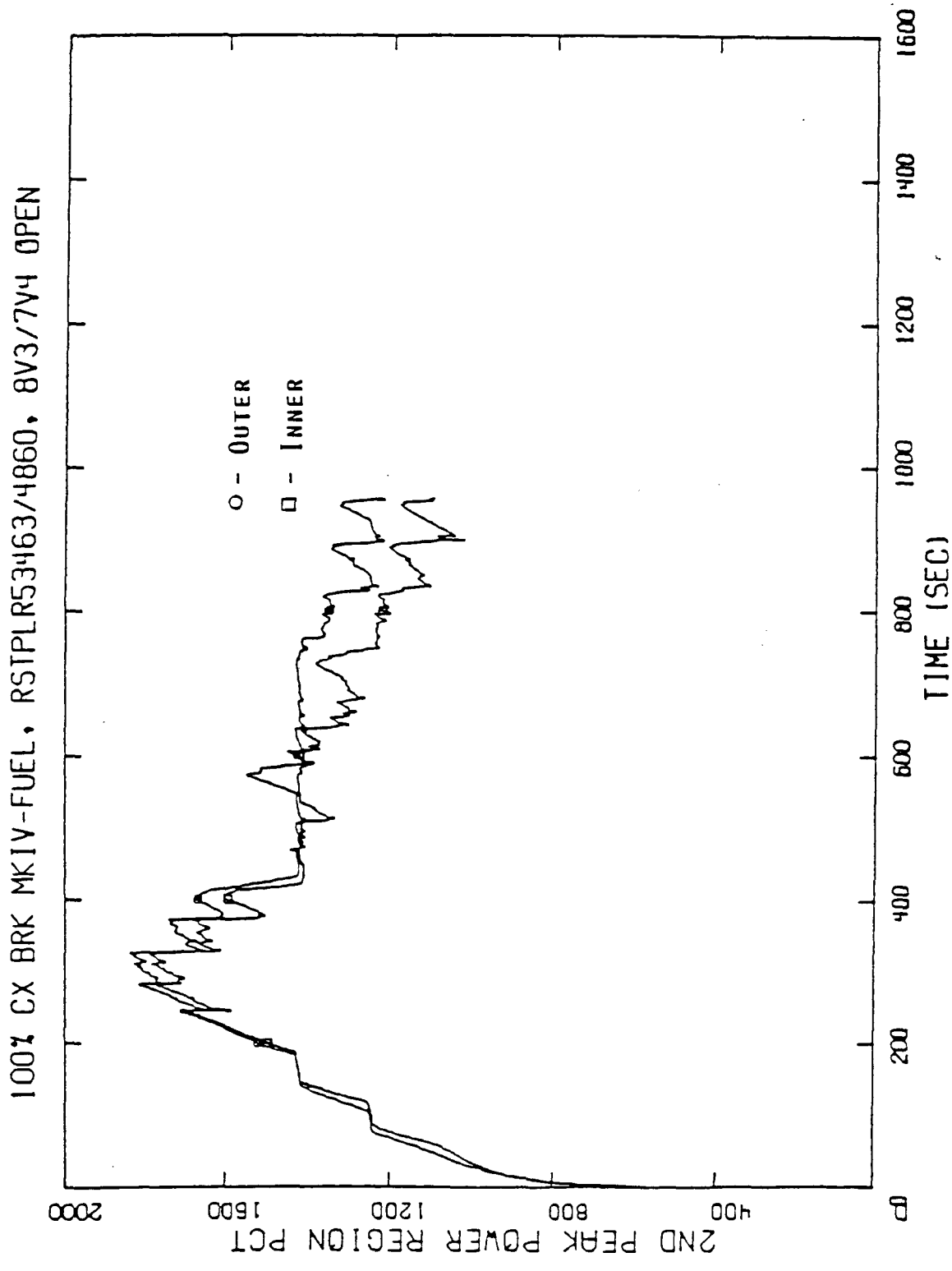
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Intact Loop CV-5 Flow
	Figure 15.6-196



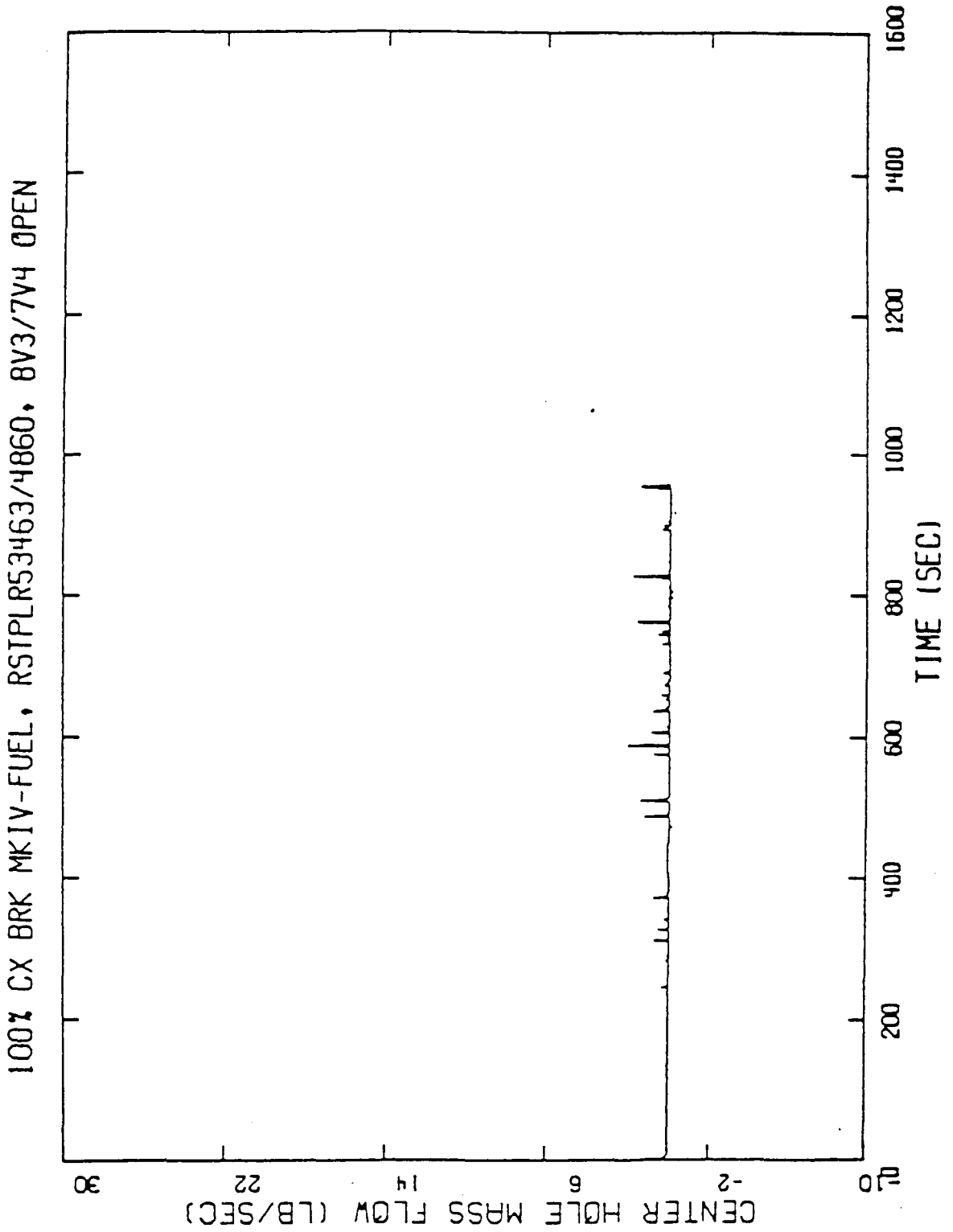
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Highest Power Region PCT Figure 15.6-197



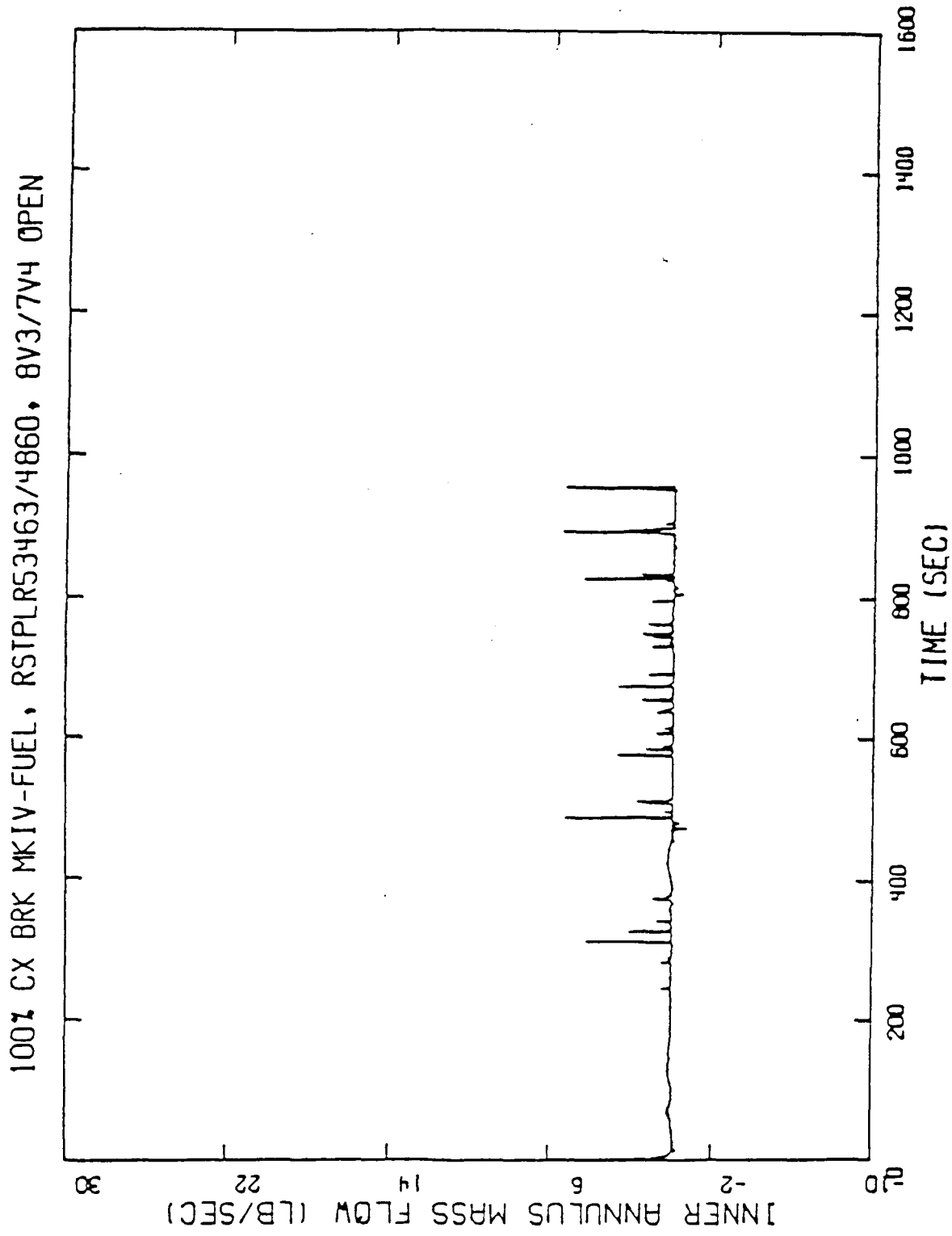
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Second Highest Power Region PCT Figure 15.6-198



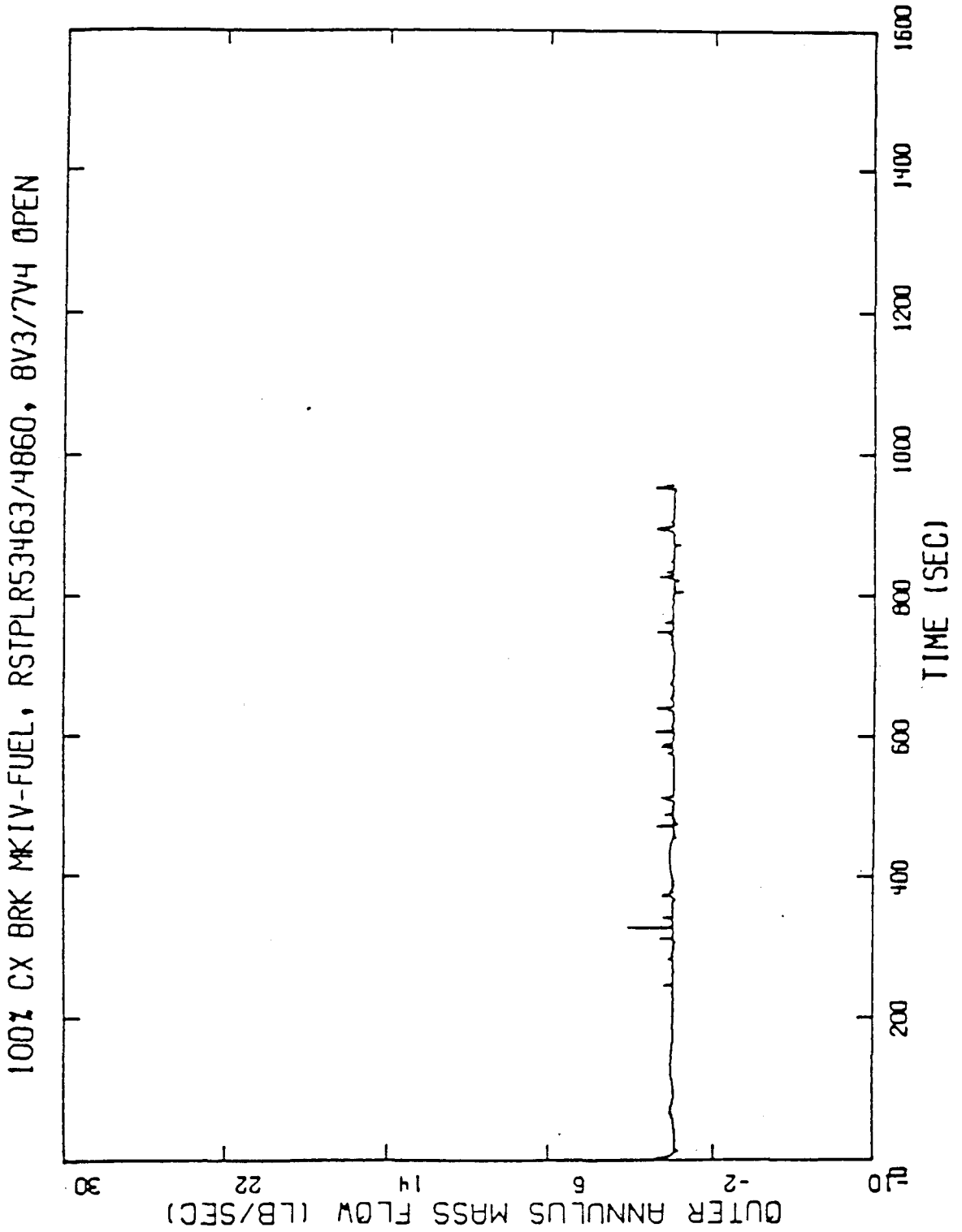
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Center Hole Mass Flow Figure 15.6-199



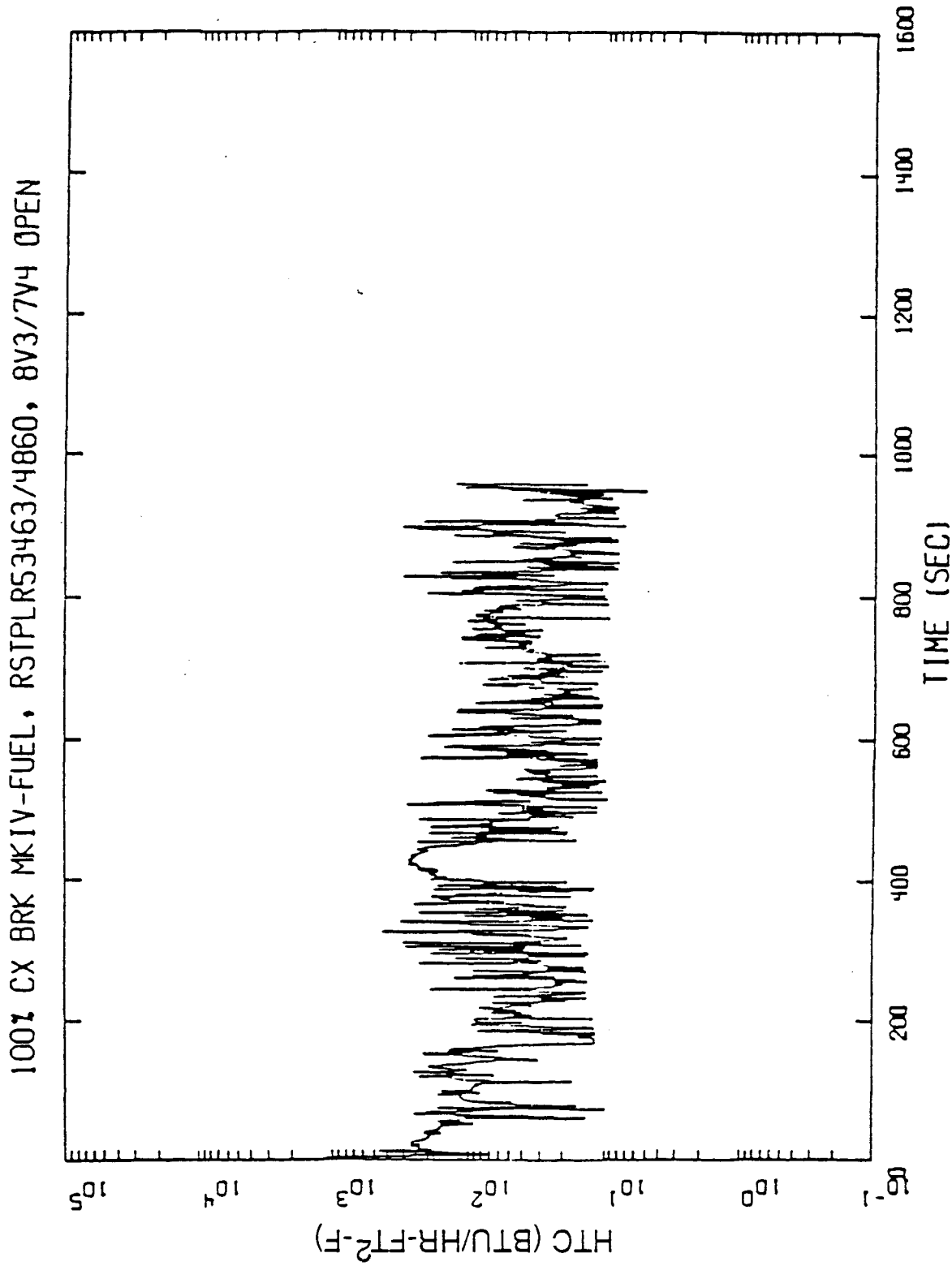
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner Annulus Mass Flow
	Figure 15.6-200



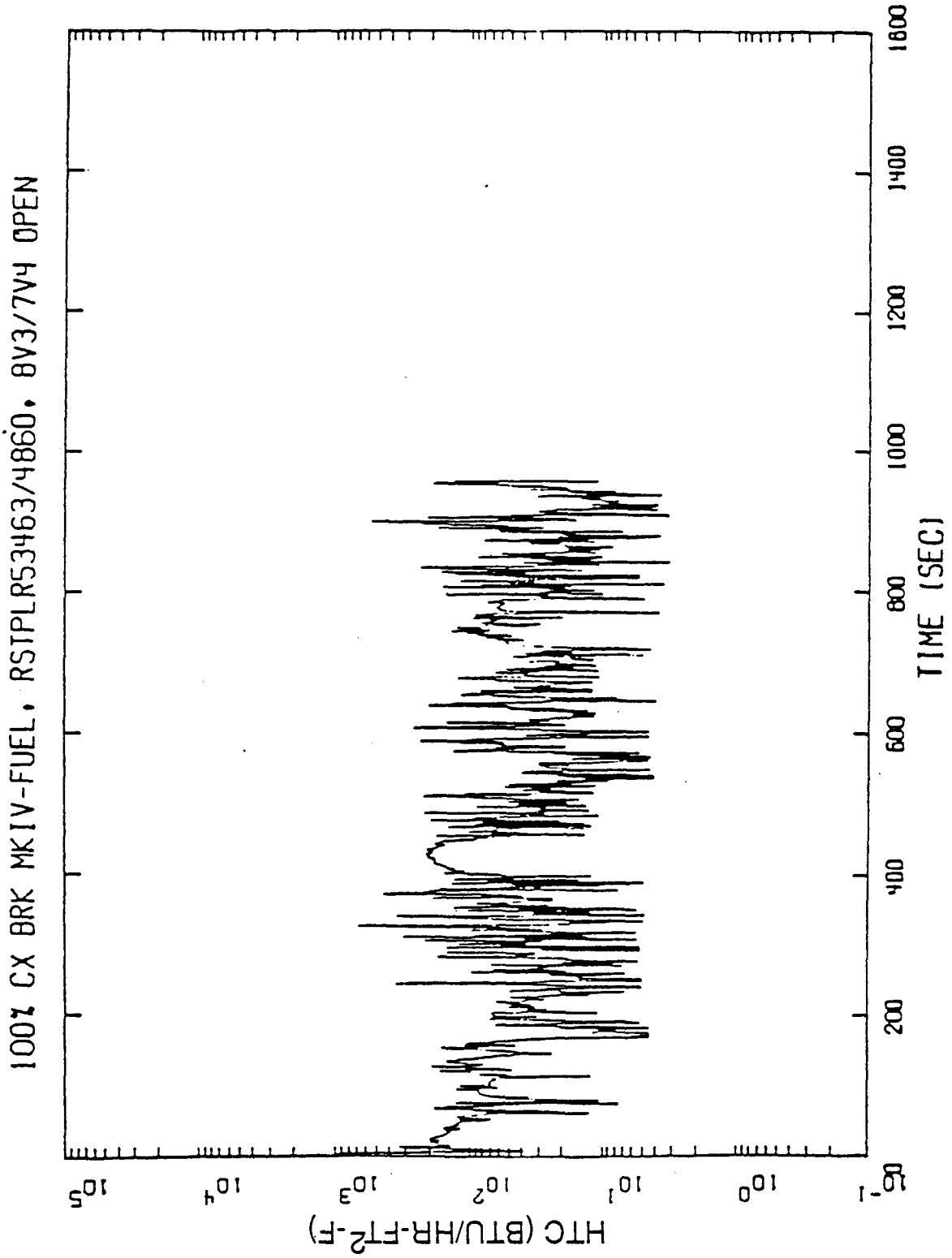
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outer Annulus Mass Flow
	Figure 15.6-201



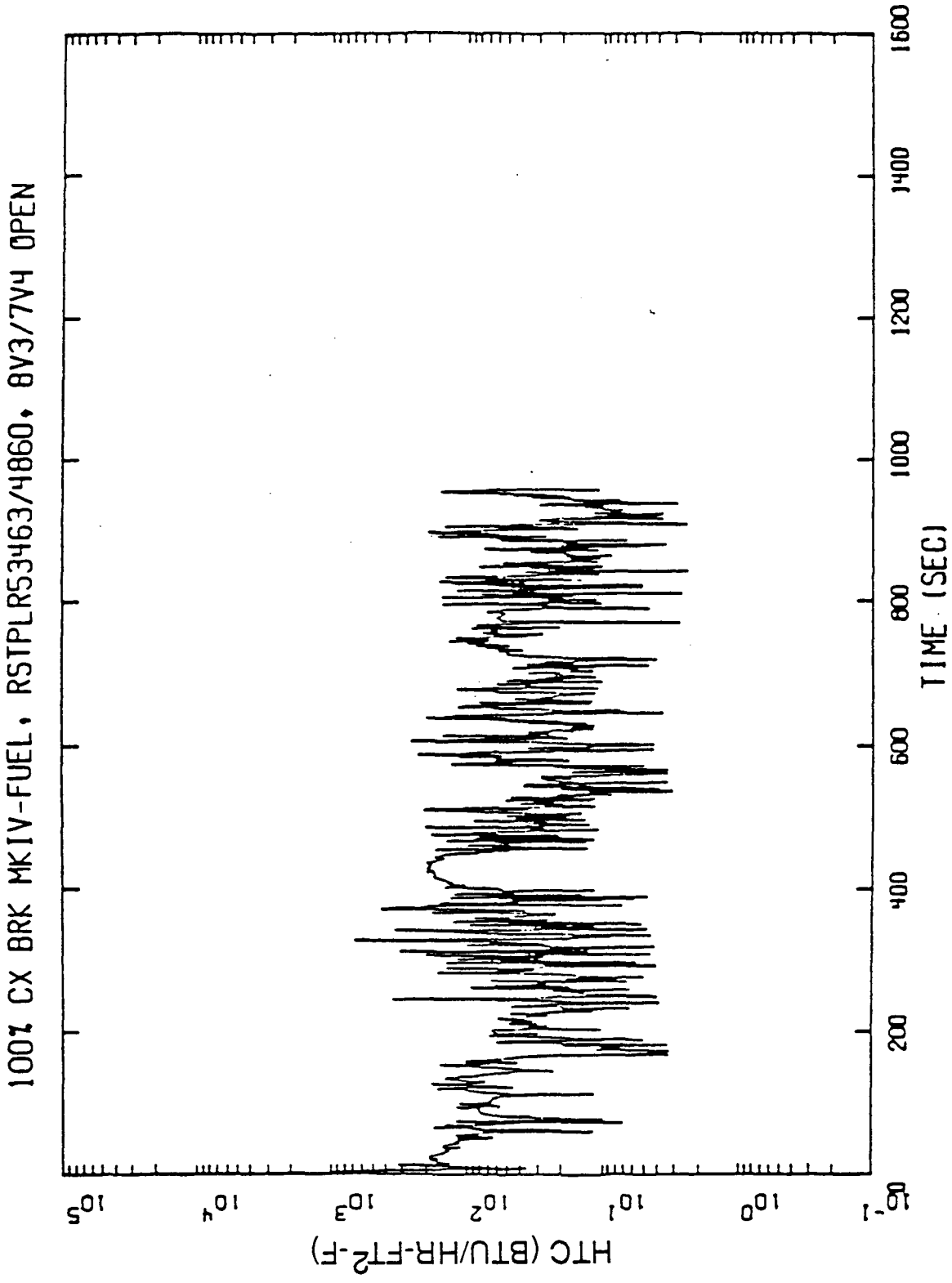
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner Fuel Inner Surface HTC
	Figure 15.6-202



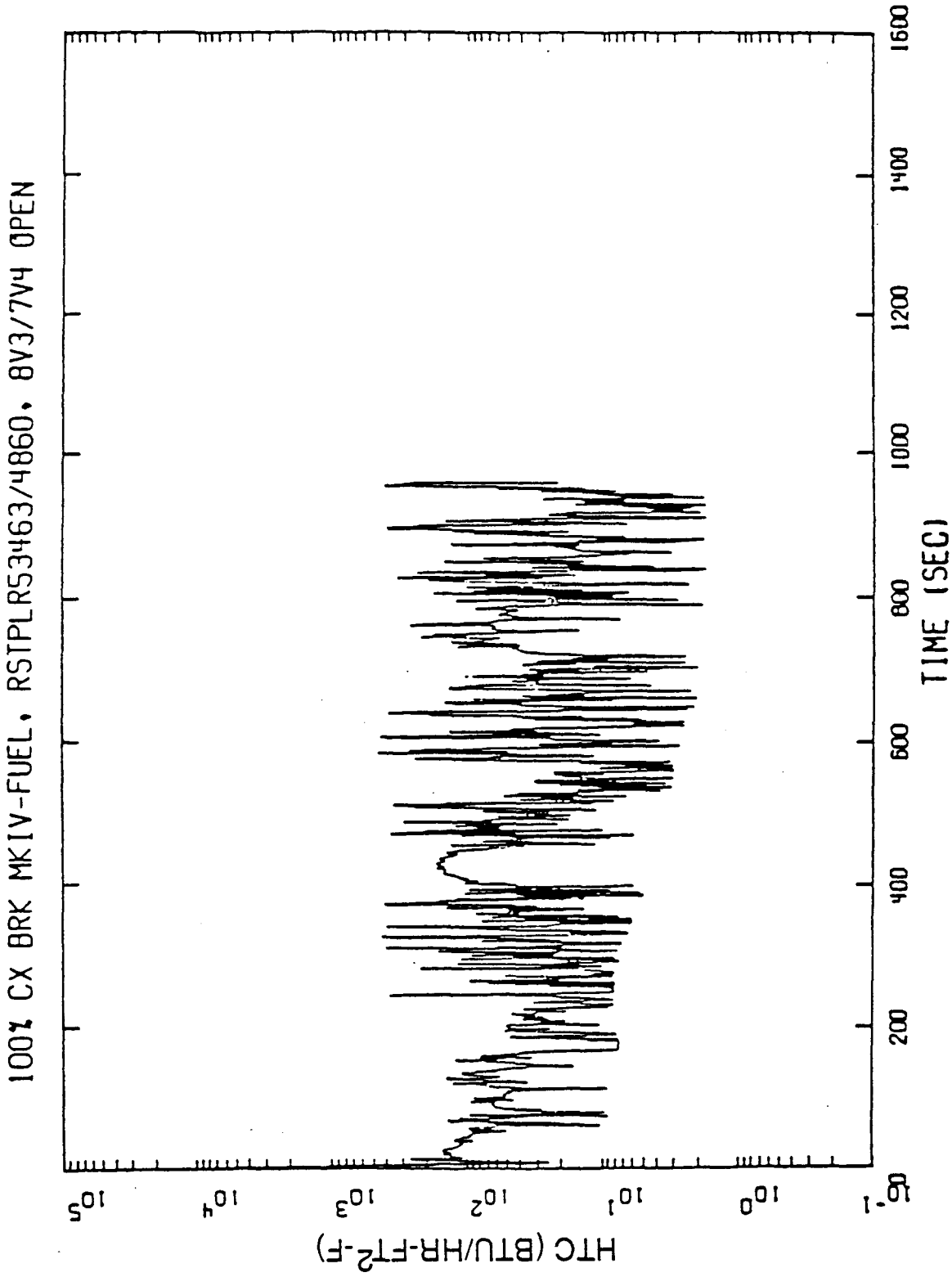
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner Fuel OuterSurface HTC Figure 15.6-203



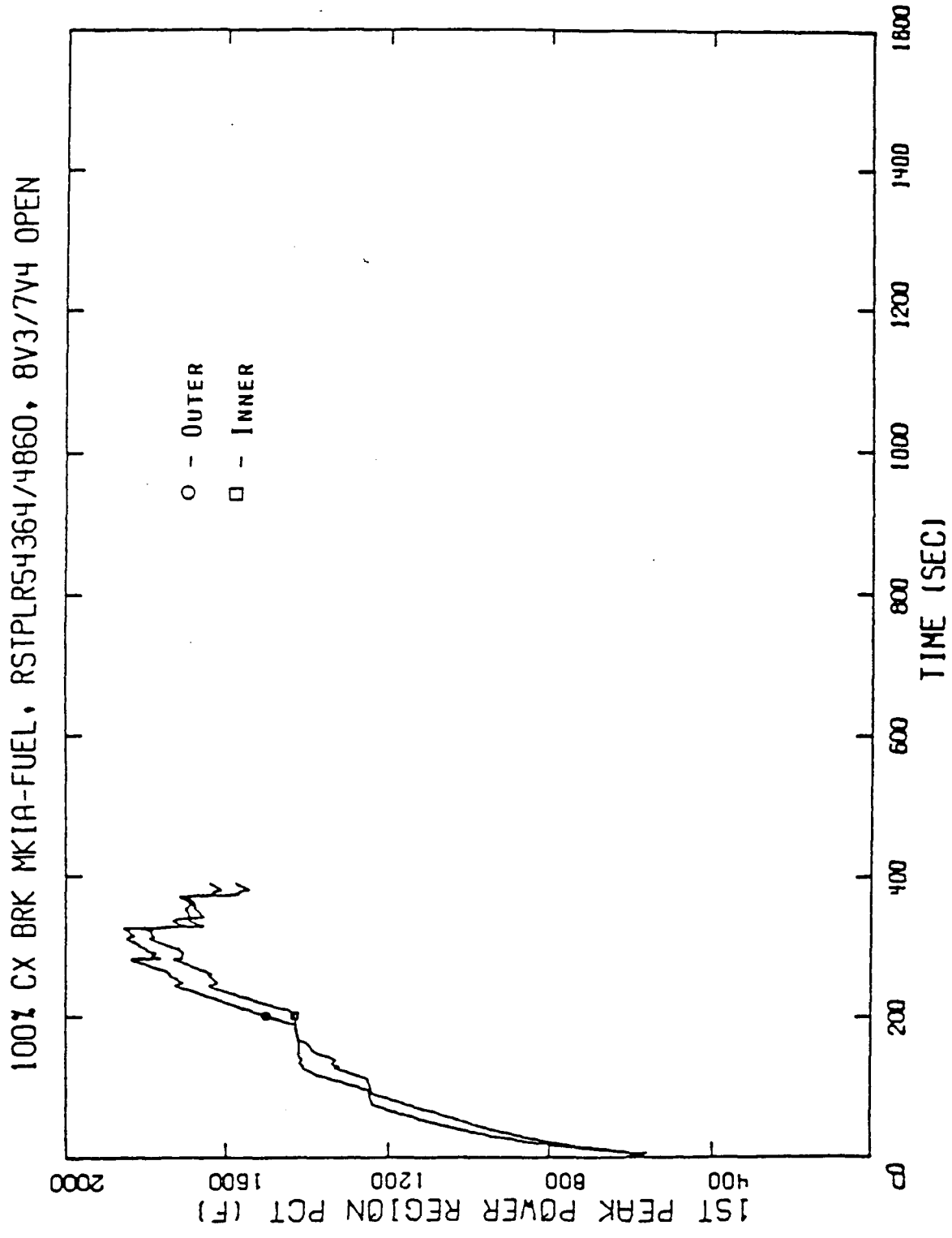
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outer Fuel InnerSurface HTC
	Figure 15.6-204



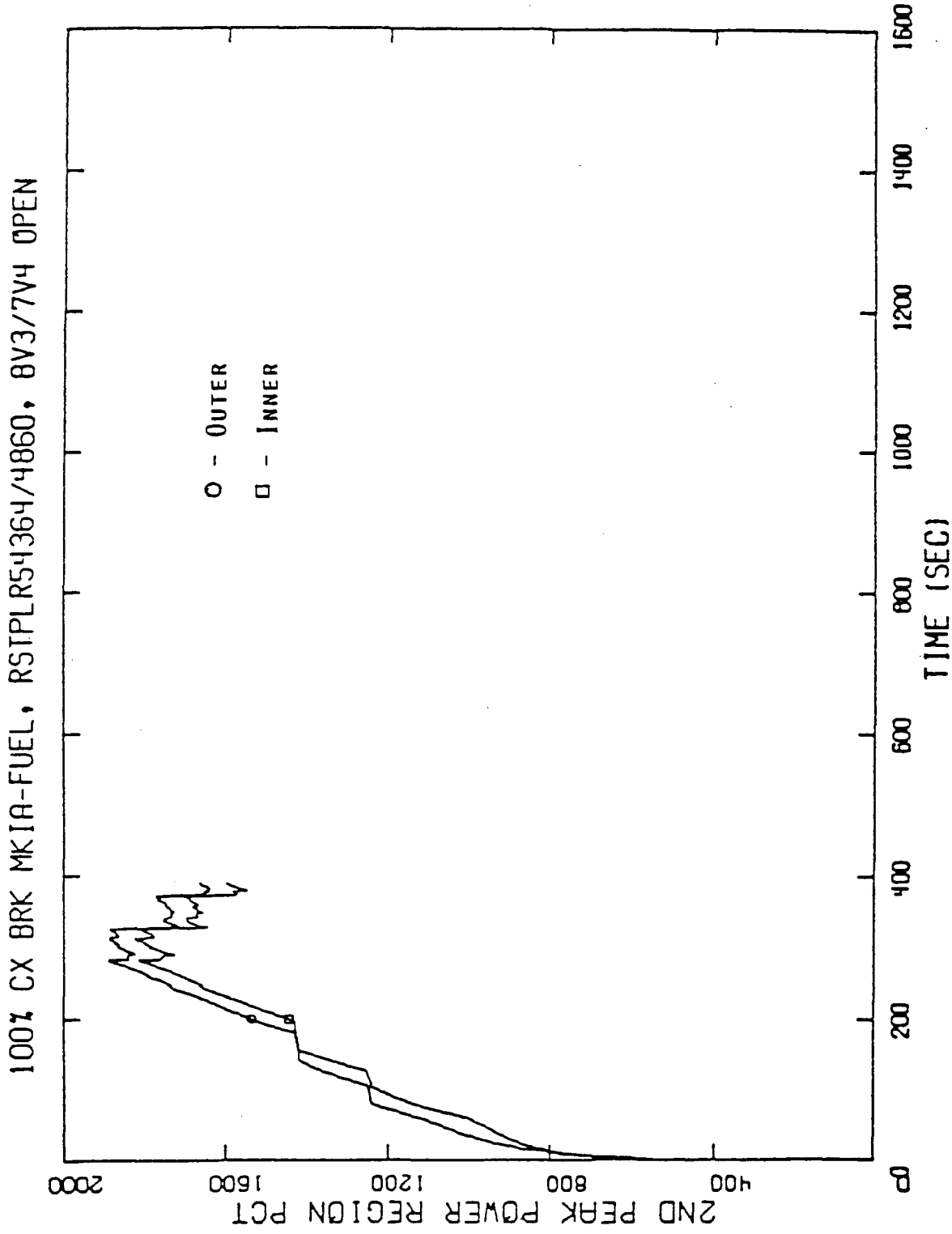
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outer Fuel OuterSurface HTC
	Figure 15.6-205



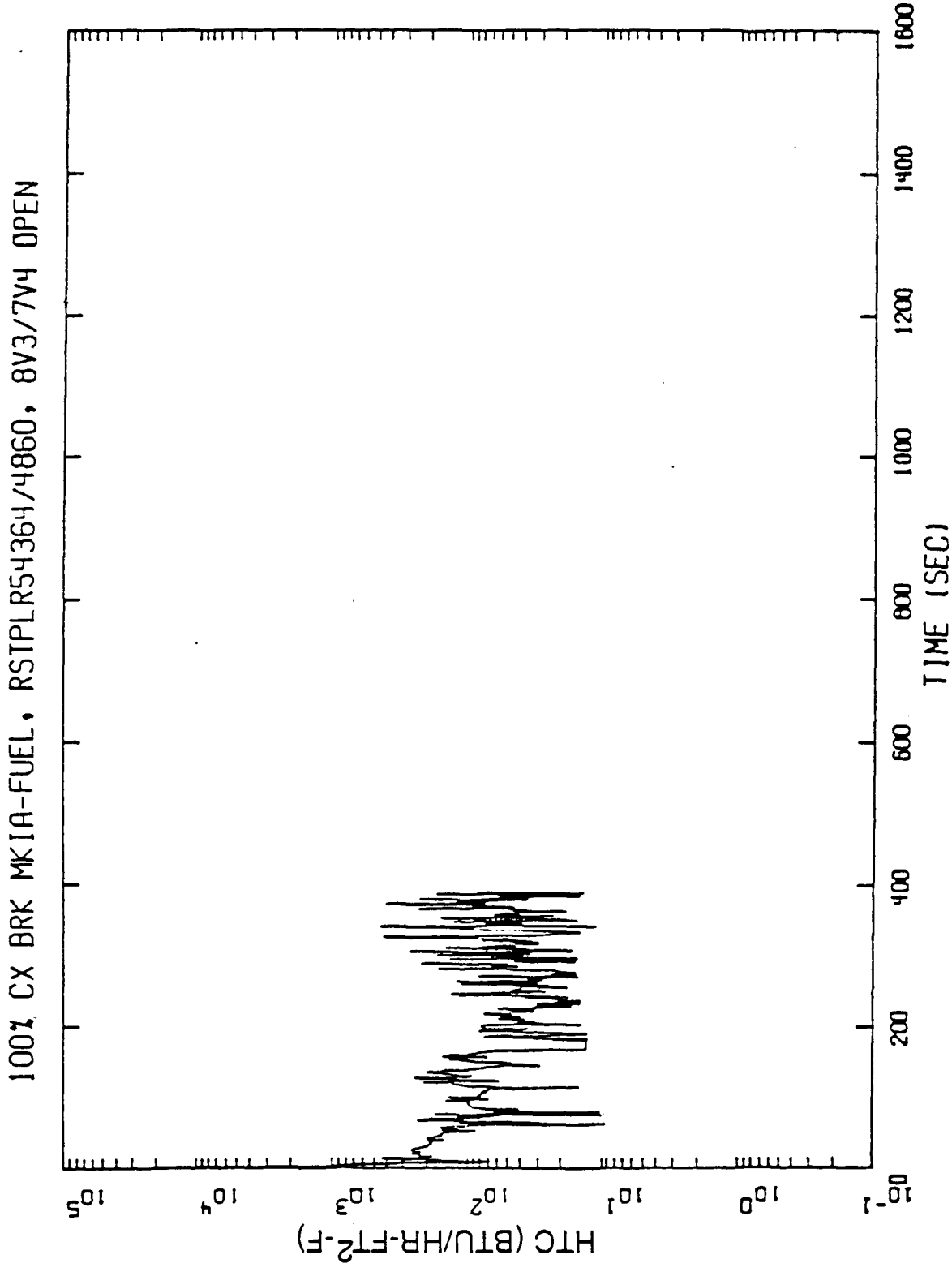
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Highest Power Region PCT
	Figure 15.6-206



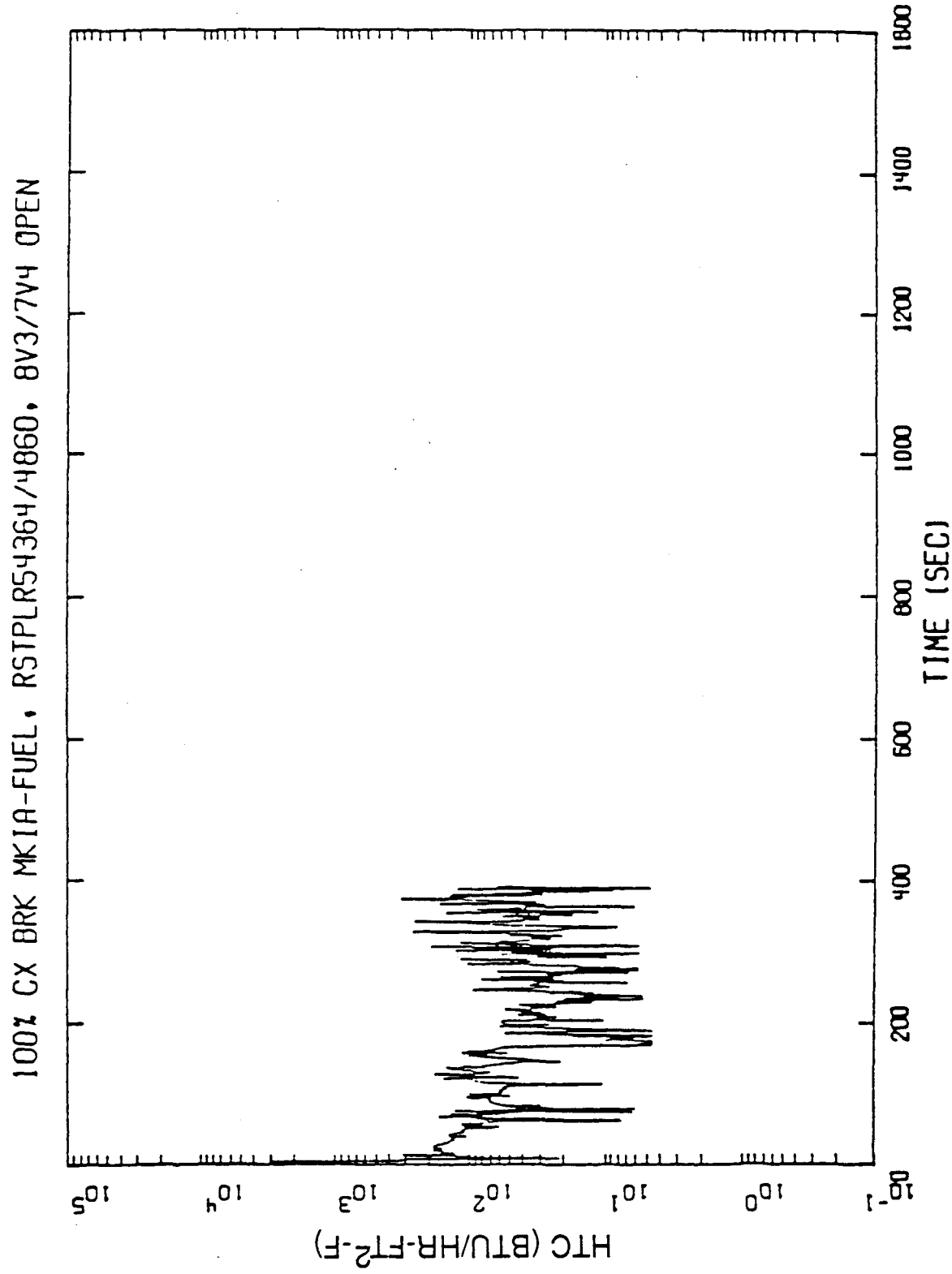
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Second Highest Power Region PCT Figure 15.6-207



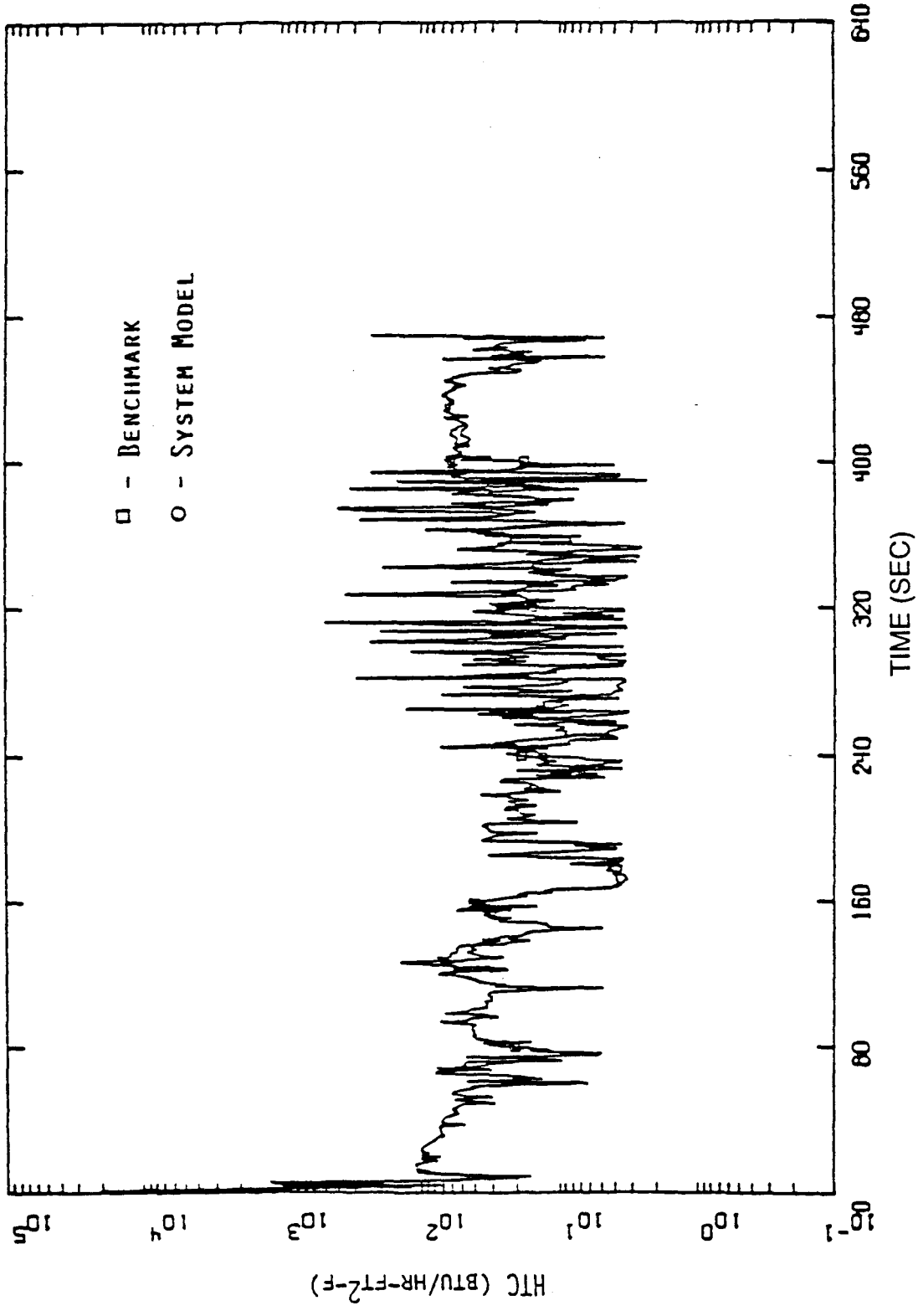
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner Fuel Inner Surface HTC
	Figure 15.6-208



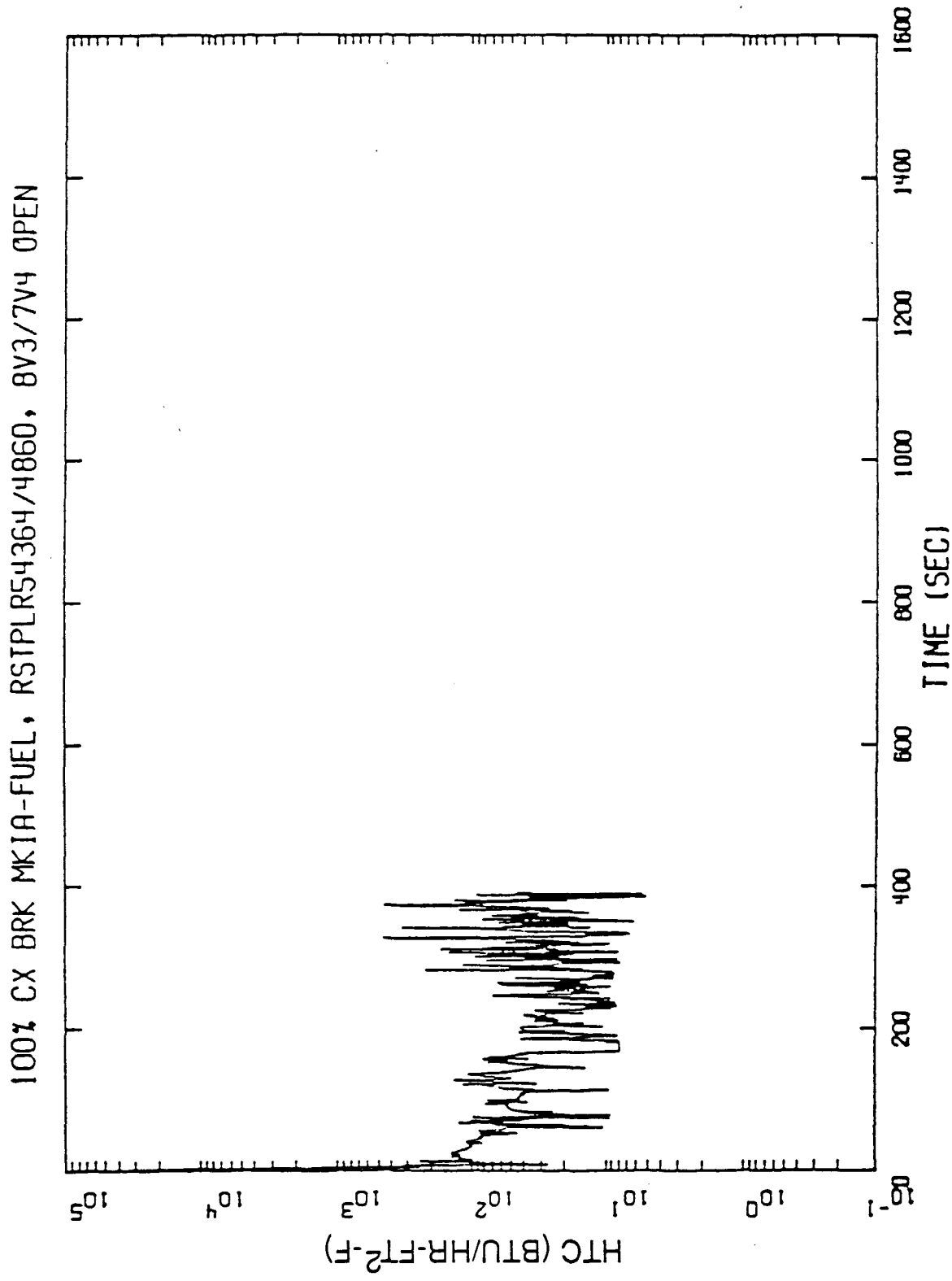
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner Fuel Outer Surface HTC
	Figure 15.6-209



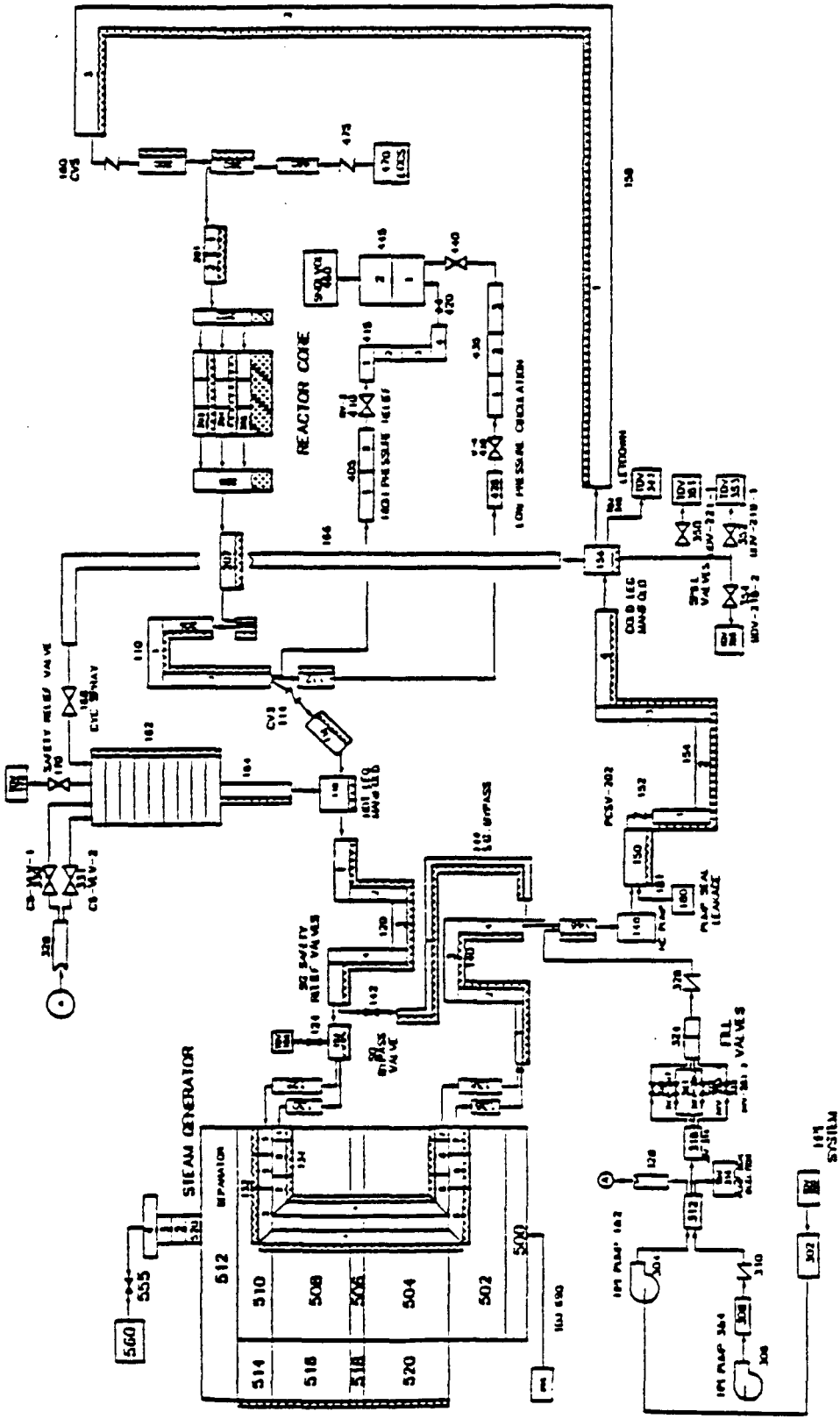
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outer Fuel Inner Surface HTC Figure 15.6-210



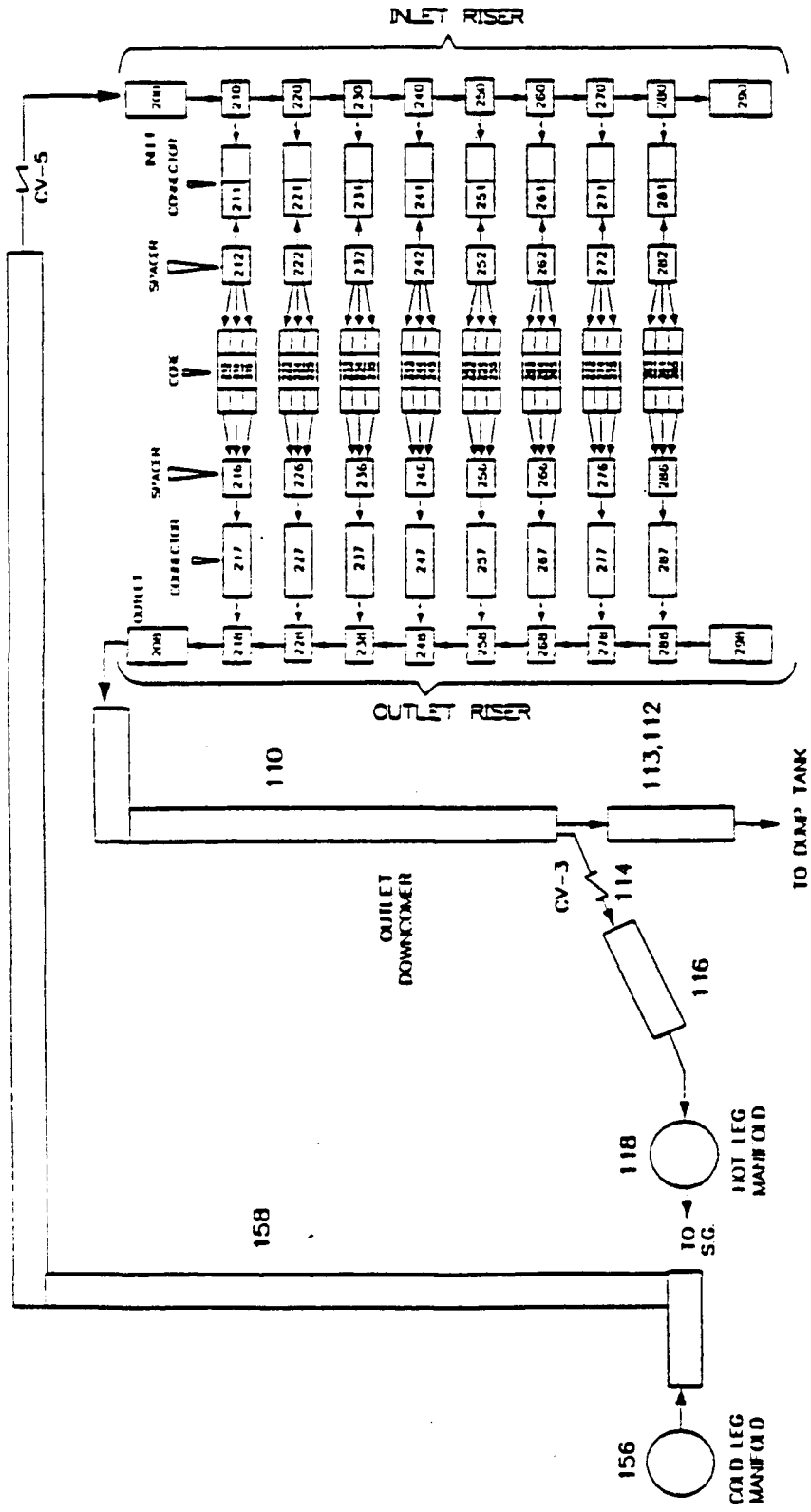
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outer Fuel Outer Surface HTC Figure 15.6-211



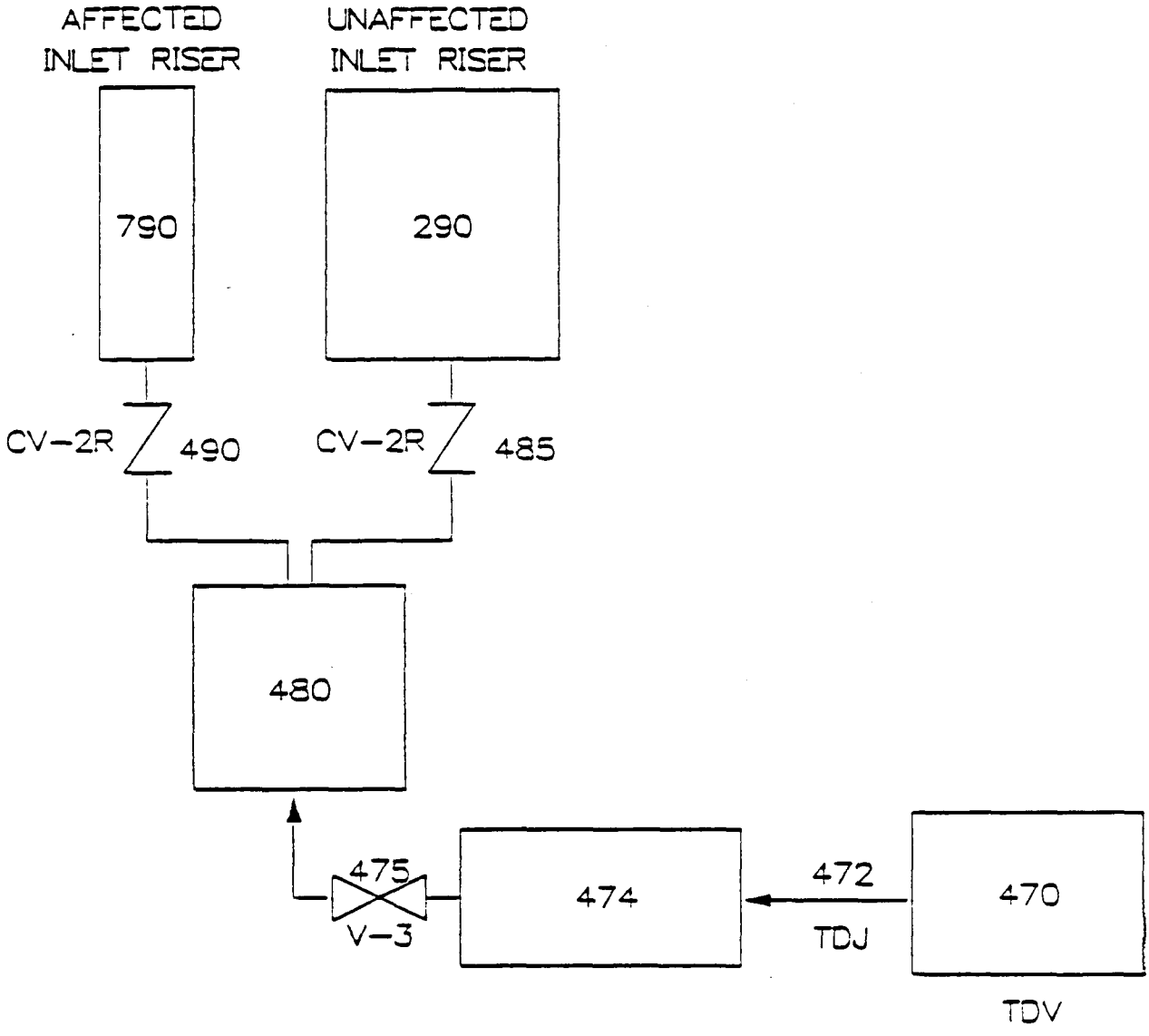
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Nodalization of the RELAP5 N Reactor Model for the Cold Leg and Hot Leg Riser Break Calculations
	Figure 15.6-212



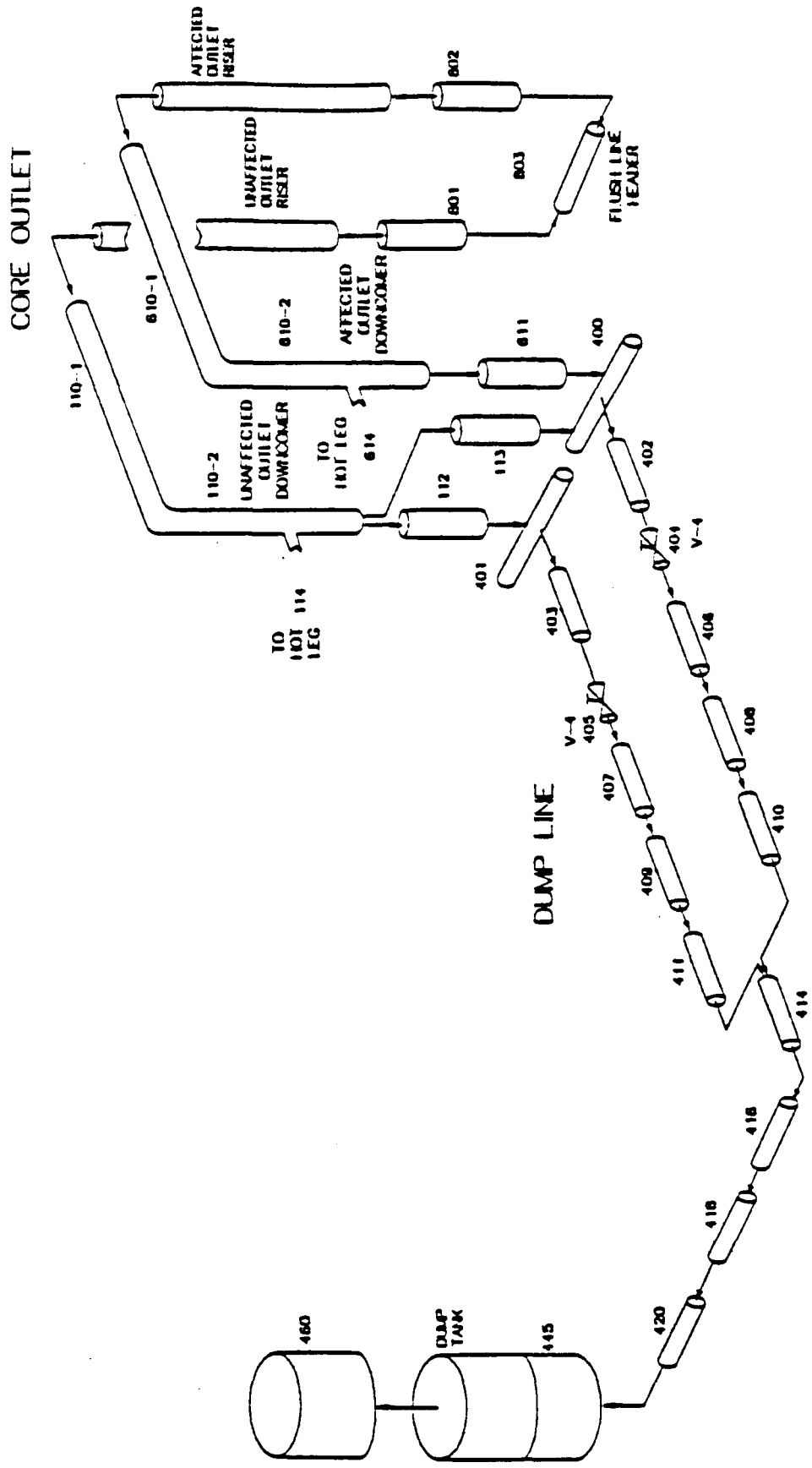
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected and Affected Core Nodalization Scheme (Unaffected Core Shown) for the Cold Leg and Hot Leg Riser Break Calculations Figure 15.6-213



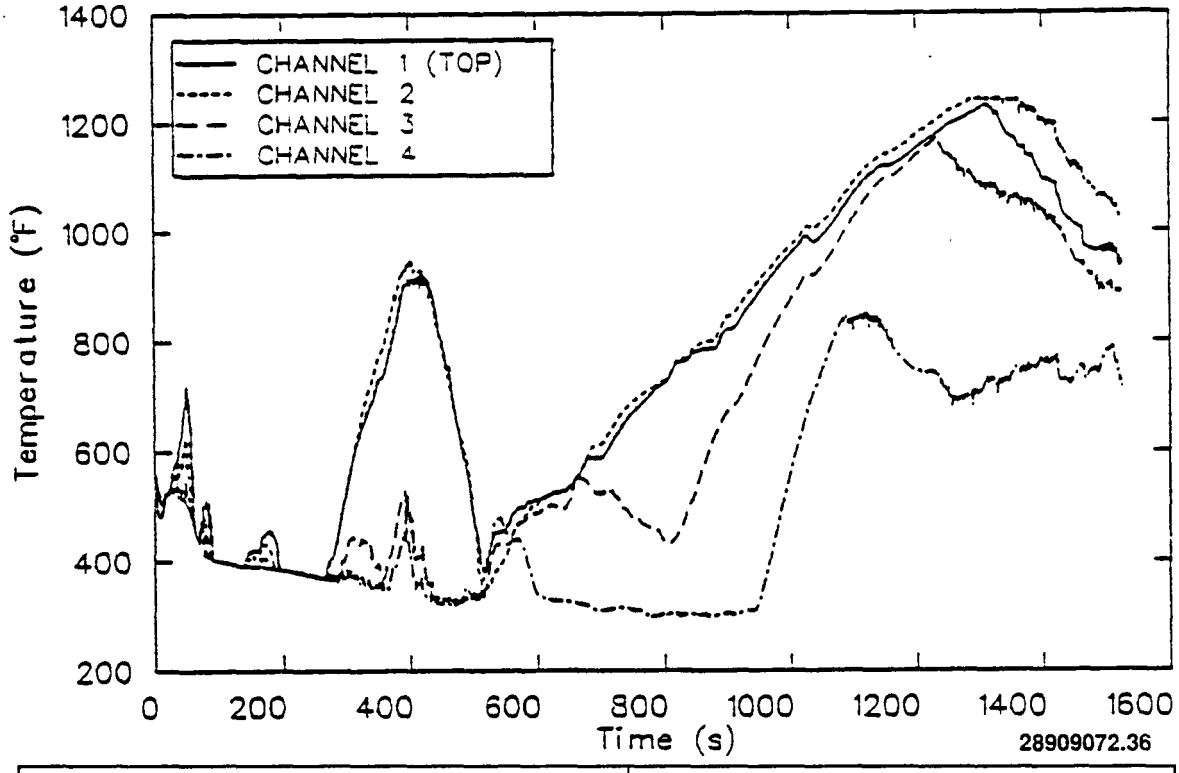
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Modified ECCS Nodalization for the Cold Leg and Hot Leg Riser Break Calculations
	Figure 15.6-214

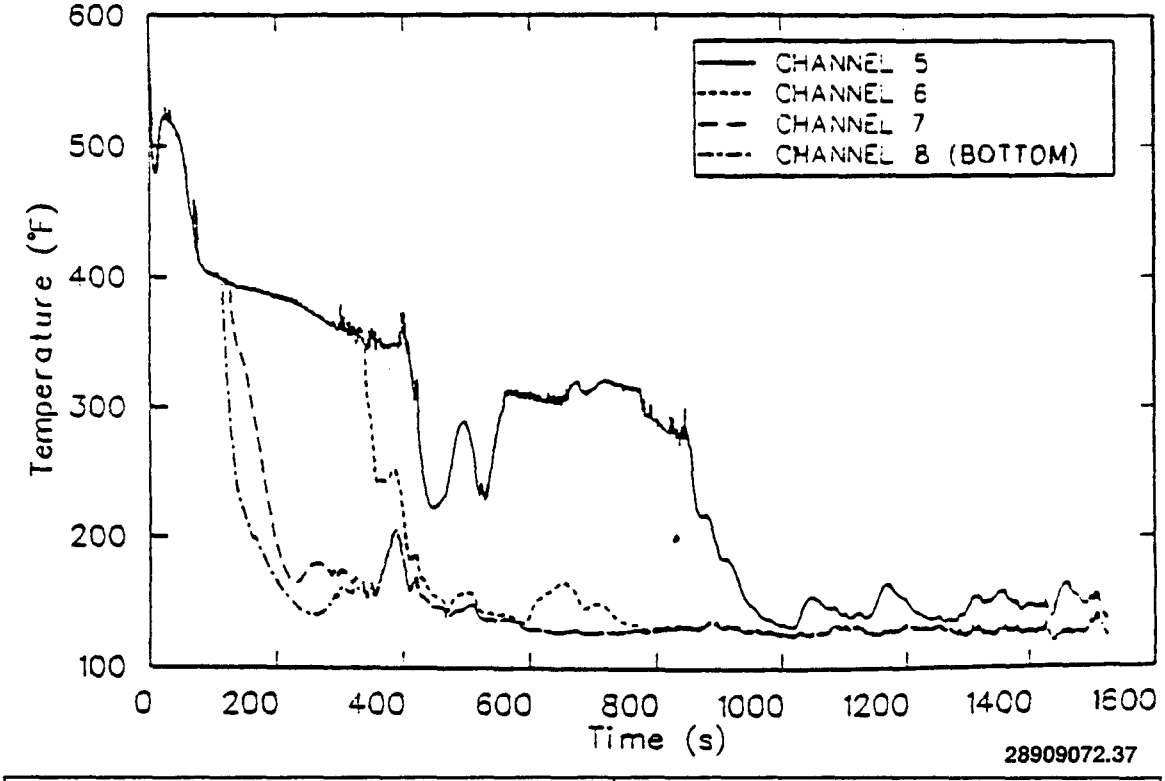


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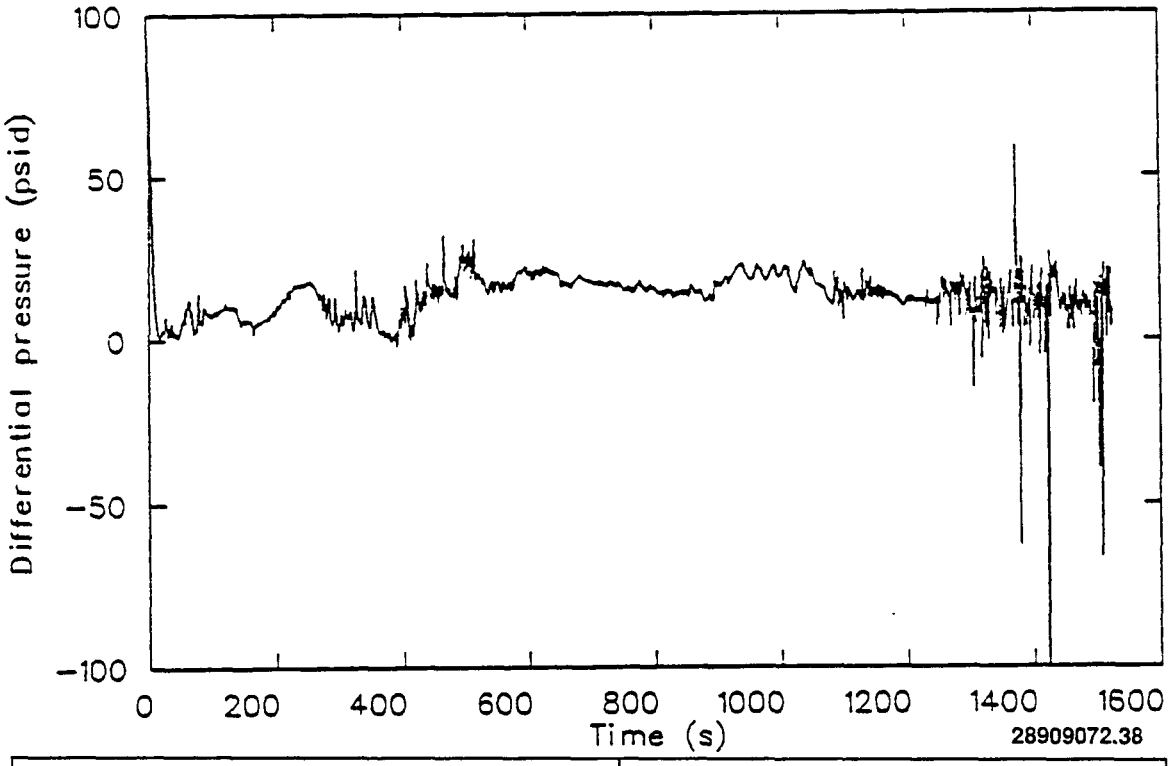
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Core Outlet and Dump Line System Modeling for the Cold Leg and Hot Leg Riser Break Calculations
	Figure 15.6-215



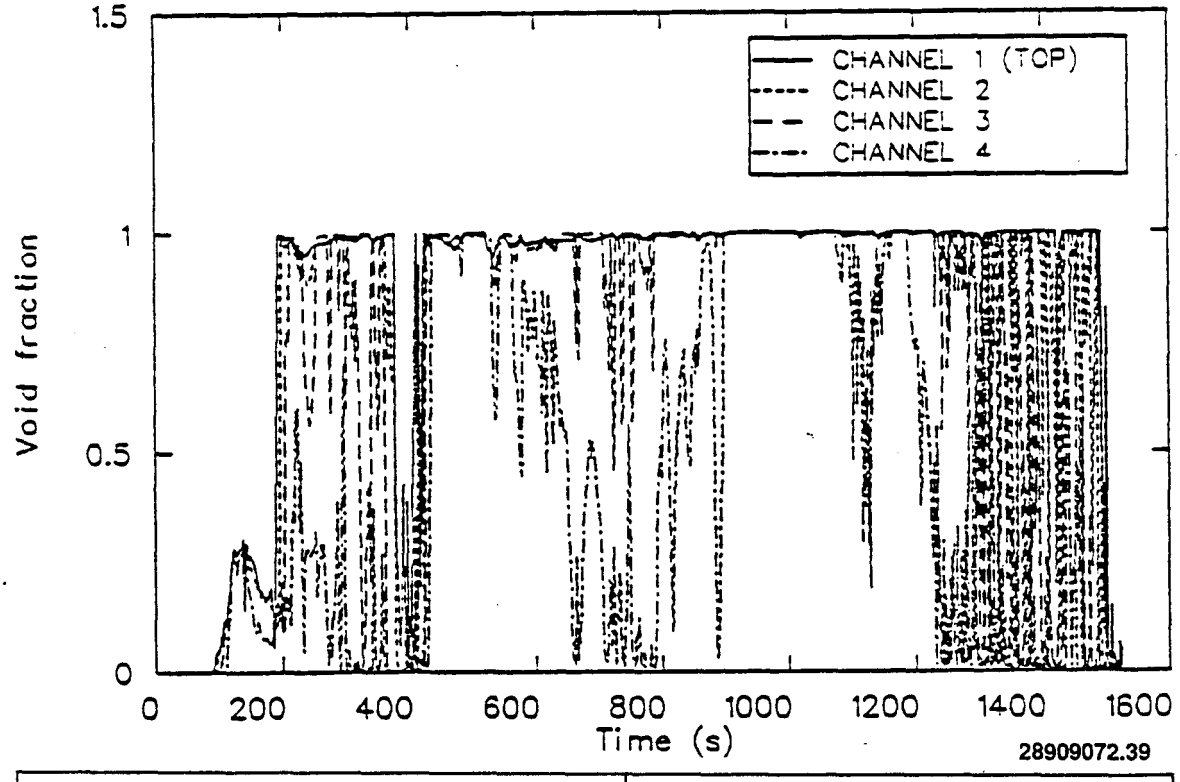
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Upper Unaffected Core: Inlet Riser Break, Base Case Figure 15.6-216



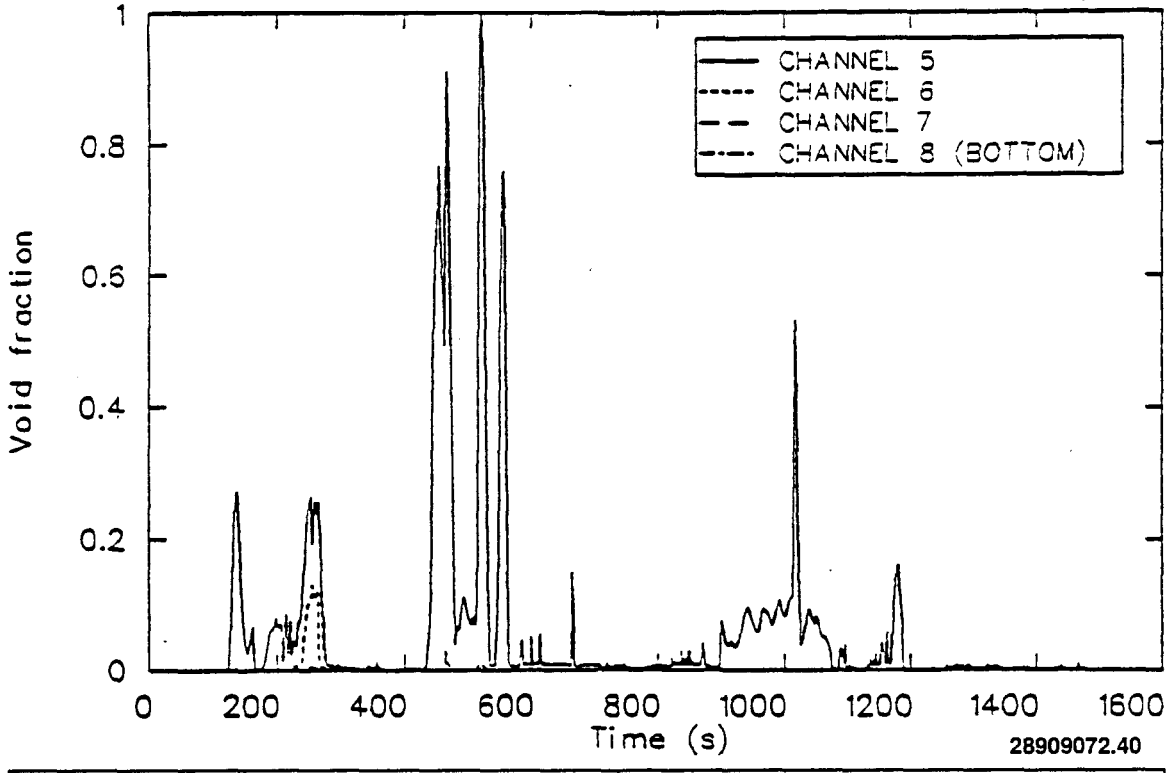
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Lower Unaffected Core: Inlet Riser Break, Base Case Figure 15.6-217



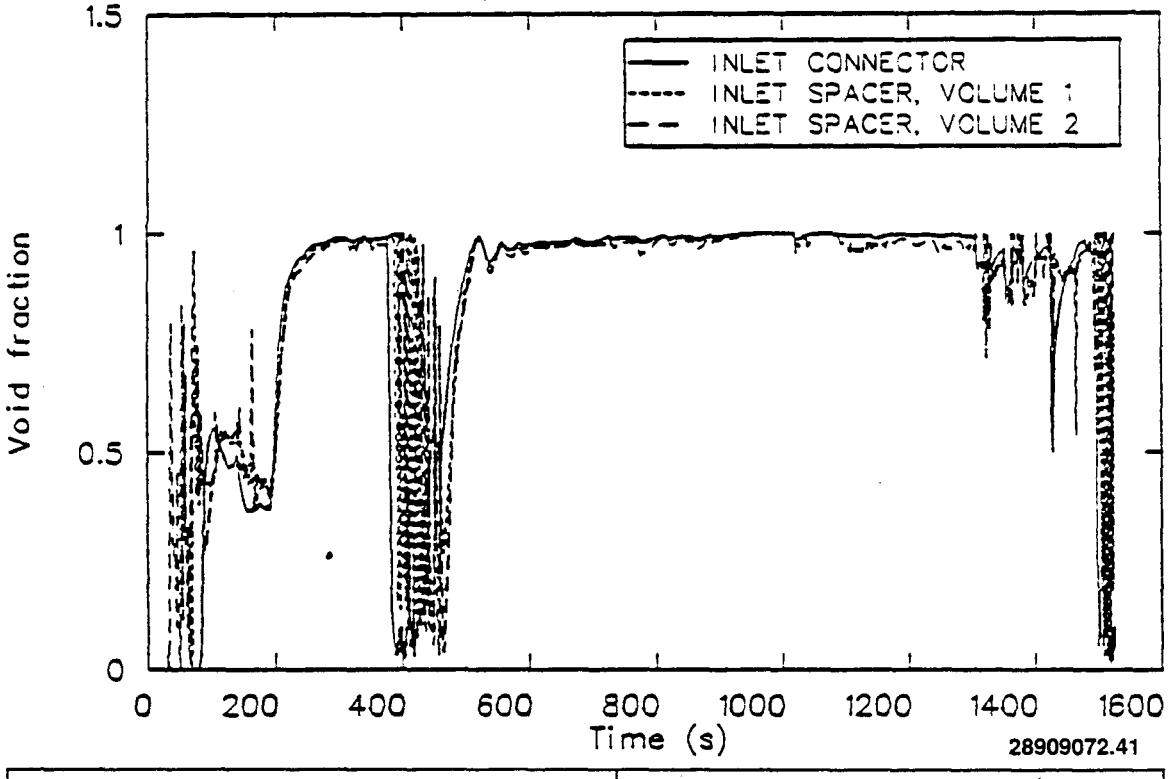
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Core Differential Pressure: Inlet Riser Break, Base Case Figure 15.6-218



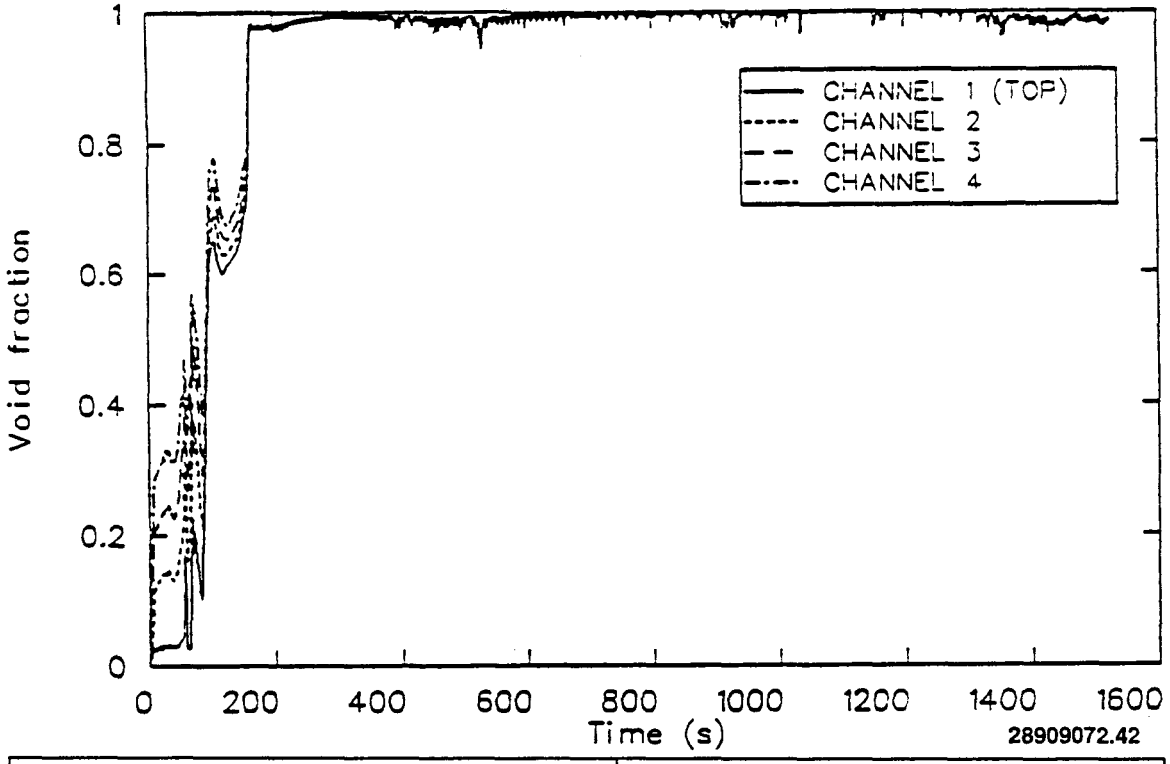
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Unaffected Inlet Riser Void Fraction: Inlet Riser Break, Base Case Figure 15.6-219



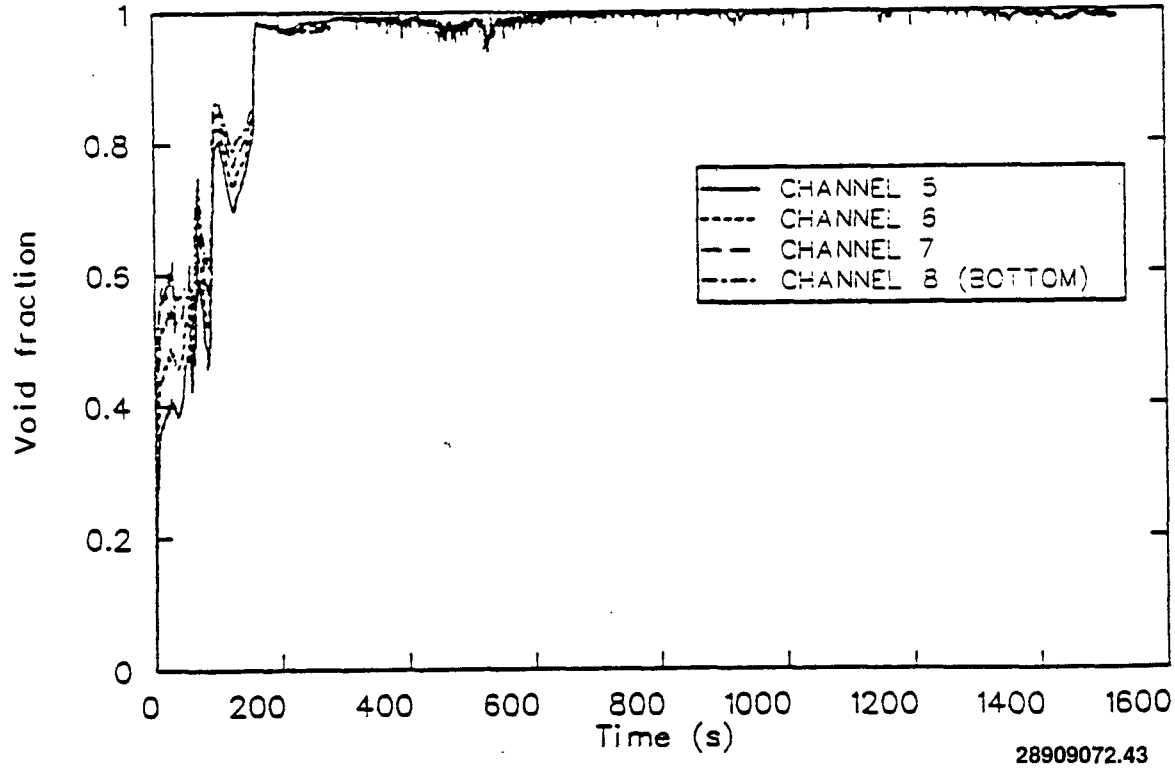
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Unaffected Inlet Riser Void Fraction: Inlet Riser Break, Base Case Figure 15.6-220



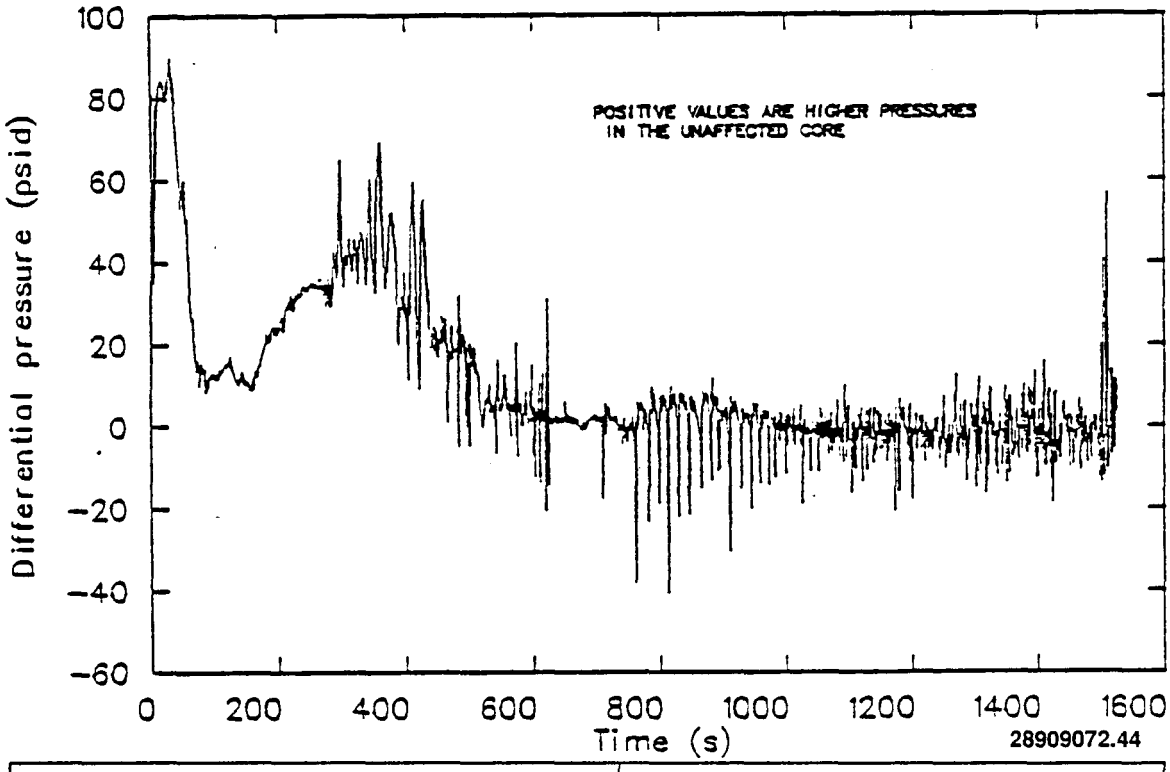
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Core Top Channel Inlet Connector and Spacer Void Fraction: Inlet Riser Break, Base Case Figure 15.6-221



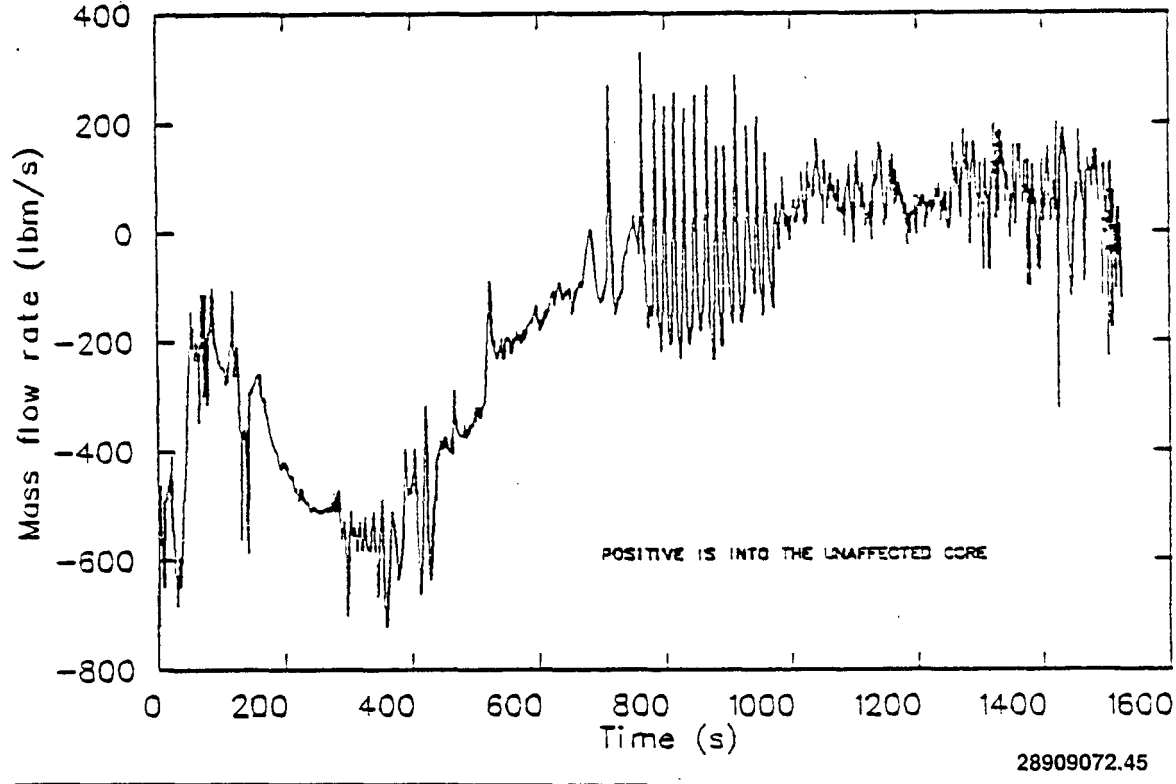
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Core Upper Inlet Riser Void Fraction: Inlet Riser Break, Base Case Figure 15.6-222



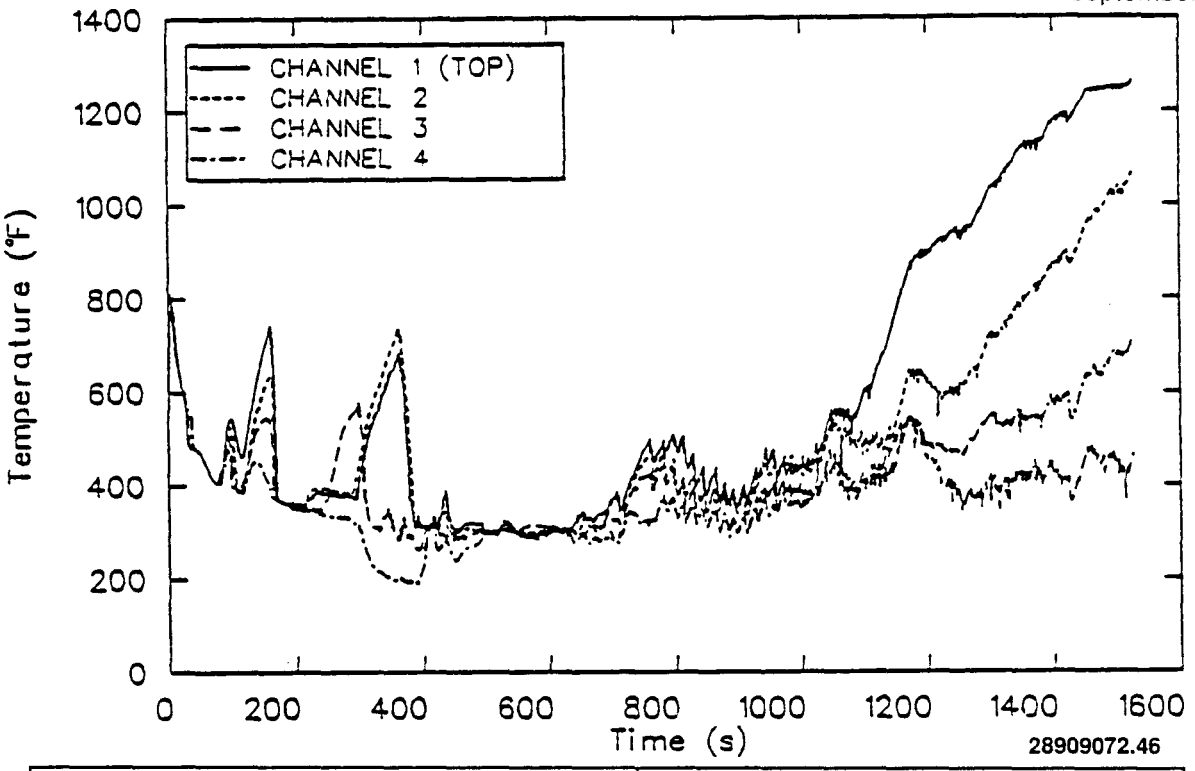
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Core Lower Inlet Riser Void Fraction: Inlet Riser Break, Base Case Figure 15.6-223



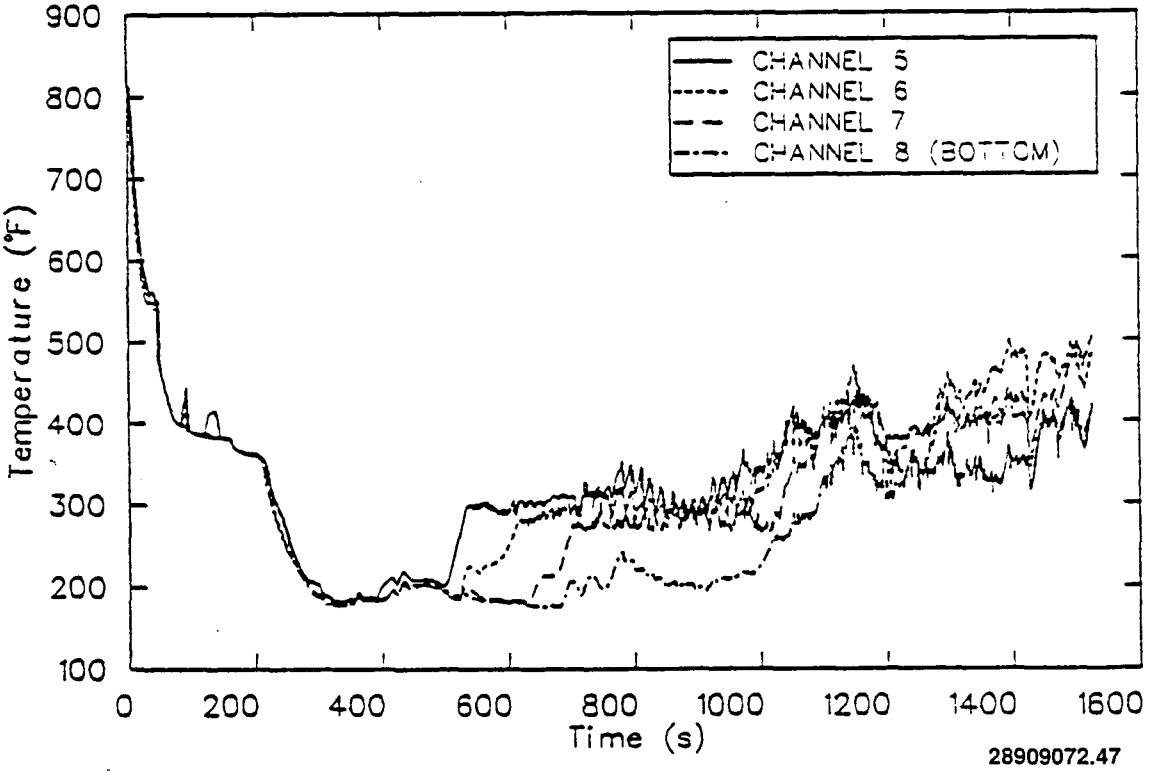
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Differential Pressure Between the Unaffected and Affected Core Outlet Riser: Inlet Riser Break, Base Case Figure 15.6-224



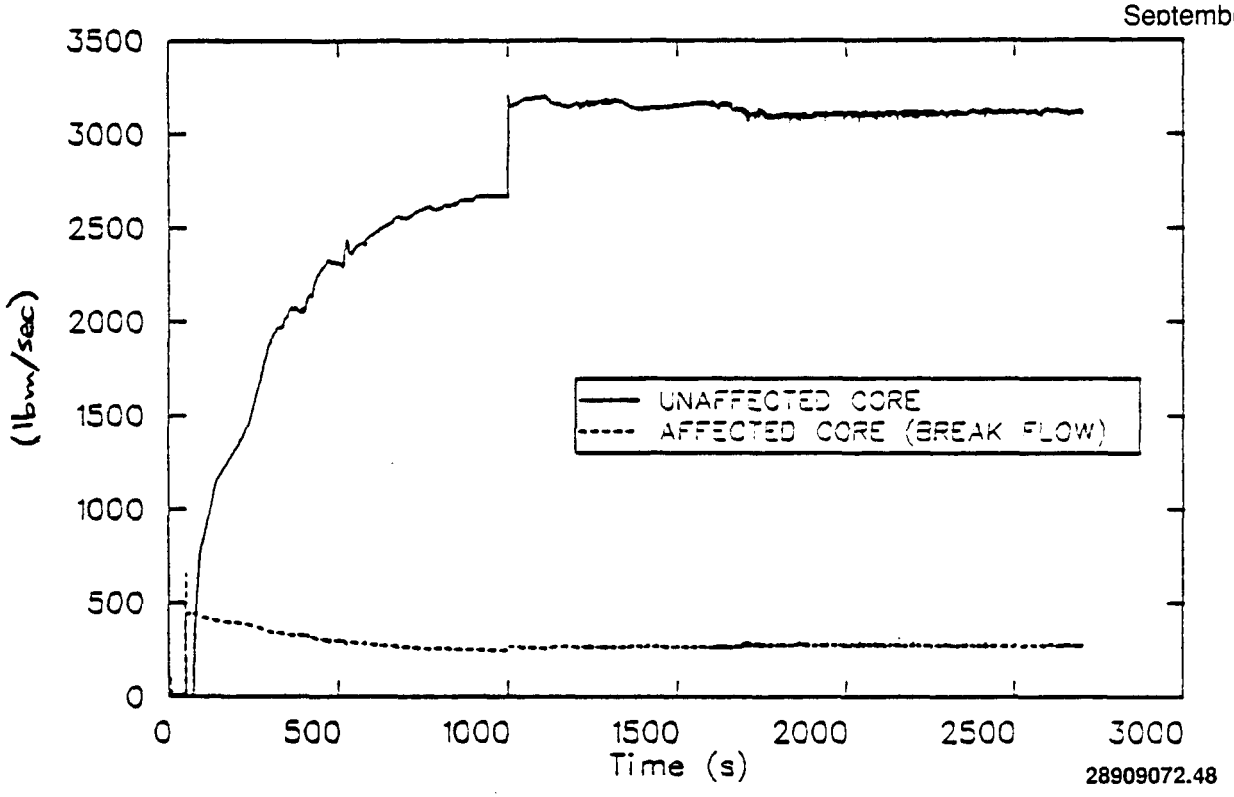
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Outlet Riser Flush Line Mass Flow Rate: Inlet Riser Break, Base Case Figure 15.6-225



Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Upper Affected Core: Inlet Riser Break, Base Case Figure 15.6-226

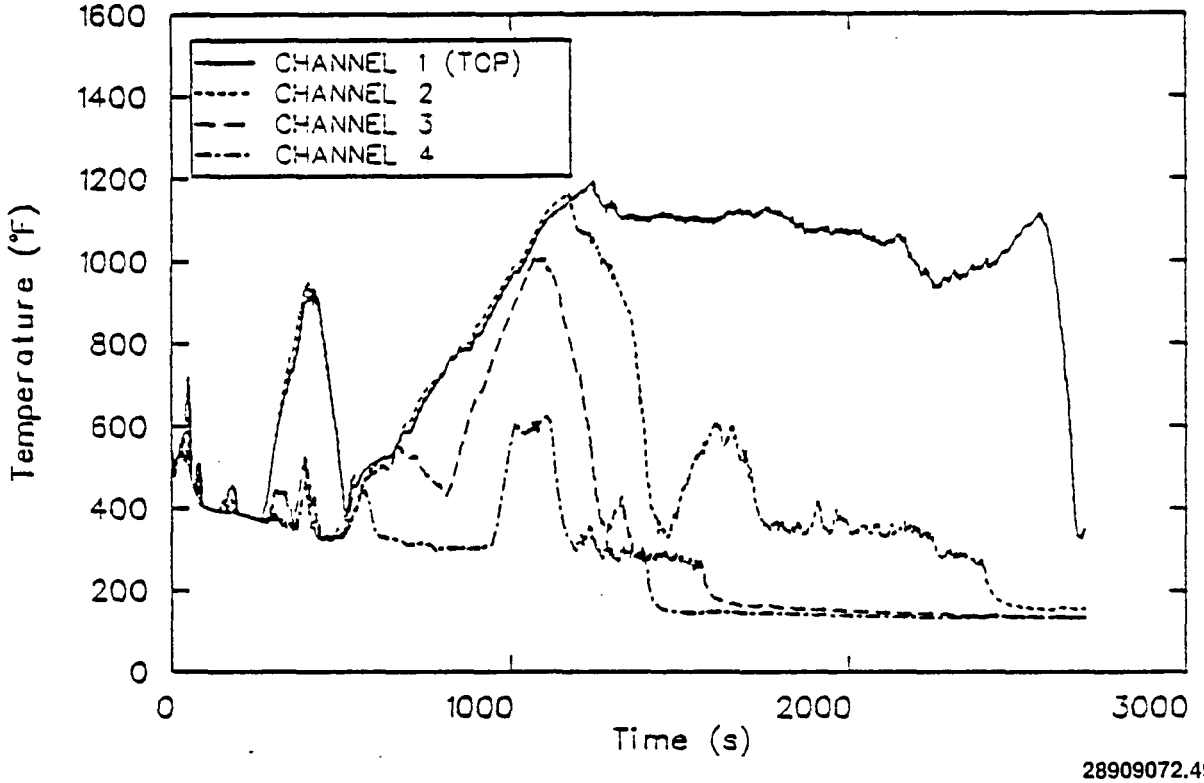


Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Lower Affected Core: Inlet Riser Break, Base Case Figure 15.6-227



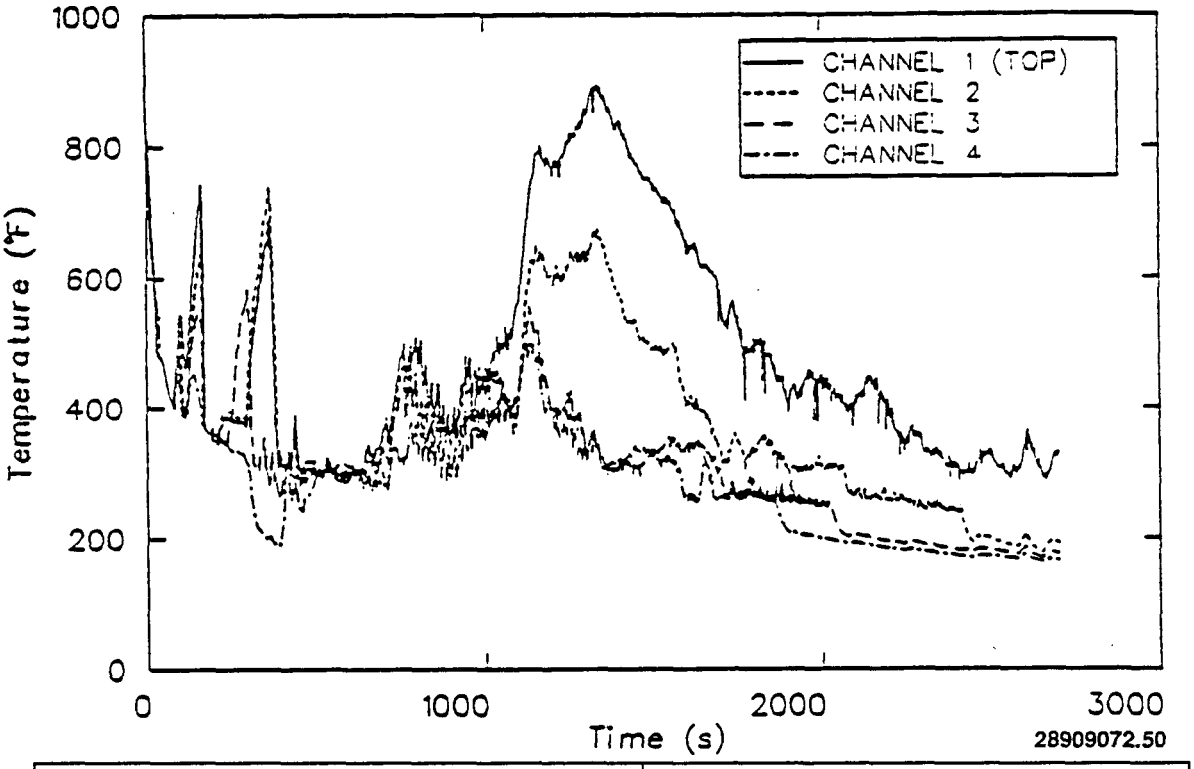
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	ECCS Mass Flow Rate: Inlet Riser Break, Sensitivity Figure 15.6-228

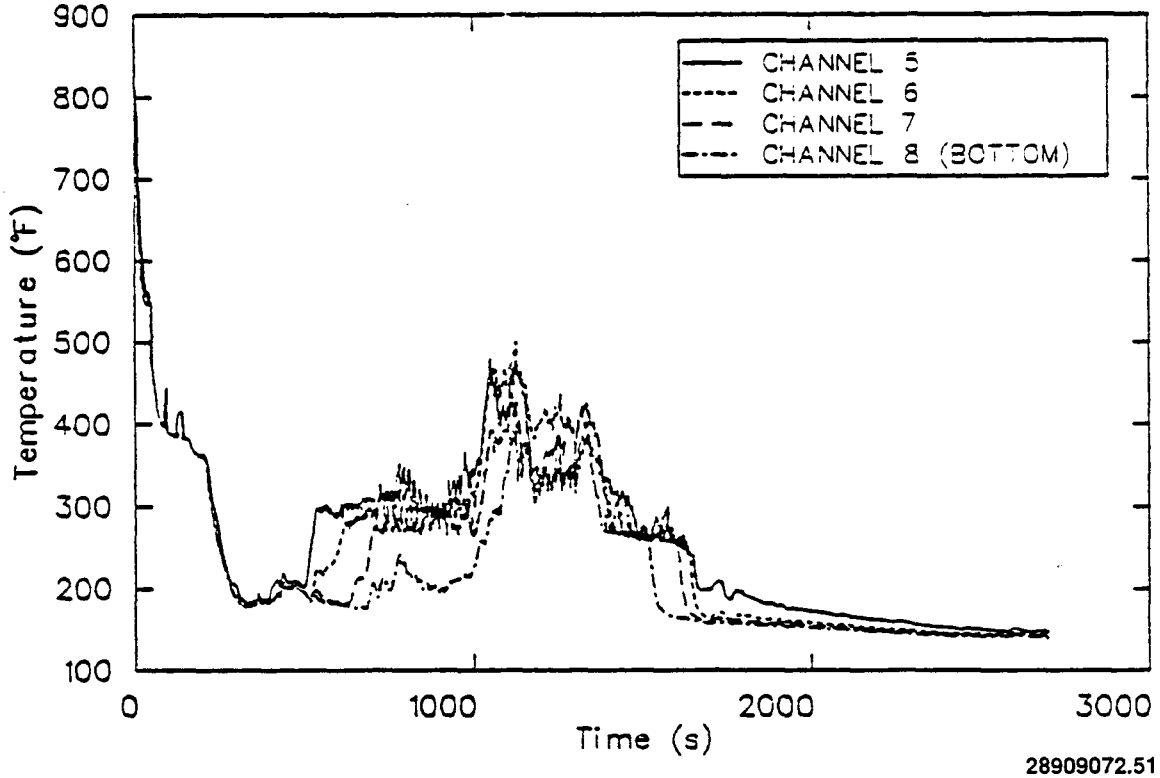


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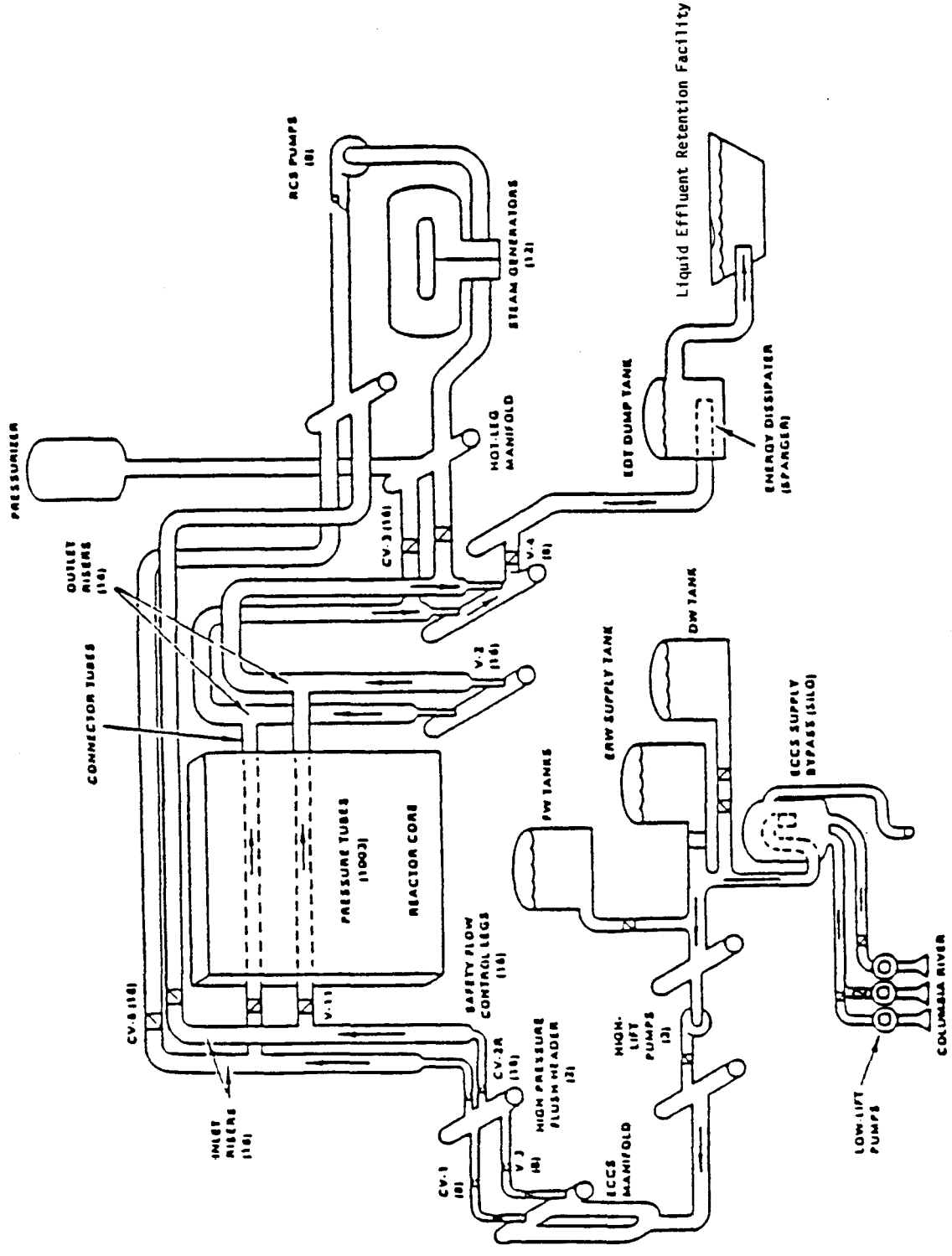
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Upper Unaffected Core: Inlet Riser Break, Sensitivity Figure 15.6-229



Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in Upper Affected Core: Inlet Riser Break, Sensitivity Figure 15.6-230

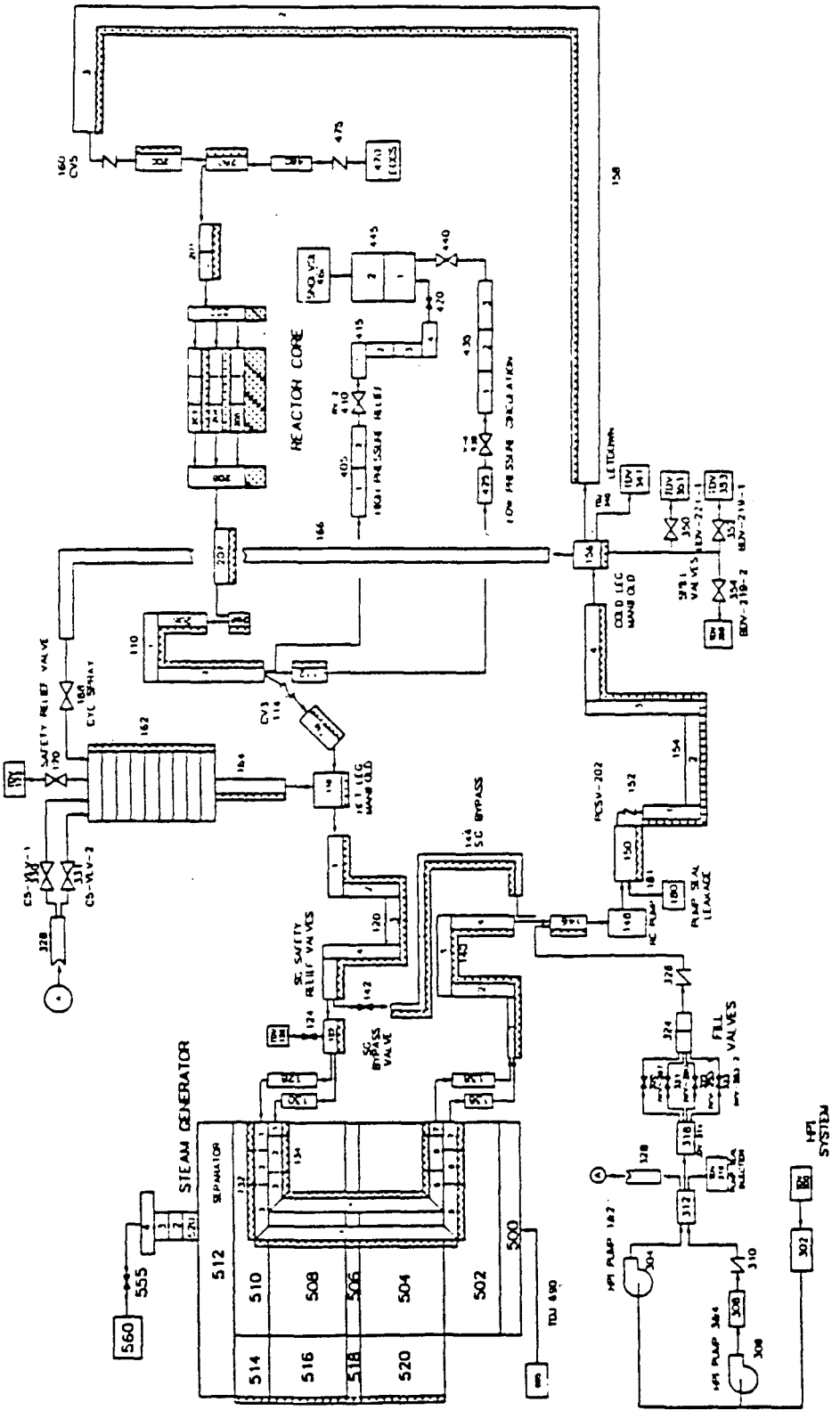


Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in the Lower Affected Core: Inlet Riser Break, Sensitivity Figure 15.6-231



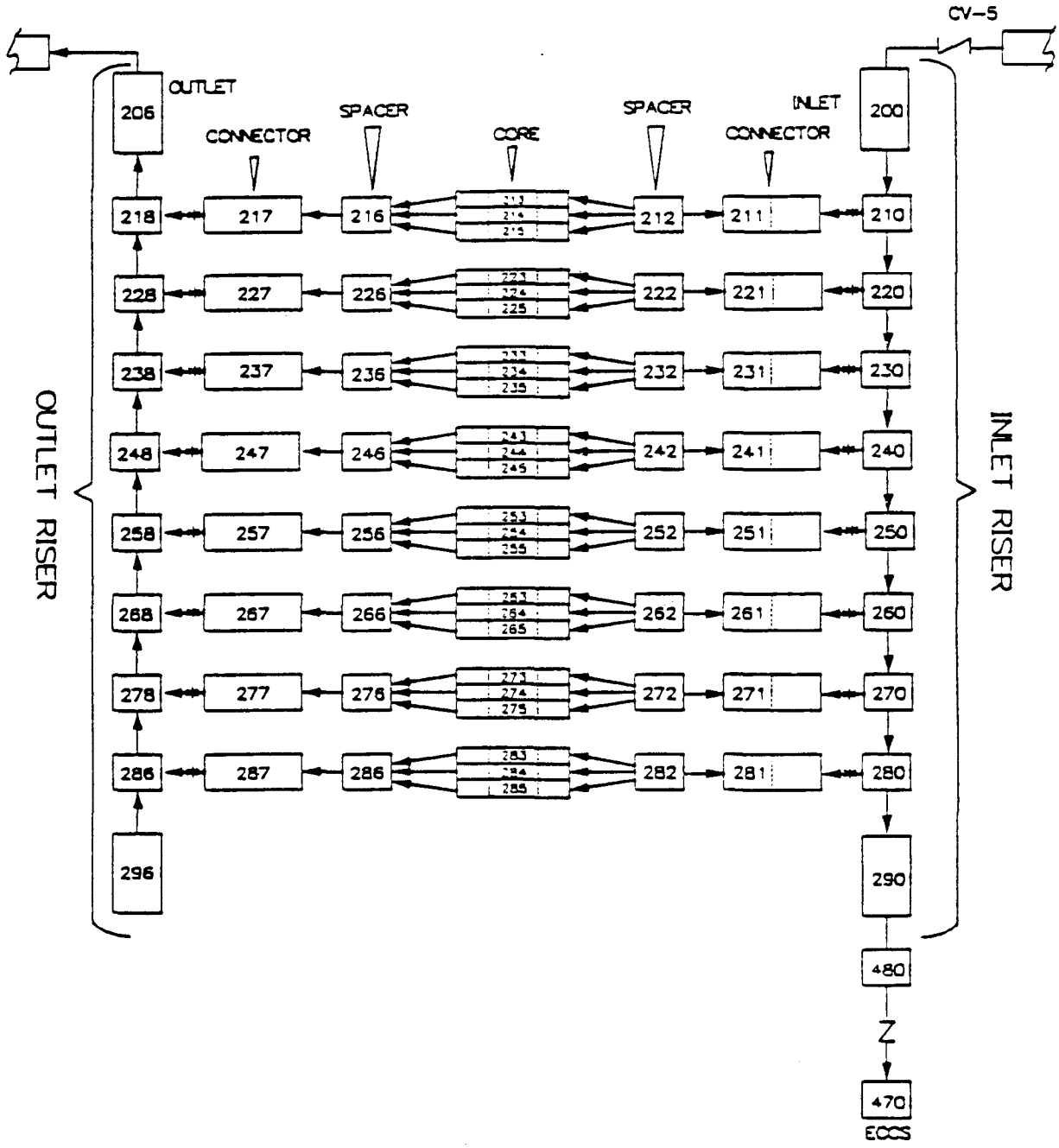
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Layout of the N Reactor
	Figure 15.6-232



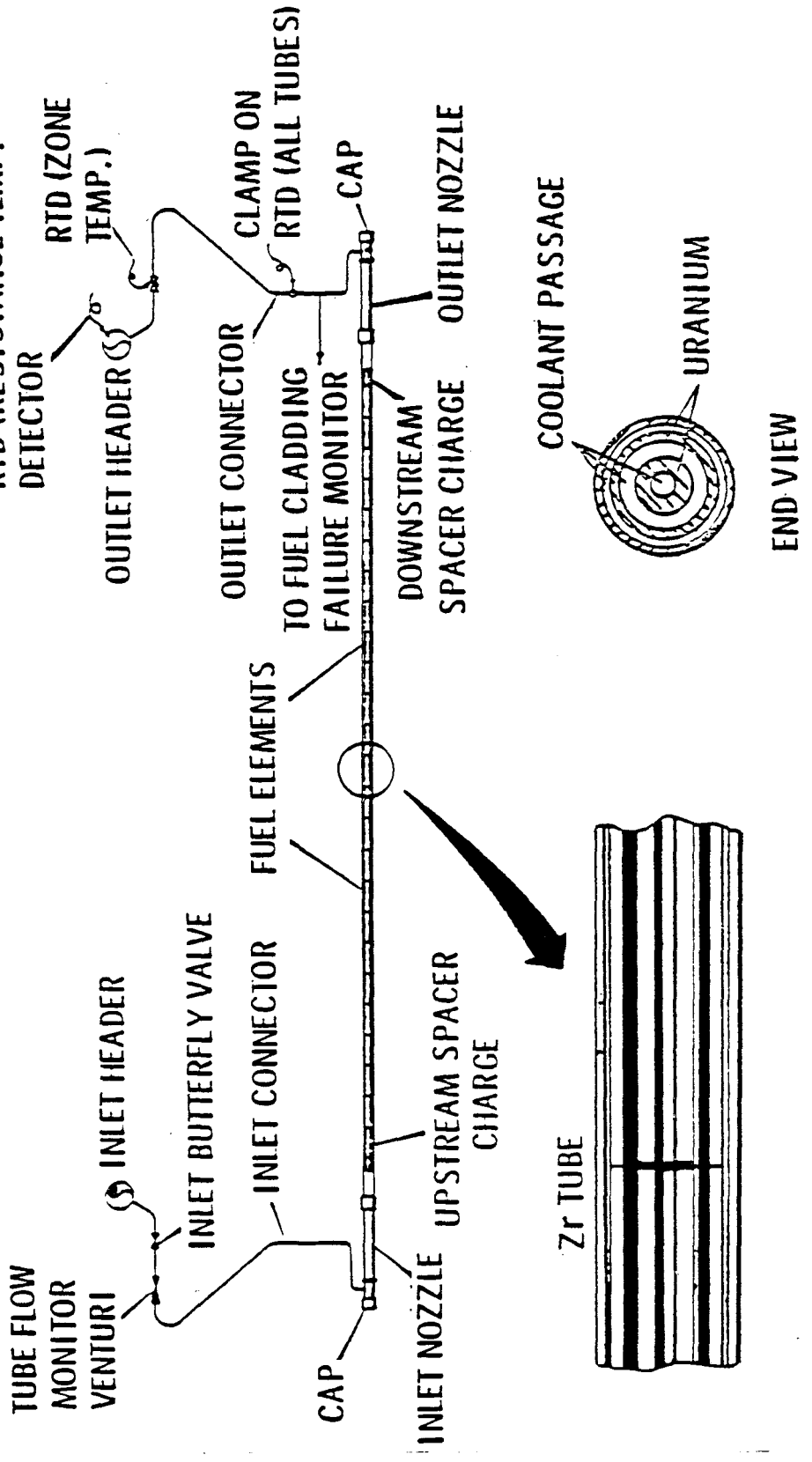
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	RELAP5 Nodalization of N Reactor Coolant Loop Piping for Cold Leg Manifold Break and Station Blackout Events Figure 15.6-233



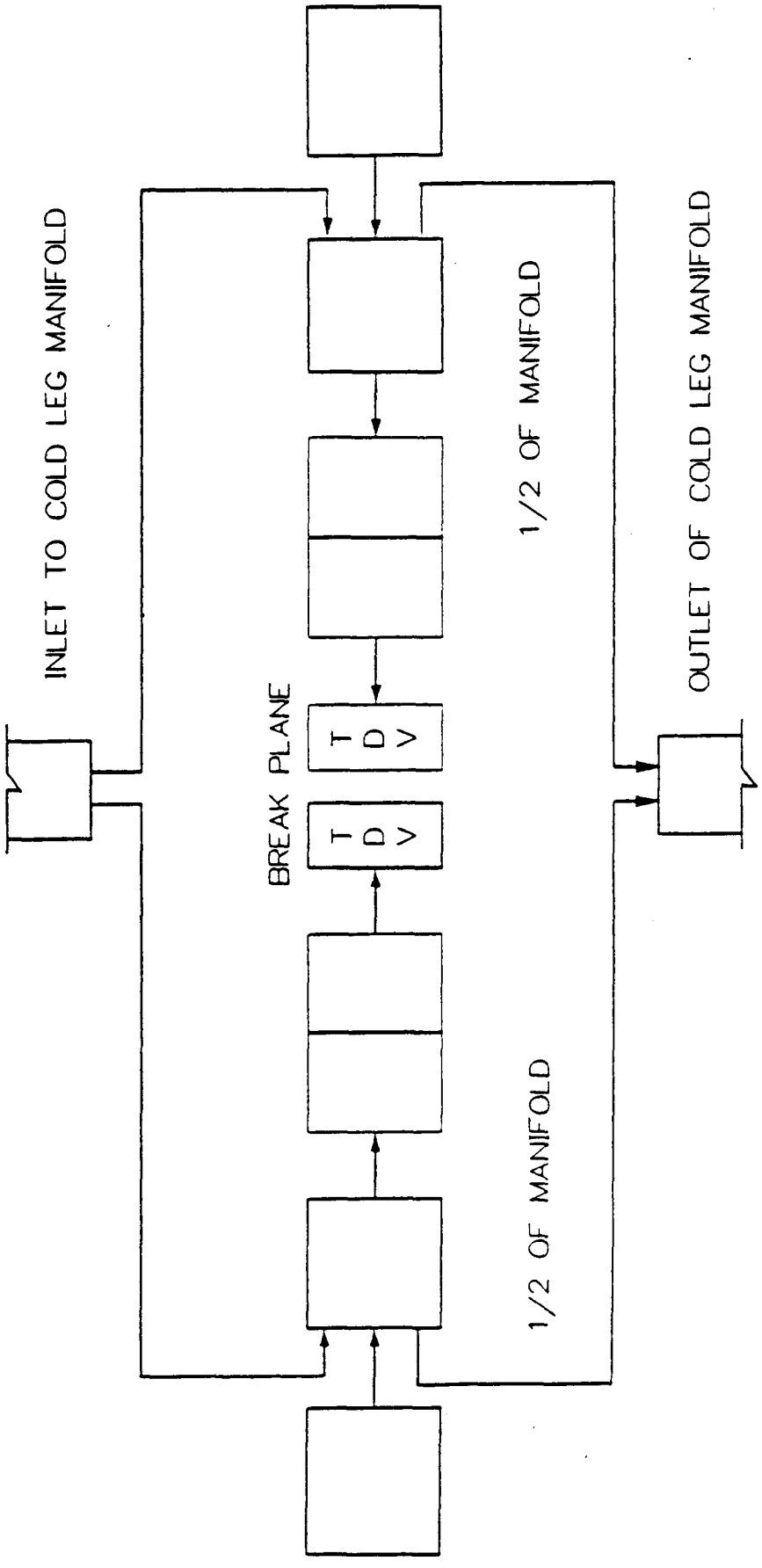
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Expanded RELAP5 Core Nodalization for Cold Leg Manifold Break and Station Blackout Events
	Figure 15.6-234



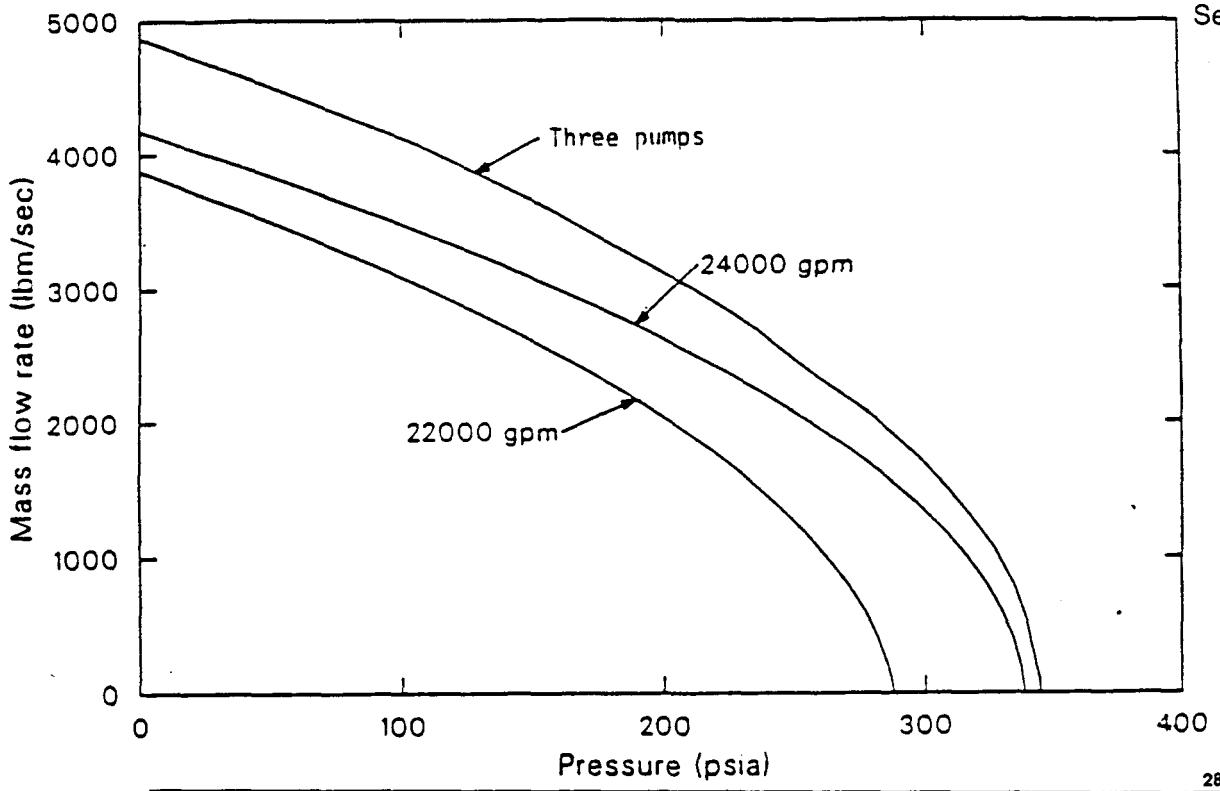
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Layout of Pressure Tube Figure 15.6-235



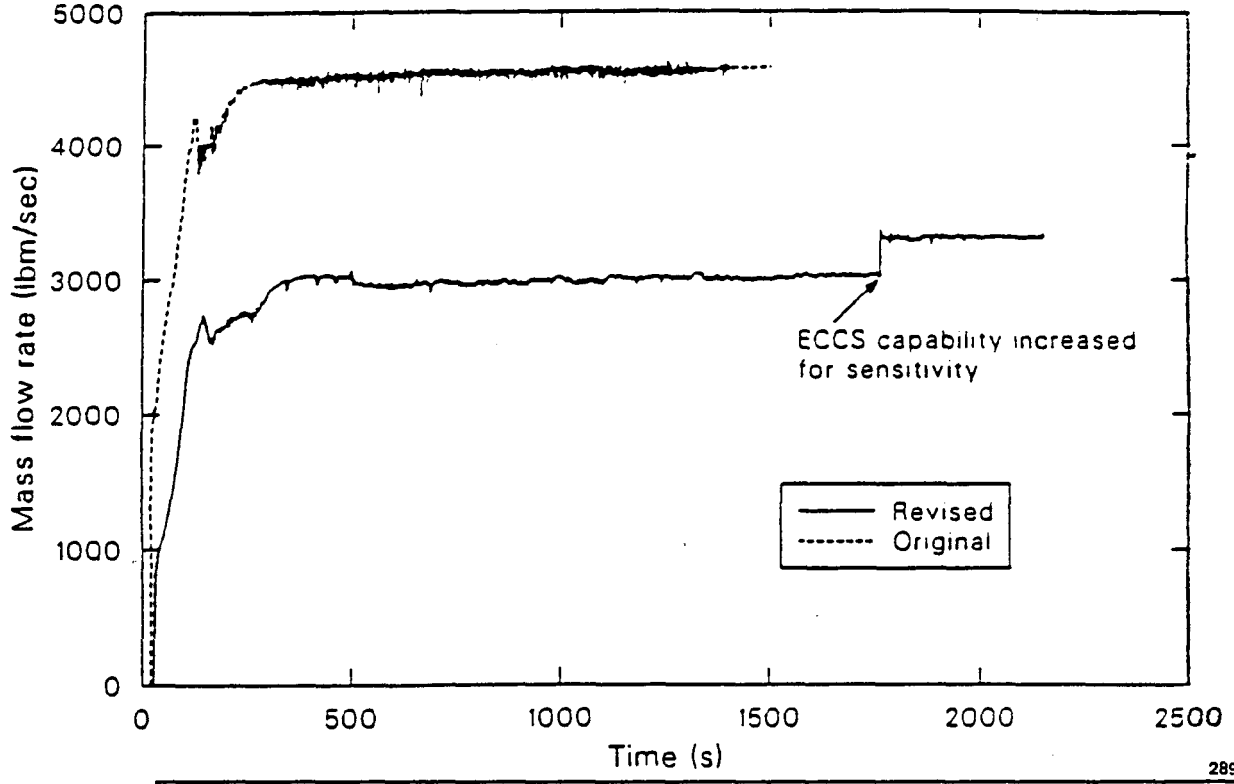
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Nodalization of Broken Cold Leg Manifold Figure 15.6-236



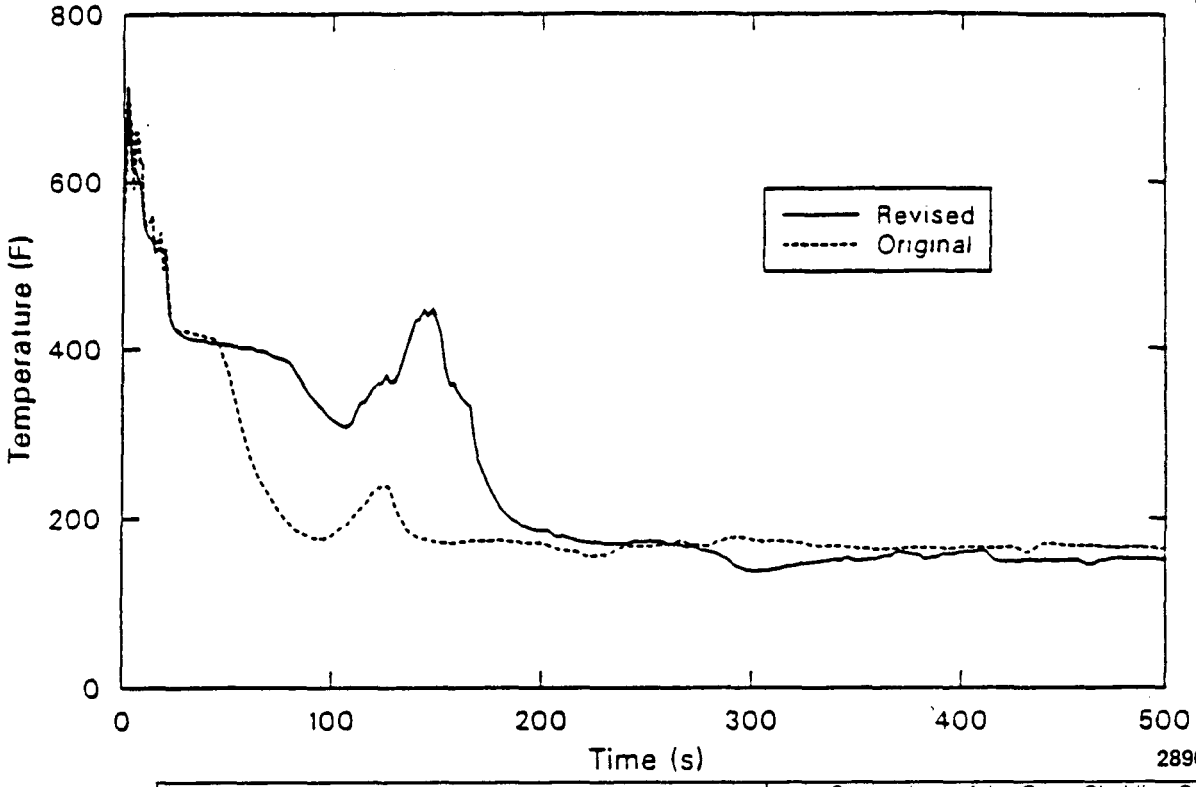
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the ECCS Mass Flow Rates as Functions of Pressure for the Original, Revised, and Sensitivity Calculations for Cold Leg Manifold and Station Blackout Events Figure 15.6-237

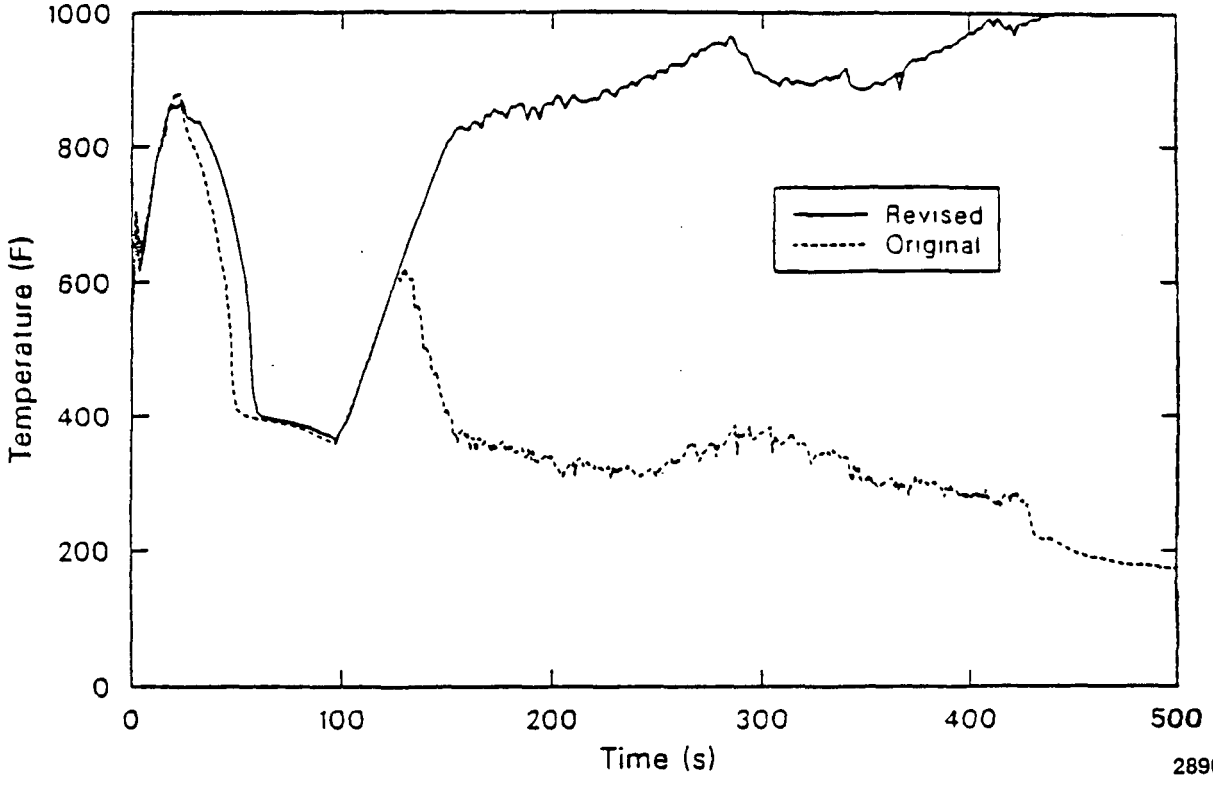


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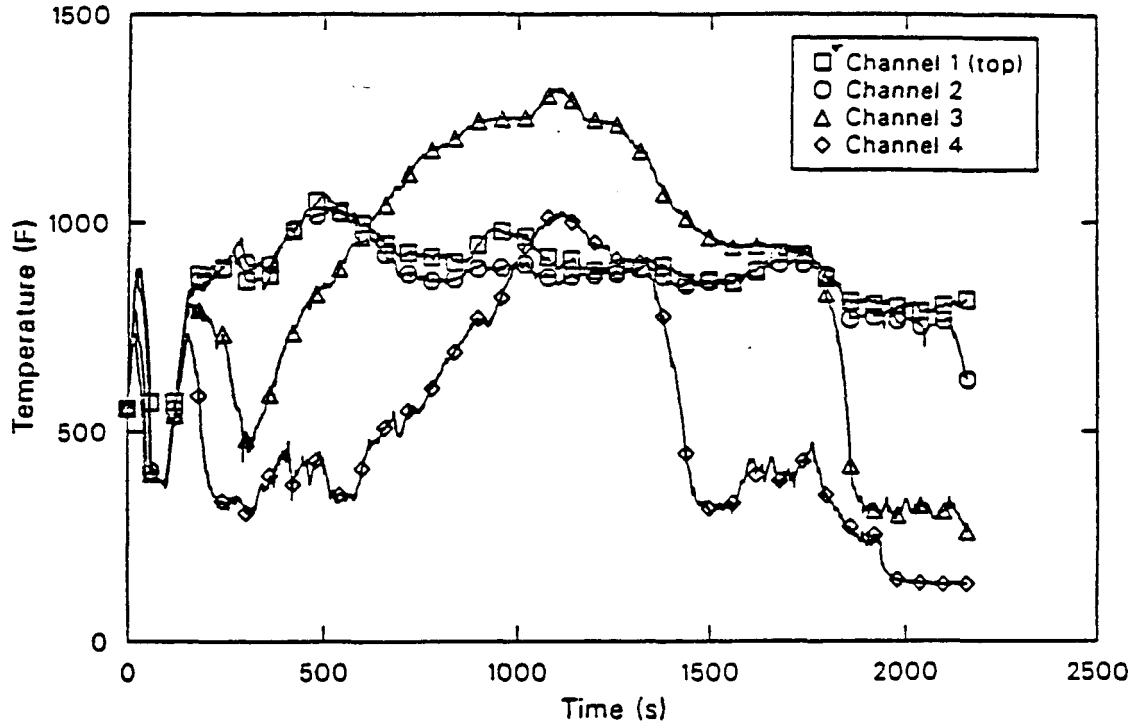
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the ECCS Mass Flow Rates of the Original and Revised Cold Leg Manifold Break Calculations Figure 15.6-238



Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the Outer Cladding Surface Temperatures in the Center Cells of the Seventh Core Channel from the Original and Revised Cold Leg Manifold Break Calculations Figure 15.6-239

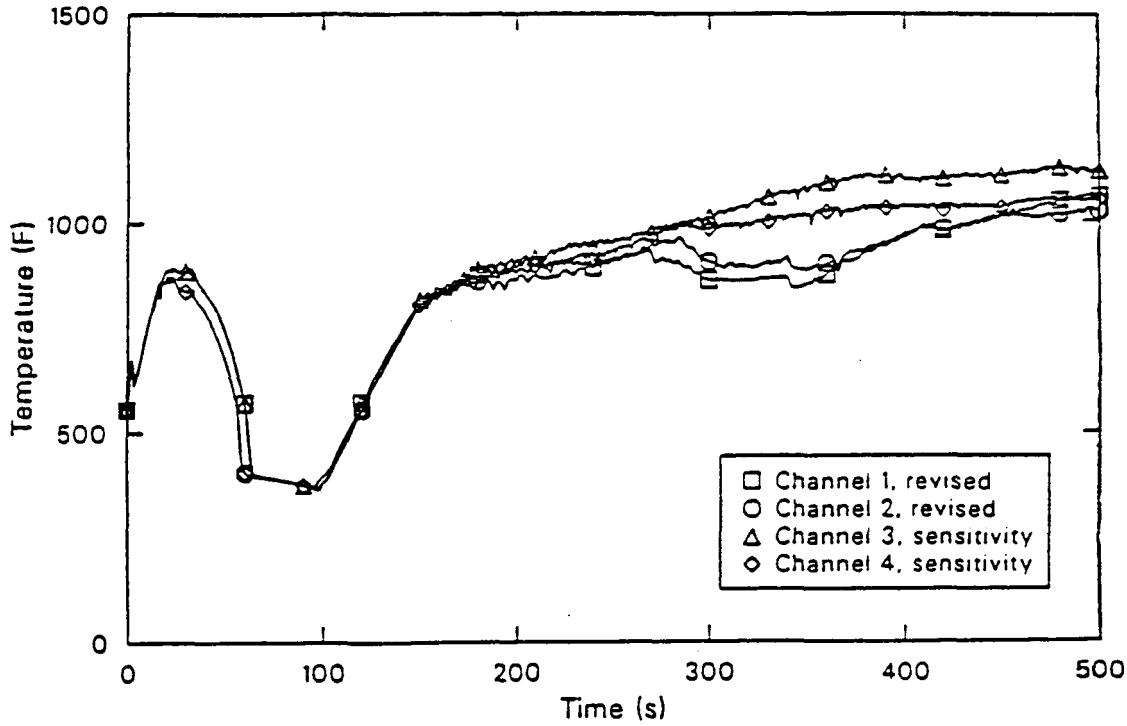


Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the Outer Cladding Surface Temperatures in the Center Cells of the Second Core Channel from the Original and Revised Cold Leg Manifold Break Calculations Figure 15.6-240



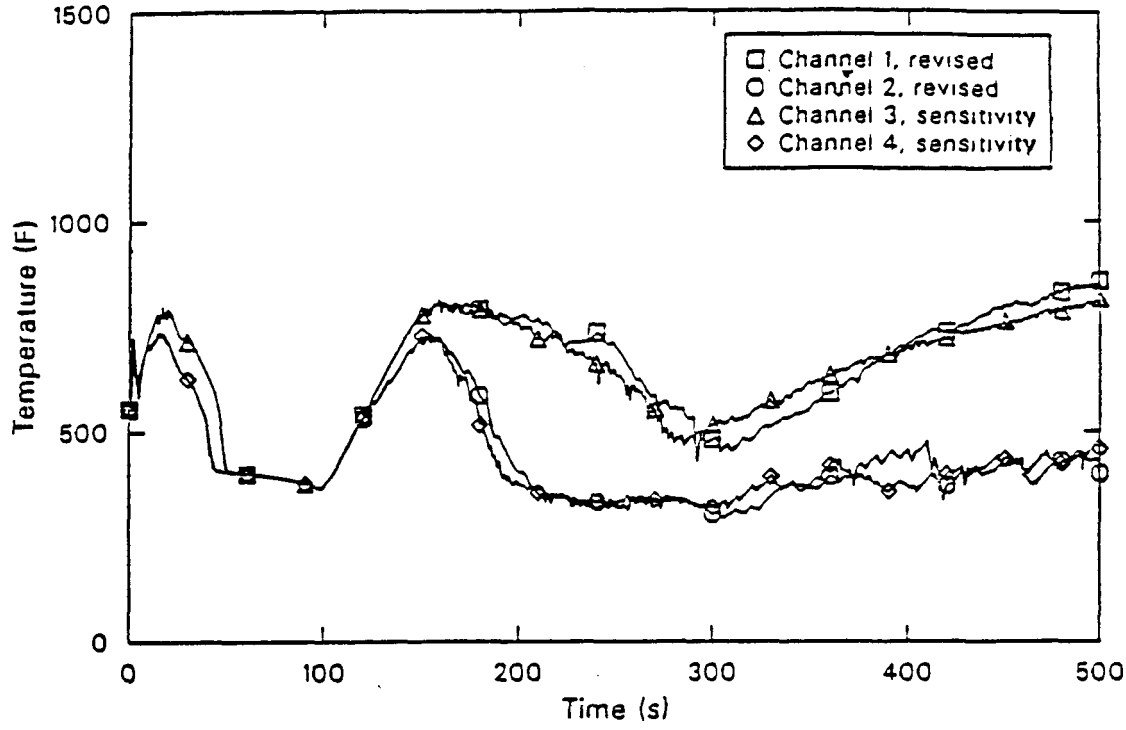
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Top Four Core Channel Fuel Cladding Surface Temperatures, Revised Cold Leg Manifold Break
	Figure 15.6-241



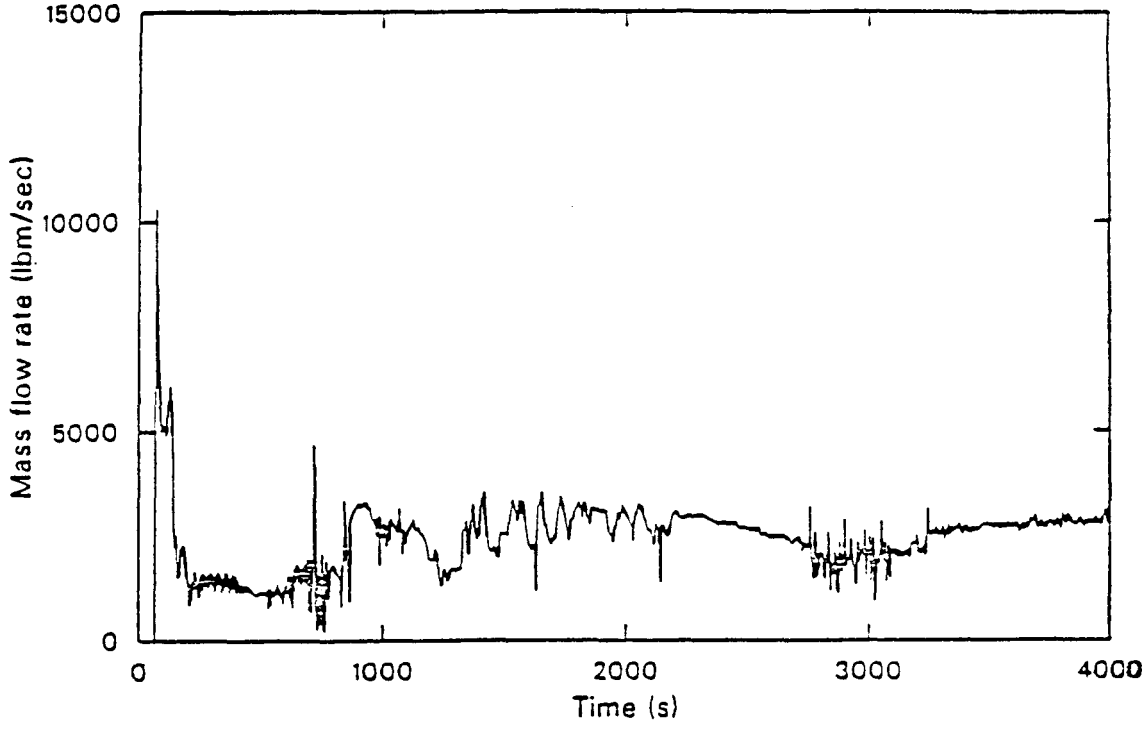
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the Outer Fuel Cladding Surface Temperatures of the Center Cells of the Top Two Core Channels for the Revised Cold Leg Manifold Break and the Sensitivity Calculation
	Figure 15.6-242



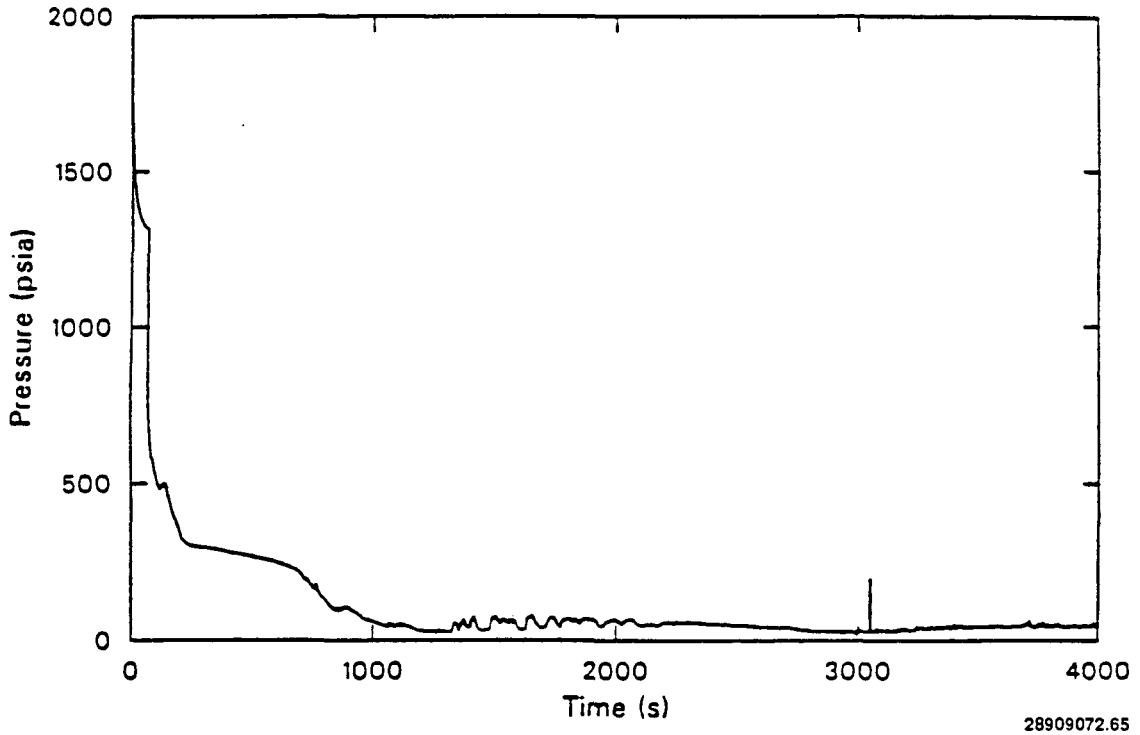
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of the Outer Fuel Cladding Surface Temperatures of the Center Cells of Core Channels 3 and 4 for the Revised Cold Leg Manifold Break and the Sensitivity Calculation Figure 15.6-243



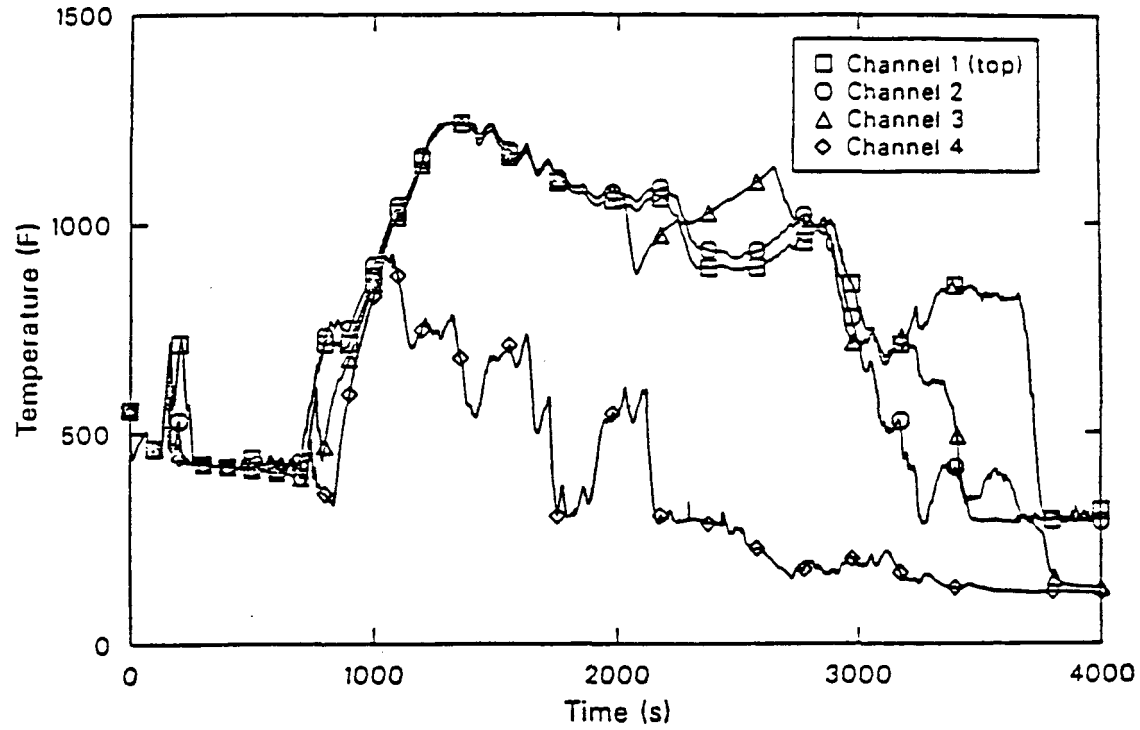
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	V-4 Valve Mass Flow Rate, Station Blackout Figure 15.6-244



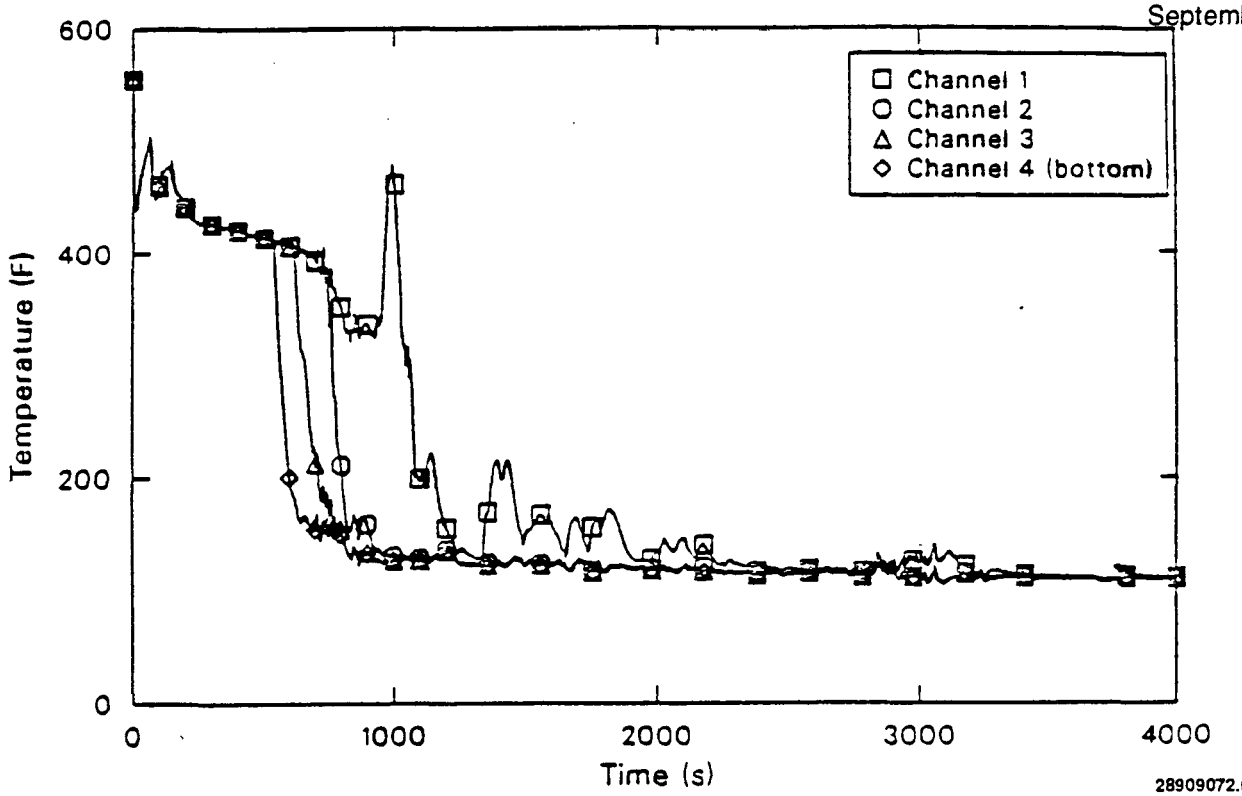
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Core Inlet Riser Pressure, Station Blackout
	Figure 15.6-245

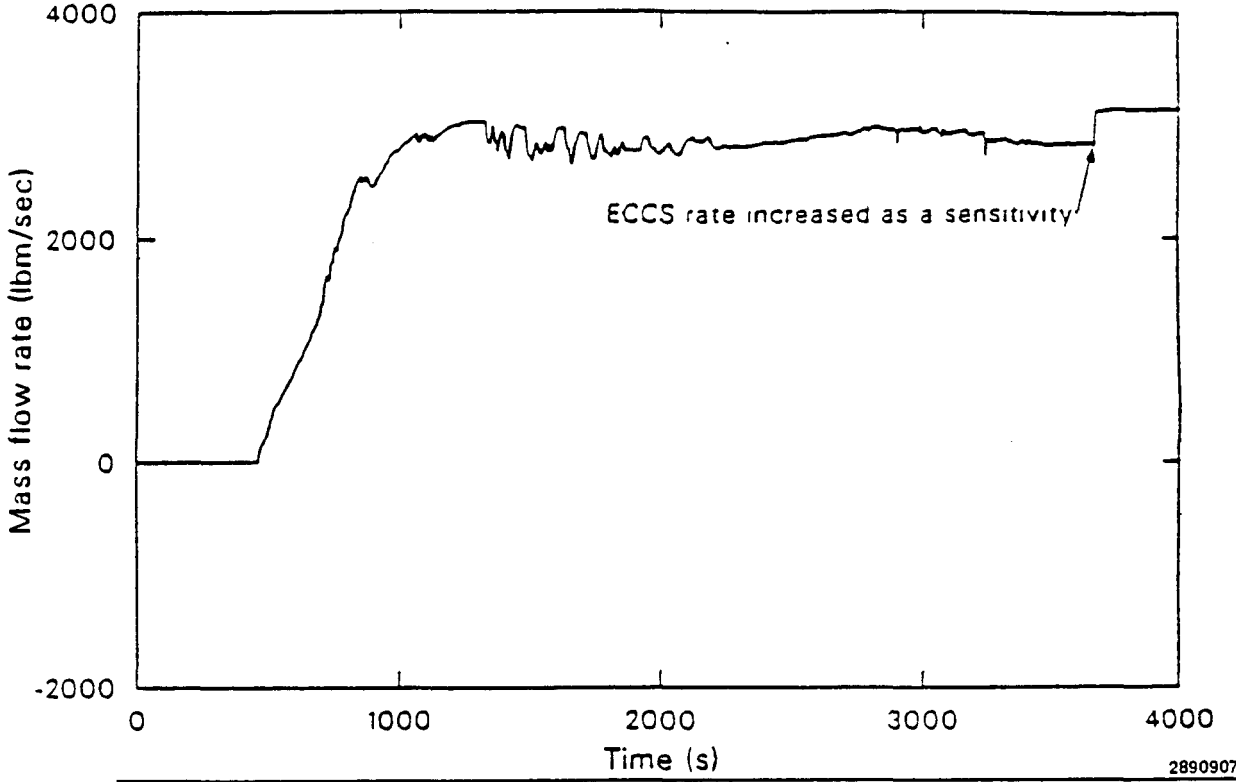


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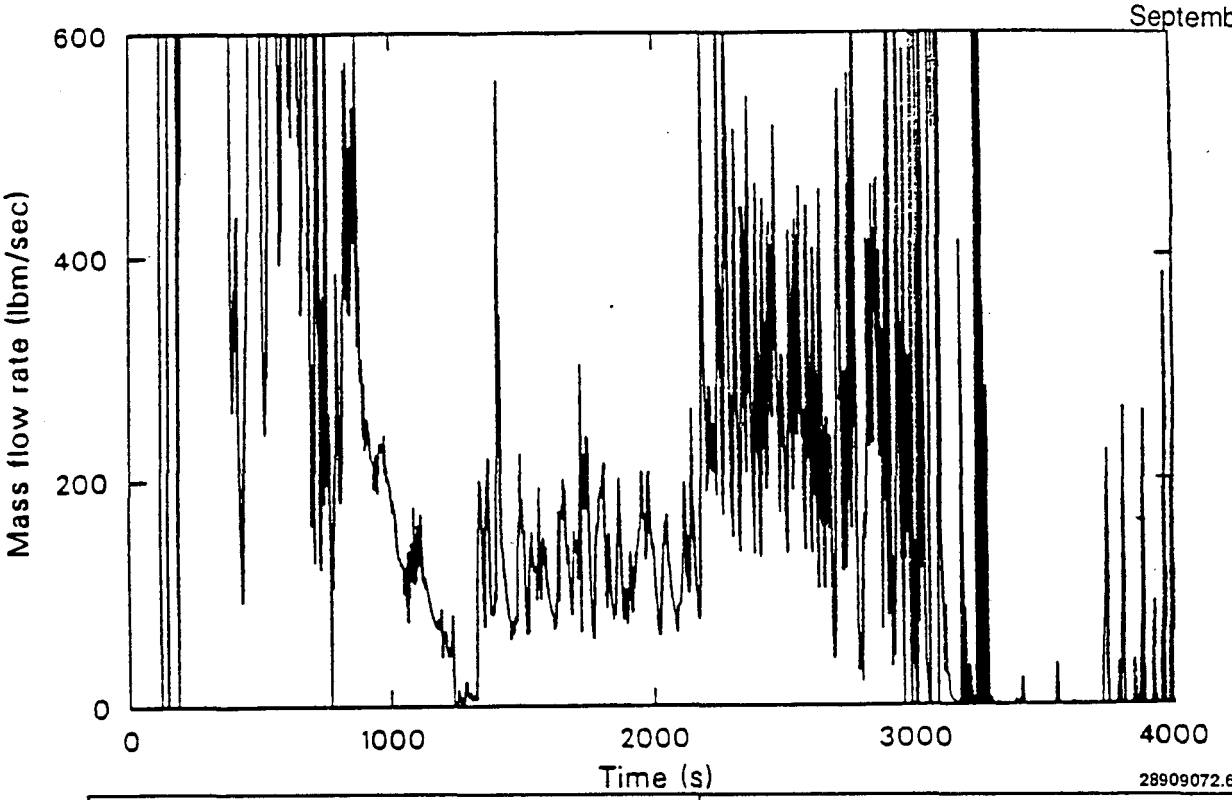
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Core Fuel Element Temperatures, Station Blackout
	Figure 15.6-246



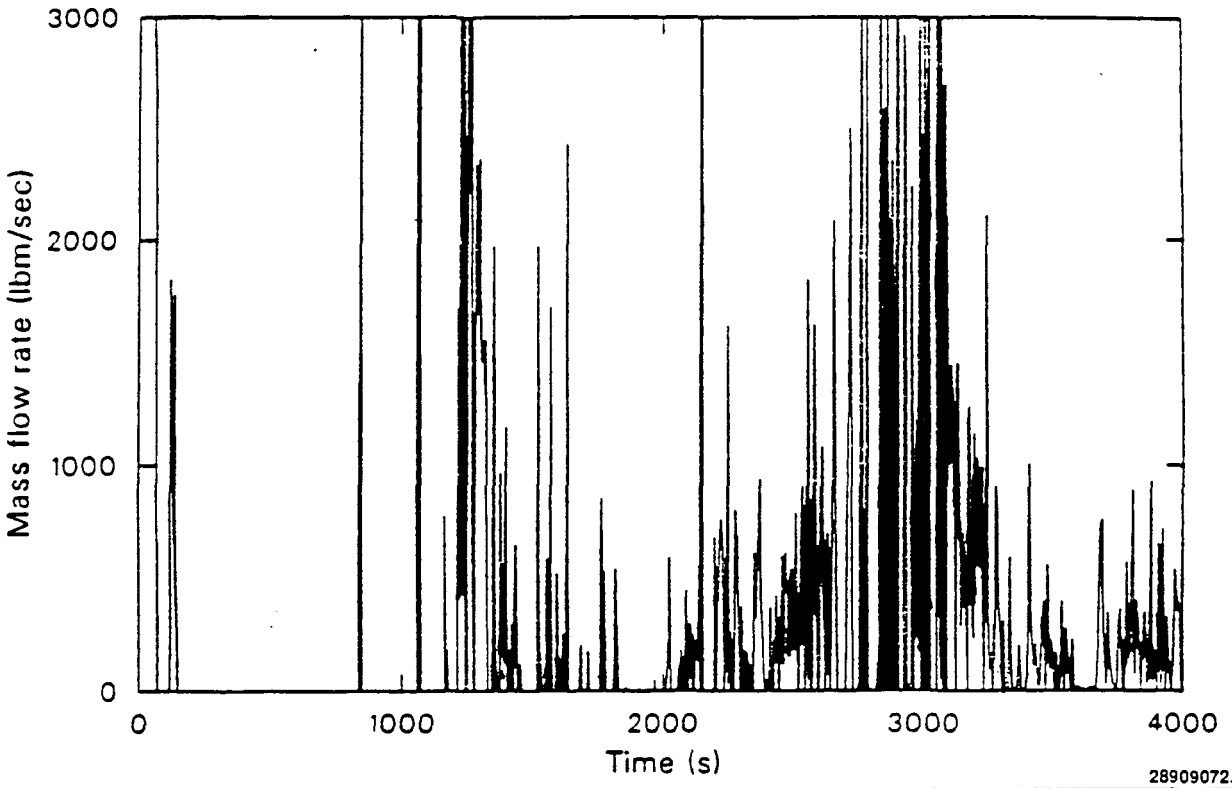
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Core Fuel Element Temperatures, Station Blackout Figure 15.6-247



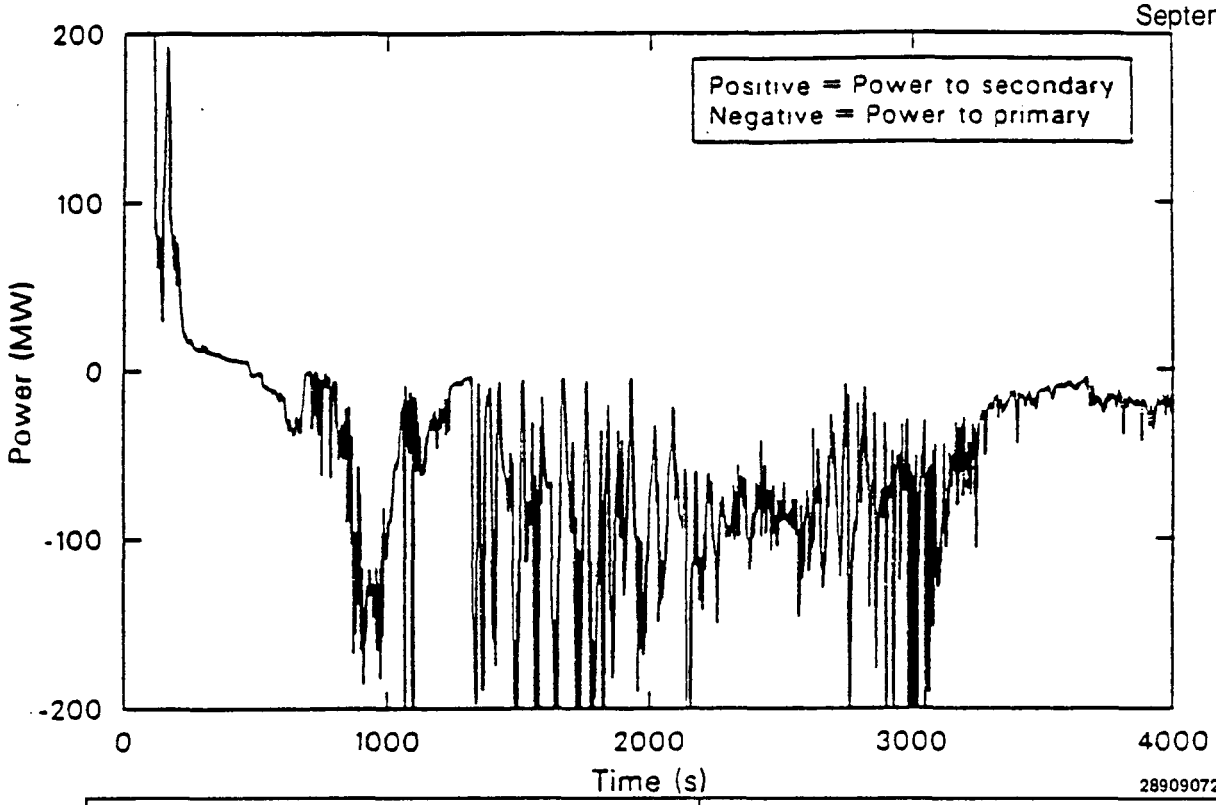
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	ECCS Mass Flow Rate, Station Blackout Figure 15.6-248



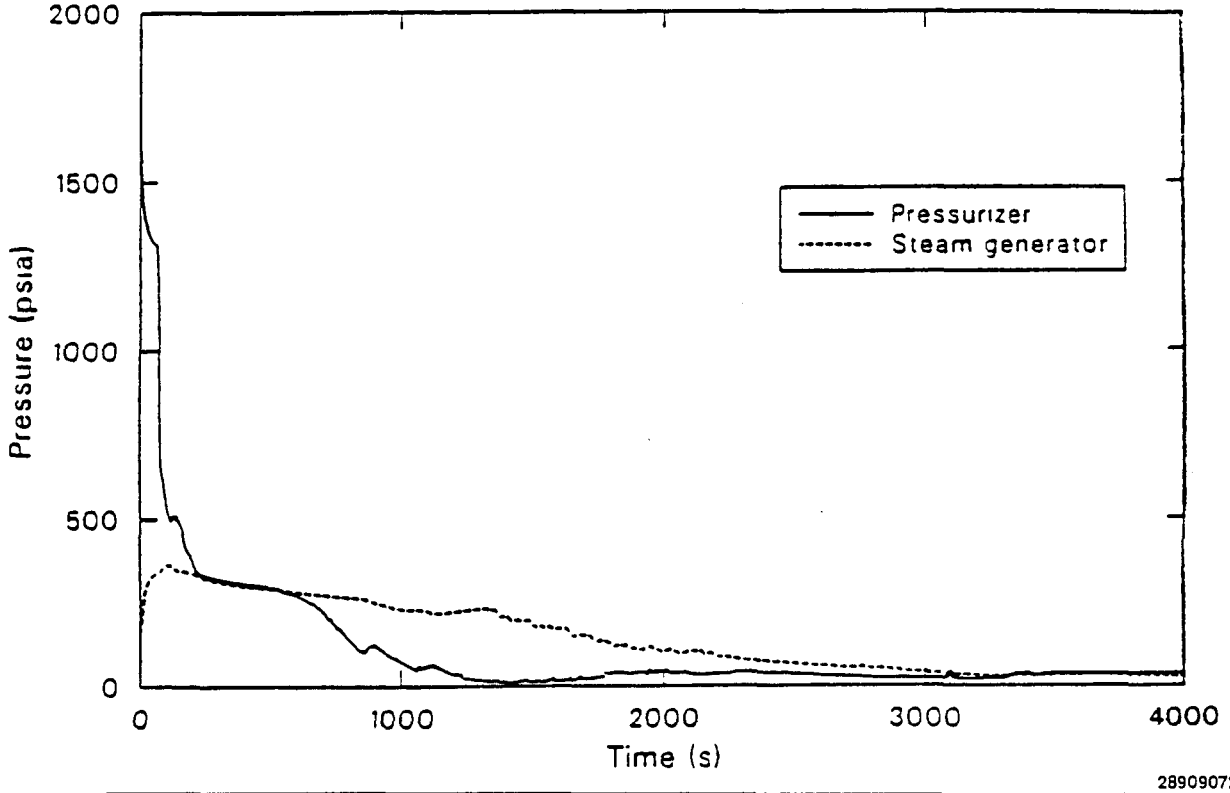
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	CV-5 Valve Mass Flow Rate, Station Blackout Figure 15.6-249



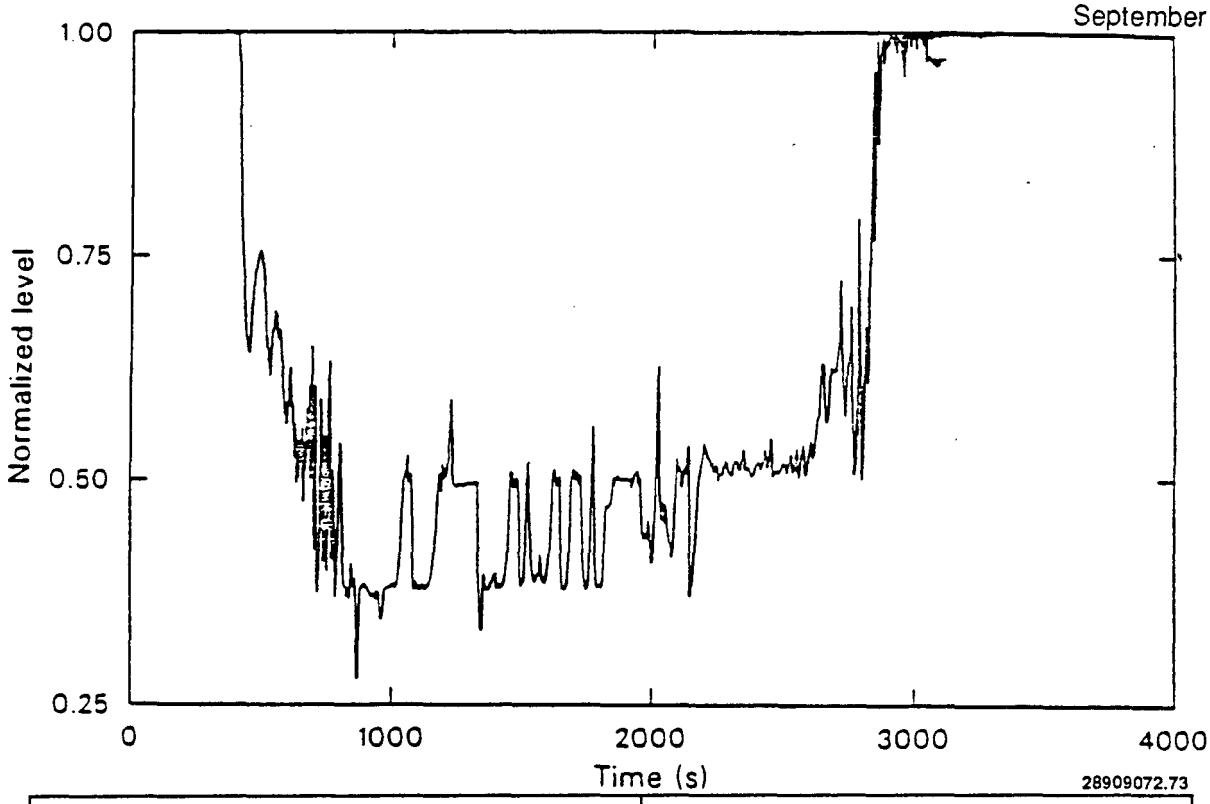
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	CV-3 Valve Mass Flow Rate, Station Blackout Figure 15.6-250



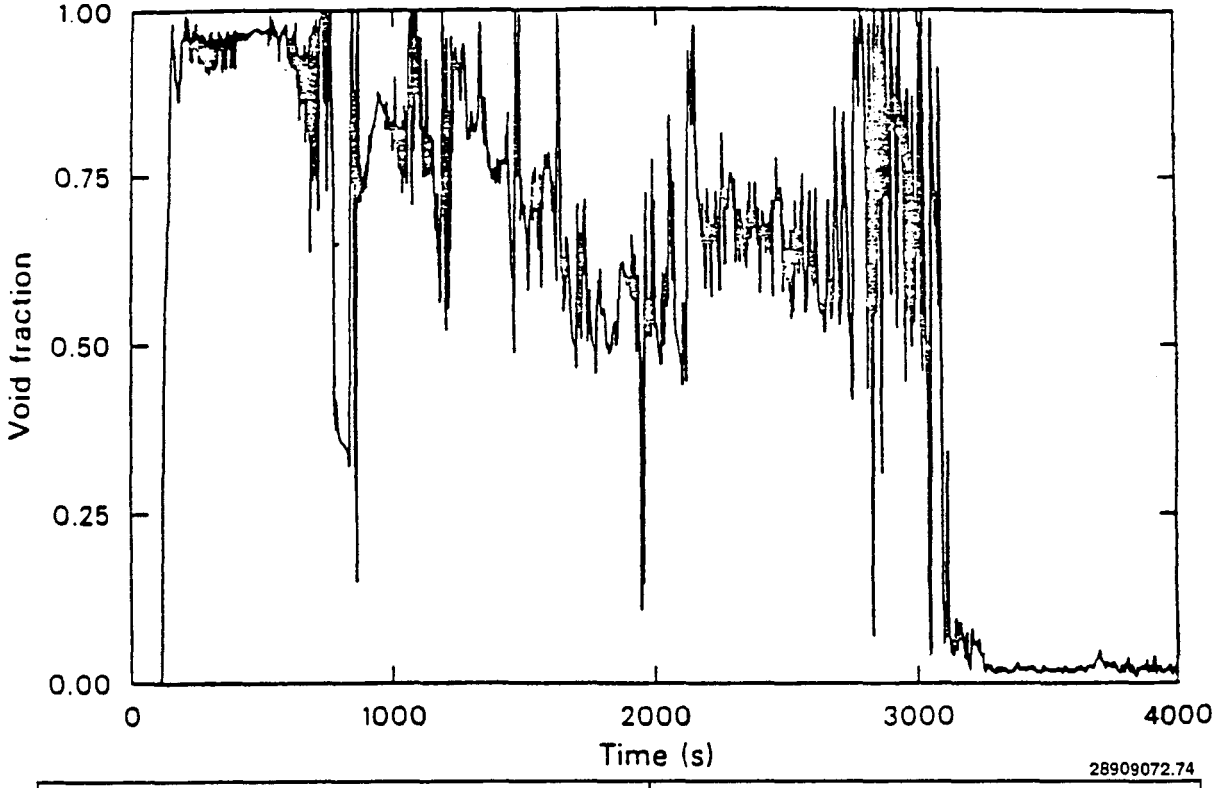
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Steam Generator Power, Station Blackout Figure 15.6-251



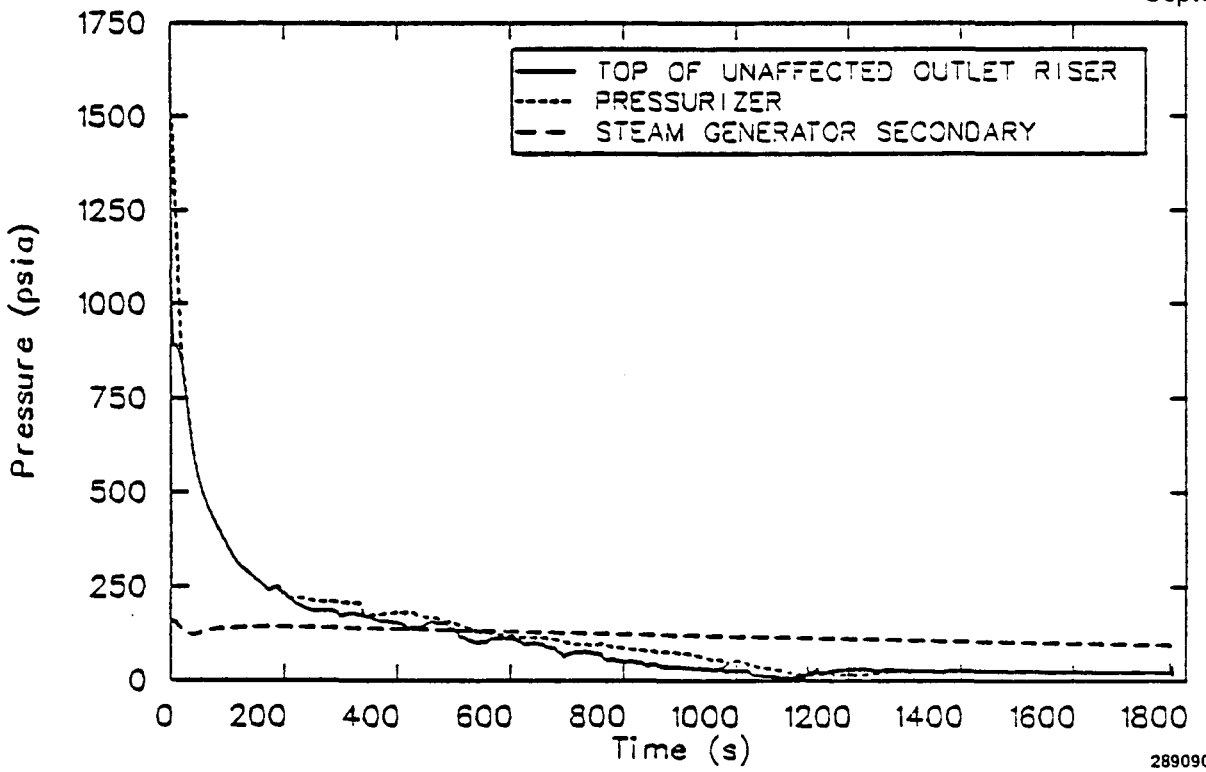
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Pressurizer and Steam Generator Secondary Pressures, Station Blackout Figure 15.6-252



Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Normalized Liquid Level in Core Inlet Riser, Station Blackout Figure 15.6-253

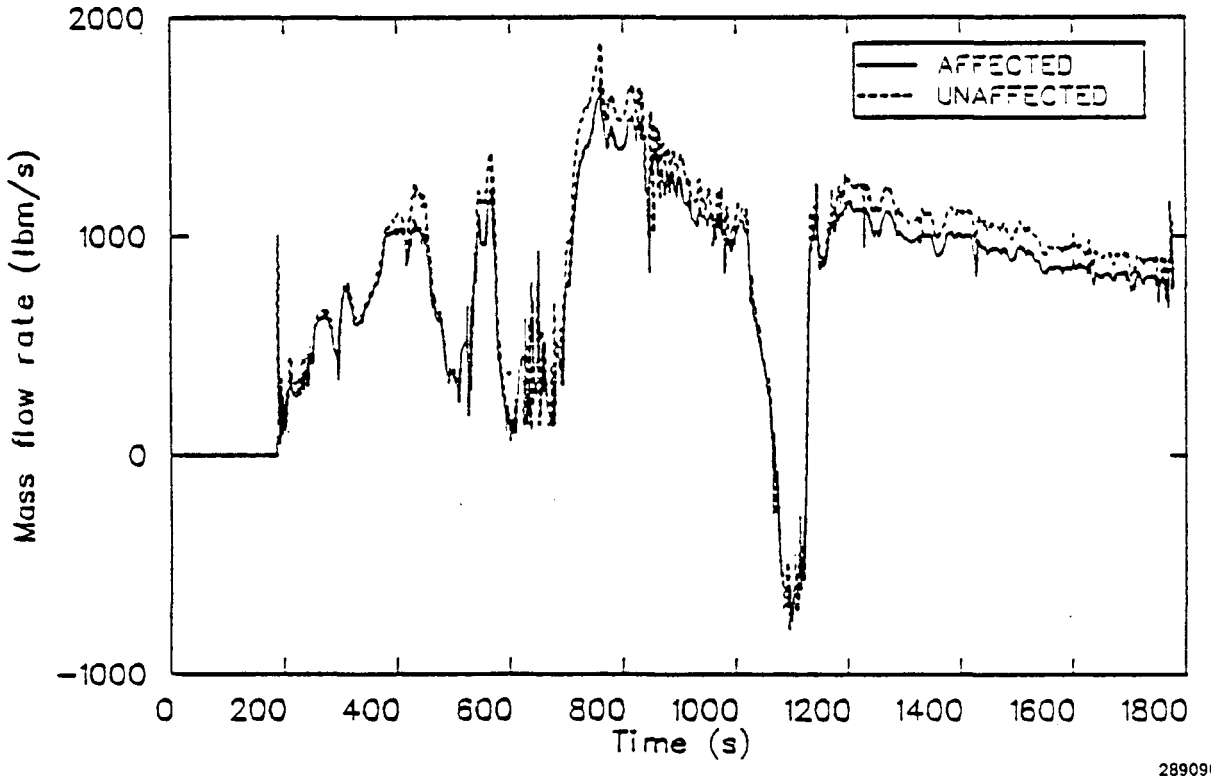


Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Void Fraction in Core Outlet Riser at Top Core Channel Elevation, Station Blackout Figure 15.6-254



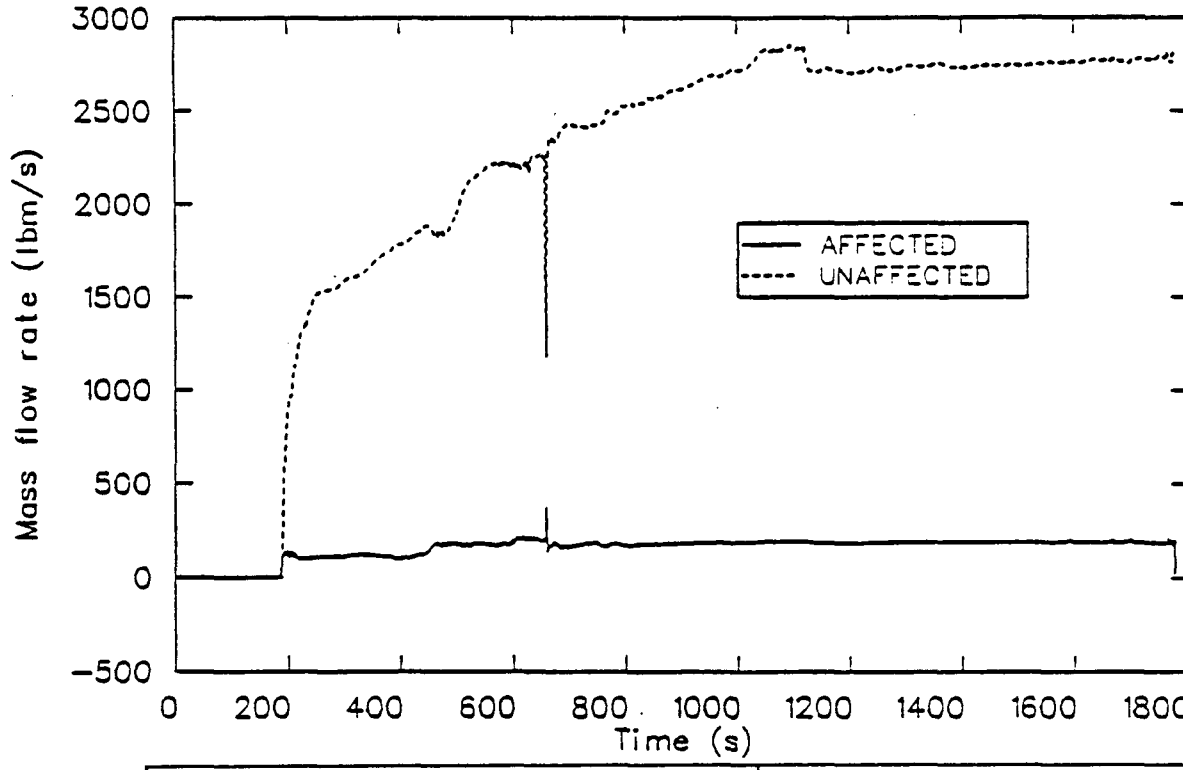
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Primary and Secondary System Pressures; Outlet Riser Break, Base Case
	Figure 15.6-255



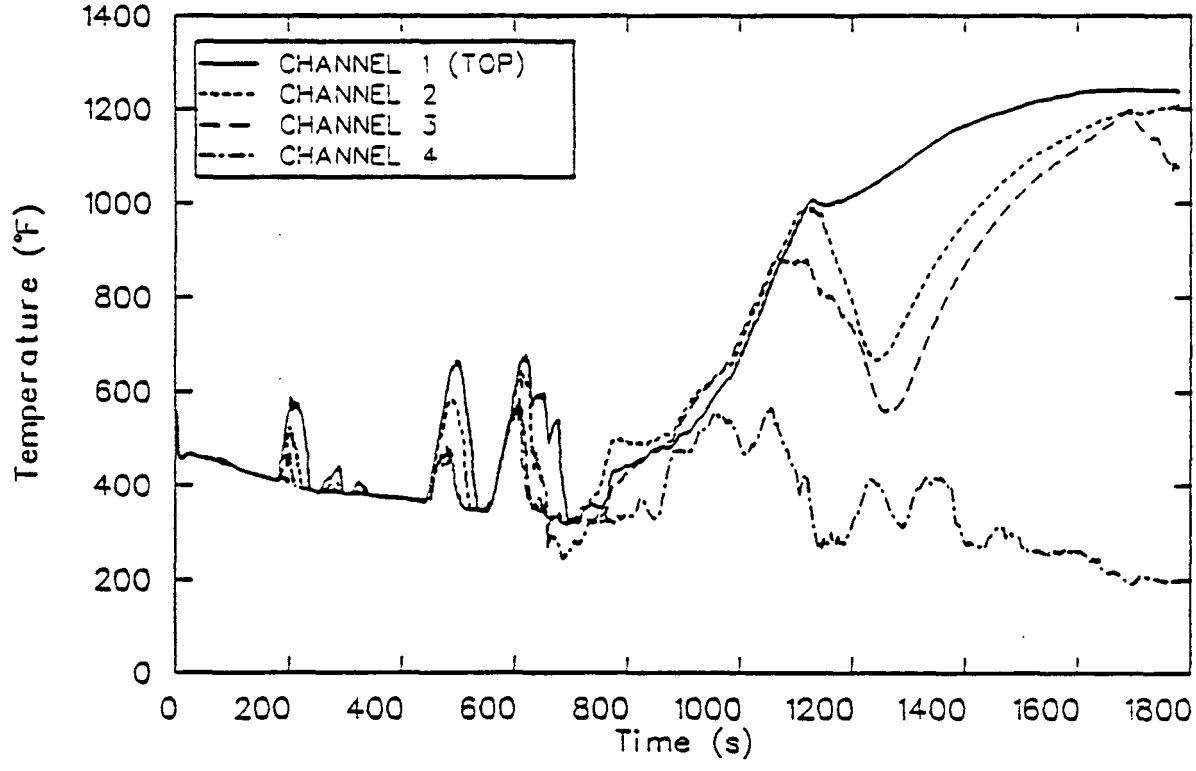
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	V-4 Valve Mass Flow Rates; Outlet Riser Break, Base Case
	Figure 15.6-256



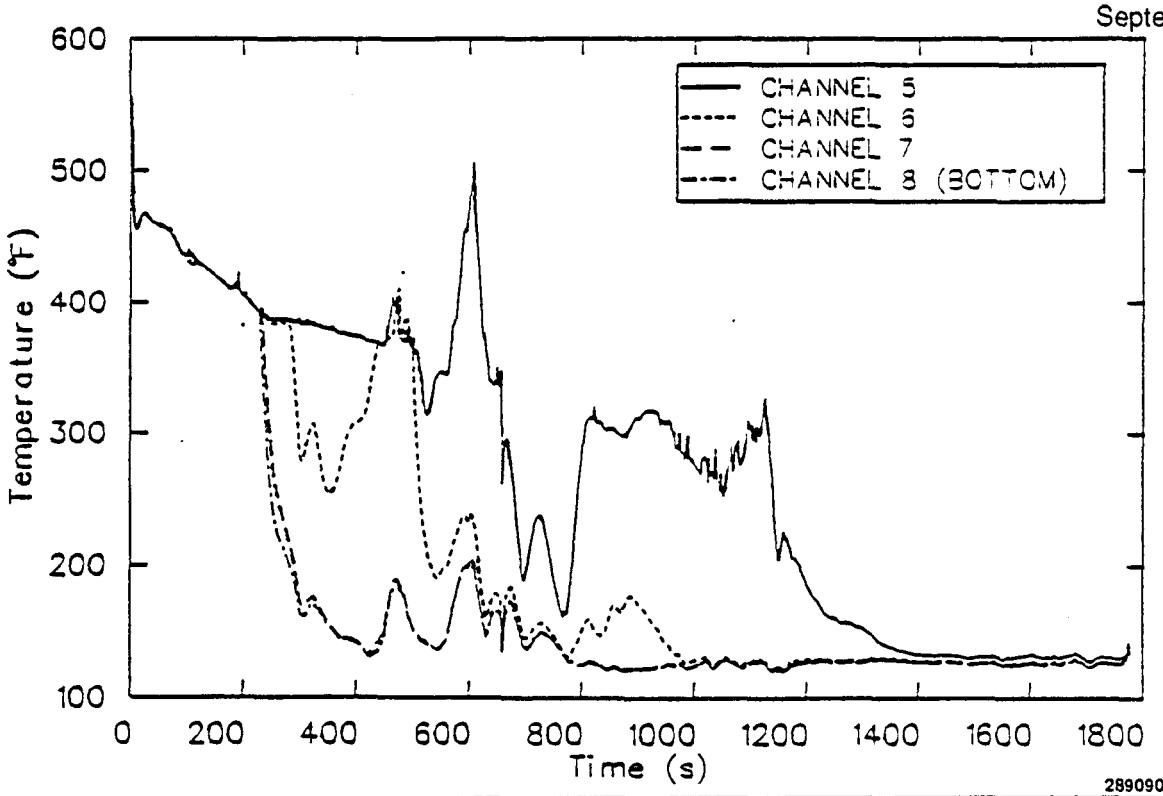
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	ECCS Injection Mass Flow Rates; Outlet Riser Break, Base Case Figure 15.6-257



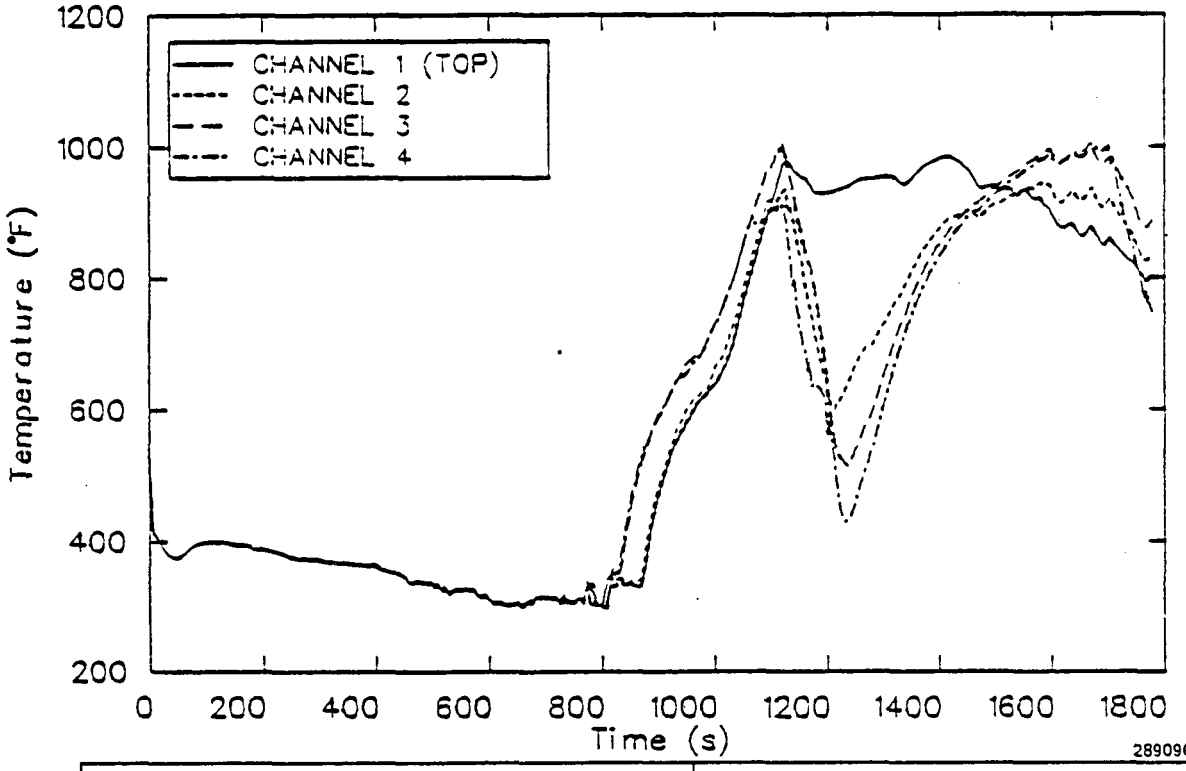
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in Upper Unaffected Core; Outlet Riser Break, Base Case Figure 15.6-258



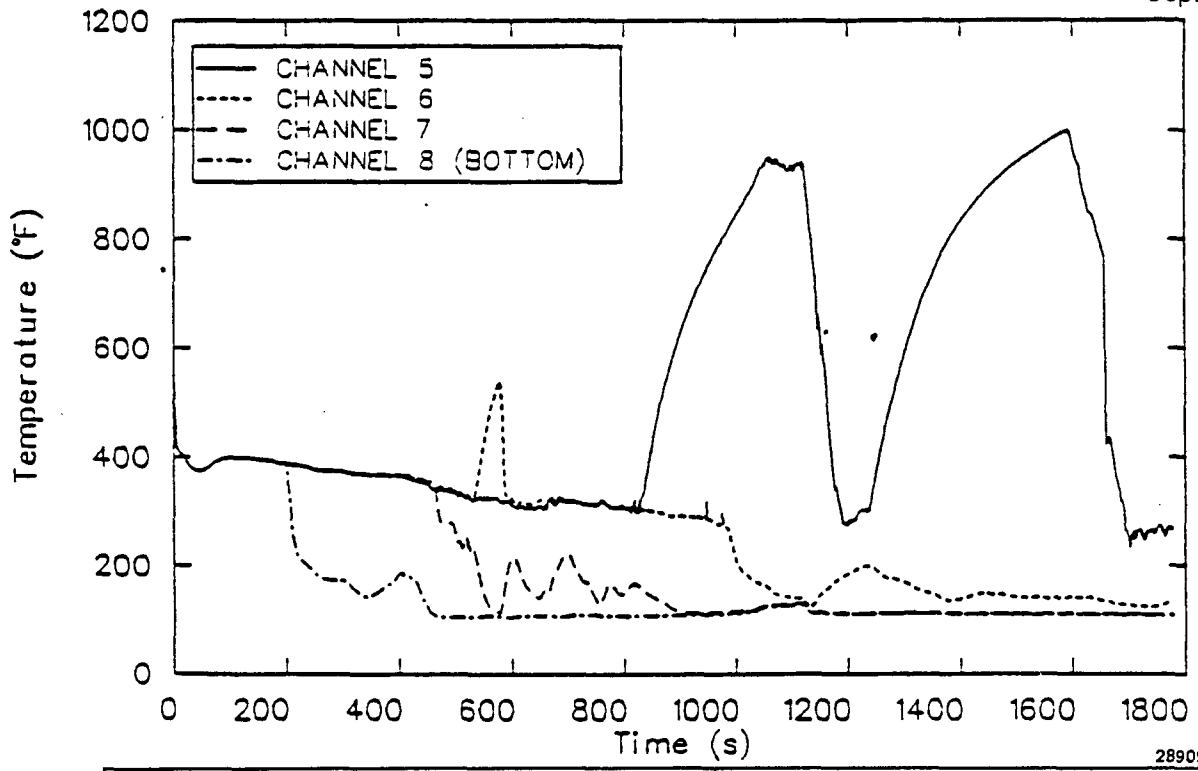
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in Lower Unaffected Core; Outlet Riser Break, Base Case
	Figure 15.6-259



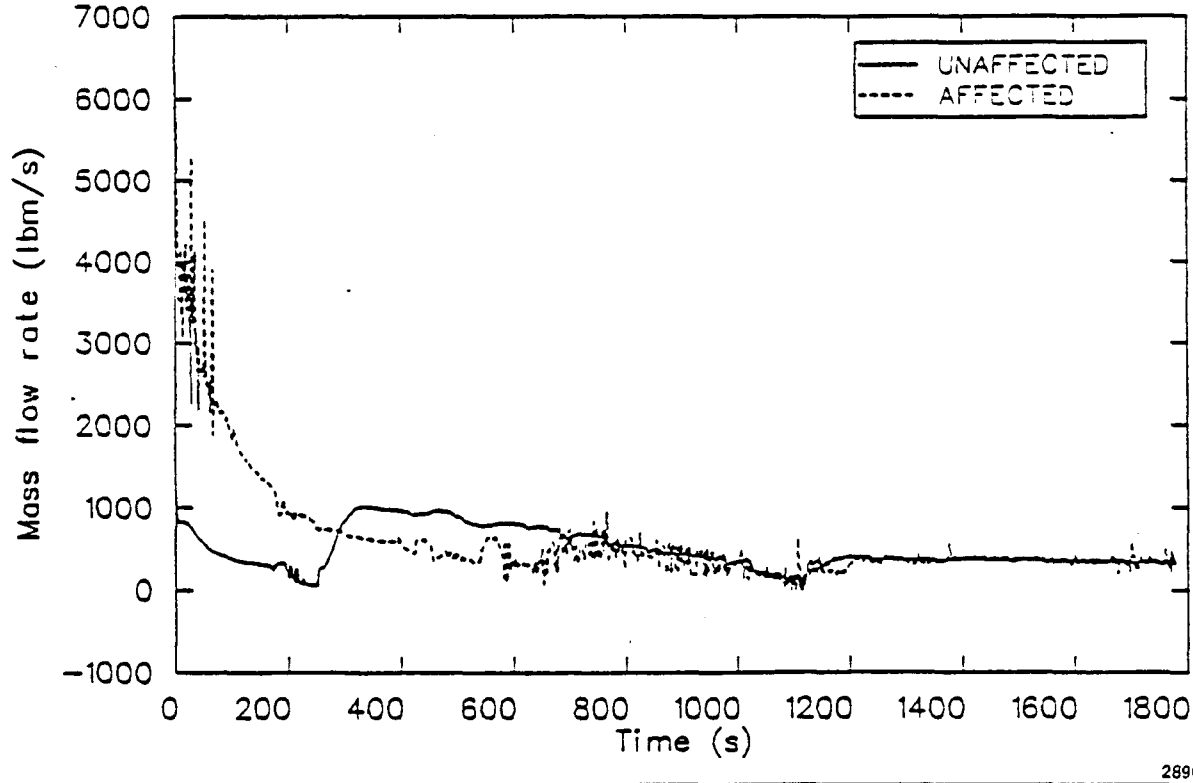
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in Upper Affected Core; Outlet Riser Break, Base Case
	Figure 15.6-260



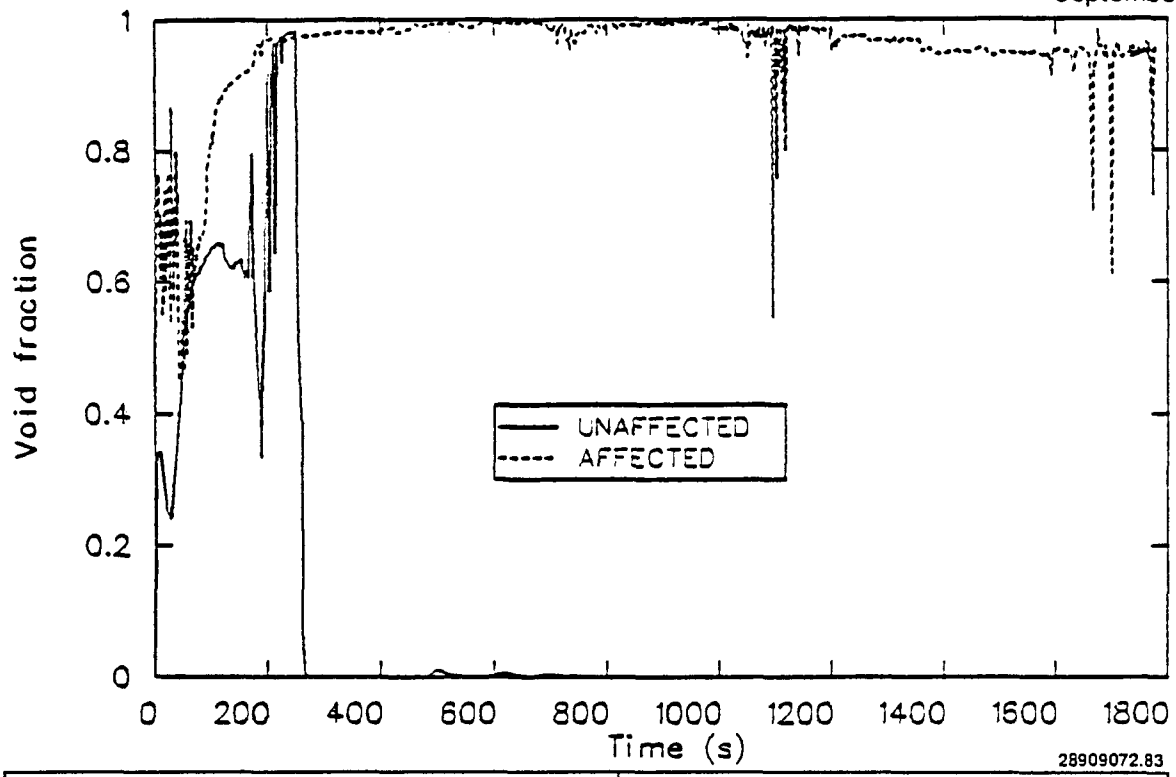
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Fuel Temperatures in Lower Affected Core; Outlet Riser Break, Base Case
	Figure 15.6-261

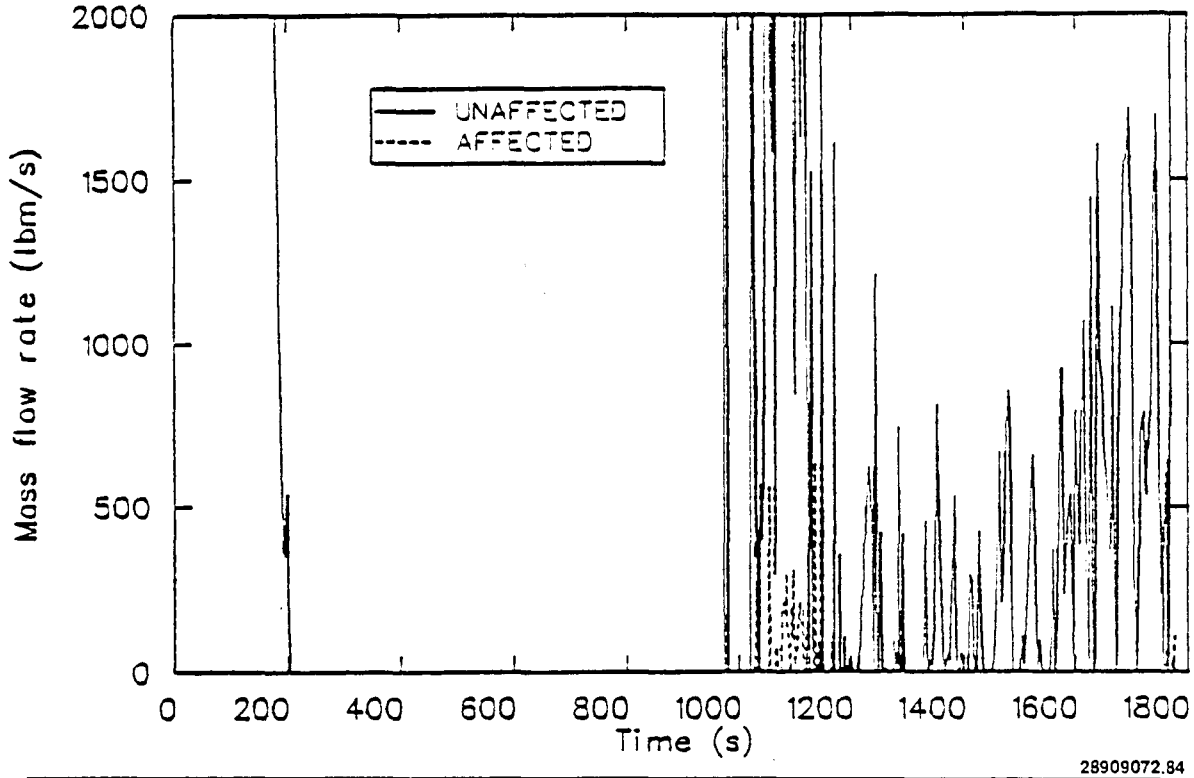


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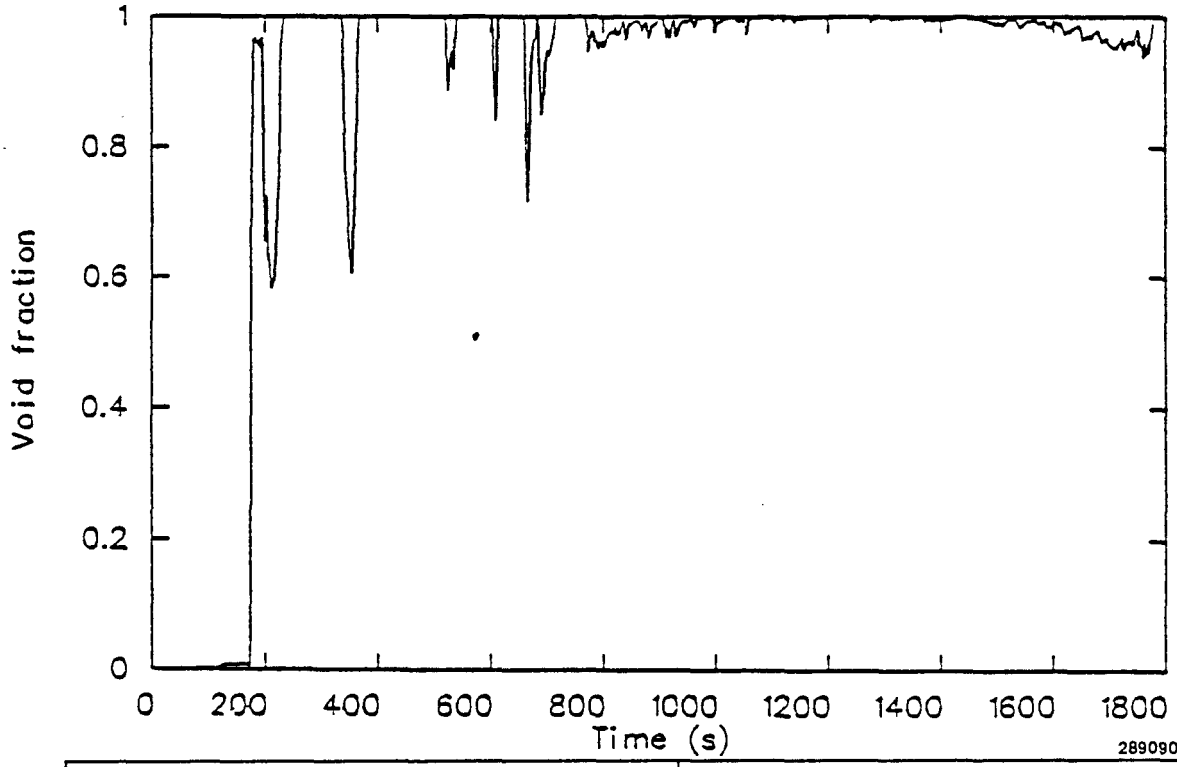
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Break Flow Rates; Outlet Riser Break, Base Case
	Figure 15.6-262



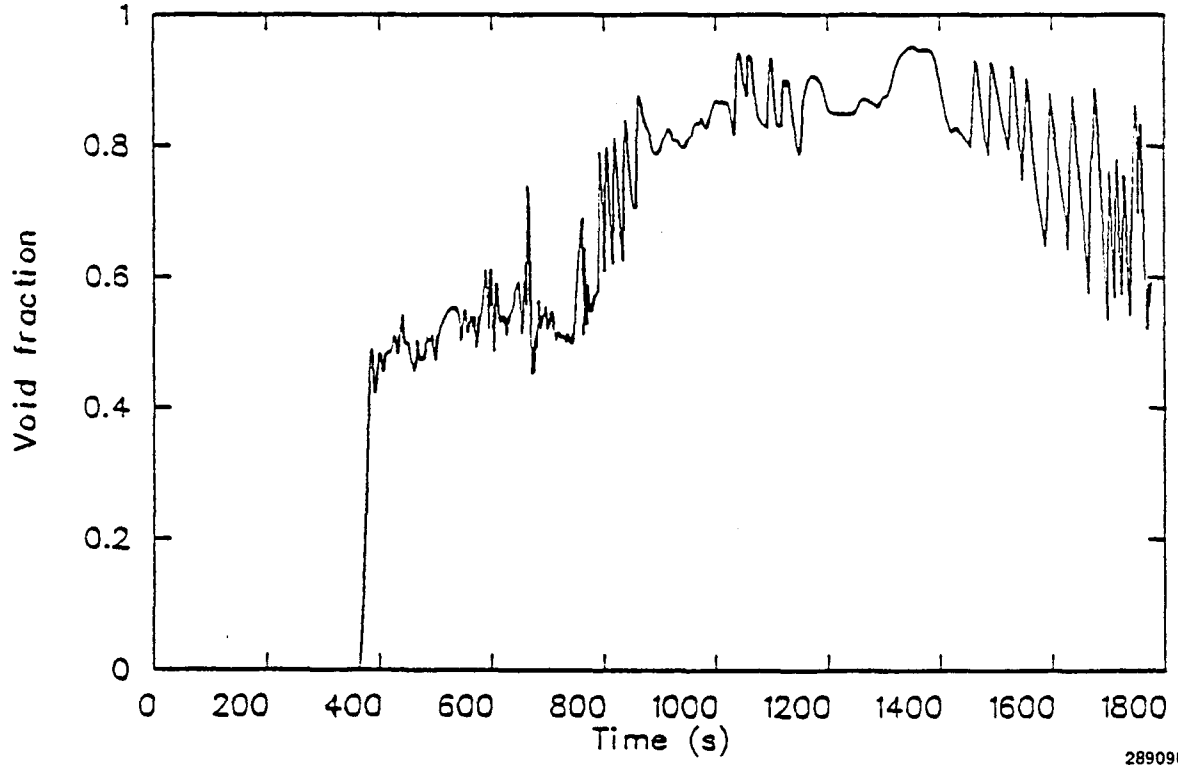
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Void Fractions Upstream of Breaks; Outlet Riser Break, Base Case Figure 15.6-263



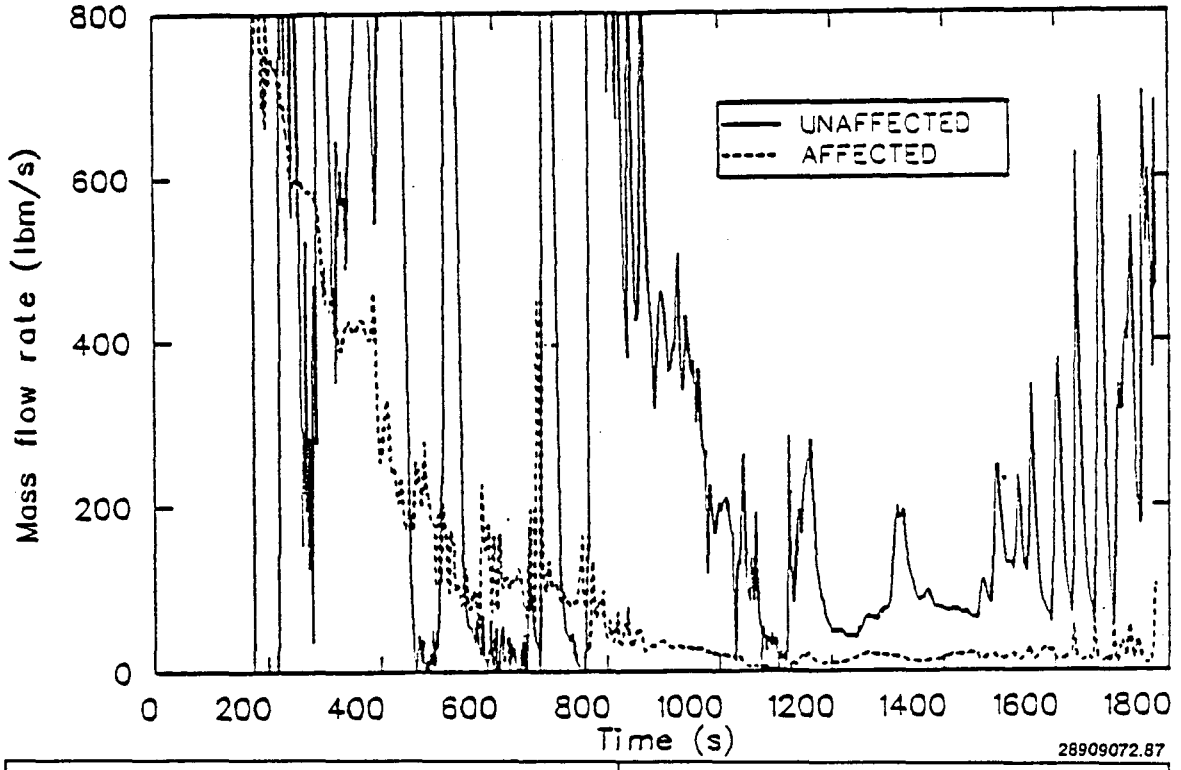
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	CV-3 Valve Flow Rates; Outlet Riser Break, Base Case Figure 15.6-264



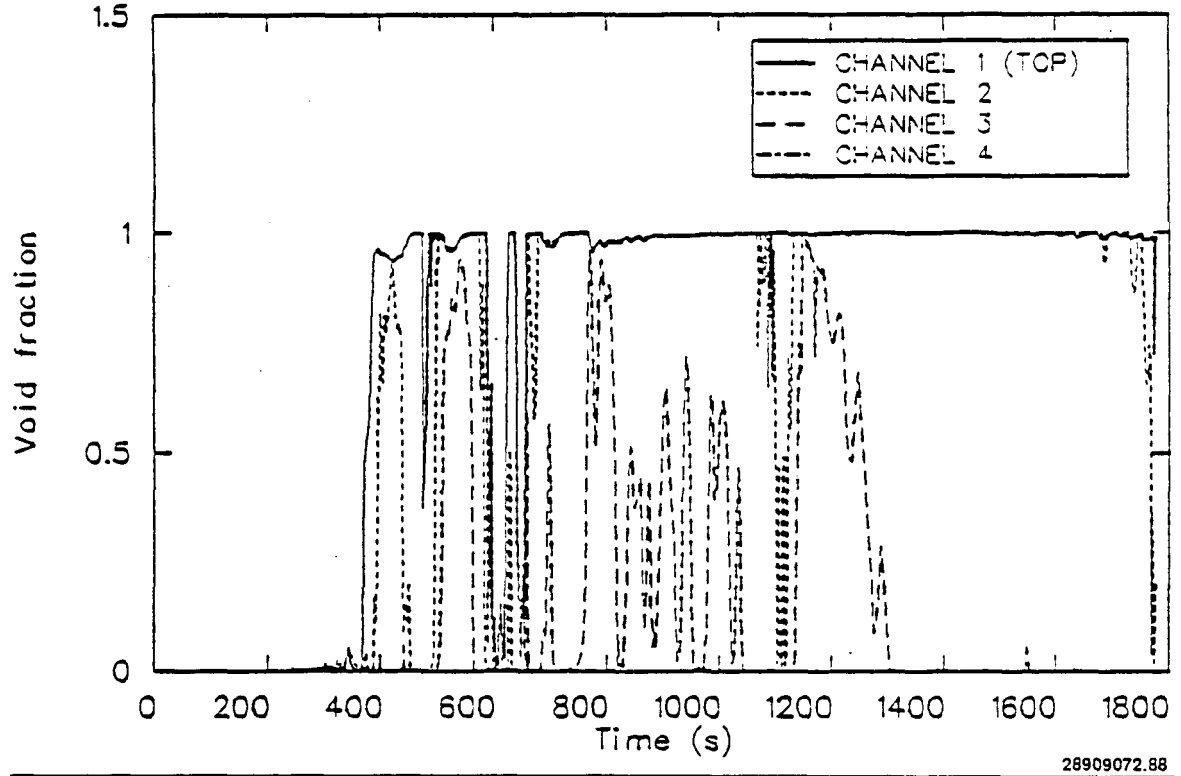
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Primary Coolant Pump Void Fraction; Outlet Riser Break, Base Case
	Figure 15.6-265



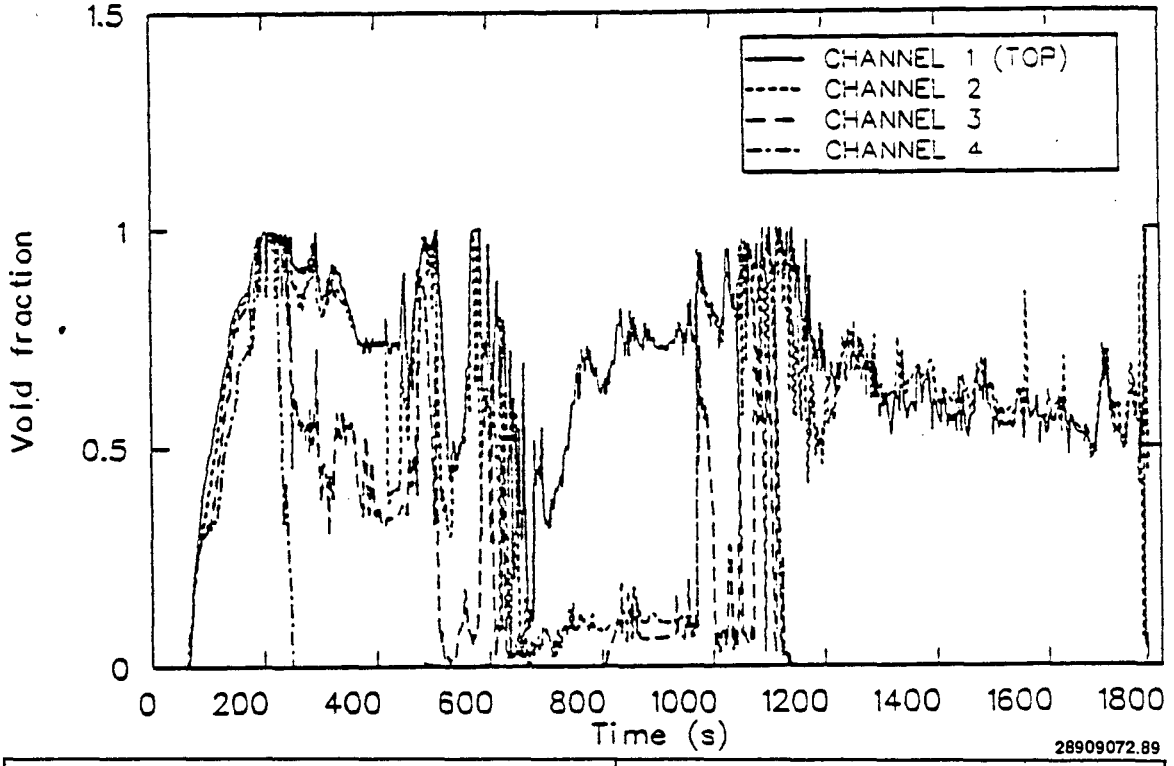
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	Figure 15.6-266



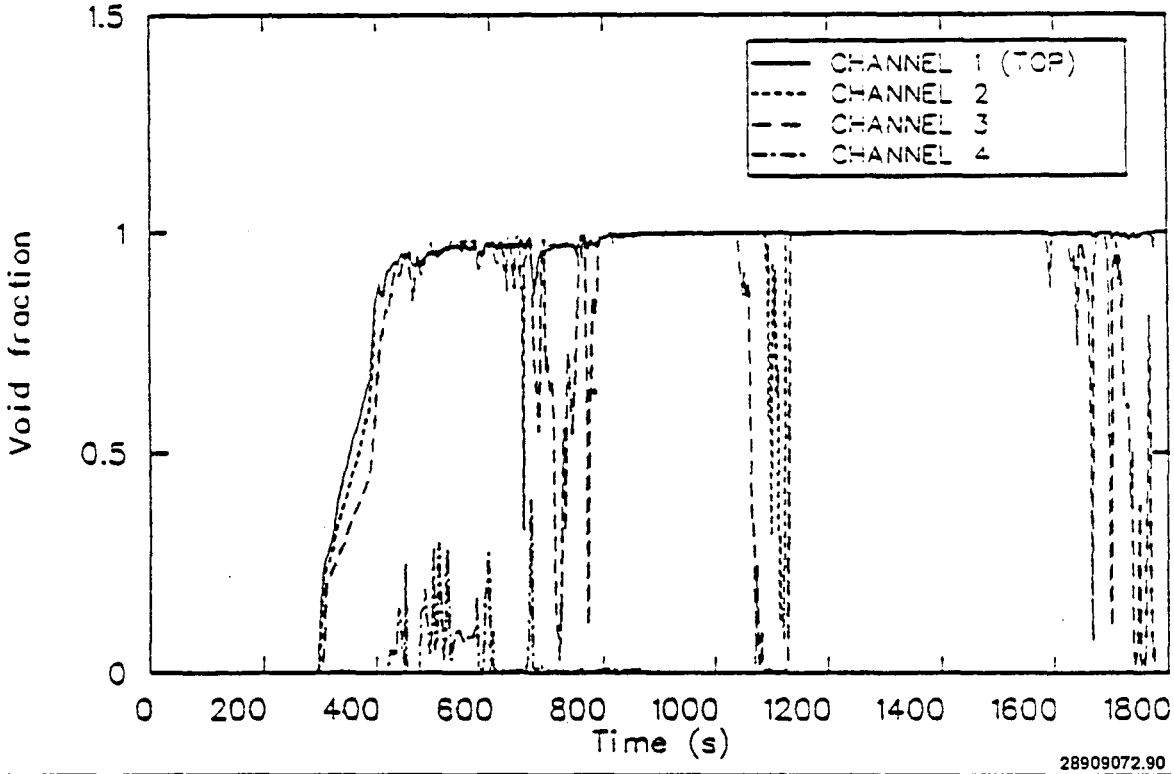
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	Figure 15.6-267



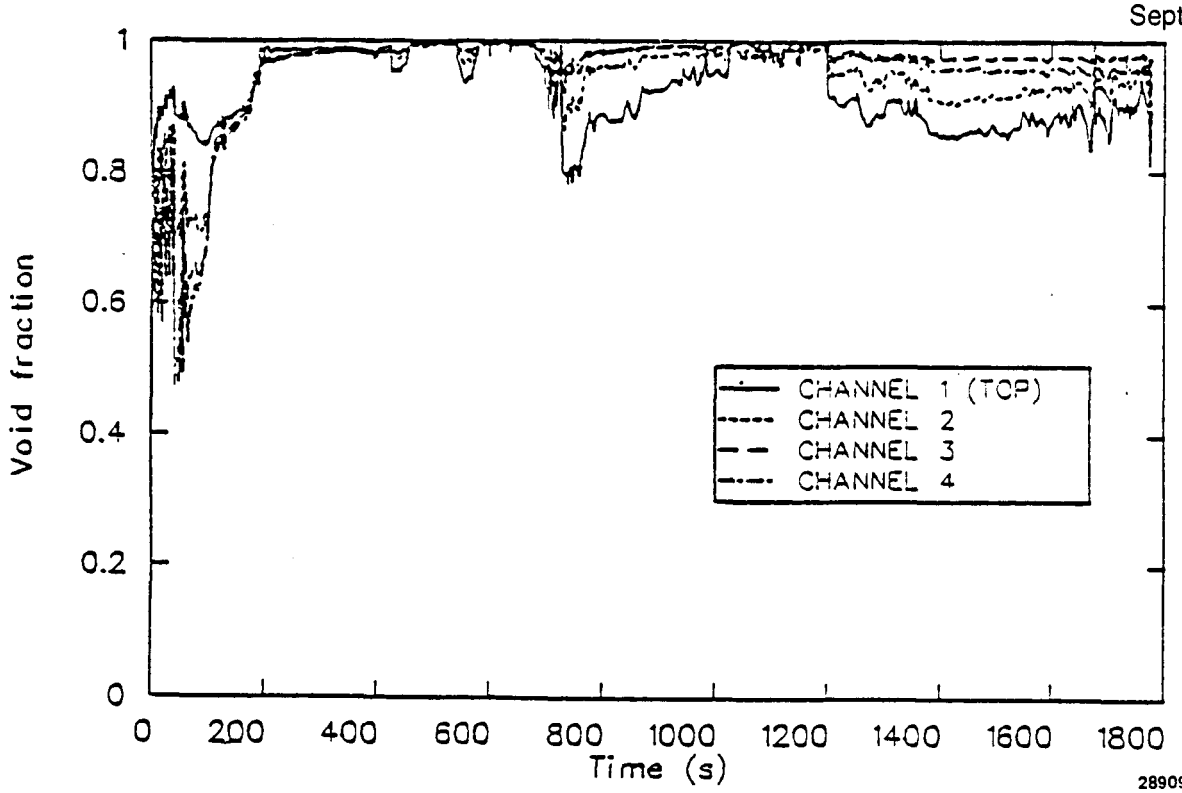
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Inlet Riser Void Fractions; Outlet Riser Break, Base Case
	Figure 15.6-268



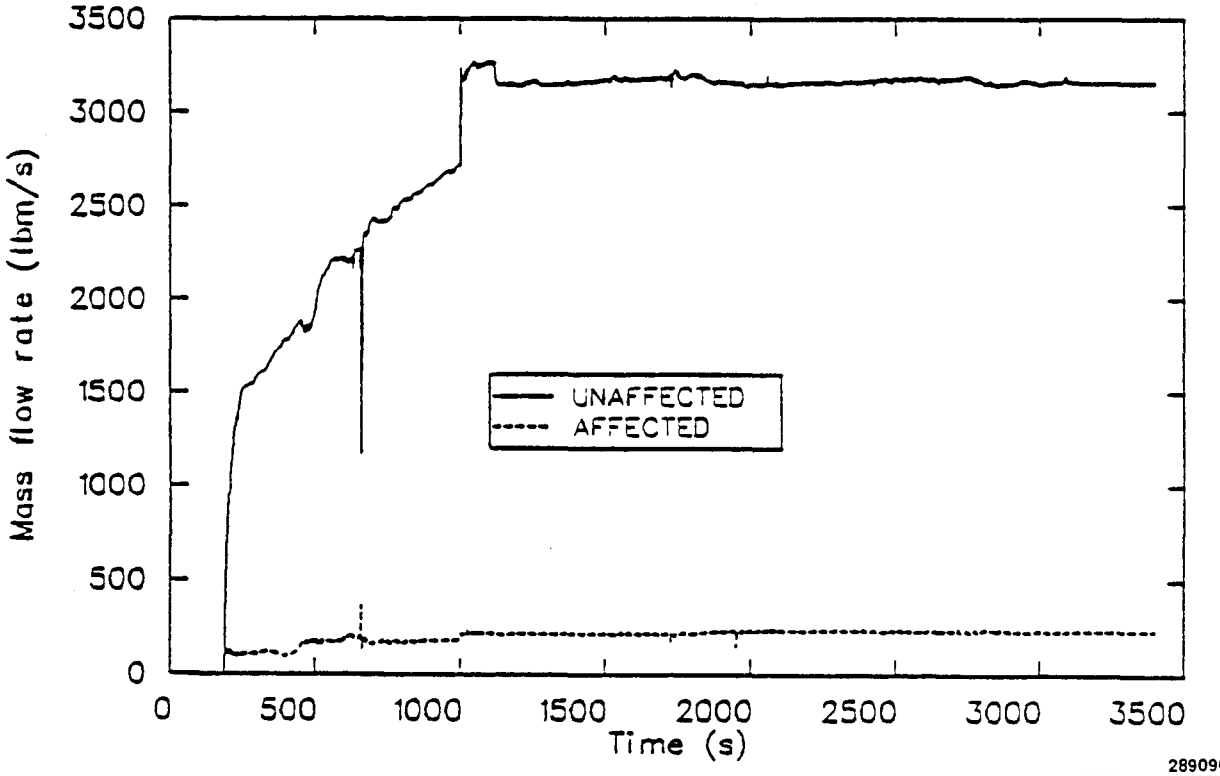
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Outlet Riser Void Fractions; Outlet Riser Break, Base Case Figure 15.6-269



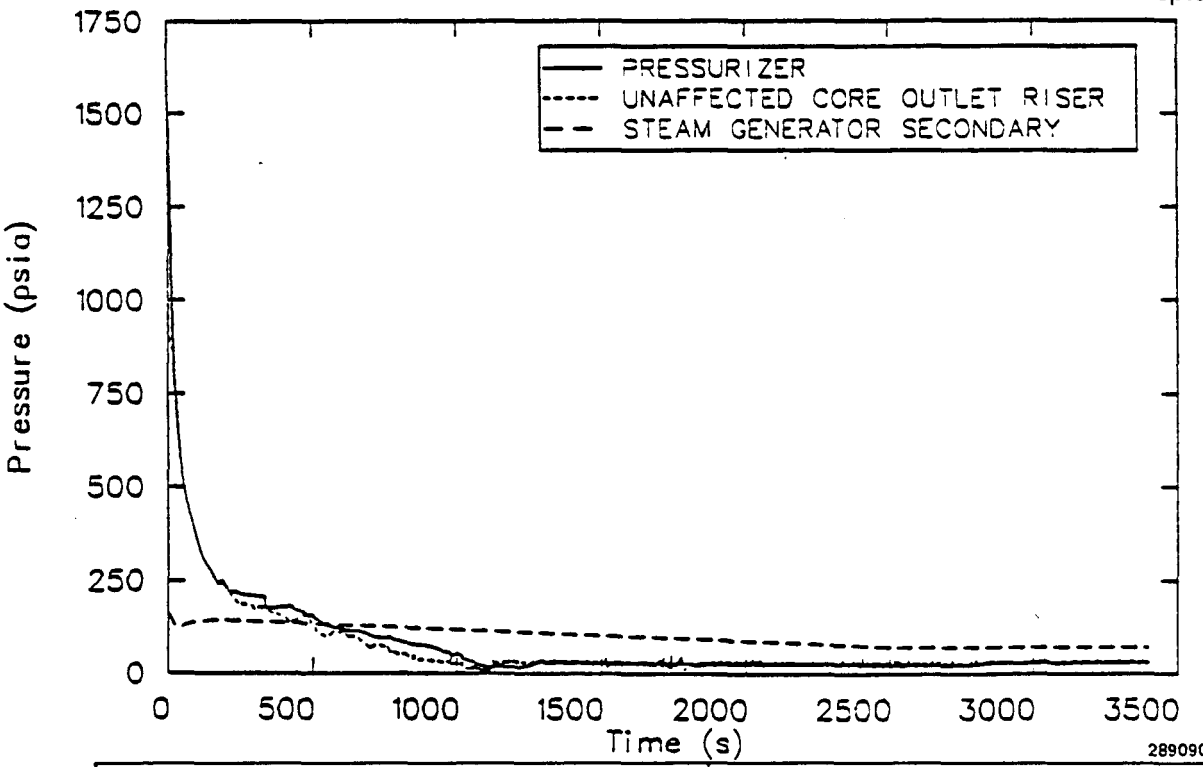
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Outlet Riser Void Fractions; Outlet Riser Break, Base Case Figure 15.6-270



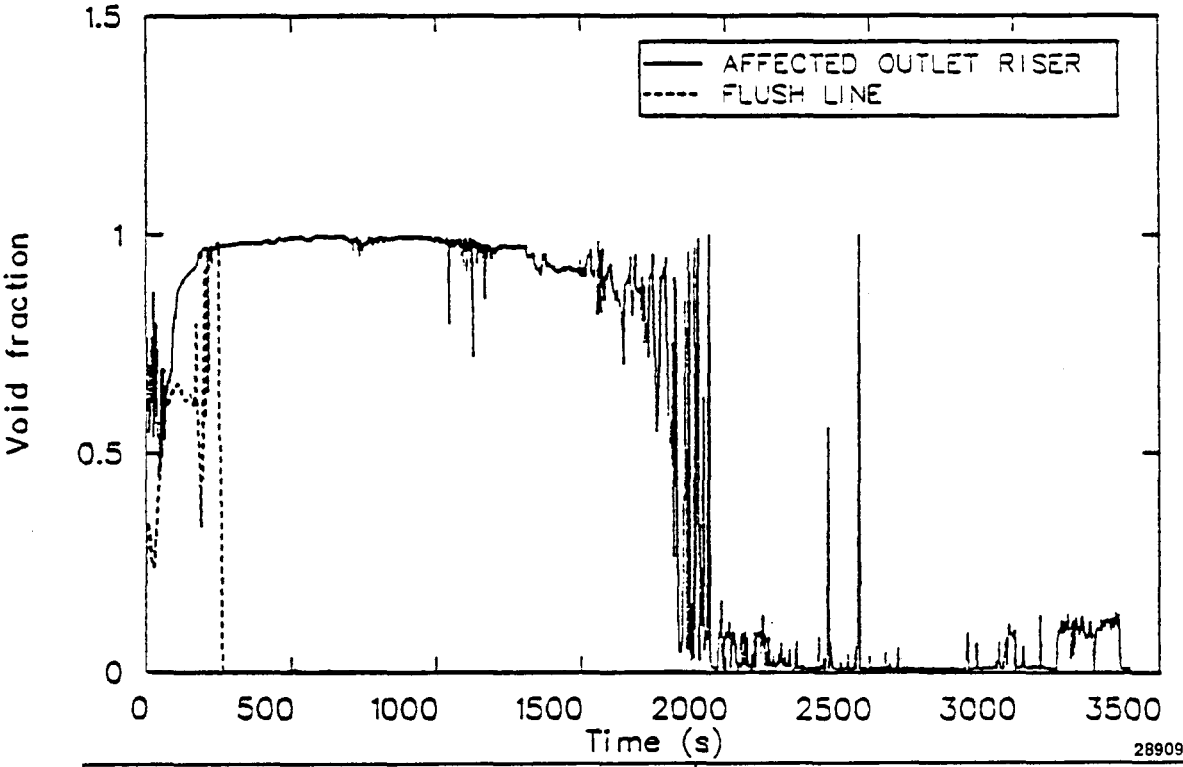
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Outlet Riser Void Fractions; Outlet Riser Break, Base Case
	Figure 15.6-271



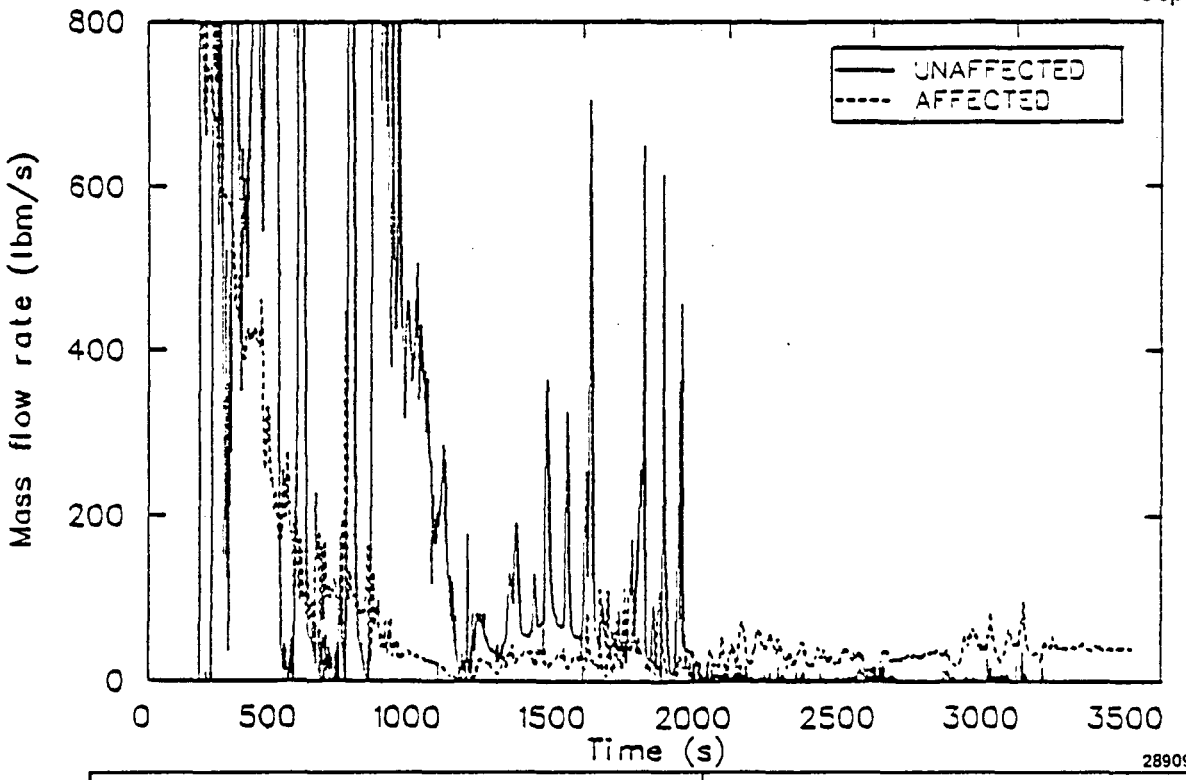
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	Figure 15.6-272



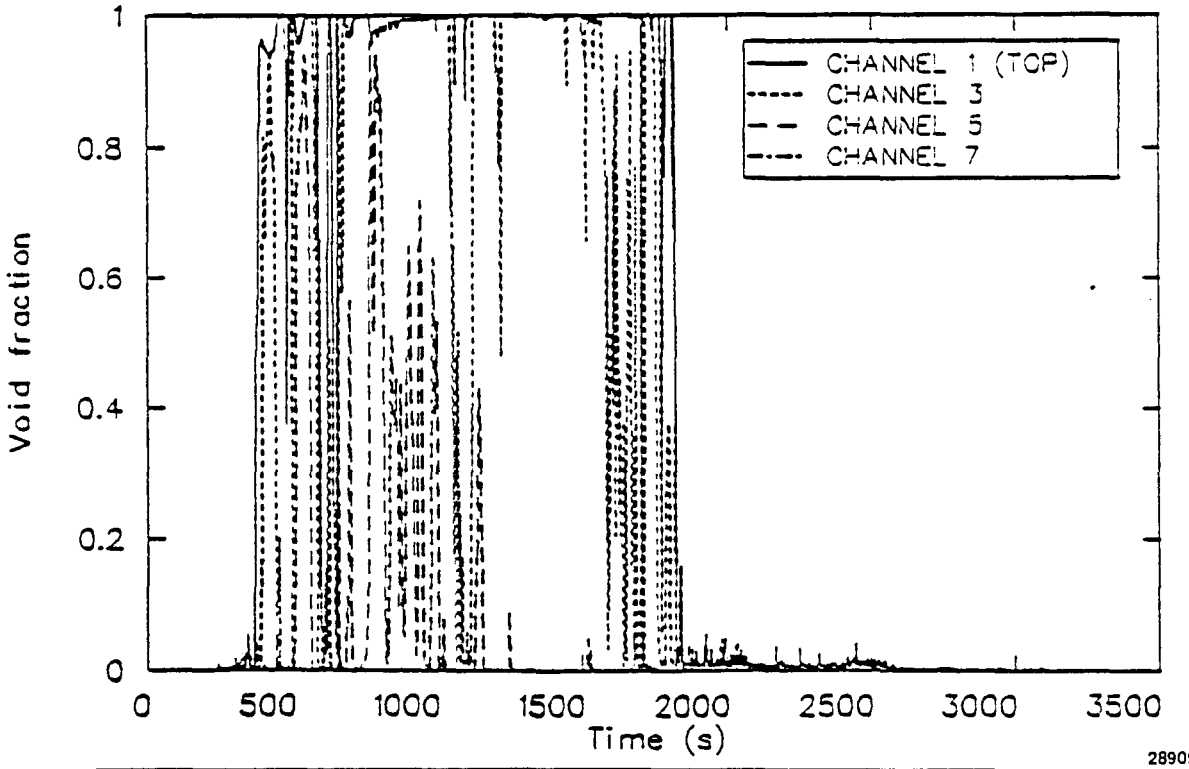
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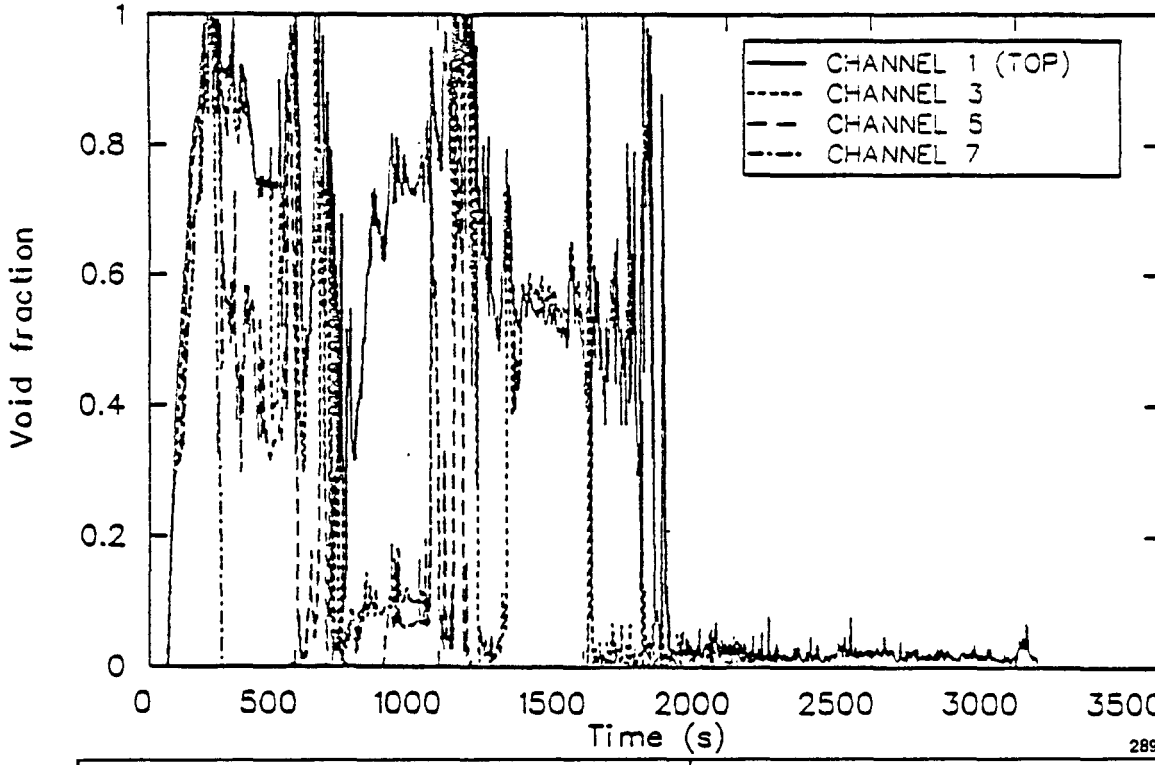
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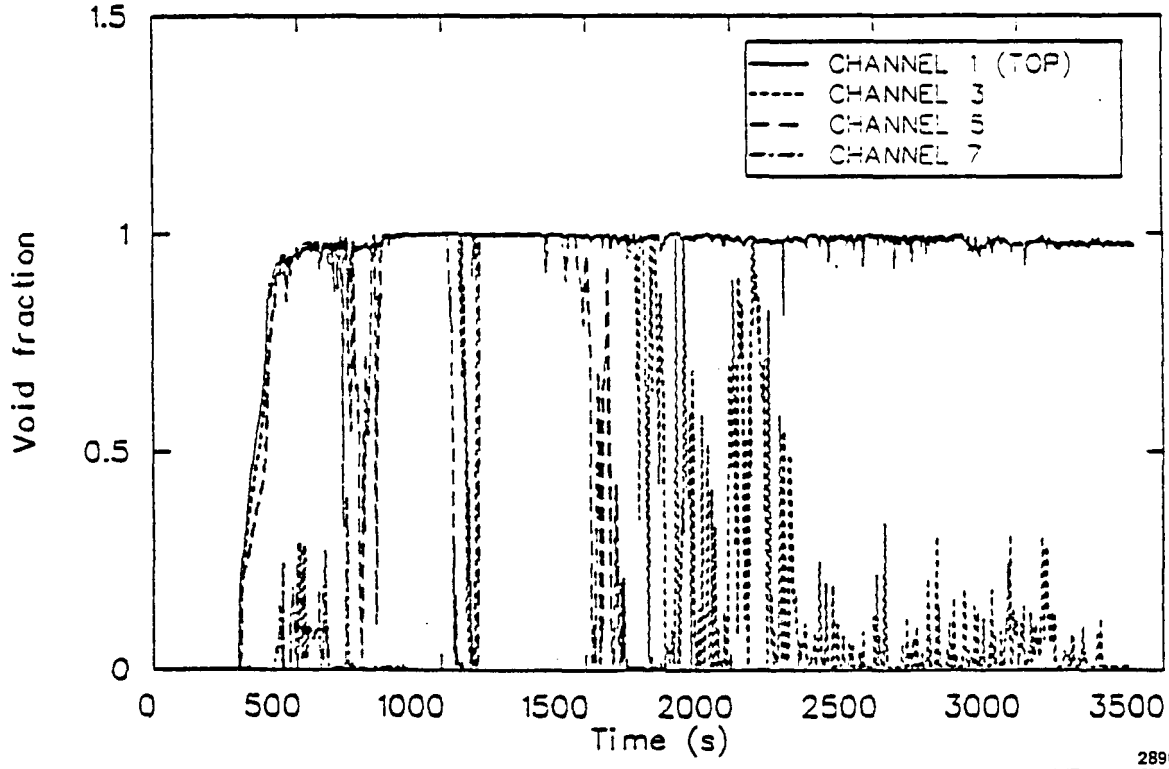
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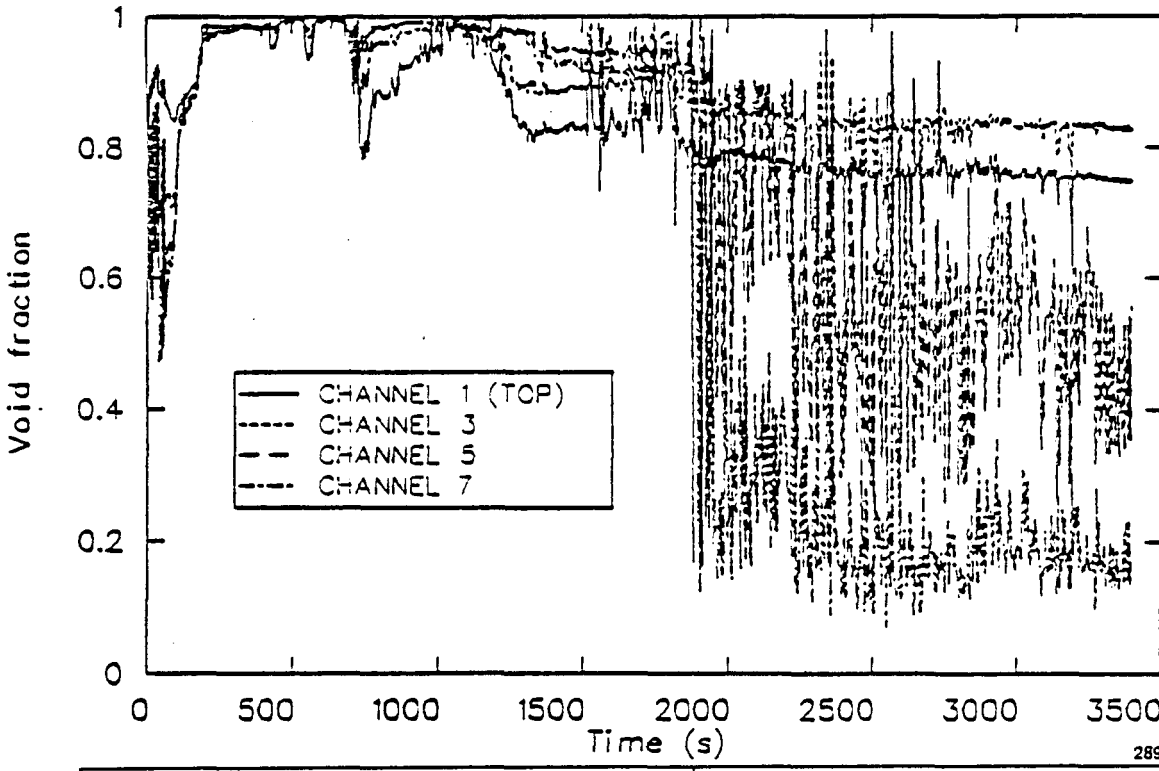
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Core Inlet Riser Void Fractions; Outlet Riser Break, Sensitivity Figure 15.6-276



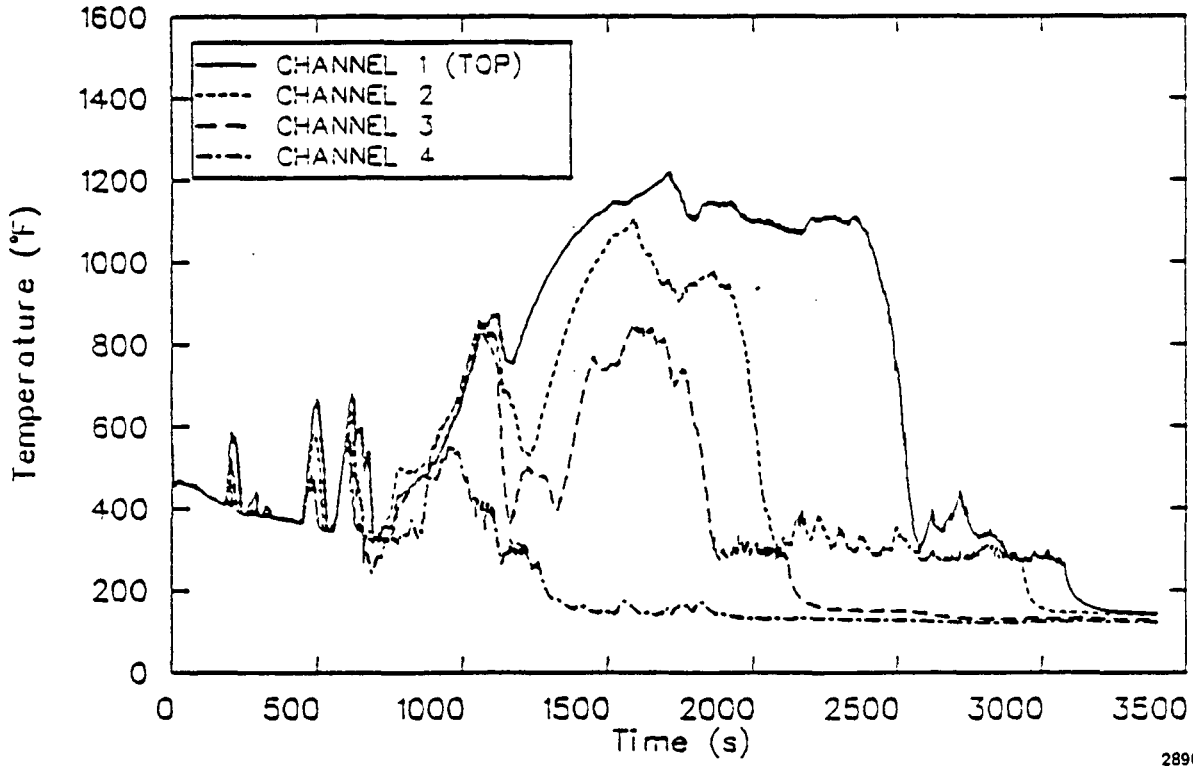
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Unaffected Core Outlet Riser Void Fractions; Outlet Riser Break, Sensitivity Figure 15.6-277



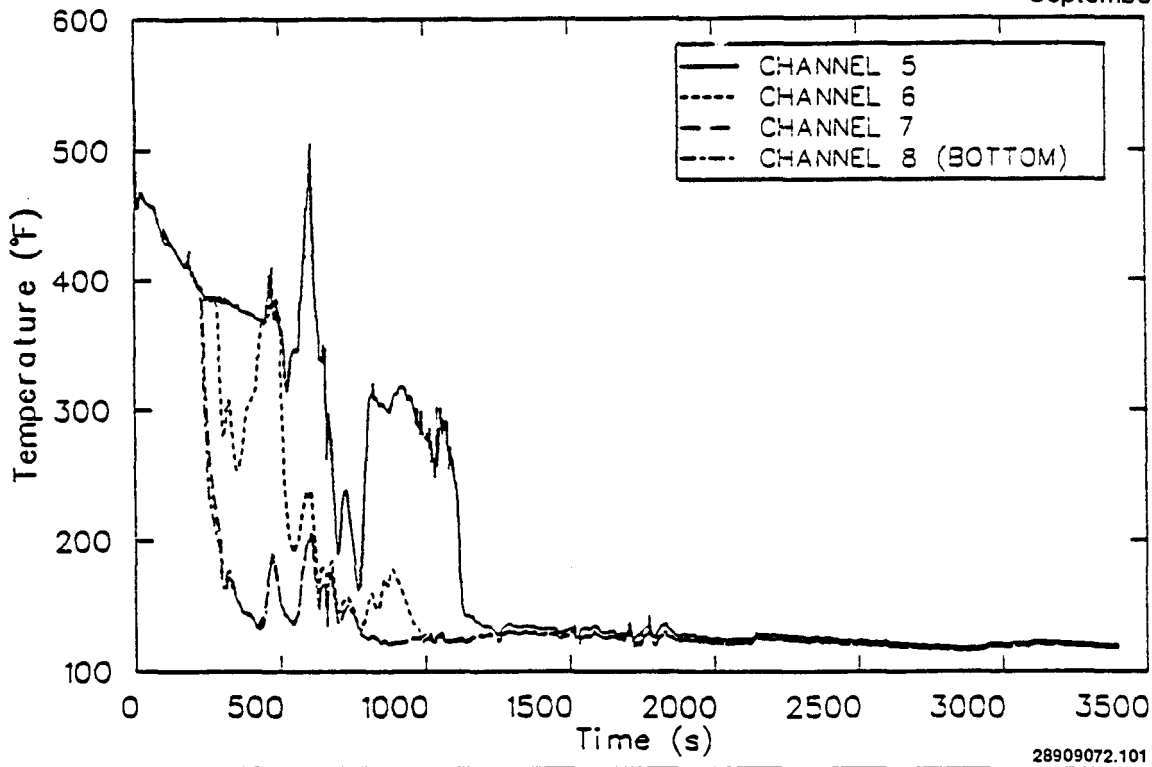
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Core Inlet Riser Void Fractions; Outlet Riser Break, Sensitivity Figure 15.6-278



Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Affected Core Outlet Riser Void Fractions; Outlet Riser Break, Sensitivity
	Figure 15.6-279

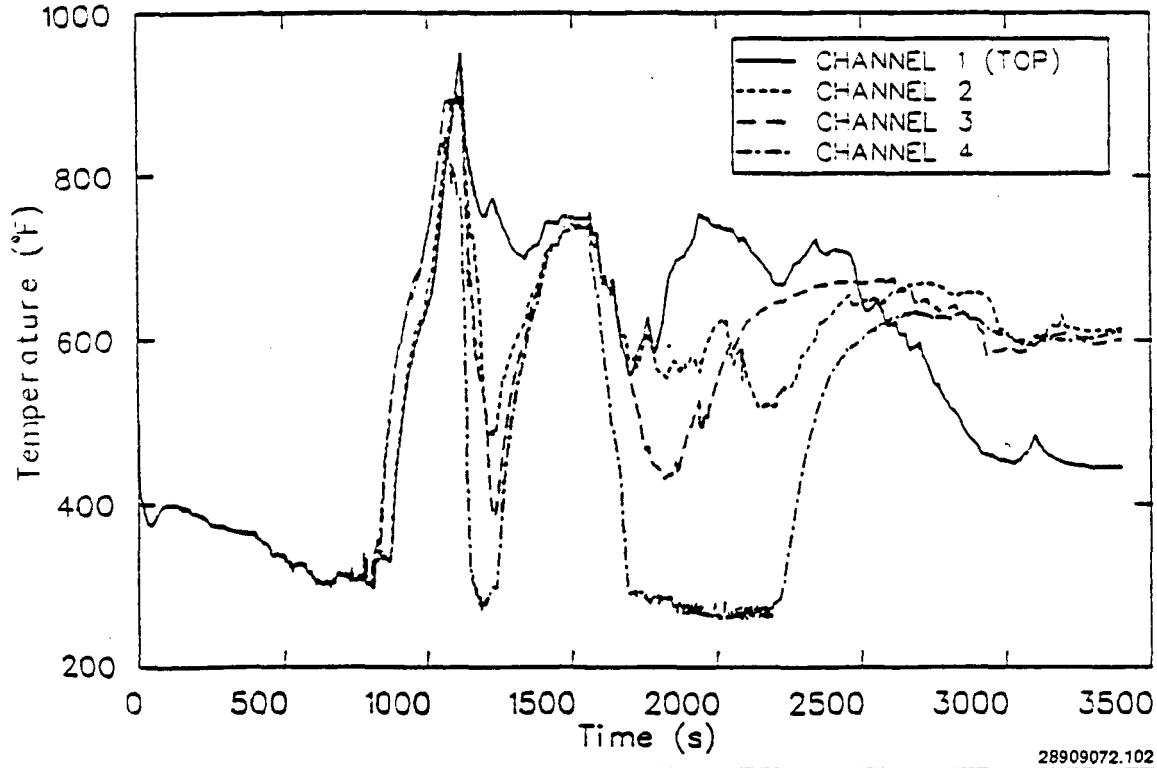


Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Unaffected Core Fuel Temperatures; Outlet Riser Break, Sensitivity
	Figure 15.6-280



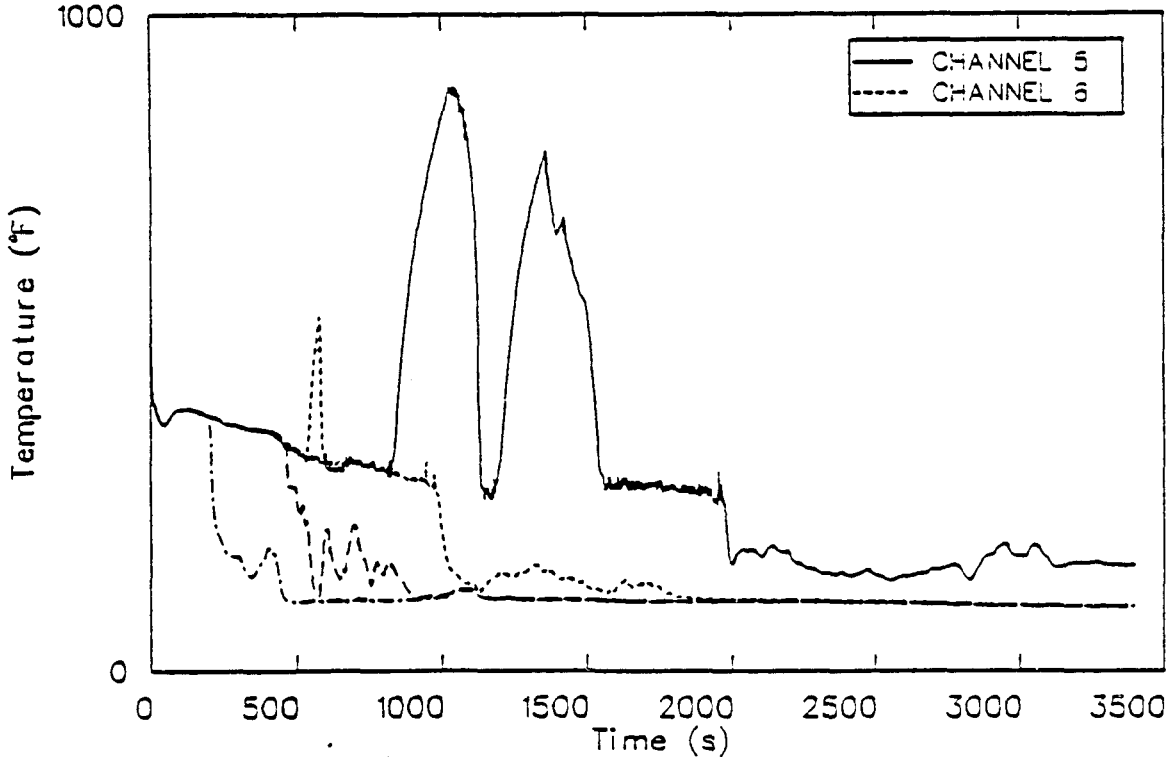
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Unaffected Core Fuel Temperatures; Outlet Riser Break, Sensitivity
	Figure 15.6-281



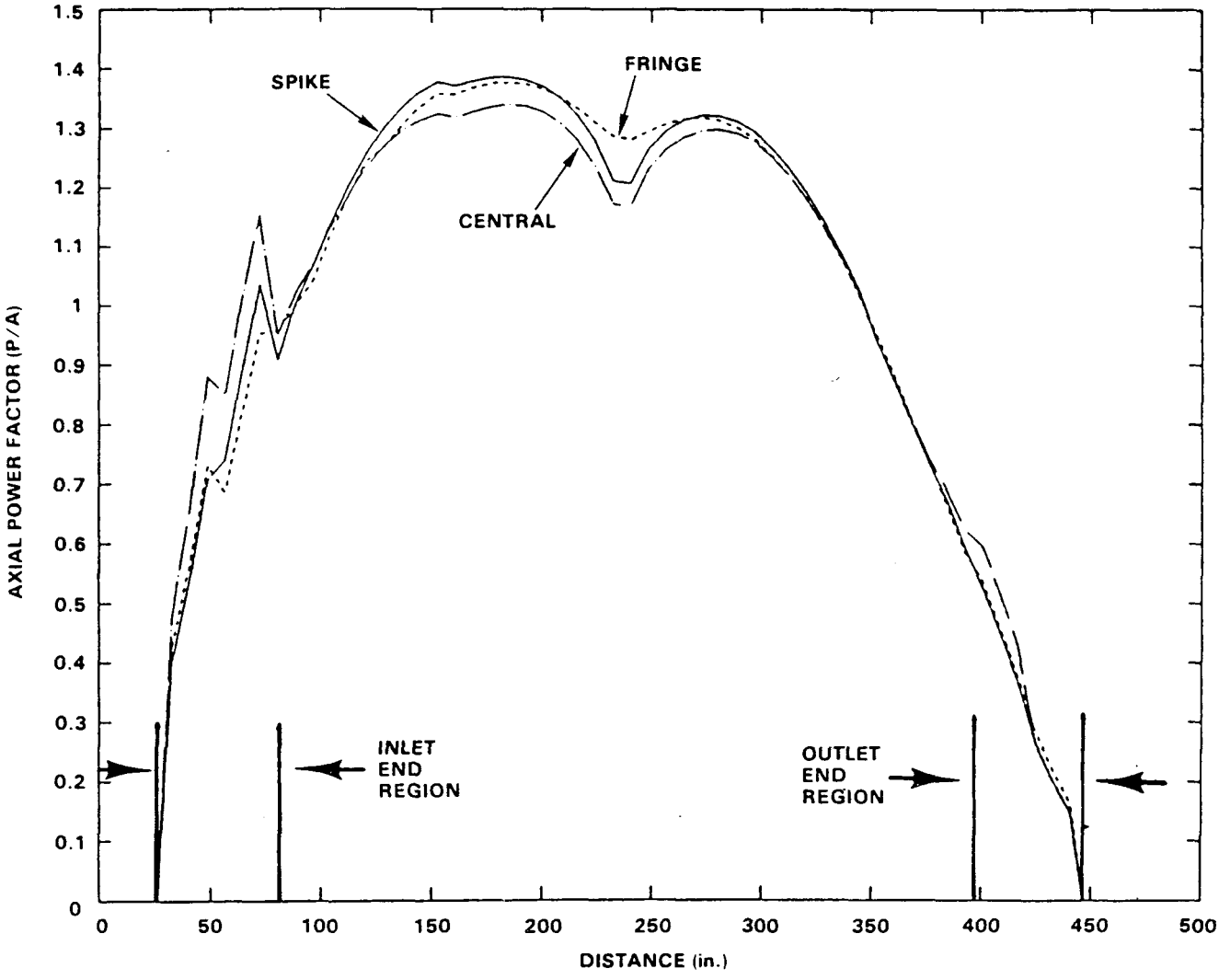
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Upper Affected Core Fuel Temperatures; Outlet Riser Break, Sensitivity
	Figure 15.6-282



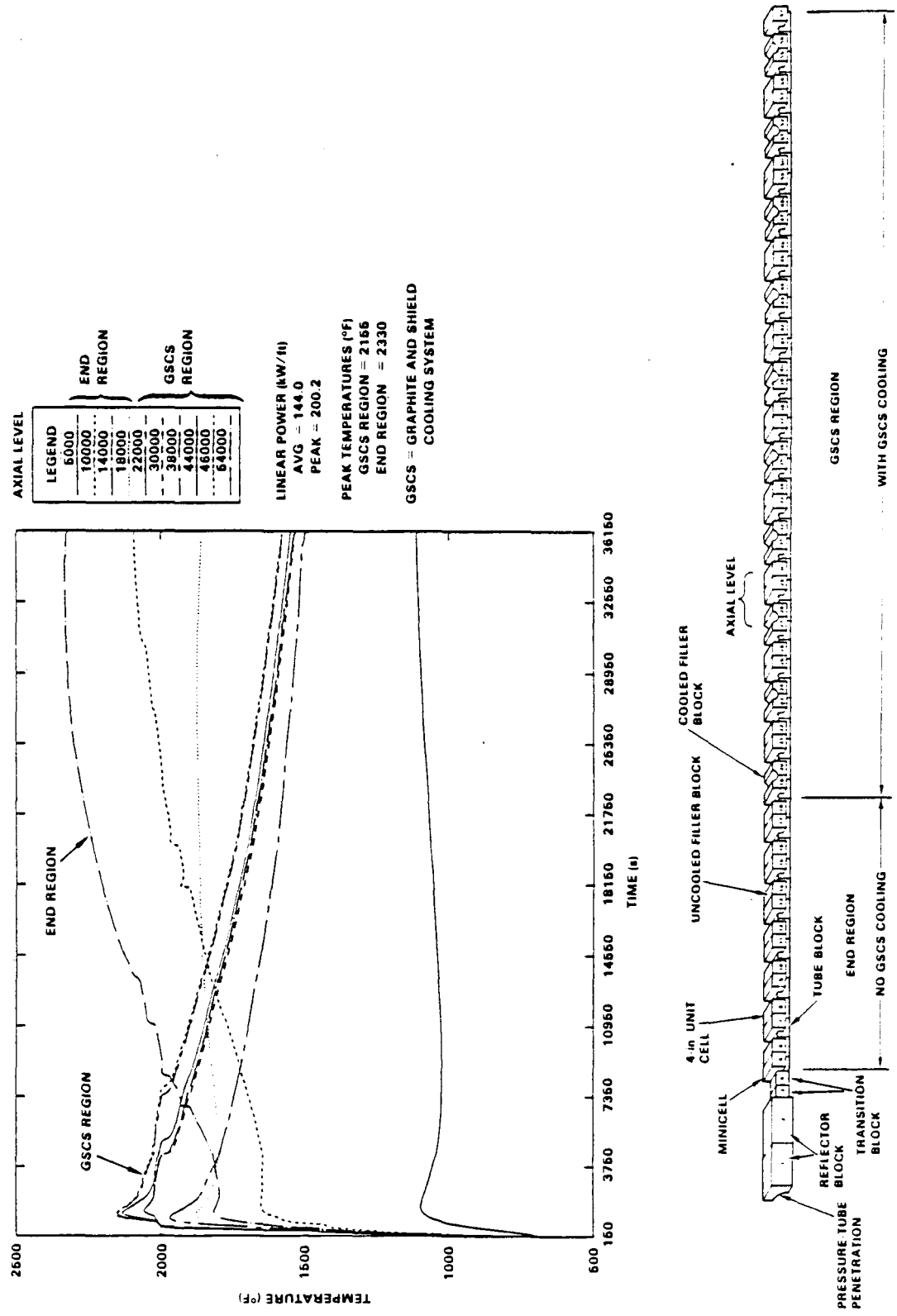
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Lower Affected Core Fuel Temperatures; Outlet Riser Break, Sensitivity
	Figure 15.6-283



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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Comparison of Fringe-, Spike-, and Central-Zone Adjusted Axial Power Profiles from TRUMP Analysis of Maximum Hydrogen-Mitigation Design Basis Accident Figure 15.6-284



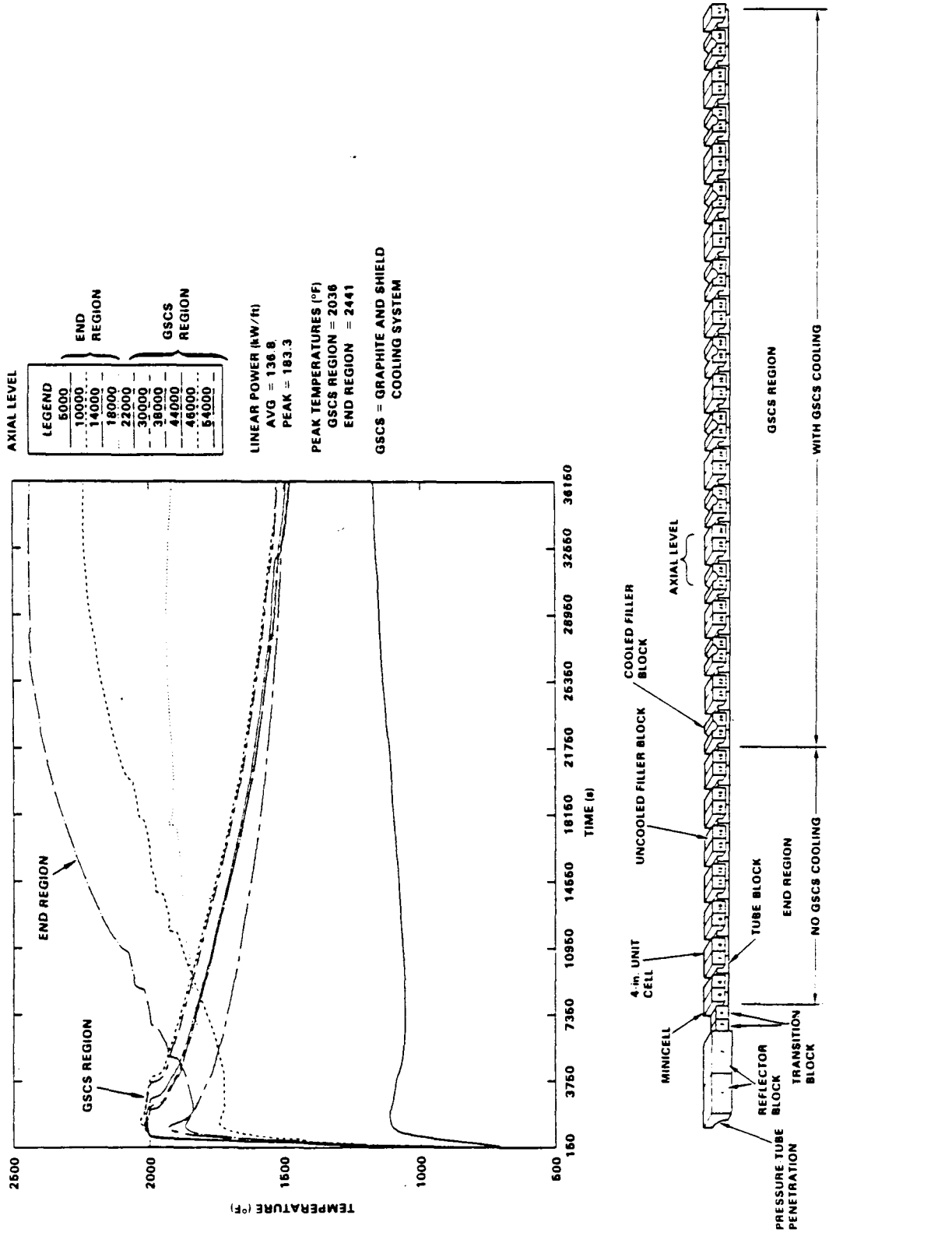
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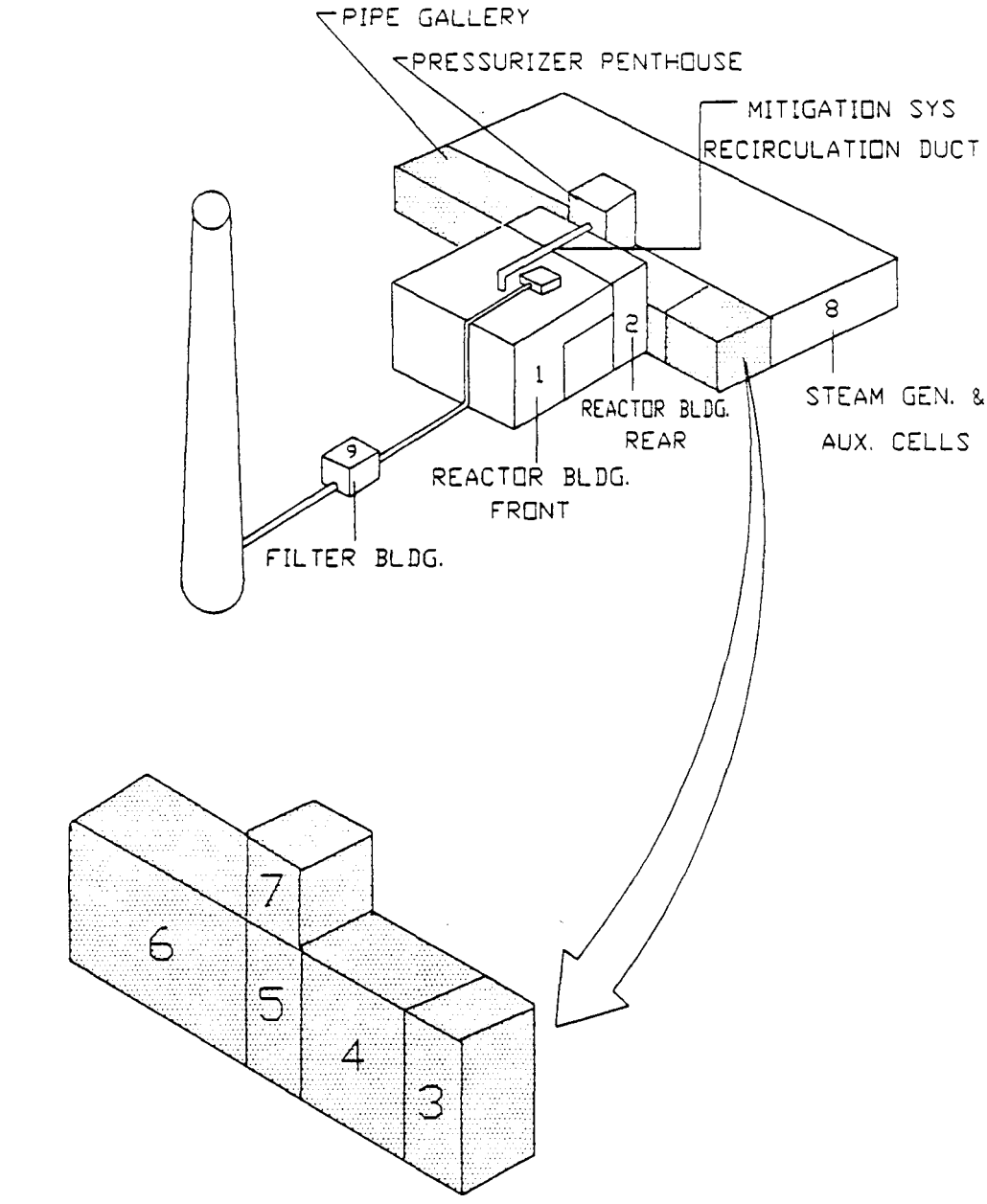
Inner-Fuel Temperatures versus Time for Spike-Zone, Inlet-Half, Maximum-Power Tube. TRUMP-BD Code Analysis of Hydrogen Mitigation Design Basis Accident

Figure 15.6-285



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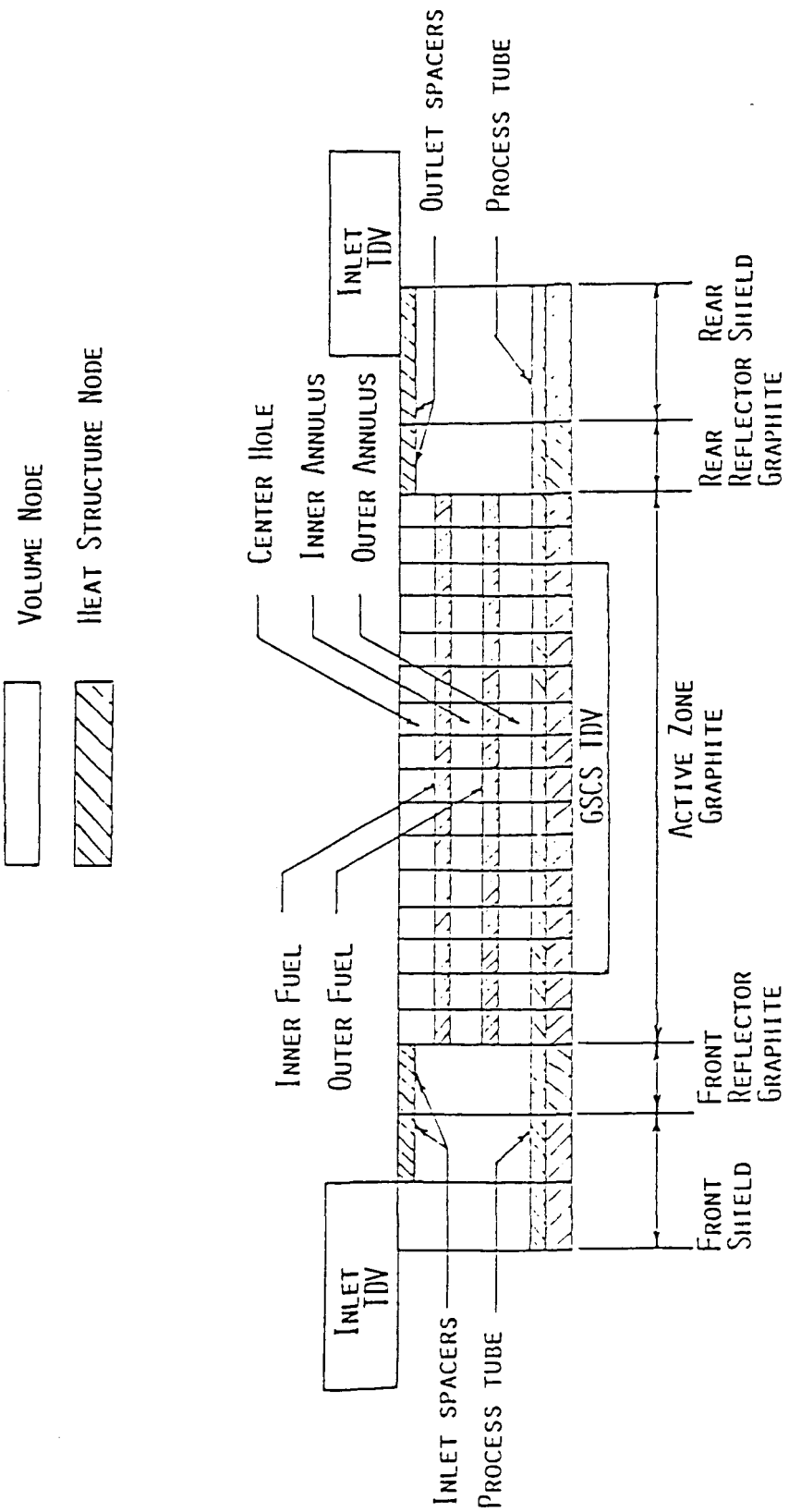
Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Inner-Fuel Temperatures versus Time for Central-Zone, Inlet-Half, Maximum-Power Tube. TRUMP-BD Code Analysis of Hydrogen Mitigation Design Basis Accident Figure 15.6-286



PIPE GALLERY NODALIZATION

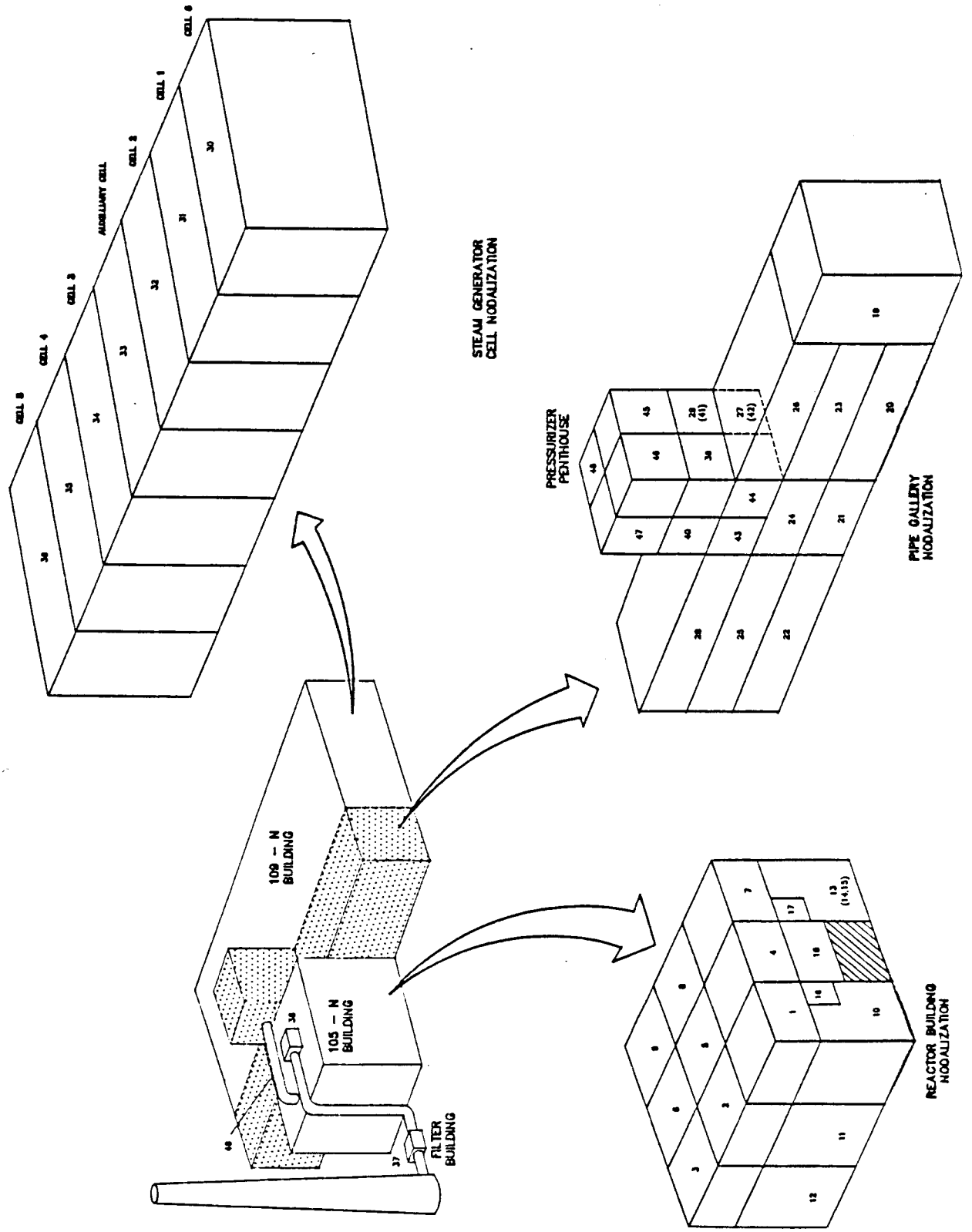
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	CONTAIN N Reactor Model Cell Locations
	Figure 15.6-287



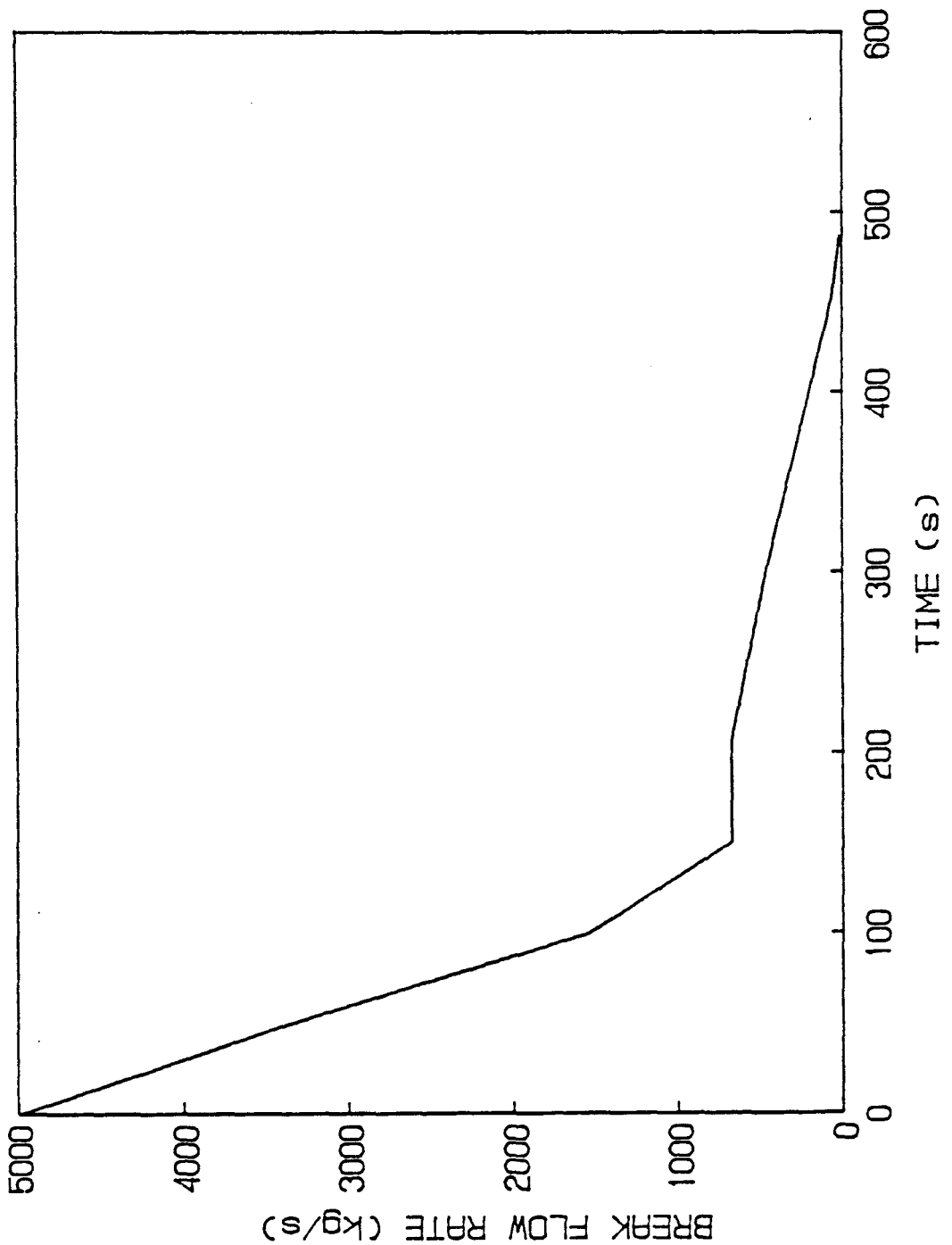
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	HCM Nodalization
	Figure 15.6-288



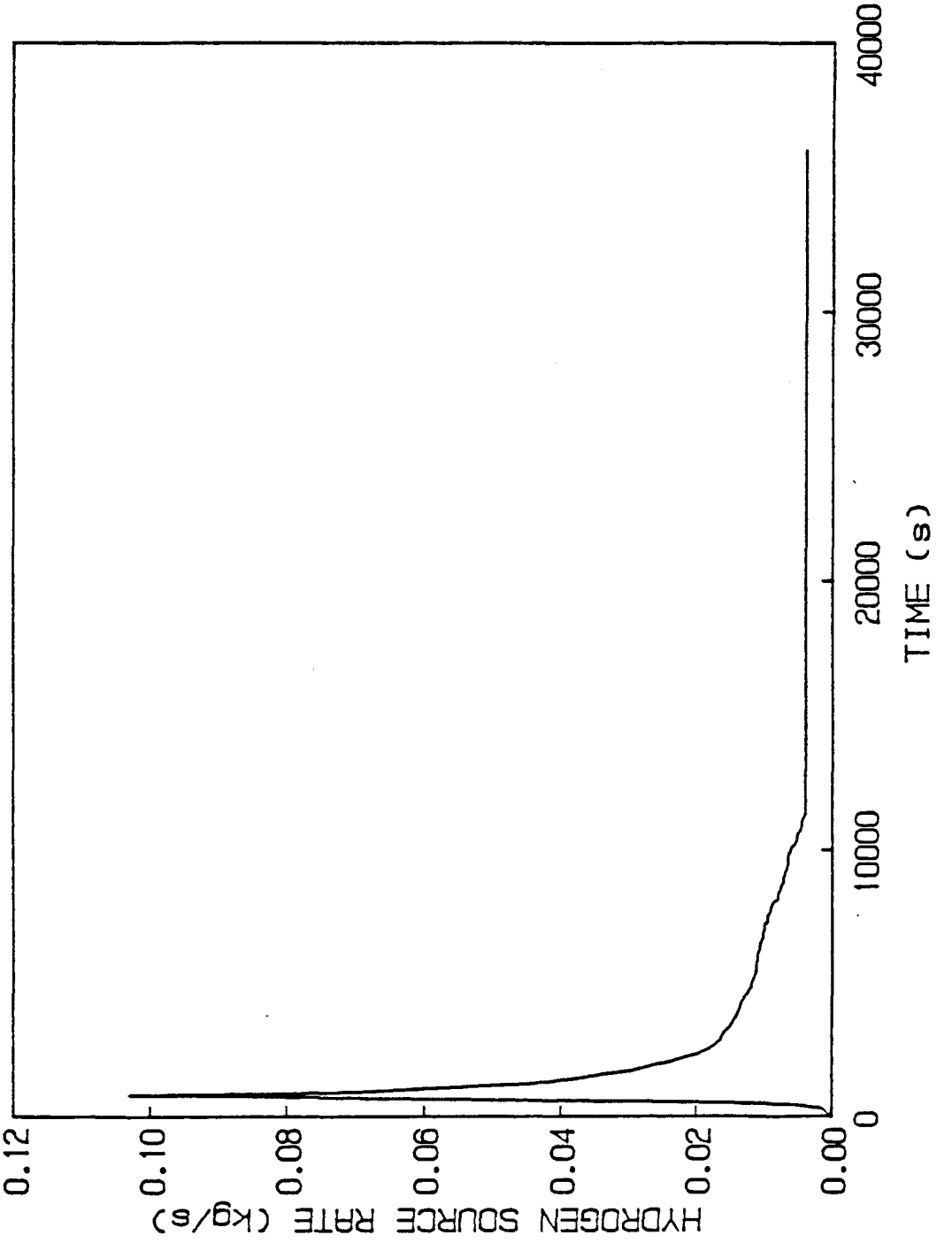
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	HECTR 49-Volume Model, Limiting Hydrogen Mitigation Analysis (Front Pipe Barrier Space Release)
	Figure 15.6-289



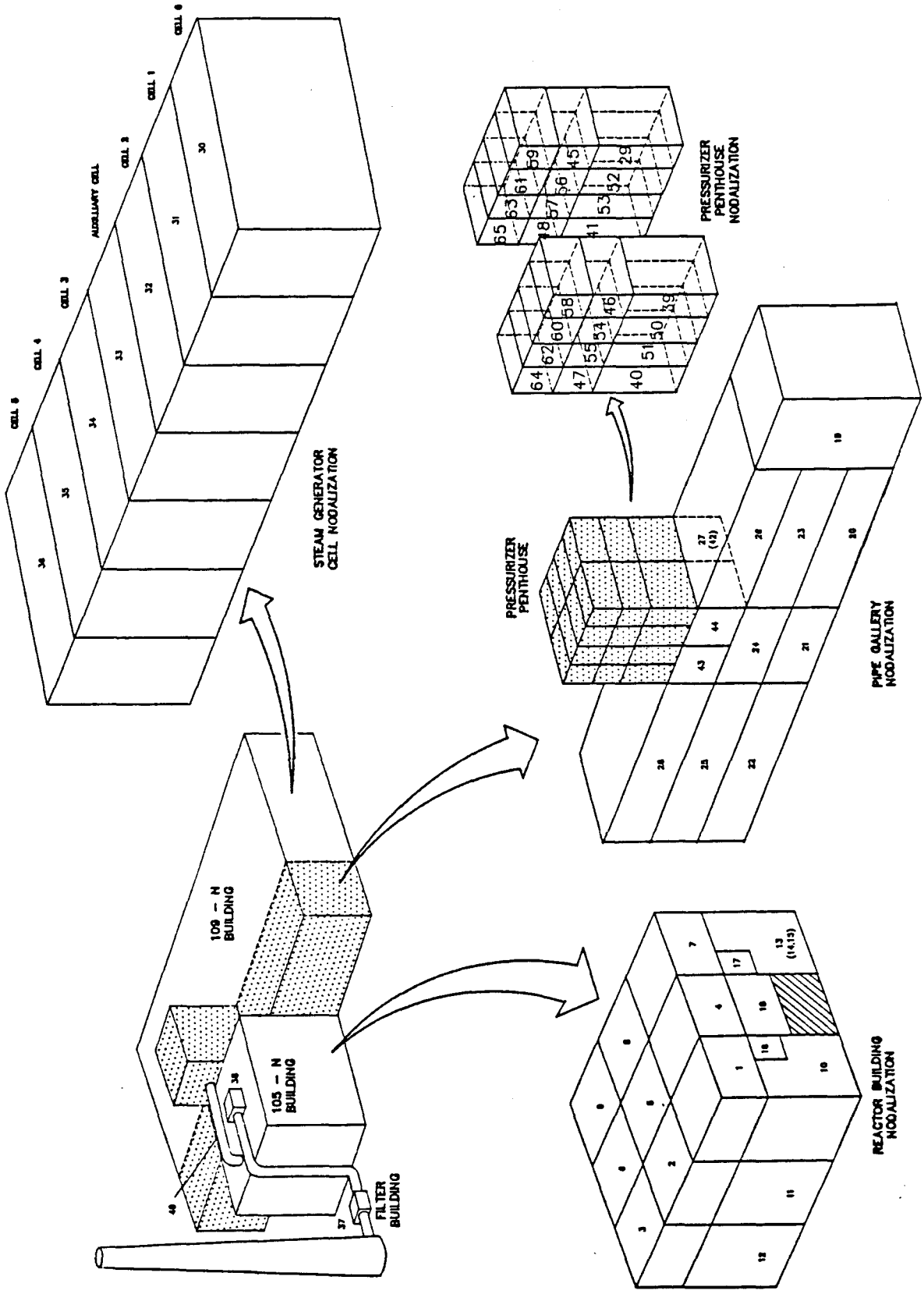
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Front-Riser-Break Flow Rate for Limiting Hydrogen Mitigation Analysis (Front Pipe Barrier Space Release) Figure 15.6-290



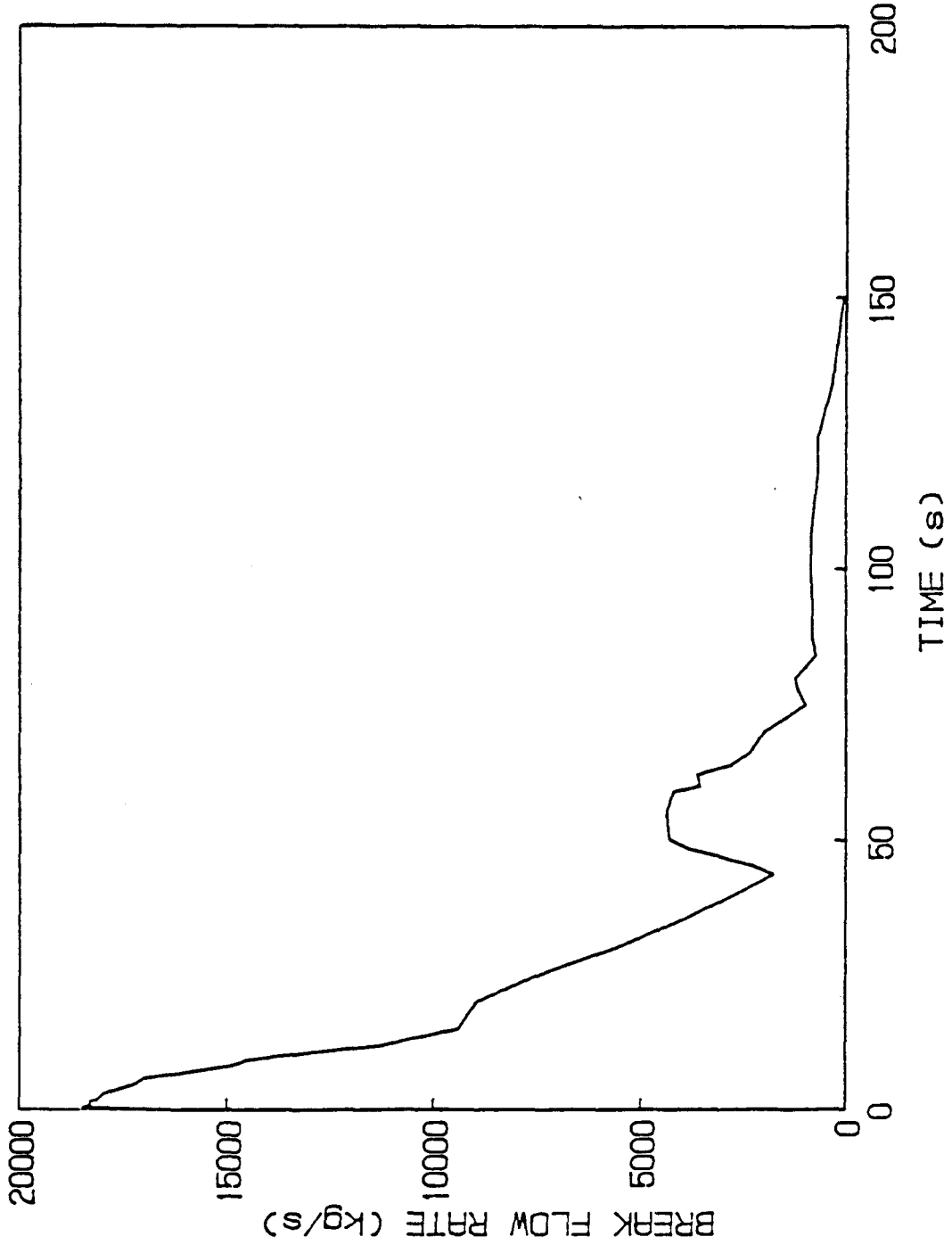
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Hydrogen-Mitigation System Source Term for Limiting Hydrogen Mitigation Analysis (Front Pipe Barrier Space Release, Pressurizer Penthouse Release)
	Figure 15.6-291



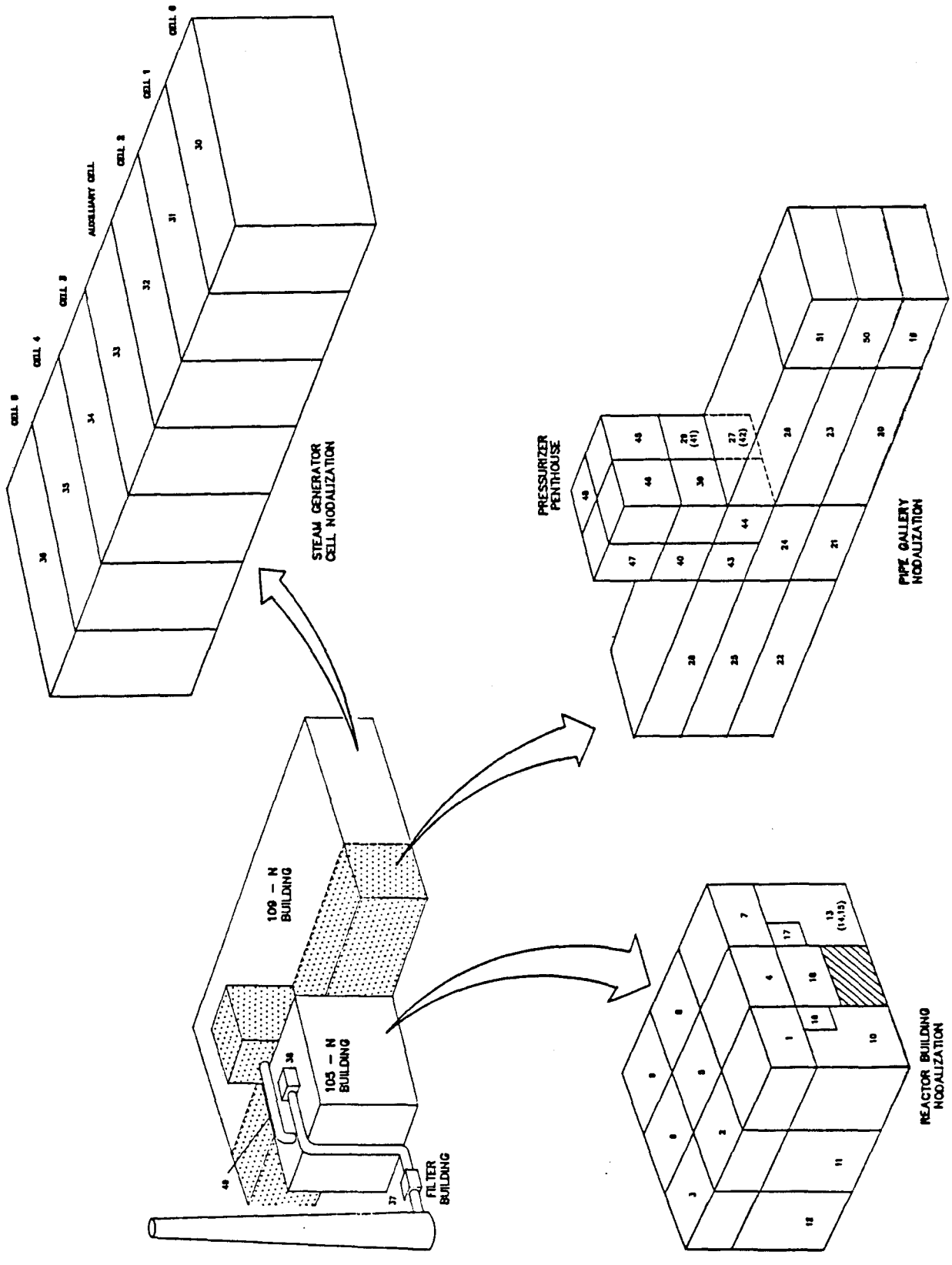
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	HECTR 65-Volume Model for Limiting Hydrogen Mitigation Analysis (Pressurizer Penthouse Release) Figure 15.6-292



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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Cold-Leg Manifold Break Flow Rate, Limiting Hydrogen Mitigation Analysis (Pressurizer Penthouse Release)
	Figure 15.6-293



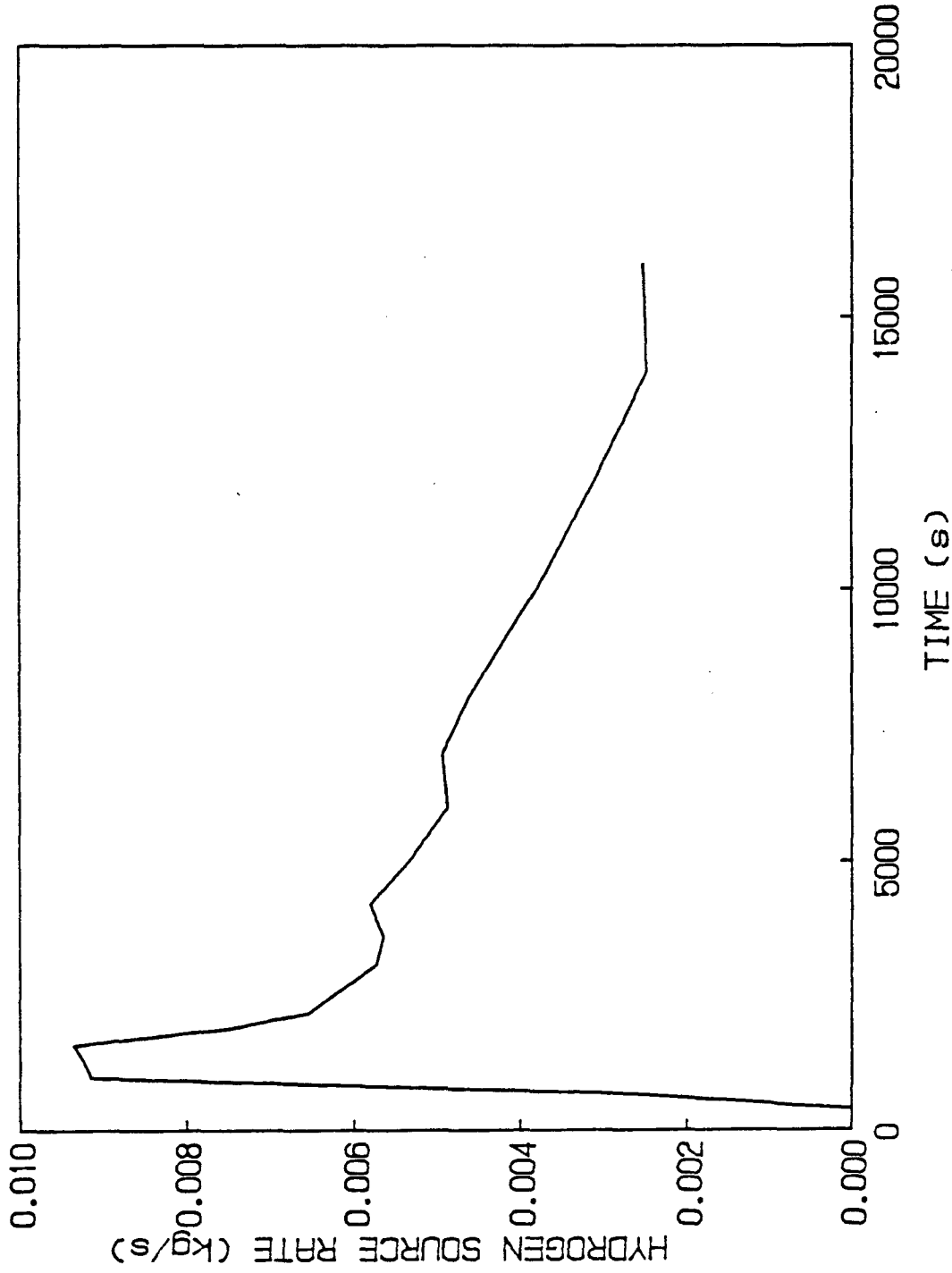
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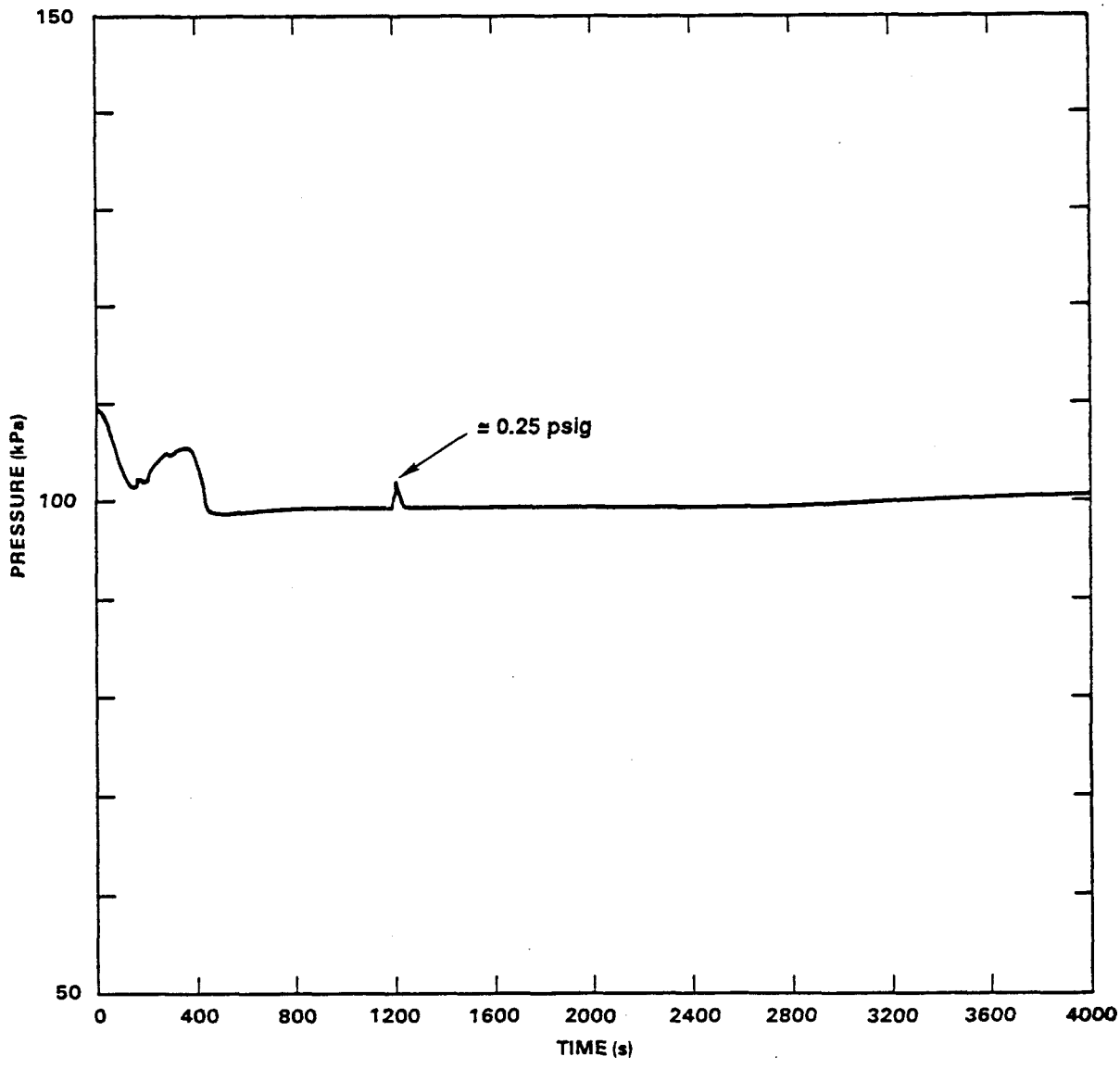
HECTR 51-Volume Model,
Limiting Hydrogen Mitigation Analysis
(Emergency Dump-Tank Vent Release)

Figure 15.6-294



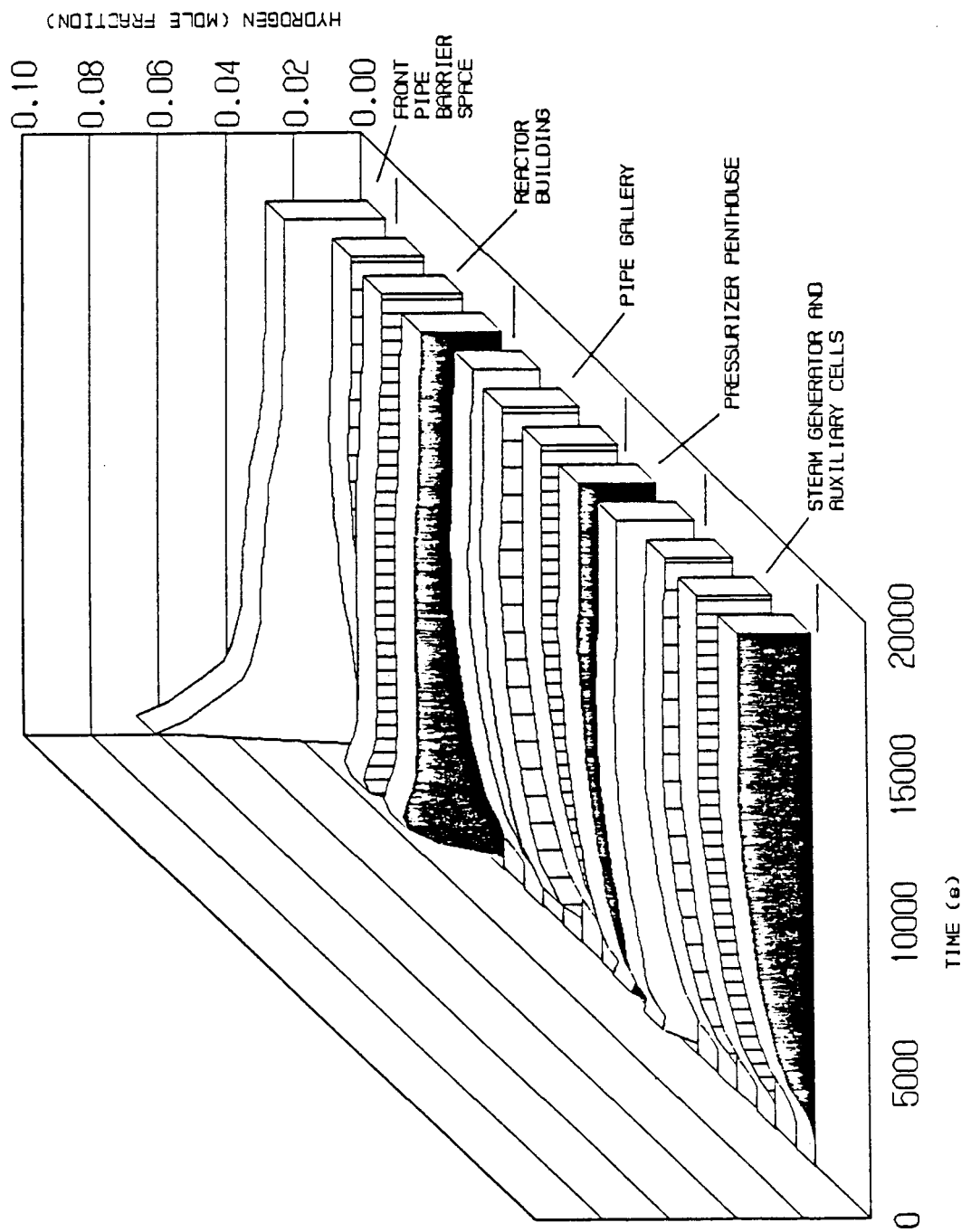
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Emergency Dump-Tank Vent Hydrogen Source, Limiting Hydrogen Mitigation Analysis
	Figure 15.6-295



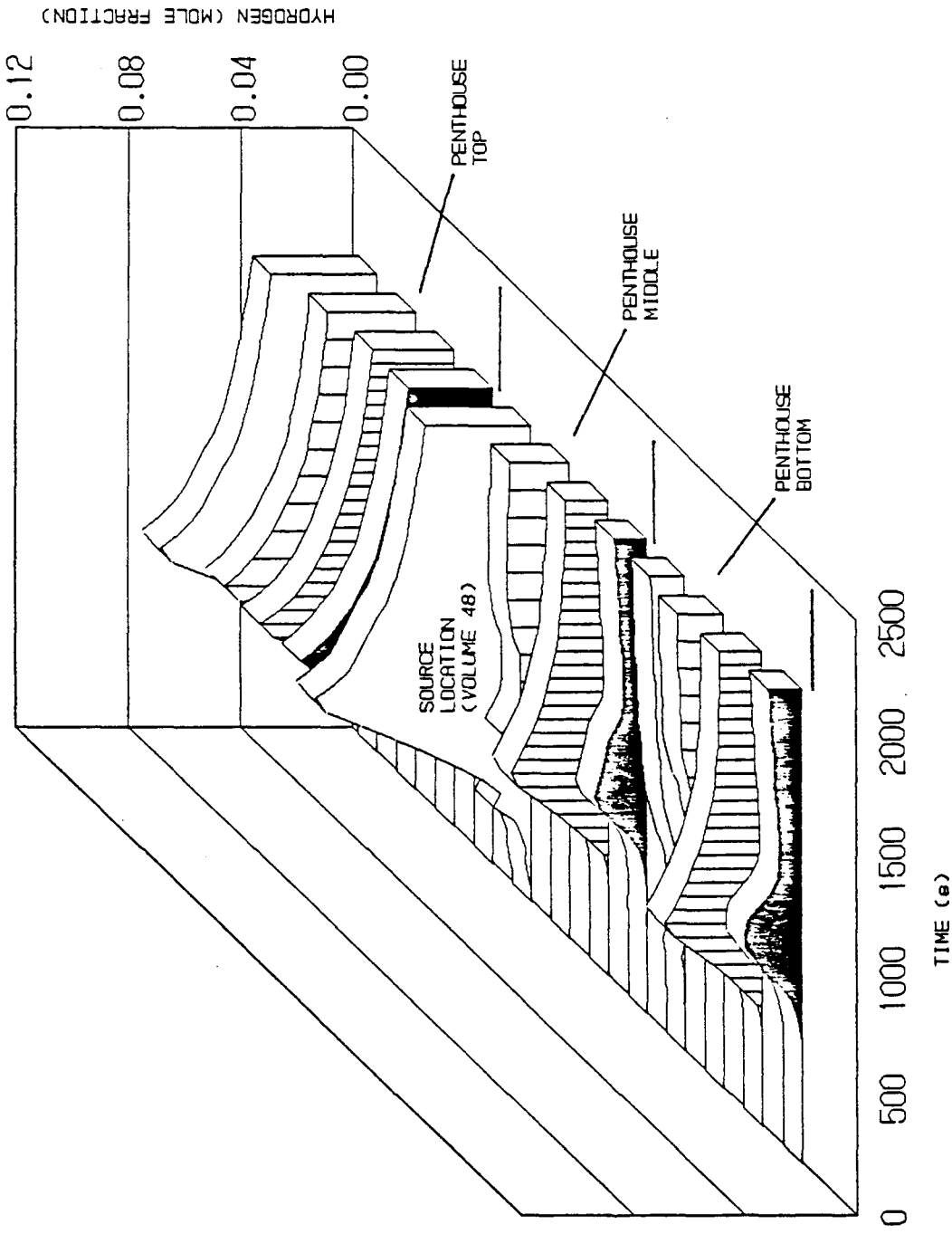
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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	HECTR Case S4 Pressure Predictions in Front-Pipe-Barrier Space. (Reference 59) Riser Break in Pipe-Barrier Space with Hydrogen Burn Figure 15.6-296



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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Hydrogen Distribution for Release in Front-Pipe-Barrier Space
	Figure 15.6-297

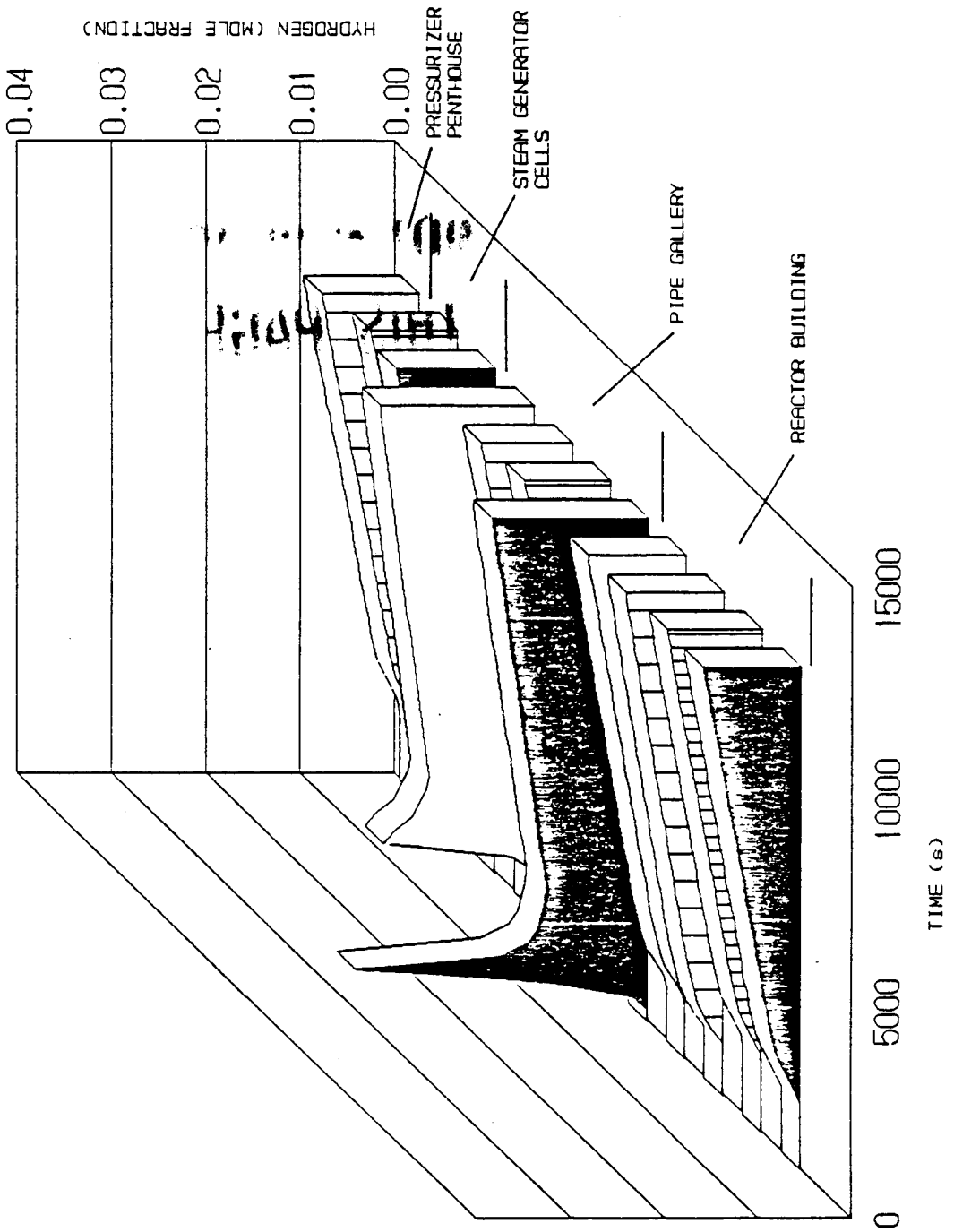


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Westinghouse Hanford Company
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Hydrogen Distribution
in the Pressurizer Penthouse

Figure 15.6-298



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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Hydrogen Distribution for Vent-Line Source Release at Bottom of Pipe Gallery Extension Figure 15.6-299

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APPENDIX 15.6A
THE RESPONSE AND BEHAVIOR OF N REACTOR FUEL ASSEMBLIES TO LOSS OF COOLANT TRANSIENTS

Introduction

The N Reactor fuel assembly consists of an inner and outer fuel element with the inner element nested and secured within the outer to form two concentric hollow right cylinders. From 16 to 21 separate fuel assemblies comprise the fuel column length of 34.8 feet, depending on the length of fuel assemblies involved. As discussed in Chapter 4, the design features slightly enriched metallic uranium fuel with 100 percent metallic bonding to all cladding components.

Nuclear safety criteria for N Reactor requires operational limits which will prevent fuel failures in the event of credible accidents. The maximum credible accident which maximizes the fuel temperature transients is a break of the cold-leg manifold crossover line. In practice, fuel assembly specific power levels are restricted to prevent reaching transient fuel failure temperatures prior to the introduction of emergency core cooling. Establishing these limits requires knowledge of fuel element failure temperatures. Assumed failure to provide direct cooling represents the hypothetical accident condition. For assumed failure to provide direct cooling following postulated pipe breaks, the fuel failure temperature must be accurately known to assess confinement system performance and the associated radiological releases.

The performance of N Reactor fuel to hypothesized loss-of-coolant accidents has been experimentally determined. The experimental results were obtained by subjecting irradiated N Reactor inner and outer fuel elements to calculated accident-case thermal transients. The tests were conducted in the absence of external pressures present in the fuel channels during blowdown. These results provide the bases for fuel performance during the hypothetical accident. The ex-reactor test results were conservatively corrected for the effects of external pressure applicable to credible accident cases involving an operable ECCS.

Summary

Maximum Fuel Thermal Transient Involving An Operative ECCS

An assumed reactor cold-leg crossover line break upstream of the CV-5 check valves imposes the maximum thermal transient on N Reactor fuel assemblies prior to reflood. Following an assumed break, the fuel experiences heat up from operating temperatures to a fairly uniform volumetric temperature. A maximum temperature of 1890°F occurs in the outer element of the maximum specific power fuel assembly in approximately 300 seconds. Fuel behavior for both inner and outer elements is summarized briefly as follows:

- Fuel heatup progresses with thermal expansion producing the only significant dimensional changes to a temperature of 1450°F.
- For fuel burn-ups consistent with the production of 6% Pu 240, good contact between the cladding and metallic uranium is assumed up to the fuel failure temperature. Fission gas pressure within the fuel begins to produce localized unbonding of intermediate and high burnup fuel which reaches 1450°F and continues to progress in extent as the temperature increases.
- For intermediate and high exposure fuel, radial gaps up to 0.005 inches maximum occur between the clad and core in areas where unbonding occurs and slightly increase the heat-up rate to the maximum fuel temperature of 1890°F experienced during reflood. Gap widths are limited by external pressures imposed on the fuel cladding from RCS system pressures during blowdown and the ECCS cooling mode.
- Physical cladding distortion does not impair proper ECCS coolant distribution in the fuel assembly subchannels.
- Reflood terminates the thermal transient and the approach to fuel failure conditions by reducing internal gas temperatures, and thus internal gas pressures, and restoring higher cladding strengths associated with the lower temperatures.
- Cladding oxidation during the blowdown and reflood stages is not a significant factor with respect to fuel cladding integrity.

- Physical restraint of the cladding by RCS system pressure during blowdown and reflood would result in a minimum fuel failure temperature of 1900°F. | 21

Hypothetical Accident Conditions

Failure to provide direct cooling of the fuel columns is incorporated into assessment of the hypothetical accident. The resulting thermal transient produces fuel failure only in those fuel elements operating at above average specific power conditions. The behavior of N Reactor fuel under such conditions is briefly summarized as follows: | 1

- Fuel heatup progresses from pre-accident conditions to 1450°F with thermal expansion producing the only significant dimensional changes.
- For intermediate and high exposures, localized clad unbonding appears at 1450° F and bond separation becomes complete at the fuel failure temperature. | 21
- Failure temperatures decrease linearly from the 2030°F uranium solidus temperature for unirradiated fuel to 1830°F at fuel element exposures of 4500 MWD/MTU. Above fuel element exposures of 4500 MWD/MTU, the failure temperatures decrease to a minimum of 1480°F at exposures of 6490 MWD/MTU, the maximum fuel element exposures realized under the 13 percent Pu-240 product operational mode.
- The total core fuel failure fraction is based on a uniformly applied fuel failure temperature of 1900°F.
- With outer-element and inner-element peak burnup corresponding to 6% Pu 240, fuel failure temperature limit of 1900°F is assumed. | 21

Discussion

Experimental Testing Program

Ex-reactor tests have been conducted on irradiated N Reactor fuel elements to establish their behavior under loss-of-coolant thermal transients (Ref. 1, 2, 3 and 6). The tests involved both inner and outer fuel elements with exposures from 2000 MWD/MTU to 7600 MWD/MTU. The tests were conducted in either steam or air environments at one atmosphere pressure. | 21

The fuel elements were inductively heated at a rate established by the calculated transient for maximum specific power conditions. The heating coil design was established to provide uniform axial and radial temperatures between 1450°F and the failure temperature. Tests with unirradiated fuel elements were conducted to confirm a slight preferential inductive coupling to the uranium simulating the decay heat generation within the fuel material. These tests also showed that the fuel element outer and inner clad surface temperatures could be maintained within 9°F of each other throughout the heating cycle. Axial temperature variations up to 72°F occurred during the early high power input phases of the heating cycle but were reduced to negligible values at temperatures above 1450°F by proper coil designs for inner and outer elements, respectively.

Based on heat distribution tests with unirradiated fuel, temperature monitoring was accomplished using a single thermocouple attached to the exterior surface of the inner cladding at the axial midpoint of the elements. Determination of the failure temperature was originally accomplished using gamma analysis of the gas stream corrected for transport times. Fuel element failure was characterized by a cloudy gaseous expulsion from the elements. Visual detection of the expulsion was found to provide for just as accurate definition of the failure temperature and was utilized as the indicator in all but the initial tests.

The tests were conducted within open-ended quartz cylinders, which provided for positioning and electrical insulation of the fuel elements from the coil. The quartz cylinders were sized to the inside diameters of the pressure tube and outer element inside diameter for outer and inner element tests, respectively.

Fuel Performance Under the Maximum Thermal Transients Involving Direct Cooling by the ECCS and Without ECCS Cooling 21

For exposures up to 4500 MWD/MTU, a minimum fuel element failure temperature of 1830°F was established by the atmospheric pressure tests. For exposures corresponding to 6% Pu-240, a fuel element failure temperature can be deduced from Reference 6. Correction of clad strains for the effect of external pressure establishes a minimum failure temperature of 1900°F as discussed in the next section. 21

The tests showed bond separations initiating at a fairly uniform temperature during the heat up of 1450°F. These separations represented the first indication of physical distortion of the cladding other than the uniform changes caused by thermal expansion. Bond separation is attributed to formation of macroscopic fission gas bubbles which proceeds rapidly above the β-γ uranium transformation temperature (1420°F). The gas achieves access to the clad-core bond as pores within the uranium become interconnected. Internal pressures exerted on the cladding increase as the release fraction and pressure increase with increasing temperature. Bond separations increase in proportion as a result and become complete shortly before the failure temperature is reached. The availability of greater quantities of fission gases with increasing fuel element exposure produces a slight negative exposure dependency to the failure temperature data below 4500 MWD/MTU.

For low fuel burn-ups, References 1 and 2 indicate that performance at these burnup levels, typical of current N Reactor operation during postulation high-temperature accident conditions, does not lead to cladding debonding with the character of "ballooning" as in light water reactors prior to fuel rupture. After fuel rupture, the metallic uranium is expelled from the cladding, relieving the internal pressure. 21

Based on the experimental data for fuel with burnups corresponding to 6% Pu-240 the cladding remains bonded for inner fuel elements up to the rupture temperature. For outer elements, some debonding occurs prior to rupture of the cladding, but at temperatures well above 1435°F. Reference 2 indicates

that isolated spots of separation were observed; however, complete fracture of the bond, typical of the higher burnup tests was not observed.

Reference 2 indicates that for the burnup level, corresponding to 6% Pu-240, bond separation is at isolated spots and does not propagate over the entire fuel length prior to rupture. The gap between the debonded cladding and the fuel is not defined in the references, except for the indication of dark spots. Since spots and not bubbles were called out, the gap was probably small; i.e., no substantial cladding "ballooning." No significant change in the coolant flow channels is indicated prior to fuel rupture, because of the minimal debonding noted for burnups for 6% Pu-240. The appearance of the black spots may be explained as cool spots on the cladding surface. The lower local temperature, being a result of the temperature drop across the local gap, would show up as a black spot on the heated elements. The key is that at the lower burnups the few isolated spots of cladding separation from the uranium that were observed do not represent the complete fracture of the bond observed for the high-burnup fuel.

Thus there are no data to support the need for a debonding or ballooning model at the burnup levels expected for the 6% Pu-240 production mode in N Reactor. Therefore, for the limiting loss of coolant and HMDBA, bond separation is not assumed prior to fuel rupture.

Bond separation is of concern for intermediate and high exposure fuel with respect to the integrity of ECCS flow paths and heat transfer from the fuel elements above bond separation temperatures. The accident condition fuel behavior evaluation program included tests on two outer fuel elements which had experienced an exposure level of 5940 MWD/MTU. These fuel elements exhibited identical failure temperatures of 1616°F. A third outer fuel element, irradiated to an exposure of 4620 MWD/MTU, was heated to 1650°F without incurring failure. The tests on high exposure fuel elements, terminated at temperatures just above 1600°F, provided an excellent basis for assessing the effects of external RCS system pressure on bond separation behavior during credible accidents involving an operable ECCS.

Table 15.6A-1 presents data on the high exposure outer element heating tests. All outer element transient heating test fuel failures were due to the occurrence of plastic instability of the cladding near its juncture with the fuel element end cap-to-cladding fusion welds. Due to high localized bending strains, failure occurred at stresses below cladding burst strength and, accordingly, at internal pressures below that required for burst type failure.

To assess the effect of external RCS system pressure on strains observed in the test program at atmospheric pressure, the clad was assumed to be at burst failure conditions. The burst strength of N Reactor Zircaloy-2 pressure tubes as a function of temperature is presented in Figure 5.3-9 and was used to calculate the failure stress of Zircaloy-2 at 1616 F. This failure stress was then assumed to exist in the cladding of the elements listed in Table 15.6A-1 at the maximum test temperatures. The cladding was treated like a hollow cylinder since bond separation had occurred, and an internal pressure at burst failure conditions was calculated for the outer clad, the weakest component, after correcting the post-test diameters for thermal expansion. The calculated internal pressure was 43 psi and is considered quite conservative since failure occurred due to localized bending strains rather than burst conditions on two elements and failure did not occur at all on the third test element analyzed.

Clad strains were then corrected for RCS system pressure, to simulate those events which ECCS is relied upon, which reaches a minimum value of 300 psig at reflood. The post test fuel element dimensions were first corrected for thermal expansion. The number of moles of gas present was then calculated assuming an ideal gas in the absence of VanderWaals Constants for the fission gas species. A new void volume was calculated, assuming the only internal voids were between the uranium outside diameter and the outer clad inside diameter and using an internal pressure of 300 psig. The resultant calculated uniform radial gap was 0.007 inches. 21

Since bond separation of the inner cladding had been ignored to provide for a simplified analysis, the calculated 0.007 inch gap was arbitrarily assigned a value of 0.005 inches between the uranium and both the inner and outer

cladding. Reanalysis of the fuel heat up transient for the limiting LOCA analysis which relies on ECCS flow between 1450°F and reflood indicated a maximum fuel temperature of 1890°F would occur in a hot channel maximum specific power fuel assembly at reflood. The effect of external RCS system pressure also limits clad strains to a value that has no significant effect on the coolant subchannel geometry of the fuel assemblies.

Significant clad strains developed between the onset of bond separation (1450°F) and failure in the tests conducted under atmospheric pressure. The minimum system pressure experienced in the RCS, for events which rely on ECCS flow prior to reflood and reversal of the fuel thermal transient, is 275 psig which serves to limit clad strains and produce a higher failure temperature than that observed in the tests conducted at atmospheric conditions.

The volatile fission products were treated as an ideal gas to find the volume of released fission products for those tests described in the previous section which were terminated at 1616°F to 1652°F. Hilliard, et al, have examined the fractional release temperature dependency of select fission product gases from metallic uranium (Ref. 4). While the release fractions for various species vary at a given temperature, the temperature dependency of the various species below the melting point of uranium is essentially identical. The data of Hilliard, et al, (Ref. 4) was used to proportionally correct the volume of fission gas released between the conditions examined at 1616°F to 1650°F in the preceding section and 1900°F. This correction produced a released gas volume at 1900°F that was 1.5 times as great as the release which had occurred at 1616°F to 1650°F. With the fission gas quantity available for internal pressurization at 1900°F, the basic methods described in the preceding section were utilized to calculate the effects of external pressure, for events which rely upon ECCS flow, on limiting the cladding strains expected at 1900°F. All clad strains were assumed to occur in the outer surface cladding for conservatism, i.e., available expansion volume was limited to between the uranium core and the outer cladding. Resultant clad strain, using the post irradiation clad diameter as the basis, were less than 5 percent at the attainment of 1900°F temperatures.

Analysis of conditions up to 1900°F suggests no reason for fuel element failure to occur in the presence of RCS system pressure and this temperature is presented as the minimum failure temperature. | 21

Cladding Effects Due to Reflood Quenching

An experimental testing program was also conducted to examine the effects of the rapid quenching that would occur during reflood on fuel cladding integrity (Ref. 5). N Reactor fuel elements (two inner and two outer elements), irradiated to approximately 1900 MWD/MTU, were heated as in the previously described tests to a temperature of approximately 1800°F. The fuel was then quenched with 142°F to 146°F water simulating ECCS reflood conditions and under maximum system pressure of 8 psig. The cladding temperature data displayed an immediate decrease to essentially the quench water supply temperature. The quench rate experienced by the test fuels was much more severe than those calculated for reflood since no percolation occurred in these tests and only minimal tempering of the initial ECCS coolant occurred prior to its contact with a simulated high specific power fuel element.

The post test fuel cladding of the outer fuel elements exhibited a "bumpy" appearance in contrast to the rather uniformly strained condition displayed by the inner elements and nonquenched fuel elements involved in the previously described tests. The bumpy appearance is most likely due to distortions incurred upon rapid cooling. Bond separation behavior of these test fuels paralleled that of previously described testing.

The quenching did not produce failure of the cladding based on the absence of fission products at a sampler downstream of the test elements. The test assemblies were examined visually and metallographically following testing, and no imminent clad failure conditions were observed.

The cladding oxidation condition did not appear significantly different than that displayed by normal irradiated fuel. Metallographic examination of cladding areas incurring the highest strains revealed no cracking of any type and no unusual corrosion behavior.

The Failure Temperature of N Reactor Fuel Elements During the Fuel Ejection Accident

Failure temperature determination tests were conducted on irradiated N Reactor fuel elements (Ref. 1, 2, 3 and 6) including both inner and outer elements with exposures ranging from approximately 2000 to 7600 MWD/MTU as previously described. Table 15.6A-2 and Figure 15.6A-1 display the results of the fuel failure determination tests conducted at atmospheric pressure conditions. Failure of outer test elements occurred due to high localized clad strains and the occurrence of plastic instability at the juncture of the clad and the end cap in the area of the end closure fusion weld. Failure of inner test elements occurred by burst failure of the outer cladding.

The exposure dependency evident in Figure 15.6A-1 for exposures to 4500 MWD/MTU is believed to result from the exposure dependency of fission product gas quantities produced within the fuel. For a given transient temperature and corresponding fractional fission gas release value, the "free" fission gas volume is greater than for higher exposure fuel. Internal pressures sufficient to produce failure thus occur at slightly lower temperatures for high exposure fuel.

The exposure dependency of the fuel failure temperature is indicated to change rapidly above 4500 MWD/MTU, as indicated by Figure 15.6A-1. Uranium and dilute uranium alloys experience extreme loss of room temperature ductility at high exposures. This embrittlement is believed to reduce the ability of the uranium to withstand impact forces associated with fuel discharges. The occurrence of uranium cracking during discharge of the high exposure test fuel would lead to a failure temperature dependent not only on the gaseous fission product inventory but on the extent of mechanical damage occurring to the uranium during discharge.

The drastic change in the fuel failure temperature exposure dependency is discussed for two reasons. First, it may well be very pessimistic if applied to the in-reactor fuel which will not have experienced the impact loads associated with fuel discharge activities and which all test fuel experienced.

Secondly, N Reactor is fueled with the base 0.95 weight percent U-235 enriched Mark IV fuel design and the "spike" 1.25 weight percent U-235 enriched Mark IA design. Failure temperature determination tests on the Mark IV design were conducted only on fuel above 4360 MWD/MTU. Comparison of the Mark I and Mark IV data on Figure 15.6A-1 confirms that the two fuel designs behave very similarly and that the lower bound curve adequately represents the minimum failure temperature - exposure relationship for both designs.

Table 15.6A-3 presents maximum fuel assembly and fuel element exposure levels applicable to the 13 percent Pu-240 product operational mode. Maximum theoretical exposure levels for Mark IV and Mark IA outer fuel elements exceed the 4500 MWD/MTU knee in the fuel failure temperature - exposure curve of Figure 15.6A-1. Failure temperatures for the fuel ejection accident are derived directly from Figure 15.6A-1 assuming appropriate fuel exposures. 21

The clad separation model for intermediate and high exposure fuel employed results in increased heatup rates above 1450°F where clad separation initiates and particularly above 1900°F where clad separation becomes essentially complete. Clad separation was observed to become complete at the failure temperature. In addition, the radial gaps assumed are based on tests involving exposure levels of 5940 MWD/MTU where the gaps were visibly much greater than for test fuels below this exposure level. Thus, the clad separation model is conservative.

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TABLE 15.6A-3

MAXIMUM FUEL ASSEMBLY AND FUEL ELEMENT
EXPOSURE LEVELS FOR THE 13% Pu-240 OPERATIONAL MODE

Assembly Exposure - MWD/MTU

	MKIV	MKIA
Exposure at 13% Pu-240 Product	3080	3553
120% of goal at 13% Product	3696	4268
(1.4 Axial Peaking)* (120% goal @ 13% Pu-240)	5170	5753

21

Fuel Element Exposure

Exposure at 13% Pu-240 Product		
Inner Element	2596	2585
Outer Element	3278	4015
120% of exposure at 13% Product		
Inner element	3135	3135
Outer Element	3960	4840
(1.4 Axial Peaking)* (120% exposure @ 13% Pu-240)		
Inner Element	4367	4202
Outer Element	5555	6490

21

*If the average exposure for MKIA in a spike charge is used, the appropriate peaking factor over the Mark IA section of the fuel charge is 1.18.

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APPENDIX 15.6B
ASSUMPTIONS AND CALCULATIONAL TECHNIQUES USED TO EVALUATE RADIOLOGICAL CONSEQUENCES

Introduction

This appendix describes the assumptions, calculation models and analytical computer codes used to evaluate the radiological impact of the accidents that are assumed to release fission products with the exception of the maximum hypothetical accident described in NUSAR Section 15.6.6.

Techniques used to calculate the radiological consequences of a postulated accident that releases activation products are described separately.

Calculational Techniques

In the accidents that are assumed to release fission products, only a portion of the fuel elements in the affected pressure tube or tubes would reach the clad failure temperature. See Reference 1, for analysis and discussion of the confinement response and dose calculations of the postulated accidents. The fuel heatup transients were calculated using the MELT code described in Appendix 15.6C. The fuel cladding failure temperature of 1900°F is based upon experiments described in Appendix 15.6A.

Only that fuel exceeding 1900°F would fail and release fission products. The fission product burden of the fuel that reached or exceeded 1900°F was calculated using the RIBD code and measurements of axial and radial specific power distribution within the reactor.

Of the fission product burden available to be released from the failed fuel, 100 percent of the noble gases and 50 percent of the iodine are assumed to be released. Fifty percent of the iodine released from the failed fuel would be plated out prior to release to the confiner atmosphere. Of the iodine released to the confiner atmosphere, 4 percent would be converted to organic iodine, and 5 percent would be in particulate form. All of the noble gases would be released to the confiner atmosphere. These fission product release fractions are based upon guidelines in Regulatory Guide 1.4.

The CORRAL code was used to calculate the time dependent transport and distribution of the fission products within the confiner. The code calculates: (1) the plate-out on the confiner surfaces based upon the path the fission products would take prior to filtered release, (2) the removal efficiencies of the filter sprays based upon spray characteristics and the path followed by the fission products in reaching the filters, (3) the building leakage rate based upon the confiner pressure transient, (4) the filter efficiencies and (5) the release rate out the ventilation stack.

RIBD is used to calculate the initial curies in the failed portion of fuel. A code called RELEASE determines the fission product release and doses from the RIBD and CORRAL results by integrating the releases over finite time intervals and multiplying by appropriate meteorological dilution factors and dose conversion factors.

Dose calculations using Regulatory Guide 1.4 assumptions do not include radioactive decay following release from the confiner. Realistic dose calculations include radioactive decay based upon the time required for the fission products to reach the site boundary or the low population zone.

Other assumptions used in the calculations are listed in Table 15.6B-1.

The calculations of atmospheric dispersion of the radionuclides released from the confiner were based upon:

1. the distances to the site boundary and low population zone (LPZ) in both realistic and regulatory guide cases of 5.5 miles (8900 meters) and 20 miles (32,000 meters), respectively;
2. the X/Q values selected from values published in Regulatory Guide 1.4 for cases based upon Regulatory Guide assumptions - see Table 15.6B-2;
3. the X/Q values that would not be exceeded for 95 percent (median) of the time based upon 12 months of onsite and 15 years of nearby meteorological data for cases based upon realistic assumptions - see Table 15.6B-2.

Wind speed becomes a significant variable for large distances to the site boundary. The 1 m/sec used in Regulatory Guide 1.4 would be insufficient to move the radioactive cloud to the 5.5-mile site boundary during the first 2 hour after fission product release from the fuel following any of the postulated accidents. In calculating a modified regulatory guide dose at the site boundary, a wind speed was calculated that would maximize the dose rate for the first 2 hours after fission product release from the fuel. This calculation considered the increased dose from the more rapid arrival of the radioactive cloud and the decreased dose from lower X/Q's which are inversely proportional to wind speed. This modified regulatory guide dose is the highest dose that can result at the site boundary within the first 2 hours after fission product release from the fuel.

Whole body dose calculations assumed that the receptor was immersed in a semi-infinite plume of uniform concentration. The equations used were:

$$D_\beta = 0.23 \cdot X/Q \cdot \Sigma_i Q_i \cdot \bar{E}_i$$

$$D_\gamma = 0.25 \cdot X/Q \cdot \Sigma_i Q_i \quad \bar{E}_i$$

where:

D_β is the whole body beta dose rate (Rem/sec)
D_γ is the whole body gamma dose rate (Rem/sec)
X/Q is the atmospheric dispersion parameter (sec/m^3)
Q_i is the rate of isotope i released (curie/sec)
$\bar{E}_i$ is the decay energy from isotope i (Mev/disintegration).

The thyroid inhalation doses were calculated using the following expression during each period of interest:

$$D_t = X/Q \cdot B \cdot \Sigma_i Q_i \cdot DCF_i$$

where:

D_t is the thyroid inhalation dose rate (Rem/sec)
X/Q is the atmospheric dispersion parameter (sec/m^3)
B is the breathing rate (m^3/sec)
Q_i is the rate of iodine isotope i released (curie/sec)
DCF_i is the dose conversion factor for iodine isotope i (Rem/curie inhaled).

Specific physical data for the radionuclides that would be released, dose conversion factors and breathing rates used in the calculations are listed in Table 15.6B-3.

Computer Codes

1. CORRAL, (Ref. 3), solves exactly a set of coupled, first-order differential equations which describe the transport of fission products within a multi-compartment containment. Fission product releases to the atmosphere are calculated for use in offsite dose calculations.

   Natural deposition and spray washout of radionuclides in CORRAL mass transfer models and empirical data were verified by extensive experimentation during the containment system experiment program.

   The fission product source terms for a postulated accident are the materials that escape the reactor confinement system through building leakage or the confinement system. To describe the airborne source term, as an accident progresses, requires a mathematical model of the internal transport, deposition and leakage of airborne fission products. Simple compartment models do not adequately describe the fission product behavior in multi-compartment systems whose thermodynamic conditions, flow rates, and fission product cleanup mechanisms are complex functions of time. In realizing this complexity, the Rasmussen Reactor Safety Study decided to develop computational models that would handle the many proposed detailed accidents in two containment systems (boiling water and pressurized water reactors) and provide source terms for the

release of fission products through building leakage and filters. The resulting computational models or computer codes were BWR and PWR versions of code CORRAL.

CORRAL is a multi-compartment code patterned after requirements of the Rasmussen Study; because these needs were variable, a high degree of flexibility was provided. This flexibility includes variable fission product source terms, leakage calculations from any compartment as a function of time, inter- and intra-compartment filtering, natural deposition in all compartments, water, spray washing and filtration prior to stack exit. General inputs required are the time-dependent compartment thermodynamics and flow rates, spray characteristics, compartment parameters and the times of the various events. General time-dependent output is accumulative fractions of the radionuclides released to the atmosphere via the filters or direct building leakage.

CORRAL calculations chronologically proceed through an accident sequence at time intervals specified in the input data. During this time interval, CORRAL solves exactly a set of coupled, first-order differential equations for each inert gas, I_2 and particulate release, during which all thermo-dynamic parameters, washout rates, leak rates and flow rates remain fixed. The solution provides the deposition and transport of the radionuclides within the confiner and leakage of the radionuclides from confiner. During the next time interval, if conditions change or if new releases occur, the data are all automatically incorporated into a new set of differential equations with the final radionuclide concentrations from the previous interval as the initial conditions for the new interval. Radionuclides leaked during the interval are added to those leaked during previous intervals. CORRAL prints the status of radionuclides contained, released and deposited for each time interval.

2. RIBD (Ref. 4 and 5) calculates isotopic concentrations resulting from two fission sources with down-chain decay by beta emission and isomeric transfers and inter-chain coupling resulting from neutron-gamma reaction.

Decay heat generation is calculated for gamma and beta decay of the fission products. See Table 15.6B-4 for the half-lives of iodines and noble gases as provided by the RIBD code.

A comparison between RIBD calculations and fission product decay heating experiments has shown excellent agreement (Ref. 6).

In general, the code is a grid processor which calculates radionuclide concentrations resulting from two fission sources (for example, U-235 and Pu-239) with normal down-chain decay by beta emission and isomeric transfers and inter-chain coupling resulting from n-gamma reactions. The calculations follow the irradiation history through an unlimited number of step changes of unrestricted duration and variability including shutdown periods, restarts at different power levels and/or any other level changes.

RIBD incorporates a library containing pertinent nuclear data for 818 thermal fission product nuclides, including both ground states and isomeric states where they are known to exist whether formed directly from fissions of U-235 and/or Pu-239 or, subsequent to fission, as a result of neutron capture and/or radioactive decay of precursors.

RIBD organizes the various members described by the library into a grid with the various linkages established from chain branching data, yield data and neutron capture cross-sections with their branching ratios. Radioactive decay includes not only the simple member-to-member cascade but also the more complex forms where branching may partially or completely skip one or two intervening members. Ground states are allowed to have up to seven source paths and isomeric states, up to five source paths. Concentrations of the various fission products represented by the boxes of the grid are then computed as a function of time in accordance with the exposure history supplied as input.

The output report from the program includes six optional combinations of activity levels (expressed in curies), concentrations (expressed in

gram-atoms), heat generation rates (both beta and gamma) and integrated energy releases between time points for any set of points (up to 90) following shutdown that the user desires. Complete information is presented for each nuclide arranged element-by-element with summary sheets of element subtotals and grand total. A class summary is also written for noble gases, halogens, volatile solids (Se, Te and Cs) and all remaining fission products. Heat generation from decay of U-239 and Np-239 (if applicable) is also detailed and included in the totals, as is the fission energy liberated during the scram transient.

The fission product library presently in use with RIBD has been scanned and updated five times, the latest revisions having been made in 1975. The fission yield data were obtained from the work of Yu A. Zysin; the data library of fission products yielded and their release modes was generated with the latest data available (Ref. 7).

Comparison between RIBD and experimental data show that decay heat calculations agree within 1 percent.

3. RELEASE calculates the release of fission products to the environs from initial fission product inventories from RIBD and release fractions from the CORRAL code. An increment of fission product release is determined by first accounting for time decay before each isotope is retained by the confinement. Each isotope is then identified and proportioned by group depending on whether it is a noble gas or methyl iodine, elemental iodine or particulate. The appropriate amount of iodine entering the confinement atmosphere and forming elemental iodine, methyl iodine and particulates is consistent with specifications of Regulatory Guide 1.4. The release of each fission product is then calculated from the incremental change in CORRAL output between time steps for filtered release and building leakage. An increment of dose increase is then determined by multiplying the curies released from each isotope by the appropriate dose conversion factors and meteorological diffusion factors.

The code will also account for radioactive decay in transit to the dose point which is used for the realistic cases only.

Techniques Used to Evaluate Effect of Activation Product Release

The dose in the control room, at the site boundary, and at the low population zone boundary due to a radioactive dust blown from the reactor following a severe pressure tube break was calculated using the same methods as used for evaluating fission product releases. The dust was determined to contain the following radioactive isotopes: Sm-153, Sm-155, Eu-154, Eu-155, Eu-156, Gd-159 and traces of Co-60, Cr-51 and Fe-59 in Table 15.6B-5. Decay and transmutation of the specific isotopes was included in the calculation. The release fraction of the dust from the confiner was determined by the CORRAL code over a 30-day period. Dispersion and migration of the dust to the boundaries was treated according to Regulatory Guide guidelines, as applicable to N Reactor. Dose commitment to the internal body was based on dose deposition factors developed by BNW for the particular isotopes (Ref. 8).

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7. The design staff of Vitro Engineering for performing preliminary Title I design, detailed Title II design and Title III inspection on major modifications and projects and special system and component design analyses.
8. The facilities and staff of Boeing Computer Services for assistance in computer programming and analyses and for providing computer processing time.
9. Facilities and staff of J. A. Jones Construction Company for major construction and maintenance.
10. The physics and thermal hydraulics staff of BNW for assistance in analyses associated with N Reactor core and fuel storage facilities.

The availability and use of such nuclear expertise within the confines of a single project is unique in the nuclear industry.

1.1.7 Safety Analysis Report Amendments

To date, there have been twenty-one amendments added to the NUSAR. The contents of these amendments are as follows: 21

Amendment 1 (October 1978)

- The single tube flow blockage accident was reanalyzed using an improved clad separation model.
- Some reference pages were revised as a result of the review and clearance of the references. 4
- Section 9.5 was revised to incorporate a description of the lift station diesel-engine-driven pump and associated oil, cooling, and air systems.
- Table 6.2-3 was revised to include compartment pressures for the maximum pipe break within the compartment.

Amendment 2 (March 1979)

- Proper identification was made of safety, safety/relief, and relief valves which were previously referred to only as relief valves.

- Table 1.8-1 was enlarged to include the definition of more acronyms commonly used in the NUSAR.

Amendment 3 (July 1979)

- Necessary changes were made to reflect the replacement of samarium oxide balls with boron carbide balls in the ball safety system.

Amendment 4 (April 1980)

- Additional physics and thermal-hydraulics data were produced for inclusion in Chapter 4 to support four steam generator cell operation.

- The cold-leg crossover break LOCA was reanalyzed for four steam generator cell operation.

Amendment 4A (January 1981)

- Necessary changes were made to reflect updates for the confinement system.

- This is the fifth amendment of the NUSAR.

Amendment 5 (see Amendment 4A)

- Amendment 5 was skipped to avoid possible confusion with Amendment 4A which is the fifth amendment of the NUSAR. Amendment 4A is considered the de facto Amendment 5.

Missing Page(s)
from
Original Document

- Section 2.4, River Level Conditions, was revised per ECN 461037. Figure 2.4-12 has been deleted because low lift delivery tests were conducted in November 1987 that measured the river level at 100-N to be 378 feet 3 1/4 inches at a flow of 36,000 cfs. This information has made the extrapolated information in the figure obsolete. References to the figure have been deleted. Information has been added to show that 36,000 cfs was measured at N-Reactor by PT 614 to be 378 feet 3 1/4 inches elevation mean sea level.

- Sections 3.9, 6.2, 9.2, and 9.3 were revised per ECN U-000-42 which referenced DC 87230, Mechanical Isolation of the Fog Spray Accumulator. The Fog Spray Accumulator, SA-1, medium pressure system, was isolated by this design change.

- Sections 4.4 and 9.5 were revised per ECN U-000-46, which referenced DC 86150 REV1, Loose Parts Monitor System. A new section 9.5.11 was added that describes the new system and a reference to this section added by this design change.

- Sections 5.2, 11.5, Figure 5.4-11, and Table 3.9-4 were revised per ECN U-000-44 which references DC 87024, Divert Spill Cooler Raw Water Return to 10" RWR-1. A diversion of water from 1301-N spill cooler and reroute to RWR was made because it is essential to minimize the flow of any non-radioactive water to the crib so that its life can be maximized. This was identified in Project H-682 and is documented in the design change.

- Sections 4.6 and 6.7 were revised per EDT 111051 which referenced DC 87311, "Safety Circuit Improvement." This design change ensures that no common-mode failure of master trip breakers can occur which could result in the loss of both the ball system and the control rod system.

- Section 5.3 was revised per ECN 111301 to reflect the significant work on N-Reactor pressure tube integrity done in recent years. This update incorporates this work into NUSAR and the Technical Specifications (WHC-CM-5-26).

- Section 6.3 was revised per ECN U-000-41 which referenced DC 83016, "High Lift Pump Discharge Piping Upgrade." This design change replaced accommodation for piping in ECCS with fracture toughness properties.

- Section 7.4 was revised per ECN U-000-45 which referenced DC 81200, "Automated Rod Indication System." This design change replaced the description of the HCR Position Indication and Data Logging with the new Automated HCR.

- Several drawings were revised to reflect the appropriate impact level by ECN 202258. These drawings affected and updated 33 NUSAR figures. They include these NUSAR figure numbers: 3.2-1, 3.3-1, 3.5-2, 3.9-11, 3.9-17, 4.2-5, 5.5-1, 6.3-1, 6.3-2, 6.3-8, 6.3-9, 6.3-10, 6.3-11, 7.2-1, 7.2-6, 7.2-8, 7.2-9, 7.2-10, 7.2-11, 7.2-13, 7.2-16, 7.2-17, 7.2-18, 7.2-19, 7.2-20, 7.2-21, 7.4-3, 7.7-2, 7.7-3, 7.7-12, 7.7-18B, 8.2-0, 8.3-5.

- Amendment 19 was edited by a technical editor as part of issuance of WHC-EP-0240. As a result of this edit, 88 pages have been changed mainly in grammar and punctuation to conform to Westinghouse style standards. These changes are included as part of Amendment 20. The changed sections include: 1.1, 4-x, 4.3, 6.2, 7.4, 15-ix, 15.4, and 17.2 and they have not been indicated with change bars.

Amendment 21 (September 1989)

- Section 2.4, Hydrologic Engineering, was revised to reflect the existence of a recent flood analysis by Lawrence Livermore Laboratories. Corrections involved referring to the 1301-N facility as "retired" and reflecting the use of the 1325-N and 1312-N facilities. Also, greater specificity was provided in the Technical Specification references to low river level. Additionally, editorial and typographical corrections were made.

- Chapter 3, Design of Structures, Components, Equipment and Systems, was revised in it's entirety.

  The sections of Chapter 3 have been revised to reflect recently performed modifications and analyses. The major issues addressed were seismic analyses and modifications, wind loading analyses, and safety review findings and resolutions.

- Section 4.3.2.3.3, Inlet Water Temperature Coefficient, was revised to include a discussion of the void coefficient.

- Section 5.2, Integrity of Reactor Coolant Boundary, was revised to reflect recent changes in the Inservice Inspection and Testing Programs (ISI/IST).

- Section 5.3.4, Surveillance, was revised for correction of typographical and syntax errors. 21

- Section 6.2, Confinement System, was revised to reflect recent HECTR confinement analyses, fog spray piping and confinement logic modifications and deletion of the double-riser break analysis.

- Section 6.3, Emergency Core Cooling System, was revised to make reference to Section 15.6.4.5 for a modified description of plant events for which ECCS is relied upon, and added changes to the NUSAR text to incorporate the results of recent ECCS system and component tests, modifications and analyses.

- Section 6.4, Habitability Systems, was revised to reflect Project H-544 modifications which enhance control room habitability.

- Section 6.6, Inservice Inspection of Class 2 and 3 Components, was revised to reflect the new ISI/IST Plan and Procedures (WHC-CM-5-15 and UNI-M-130, respectively).

- Chapter 7, Instrumentation and Control, was revised virtually in its entirety.

  Section 7.1 was revised to provide greater clarity.

Section 7.2 was revised to reflect the conclusion of the various safety review findings and resolutions.

Sections 7.3 through 7.7 were revised for editorial clarity and inclusion of modifications and analyses performed in the comparitively recent past.

- Chapter 8, Electrical Power, was revised to reflect modifications and analyses performed during the comparitively recent past.

- Section 9.2.5, Condensate Storage Facilities, was renumbered to be 9.2.6 to permit the inclusion of a new Section 9.2.5, "Ultimate Heat Sink." This new section was added to reflect reliance on the Columbia River as the ultimate heat sink and to follow the Regulatory Guide 1.70 format.

- Section 9.5.1, Fire Protection System, was revised to reflect the completion of various modifications including the addition of sprinkler systems, deletion of CO_2 System in Room 6, and recognition of administrative controls that enhance fire prevention.

- Section 11.2, Liquid Waste Management System, was revised to reflect the retirement of the 1301-N Facility, the use of the 1325-N Facility and the completion of the 1312-N Facility (LERF).

- Section 15.6, Decrease In Reactor Coolant Inventory, was revised to reflect the completion of RELAP5, TRUMP, CORRAL and HECTR code analyses which were performed to address the National Academy of Sciences recommendations for N Reactor. The new analyses relate to both small and large break LOCAs and the radiological consequences of the same. The effects on offsite dose from the use of the Hydrogen Mitigation System have been included as well as the RELAP5 analysis of a station black-out event (loss of AC power).

In addition, these amendments also include miscellaneous editorial corrections and updating to improve the consistency with other parts of the NUSAR. Pages and tables which have been changed are marked on the border at the location of the changes.

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SITE CHARACTERISTICS

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SITE CHARACTERISTICS

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2.4 HYDROLOGIC ENGINEERING

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities

N Reactor is located on the right bank of the Columbia River in the north-central part of the Hanford Reservation at approximately river mile (RM) 380, 17 miles downstream from the Grant County PUD Priest Rapids Dam. The Hanford Reservation location is shown in Figure 2.4-1. The Columbia River is the predominant hydrologic feature of the area and provides principal drainage for the site and the surrounding area. The river bed is clearly marked in the terrain, and at N Reactor the river flows between a high right bank and a low left bank. The approximate river bottom is at elevation 350.0 ft mean sea level (MSL), the ground elevation at the plant site is approximately 453 ft MSL. All safety-related structures with the exception of the river water intake structure and 182-N pumphouse are completely sealed (flood protected) to elevation 454.5 ft MSL. Entrances into the buildings are located at 454.5 ft MSL. The intake structure, shown in Figure 2.4-2, is located on the shore of the Columbia River and is known as the 181-N building. The operating deck of this structure is at 421 ft MSL, with the emergency diesel pump drives at 426 ft MSL. For flood elevation see Sections 2.4.3 and 2.4.4. For discussion of wind wave activity see Section 2.4.3.6; for flood-force application (dynamic forces) see Section 3.4.2. For the 182-N pumphouse, the Probable Maximum Flood (PMF) elevation would put water into the sump but would be 7 ft below the basement floor.

A topographic map and contour map of the site, as they existed prior to construction, are shown in Figures 2.4-3 and 2.4-4. With the construction of the plant the natural drainage features of the surrounding area are not changed.

2.4.1.2 Hydrosphere

The Columbia River drains an area of 258,000 square miles lying to the west of the Continental Divide in the northwestern part of the United States (85%) and southwestern part of Canada (15%). The Columbia River drainage area has been divided into six areas with similar hydrometeorological characteristics (Ref. 1).

The six areas are: (1) Upper Columbia, which includes the drainage of the area in Canada and the northern part of the United States above Chief Joseph Dam; (2) Middle Columbia, which includes the area between Pasco and Chief Joseph Dam; (3) Upper and Middle Snake River; (4) Lower Snake River, the area between Weiser, Idaho and Ice Harbor Dam; (5) Lower Columbia, including the area between Bonneville Dam and Pasco; (6) the Columbia below Bonneville Dam, including the Willamette River.

N Reactor is in the Middle Columbia area. This area drains mainly the east side of the Cascades in Washington State and British Columbia, with some additional areas in the eastern part of Washington State. This is a relatively low-producing runoff area except for the mountainous region. The Wenatchee and Yakima Rivers are the major contributors. The eastern portion of this area includes the Columbia Basin Reclamation Project with the majority of its water supplied from the Grand Coulee equalizing reservoir.

The regulation of Columbia River by dams and reservoirs has improved over the past 35 years. A large portion of the main stream and major tributaries is developed to meet various functional requirements such as flood control, hydroelectric power, irrigation, municipal and industrial supply, etc. The regulation of Columbia River floods by reservoir storage is accomplished by joint use of reservoir storage space provided primarily for irrigation or for hydroelectric power utilization. The volume of usable reservoir storage space is on the general order of 20 to 25 percent.

On the main stream of the Columbia River within the United States, there are seven dams upstream and four dams downstream of the site as follows:

Dams On The Columbia River

	Dams	Owner	River Mile	Approximate River Miles From Site
Upstream:	Priest Rapids	Grant Co. PUD	397	17
	Wanapum	Grant Co. PUD	416	36
	Rock Island	Chelan Co. PUD	453	73
	Rocky Reach	Chelan Co. PUD	474	94
	Wells	Douglas Co. PUD	516	136
	Chief Joseph	Corps of Eng.	545	165
	Grand Coulee	US Bureau of Rec.	598	218
Downstream:	McNary	Corps of Eng.	292	88
	John Day	Corps of Eng.	216	164
	The Dalles	Corps of Eng.	192	188
	Bonneville	Corps of Eng.	146	234

These dams and other major hydrologic features are shown in Figure 2.4-5. Additional data on these and other dams are given in Table 2.4-1 and Table 2.4-2.

Prudent design criteria require dams to pass the PMF without major damage to the dam itself. The spillway and discharge capabilities of the upstream dams are discussed in Section 2.4.4.1.

The four dams downstream of the site include large locks to permit the passage of river vessels. The Grand Coulee, the largest of the dams in the U.S., augments the low winter flow rates for the entire system from its 9,402,000 acre-feet of available storage (of which approximately 5,100,000 acre-feet is active storage). At Grand Coulee, water is also pumped to the Columbia River Irrigation Project. Large dams above Grand Coulee assist in augmenting the low winter flows particularly Arrow, Mica and Libby. The seismic design criterion used for all U.S. dams is 0.1 g based on the dams being in Zone II of Seismic Risk Map of the U.S. and the Seismic Probability Map of the Uniform Building Code. The two Canadian dams are designed for 0.15 g.

The Columbia River in the vicinity of this site can be characterized as a fast, deep river with flow rates varying on a daily cycle due to the power-producing Priest Rapids Dam at RM 397 which contains about 45,000 acre-feet of active storage, and Wanapum Dam at RM 416 containing about 161,000 acre-feet of active storage. Flow rates in the Columbia River during the summer, fall and winter vary from a low 36,000 cfs (cubic feet per second) to as much as 160,000 cfs. During spring runoff, high flow rates ranging from 250,000 cfs to 450,000 cfs have been recorded. The average annual flow rate is 120,000 cfs. During low-flow periods flow rates typically average 60,000 cfs, Figure 2.4-6.

The river width at N Reactor varies between 400 and 600 yards and about 25 to 60 feet in depth depending on flow rate.

The local drainage area containing N Reactor comprises about 800 acres on the right riverbank. The area has about a 6000-foot river frontage and extends landward about one mile. Its elevation is about 70 feet above normal river level.

Maximum velocities vary from 3 fps (feet per second) to over 11 fps, depending on location and flow rate. Average water temperature is 51°F and may reach extremes of 68°F and 32°F, Figure 2.4-6.

To determine points of water usage from the Columbia River and riverbank wells, data have been taken from records of the Department of Ecology, State of Washington, Spokane, Washington. This information - water rights received until 1969 - is summarized in Table 2.4-3. Records for later years are not consolidated in useable form. However, some projections of future water use in the Tri-Cities Service Area have been made by the Pacific Northwest River Basins Commission (Ref. 2), based on projected population growth:

Tri-Cities Service Area Projected
Municipal Water Use

Year	Use Million Gallons Per Day
1970	22.5
1980	28.5
2000	46.0
2020	66.2

At the present time, there are no groundwater users on either side of the river in the vicinity of the site. The description and sources of groundwater are discussed in Sections 2.4.13.1 and 2.4.13.2, respectively.

2.4.2 Floods

The information in this subsection, related to flood criteria on the Columbia River, is based on extensive documented studies by the U.S. Corps of Engineers.

2.4.2.1 Flood History

The largest flood recorded on the Hanford reach of the Columbia River occurred in 1948 and had an observed peak discharge at Hanford of 690,000 cfs. The largest known historical flood occurred on June 7, 1894 with a peak flow, estimated from high water marks, of 800,000 cfs (Ref. 3). The previously used maximum flood of 740,000 cfs still appears in a number of official publications but the larger number represents a Corps of Engineers adjustment after study of the 1948 flood. These and other notable high-water periods, 1876 and 1956, occurred during the spring runoff and resulted from the melt of a large snowpack combined with the spring rain (Ref. 1). Major floods on the Columbia River are caused by climatic conditions which affect the entire Basin, such as a high water content snow pack holding over until late April in the upland areas and above-normal precipitation in May accompanied by a major chinook (warm wind) which causes a rapid area temperature rise.

There is no record of flooding in the site area due to ice jams, nor have there been any flood conditions due to dam failure.

Flood Conditions on the Columbia River at the Site

The flood conditions for the Columbia River which are applicable to the site were determined by the Corps of Engineers (Ref. 1). Figure 2.4-7 shows water surface profiles for the Columbia River in the vicinity of the site.

Plant Site Area Flooding

N Reactor is located at RM 380 with a main floor elevation of 454.5 ft MSL and is 40 feet above the water level estimated for the largest historical flood.

The 100-N area is not subject to flood from rain or fast melting snow. Adequate drainage is provided because of the plant's proximity to the river and good drainage paths to it. The entry elevation to critical facilities is higher than the surrounding limited drainage area.

Drainage from about 800 acres adjacent to and including 100-N Area is north to the river. Further south of N Reactor drainage occurs via a man-made ditch around 100-N Area. Figure 2.4-3 marks this ditch as the Hanford Ditch, originally used for irrigation. This ditch has not been used for over thirty years and thus presents no potential for flooding. No known flooding of this area has occurred.

2.4.2.2 Flood Design Considerations

To determine the susceptibility of N Reactor to flooding, an analysis was made of the highest calculated flood water level elevation, including wave effects, resulting from several hypothetical events. Several different probable maximum events are considered including the Corps of Engineers design-project flood, the most severe reasonably possible," wave action caused by storm winds, the effects of failure of upstream dams, surge flooding and ice flooding.

Analyses of the Probable Maximum Flood (PMF), as defined in Section 2.4.3, show that the river at N Reactor would reach a level of 423 ft MSL. The main floor and entrances to all safety-related facilities are at elevation 454.5 ft MSL, with the exception of the 181-N river pump house. The 181-N facility was designed to accommodate the 100-year flood of 800,000 cfs which would reach an elevation of 415 ft MSL. The operating floor of the 181-N facility is at 421 ft MSL. The calculated wind-wave would add 2 feet to the crest of the PMF flood raising the peak water level to 425 ft MSL. The operating floor would be submerged. However the diesel drive engines for the emergency cooling supply pumps are elevated 5 feet above the operating floor and are housed in a steel and concrete block building. The flood waters would enter the building but the wind-wave would have little effect within the building. The diesel drives are provided with 1 to 3 feet of freeboard and would not be stopped by the PMF. See Section 3.4 for a complete analysis of the stability of the 181-N structure to survive the PMF. 2

The emergency pumps in the 181-N building supplies water directly to the once-through emergency coolant pumps (high lift pumps) located in the basement of the 182-N building. The 182-N basement elevation is 430 ft MSL. The floordrains in the 182-N building discharge to the river bank through a 42-inch steel line from a sump. The inlet invert elevation of the drain line is 421.75 ft MSL, and the outlet invert elevation is 415.25 ft MSL. The outlet would not prevent some flood water seepage from entering the line but would minimize stop wave action. The PMF elevation would put water into the 182-N sump but would be 7 feet below the basement floor level. 21

2.4.2.3 Effects of Local Intense Precipitation

The N Reactor site and local drainage area is shown in Figure 2.4-8. The area represents a bench approximately 70 feet above the river and adjacent to it. Drainage from this bench of about 800 acres is generally toward the river along a 6000-foot section.

As shown in Figure 2.4-8, the drainage area size is limited by a topographic feature south of the site which would serve as a potential channel to drainage

flow from farther south. This channel is approximately marked by the first 3-4 miles of the Hanford Ditch which provides natural drainage to the east. A USGS bench marker, just south of the site in Section 33 near where the railroad crosses the ditch, locates the initial elevation of this potential channel at 442 ft MSL.

The Probable Maximum Precipitation (PMP) event for the site has been determined using the methodology developed by the U.S. Weather Bureau and reported in Hydrometeorological Report No. 43, "Probable Maximum Precipitation, Northwest States" (Ref. 32).

The PMP evaluation for the site drainage area is quite intense because of its small size but is not long lasting. The following tabulation shows the time sequence and quantity of precipitation constituting the PMP.

Hour	Inches Rain, Area Average
1	.4
2	.8
3	8.0
4	1.5
5	.5
6	.4
TOTAL	11.6

Because of the proximity of the nuclear plant to the river channel, the good drainage paths to it and the very limited drainage area, no flood would occur from the PMP event of 11.6 inches. The ground elevation around the Category I structures is 453 ft MSL, and the buildings entry elevation is 454.5 ft MSL.

The critical safety facility is the 182-N building because it contains the emergency coolant pumps and could possibly be subject to localized runoff. This is a reinforced concrete block structure with reinforced concrete basement

and door openings at the 454.5 ft MSL elevation. The building is provided with floor drains and a sump which are drained to the river via a 42-inch line. Any leakage would thus drain to the river. The 182-N building wall penetrations are all sealed. These penetrations are either located several feet above grade or are buried several feet below grade.

It should be noted that the PMP is nearly double the annual average precipitation of 6.28 inches at the site.

2.4.3 Probable Maximum Flood (PMF)

The Corps of Engineers states (Ref. 4) that the unregulated Probable Maximum Flood (PMF) at the site was derived by the North Pacific Division, Corps of Engineers by the methods outlined in a memorandum report dated September 1969, titled "Lower Columbia River Standard Project Flood and Probable Maximum Flood" (Ref. 1). The adjusted flow for the Hanford region is given in Reference 5 as 1,600,000 cfs for the unregulated PMF.

The methods of estimating peak flow rate and the controlled PMF hydrograph above Pasco were given in the referenced report. Further adjustment of the flow for the Hanford region is done and reported in Reference 5 establishing a regulated PMF of 1,440,000 cfs. This PMF was established by the Corps of Engineers whose methods for establishing PMFs are recommended by Appendix A of Regulatory Guide 1.59.

The water level of 423 feet at the intake structure (RM 380) was derived by interpolating the Corps of Engineers preliminary drawing dated November 1970, titled "Columbia River Washington Water Surface Profiles, RM 323 to RM 395", Figure 2.4-7.

The Corps of Engineers in establishing the PMF did not consider any potential earthquake-induced water release. A preliminary effort to quantify the risk of earthquake and random dam failures was performed by Lawrence Livermore National Laboratory (Ref. 28 and 29) and is described in Section 2.4.4. 21

2.4.3.1 Probable Maximum Precipitation (PMP)

The PMP consisting of intense heavy precipitation over a relatively short period of time is usually associated with small areas or drainage basins. As the drainage area increases, such intense precipitation cannot occur uniformily and intensely over very large areas such as the upper Columbia River watershed. Time required for a storm to progress from one edge of the upstream drainage area to the other edge is on the order of 48 hours. Thus a 12-hour PMP would proceed across the basin in essentially four increments (Ref. 1). Geological features within the basin would cause nonuniform precipitation over the area. The PMP considered for the PMF is not a single event but a 7-month period (October through April) of exceptionally cold and wet weather in the watershed followed by: (1) a heavy rain during mid-May, (2) then conditions conducive to unusually rapid snow melt and (3) then a second heavy rain timed to maximize the natural peak discharge in the lower Columbia River (Ref. 1).

2.4.3.2 Precipitation Losses

Soil absorption varies widely for this Basin Area, but for the period of October through mid-June losses are estimated at 3.5 to 8 inches. Other precipitation losses considered, and which would reduce the runoff, were the evaporation loss during snow accumulation period and evaporation loss during melting seasons. These losses were estimated at 2.4 inches and 2.6 inches, respectively (Ref. 1).

2.4.3.3 Runoff and Stream Course Model

The drainage basin for the Columbia River at N Reactor encompasses about 9600 square miles. This area varies from near desert terrain to densely wooded mountains.

There are two main hydrological features of this basin. The first is the many man-made dams used for power generation and flood control. The second is the spring runoff consisting of melted snow and spring rains. Snow accumulates at the higher elevations starting in October and continuing through April. The lower elevations have light winter snows which usually melt in a few days.

In establishing the PMF the Corps of Engineers divided the Columbia River watershed into 63 subbasins which were grouped into 6 areas. Two of these areas composed of 25 subbasins constitute the watershed affecting the river flow at N Reactor. Two additional subbasins are in one of these two areas but enter the Columbia River below N Reactor. Data on each subbasin and the basin hydrological model are presented in Reference 1.

2.4.3.4 Probable Maximum Flood Flow

The probable maximum flood flow for this reach of the Columbia River has been noted in Section 2.4.3 of this report as 1,440,000 cfs (regulated). The flood peak could occur during the period from mid-May to mid-July.

A system diagram for basin flood estimates has been developed by the North Pacific Division, U.S. Army Corps of Engineers. All comparisons and consistency checks indicate the derivation of the PMF is technically sound and that hydrographs that have been developed are in agreement with the definitions of those floods (Ref. 1).

Spillways and turbine passage of upstream dam projects have been designed to release regulated probable maximum flood flow without failing.

The large storage dams of Grand Coulee and upsteam from there provide the regulation to reduce the unregulated PMF. The dams between N Reactor and Grand Coulee provide little or no contribution nor hindrance to the regulated PMF.

2.4.3.5 Water Level Determinations

The Corps of Engineers prepared water surface profiles for artificial and real stage flows for RM 323 to RM 395 in November 1970. A portion of that information is presented in Figure 2.4-7. This information was used to develop the flow rating curve which is shown in Figure 2.4-9.

2.4.3.6 Coincident Wind Wave Activity

The computation of wind wave activity at the site is based on Project Bulletin No. 2; "Wind Wave Investigation" published by the U.S. Army Corps of Engineers

dated December 1967 (pages 14 and 15). The equations relating wind speed, wave height and fetch presented in the above publication are based on the wind wave investigations conducted on the McNary Reservoir, located on the Columbia River 88 river miles downstream of the site.

The computation of effective fetch is based on the Corps of Engineers' publication No. ETL 1110-2-8 and titled, "Computation of Freeboard Allowances for Waves in Reservoirs" (page 31). The wind set-up is computed in accordance with the equation presented on page 7 in ETL 1110-2-8.

The probable maximum flood elevation at the site is 423.0 ft MSL. The ground elevation at the site is at approximately 453 ft MSL. The runup of the wind generated waves will be very small at the vicinity of the plant because of the short effective fetch (2.0 miles) and the very steep slopes of the river shore.

The wind-generated maximum wave height above still water level coincidental with the probable maximum flood, using standard techniques for computation, is 2.0 feet for an over-water wind velocity of 45 mph. The water body is moving very rapidly, and the fetch distance is small; therefore the computed value for the maximum wave height is conservative. The static and dynamic effect of the waves on the intake structure is taken into account. At the intake structure, rip-rap slope protection is provided for protection against waves generated at PMF and lower water levels.

2.4.4 Potential Dam Failures (Seismically Induced)

In 1951, the Seattle District Corps of Engineers made a study (Ref. 8) to determine artificial flood hydrographs and flood profiles in the Columbia River Valley resulting from breaching Grand Coulee Dam. In August of 1963, the Seattle District Corps of Engineers issued another confidential report, which has since been declassified, regarding Columbia River artificial flood considerations (Ref. 7). The studies covered a spectrum of conditions in

terms of breach openings and hydrologic conditions that might prevail as a result of enemy attack. The profiles for these artificial floods shown in Figure 2.4-7 are useful in establishing the feasible natural flood elevations. Also these studies are useful in discussion of other hydrological aspects where their data are pertinent to analysis of natural events.

A study by ERDA (Ref. 6) concluded that seismic damage to Grand Coulee Dam would not produce a flow rate in excess of the PMF. A preliminary probabilistic assessment of the risk of dam failure (random or siesmic) for the Columbia River drainage basin was performed recently by Lawrence Livermore National Laboratory (Ref. 28 and 29). See following subsection for further discussion. 21

2.4.4.1 Dam Failure Permutations

Table 2.4-1 is a summary of major dams upstream from N Reactor. N Reactor is located at RM 380 and is only 17 miles downstream from the nearest dam.

There are four dams downstream from N Reactor; the nearest is 88 miles and it has no regulation effect upon the river in the vicinity of N Reactor.

All of the Columbia River dams upstream of the N Reactor are in Zone 2 of the U.S. Seismic Risk Map. Consequently they are designed for ground acceleration of 0.1 g. Mica and Arrow dams in Canada are designed for 0.15 acceleration.

Because Grand Coulee Dam was recognized long ago as being a major upstream dam with a storage capacity great enough to produce a flood potentially damaging to N Reactor facilities, it has been given thorough study. 21

The dams between N Reactor and Grand Coulee Dam have much smaller storage capacity and consequently are of lesser concern. However, they are all modern, well constructed dams located in the same Zone 2 seismic zone. All of these dams are constructed with gated overflow weir type spillways. Their ability to handle the PMF which is 1,200,000 cfs at Grand Coulee Dam and reaching 1,440,000 at N Reactor is shown in the following tabulation: 2

Flood Capacity of Individual Dams

Dam	Spillway Capacity cfs	Turbine Capacity cfs	Failure Point* Flow Rate cfs
Chief Joseph	1,250,000	116,000	1,700,000
Wells	1,180,000	230,000	1,450,000
Rocky Reach	990,000	112,000	1,500,000
Rock Island	954,000	82,000	**
Wanapum	1,400,000	180,000	2,100,000
Priest Rapids	1,400,000	187,000	2,000,000

* Dams can discharge flows in excess of design spill capacity; they may be damaged but not fail below this flow rate. Failure is not expected until flow over embankment reaches 5 foot depth. Turbine capacity is not included in failure point flow.

** Rock Island Dam is an all reinforced concrete dam and is designed for overflow.

The U.S. Bureau of Reclamation (USBR), in response to requests of the AEC for an up-to-date analysis of Grand Coulee Dam using the design basis earthquake developed for FFTF (0.25 g horizontal and 0.17 g vertical), stated that such an event would not produce damage resulting in a discharge in excess of 1,200,000 cfs, the design flood for Grand Coulee Dam (Ref. 6).

A study of the Grand Coulee Dam regional geology and seismology to establish seismic criteria for Grand Coulee Dam was made in conformance with Seismic and Geological Siting Criteria for Nuclear Power Plants, Appendix A to 10 CFR 100 by John A. Blume and Associates (Ref. 10). The Blume study stated a reasonably conservative prediction of the maximum horizontal ground motion that could be induced at the site is 0.08 g for the Operating Base Earthquake (OBE) criteria and 0.16 g for the Safe Shutdown Earthquake (SSE). This SSE is considerably less than the FFTF design basis earthquake used in the USBR damage analysis.

A study by the USBR concluded that Grand Coulee Dam and Forebay Dam are founded on competent granitic rock in an area of tectonic stability and low seismicity (Ref. 11). The study concludes a Richter magnitude 7.5 earthquake located 50 miles away is the maximum credible earthquake, and their analysis, including the normal water and dead loads, shows the dam has a minimum safety factor of 3.2.

A document titled "Evaluation of Impact of Potential Flooding Criteria on the Hanford Project", by the Richland Operations Office of ERDA, (Ref. 6) concluded from various studies that the damage (from earthquakes) to Grand Coulee under conditions judged to be conservatively severe would not be sufficient to cause a flow of greater than the 1,440,000 cfs Probable Maximum Flood.

The Grant County Public Utility District (PUD), at the request of the Federal Power Commission (FPC), had an independent safety-stability investigation performed by Cornell, Howland, Hayes and Merryfield engineering consultants on both Priest Rapids and Wanapum Dams. Their reports (Ref. 12 and 13) conclude the design earthquake ground motion of 0.1 g is appropriate and conservative. These reports further state that the dams have adequate factors of safety for the various conditions investigated including seismic. 2

John A. Blume and Associates performed a study for Exxon Nuclear Co. to predict flood levels in Richland, Washington for Columbia and Yakima River dam failures (Ref. 14). This study concludes that only two dams have the storage capacity which, if the dam failed, might cause subsequent failure of downstream dam(s) to produce excess flow at Richland. The study assumes an instantaneous failure without cause of Chief Joseph Dam occurring during a 25-year flood. A similar case involving Wanapum Dam was also studied. Their analysis shows complete failure of Chief Joseph Dam would not cause a failure of any down-stream dams. The data presented indicates the Chief Joseph Dam hypothetical instantaneous failure would result in a flow of about 850,000 cfs at N Reactor. A similar failure of Wanapum Dam could cause a failure of Priest Rapids Dam. Such a secondary failure would not be instantaneous with the arriving peak flow from Wanapum Dam but would be delayed in time producing

a longer high flow period with a peak discharge from Priest Rapids Dam of 2,000,000 cfs.

The instantaneous failure of Wanapum Dam as used in Reference 14 is a hypothetical failure which produces a flow just sufficient to cause a failure of Priest Rapids Dam. Any postulated realistic failure mechanism would produce a lower flow and no failure of Priest Rapids Dam. The more realistic flood level at N Reactor would not exceed the PMF.

The Blume-Exxon study relating to a Yakima River flood at Richland used a maximum ground acceleration of 0.13 g at three dams on the upper Yakima. These three earthfill dams constructed in 1912, 1916 and 1933 are near the center of the Cascade Mountains and, thus closer to the Puget Sound seismically active Zone 3 than the Columbia River dams. The study shows historically that similarly constructed dams have withstood earthquake ground accelerations of 0.35 g to 0.5 g. The Columbia River earthfill dam sections were constructed within the past 20 years to more modern specifications than those dams cited in the historical part of the report; consequently they are expected to withstand greater earthquake intensities.

The conclusion drawn from analysis of the foregoing reports is that the seismic induced damage to upstream Columbia River dams would not produce a flood damaging to N Reactor facilities even if the earthquake occurred at the height of a 25-year flood.

In an effort to quantify the likelihood and magnitude of severe flood events at DOE facilities Lawrence Livermore National Laboratory performed a preliminary Probabilistic Flood Hazard Assessment for 9 sites, including the N Reactor (Ref. 28 and 29). The preliminary assessment identifies the potential for water levels from random and seismically induced dam failures (particularly above Grand Coulee Dam) to exceed the N Reactor intake structure and site elevations. The preliminary assessment was based on conservative assumptions (Ref. 30) for which further analysis would be required to determine the assessments validity.

2.4.4.2 Unsteady Flow Analysis of Potential Dam Failures

The flood hydrographs developed by the Corps of Engineers are the result of extensive studies of the physical characteristics of the flood route (Ref. 7 and 8). Subsequent studies by New York University verified the routings (Ref. 9). However, the breaks of Grand Coulee Dam on which the three larger floods are predicted are the result of man-made forces (bombs) and not of any credible natural force.

2.4.4.3 Water Level at Plant Site

The water elevations associated with both natural and artificial floods shown on Figures 2.4-7 and 2.4-9 were developed by the Corp of Engineers. The PMF of 1,440,000 cfs would have an elevation of 423 ft MSL at the N Reactor site. The superimposed wind-wave would be about 2 ft. Past analysis of upstream dams indicate that damage caused by natural causes would not result in water levels higher than that caused by the PMF. See Section 2.4.4.1. However, recent preliminary analyses indicate the potential for exceeding the PMF water levels (Ref. 28 and 29). The assessments also indicate that a minimum of approximately 20 hours advance warning would occur, which would permit time for reactor shutdown. 21 21

2.4.5 Probable Maximum Surge and Seiche Flooding

2.4.5.1 Probable Maximum Winds and Associated Meteorological Parameters

Detailed meteorological data were obtained for approximately 1 year from January 1970 to January 1971, from a tower located at the N Reactor site. Wind gusts of greater than 39 mph were not recorded during this year, and the average wind speed measured for the year at the 50-foot level was 8.4 mph. Meteorologic data have also been collected at the Hanford Meteorology Station (HMS), located approximately 7.5 miles south and 300 feet higher in elevation. Wind gusts of greater than 39 mph did not occur at HMS in 1970. The average wind speed was 8.2 mph. Peak wind gust return probabilities have been calculated, based on 27 years of data (Ref. 15) at the HMS, and are shown in Figure 2.4-10. Inspection of Figure 2.4-10 shows that the maximum wind gust expected once every 100 years is about 85 mph at the 50-foot level. It should be recognized that the number represents the maximum wind gust and not a steady

blowing wind. A steady wind speed of 45 mph is the selected maximum which might occur simultaneously with a 100-year flood.

Tornado activity within a 100-mile radius of the project is discussed in Section 2.3.1.2. It is estimated that winds as high as 165 mph could occur, but with a very low probability of occurrence (Ref. 16 and 17). This is in agreement with the Hanford Standard Design criteria for Category I Facilities.

2.4.5.2 Surge and Seiche Water Levels

The site is not close to any water body which could experience surge or seiche flooding.

2.4.5.3 Wave Action

Calculation of the wave action in the vicinity of the site is difficult, because the fetch is small and the water body itself is moving quite rapidly (>3 fps). Estimates of wave height for a historical flood elevation of 415.0 ft MSL, using standard techniques for computation (see Section 2.4.3.6), show that with an over-water wind velocity of 45 mph and an effective fetch of 1.5 miles, a maximum wave height of 1.6 feet above still water could be expected. The 416.6 ft MSL flood-wave top elevation is well below the river pump facility operating floor of 421 ft MSL.

2.4.5.4 Resonance

The irregularity of the river bottom and the swift current tend to dampen all sources of resonance at the site location.

2.4.5.5 Protective Structures

The only structure that could be affected by wave action for a PMF is the 181-N river pumping facility which houses the electrically driven circulating raw water pumps and diesel-driven emergency coolant pumps. The main structure is reinforced concrete with the operating deck at 421 ft MSL. Constructed on the deck is a reinforced concrete block structure housing the diesel-driven emergency coolant system pumps. This reinforced concrete block structure was constructed prior to establishing the PMF; however, it will withstand the 2 feet of flood water plus the 2-foot waves postulated to accompany the PMF. The building

will also withstand the 1.6-foot waves which could accompany a repeat of the historical flood which reached an elevation of 415 ft MSL. The building is slightly recessed into the river bank; this provides some protection from the high velocity current carrying floating debris. However, more protection is provided by the WPPSS Hanford Generating Project pumping structure located about 800 feet upstream and which protrudes further into the river. Also the security fence and other equipment on the 181-N deck would protect the building from debris in water expected to be only 2 feet deep. Flood waters will enter the structure through small openings but wave action will not. No danger to critical equipment will occur from a PMF as the diesel drives are elevated 5 ft above the floor. 21

A recent preliminary assessment indicates the potential for exceeding the PMF water level at the plant site, including the potential for exceeding site grade elevation of 454.4 ft MSL. The assessment also indicates that approximately 20 hours advance warning would occur which would permit time for reactor shutdown. 21

2.4.6 Probable Maximum Tsunami Flooding

The N Reactor location in south-central Washington is not adjacent to any coastal area. It is therefore not vulnerable to tsunami flooding.

2.4.7 Ice Effects

Historically, the Columbia River has never experienced complete flow stoppage or significant flooding in this vicinity due to ice blockage. Periodic ice blocking has caused reduced flows and limited flooding for only relatively short periods of time. The most significant icing in recent recorded history occurred during the winter of 1936-1937. A relatively thick sheet of ice formed across the river. The minimum flow recorded near the Priest Rapids Dam site during this winter was 20,000 cfs. However, the ice forming on the river was caused primarily by the low flow. The deltaic mouths of many of the tributaries to the Columbia River are frequently blocked by ice causing backup of flood waters in the tributaries. However, no instance of complete stoppage is known to have occurred.

Ice blockage is most likely to occur when water temperatures are already low, when flows are small and when a significant cold spell occurs. With the completion of Grand Coulee and other dams on the Columbia River main stem, the seasonal temperature and flow variations have been drastically altered. The changes in the seasonal temperature and flow cycles are coupled in such a way as to reduce the intensity and timing of the conditions which may contribute to potential ice blockage and flooding situations. Average river flow rates during the winter months have been increased significantly. Peak water temperatures occur 30-45 days later now than before the dams were constructed. In addition, the low extreme temperatures measured have risen slightly.

Meteorological records show that the most severe cold spells last less than 2 weeks. Hence, the opportunities for extensive ice buildup in the reservoirs is most likely limited to a thickness of 18 inches or less. The warmer winter reservoir water temperatures coupled with higher winter flows (expected concurrently with cold spells as a result of maximum electrical demands for heating) serve to limit ice problems to those of booming ice at penstock entrances and pumphouse forebays.

The long term trends of temperatures in the Columbia River have been studied (Ref. 18) using a 37-year record of measured temperatures. The trends for the maximum, average and minimum temperatures are shown in Figure 2.4-11. The erection of dams on the upper Columbia River has caused the extreme high and low river temperatures measured at Rock Island Dam (Columbia RM 453, 68 miles above the reactor site) to converge toward the average. Winter water temperatures are considerably warmer and summer temperatures cooler with a slightly lowered average of less than 1°F occurring during the 37 years.

The winter of 1969-1970 provided the longest sustained cold period in the past 35 years. No ice problem occurred on any dams or pumphouse intakes, although some protective booming was required to minimize accumulation. At downstream Lake Wallula (McNary Reservoir), ice reached thicknesses to 18 inches one mile upstream of McNary Dam and impeded low-power navigation for a 2 to 3 week period. Some ice accumulations on shore occurred, but open water existed at all points on the upper river above RM 335 (Richland).

Based on these studies and the recorded observation of over 30 years of operation of the Hanford production plants, it is concluded that the potential for ice blockage or the combination of blockage and flooding behind ice dams is so low as to be considered negligible.

2.4.8 Cooling Water Canals and Reservoirs

No cooling water canals or reservoirs are used on this project.

2.4.9 Channel Diversions

The Columbia River flow in the Hanford reach is controlled, to a large extent, by regulation of the upstream reservoir projects. The site is located on Columbia RM 380, 17 miles downstream from the Grant County PUD Priest Rapids Dam. Within 100 miles there are three additional upstream dams controlling the flow. In the vicinity of the site the river bed is well defined, and it is very unlikely that the river would be diverted from the present location by natural causes. Any possible effect on water supply to the intake from river bed changes should come from extremely slow changes which can be corrected as they occur. The river has not frozen over in the Hanford Project reach during at least the past 30 years, and icing on the river has not been a problem at pumphouses or outfall structures associated with the Hanford Project special nuclear materials reactor plants.

2.4.10 Flooding Protection Requirements

The only safety-related facilities possibly endangered by the PMF are the diesel driven emergency coolant system pumps mounted on pedestals 5 feet above the operating deck of the river pumping structure. The river pumping structure was constructed prior to establishment of the current PMF with the operating floor 6 feet above the 100-year flood level. The 100-year flood would reach 415 ft MSL at the site; however, the PMF would reach 423 ft MSL.

The diesel driven emergency pumps are protected by a reinforced concrete block structure supported on top of the reinforced-concrete pumping-structures

operating deck at 421 ft MSL. The concrete block structure will not keep out the PMF but will keep out wind-generated waves and will withstand the other dynamic forces from 2 feet of flooding including floating debris.

The pump discharge piping is protected from the dynamic forces of flood waters by being enclosed under the operating deck and going underground as it leaves the pumping structure. The pumps discharge to the 182-N high lift pumps. A preliminary assessment (Refs. 28 and 29) indicates the potential for exceeding the PMF water level at the site, including the potential for exceeding site grade elevation of 454 ft MSL. The assessment also indicates that approximately 20 hours advance warning would occur which would permit time for reactor shutdown.

2.4.11 Low Water Considerations

2.4.11.1 Low Flow in Rivers and Streams

Reservoirs that have been constructed in the Columbia River Basin upstream of N Reactor have a usable storage capacity of more than 35 million acre-feet. This capacity alone is sufficient to maintain a flow in the Columbia River past N Reactor of 36,000 cfs for over one year with absolutely no inflow from other sources. Since the minimum annual average recorded river flow to occur in the past 55 years is over 70,000 cfs, it is extremely improbable that minimum flows below those administratively set for Priest Rapids Dam operation (36,000 cfs) could be caused by drought conditions. River levels for low flow rates have been measured down to 36,000 cfs. In November 1987, a special plant test (PT614) measured river level at N Reactor to be 378 ft, 3 1/4 in. at 36,000 cfs. These measurements have been extrapolated down to 25,000 cfs and are shown in Figure 2.4-12. The emergency pumps can operate with a river level of 370 ft MSL which is estimated to be a flow rate of 12,000 cfs.

Provisions to guard against an accidental shutoff of Priest Rapids Dam include:

1. a gate actuation button in the control room of the Dam which is used to maintain at least minimum licensed flow from the facility in the event of one or more turbine shutdowns;

2. independent motors on each gate which have redundant wiring and power supplies;
3. electrical heating on four of the gates to prevent ice buildup which might interfere with gate operation;
4. multiple offsite power sources in addition to an onsite diesel generator power backup for gate operation.

Experience has shown that following a turbine trip (the mechanism most likely to suddenly reduce flow from the dam) the unit or a replacement is returned to service or the river flow augmented by gate movement within 5 minutes.

The electrical power generated in the northwest is highly oriented to the hydroelectric dams on the Columbia River and its tributaries. However, N Reactor is a significant contributor to the BPA system. In CY-1976 it provided 6.3 percent of the BPA peak generation and 2.6 percent of the total energy. During periods of low flow, which could restrict hydro power, N Reactor's power would be either sufficiently critical that the river would be regulated to assure adequate flow for the N Reactor operation, or the plant shutdown in accordance with Technical Specification requirements. 21

There would be no effect from a downstream dam failure as the upper reach of reservoir of the first downstream dam is 31 miles below N Reactor.

2.4.11.2 Low Water Resulting From Surges, Seiches or Tsunamis

The N Reactor location is not subject to surges, seiches or tsunamis. The only significant meteorological or geoseismic conditions which could cause low water at the site would be an event which would cause the stoppage of flow over Priest Rapids Dam. A complete stoppage of flow is an unlikely event because of the redundant equipment and operational procedure in place at the dam. Calculations (Ref. 19) have shown that after a complete stoppage of flow at Priest Rapids Dam, the flow at the N Reactor site would decrease from 36,000 to 25,000 cfs within 1 hour. Stored water on the Hanford Reservation would provide adequate cooling for a substantial time period which would permit Priest Rapids Dam personnel time to reestablish river flow. 21

2.4.11.3 Historical Low Water

At the present time, the minimum flow past Priest Rapids Dam is limited by FPC license to the Grant County PUD, operators of the dam. This license states:

> "The licensee shall so regulate the flow from the Project 2114 that it will not result in flows of less than 36,000 cubic feet per second of water at the Hanford Works of the Atomic Energy Commission except when conditions are beyond the licensee's control."

In 16 years of operation, the flow has not dropped below this specified minimum. Prior to construction of major dams, historical records of the USGS have shown low daily averaged flows of 20,000 cfs (January 31, 1937) and low monthly averaged flows of 20,900 cfs (February 1937). An instantaneous low flow of 4120 cfs occurred in 1932 at the Trinidad gaging station 60 miles upstream from N Reactor, but this was due to activities connected with dam regulation of the river near Wenatchee, Washington.

2.4.11.4 Future Control

Future minimum flows in the Columbia River will be maintained at 36,000 cfs. This flow rate represents the legal minimum licensed flow of Priest Rapids Dam and as such is a parameter closely monitored and controlled by that facility. Water storage capacity in reservoirs located in the Columbia River Basin upstream of the site is presently in excess of 35-million acre-feet (see Tables 2.4-1 and 2.4-2). Because of this capacity and the great demand for the electrical energy generated on the Columbia River, it is very improbable that flows below 36,000 cfs would ever occur. There is always the possibility that a planned, approved and coordinated period of lower stream flow might come about for some reason in the future. Such a planned event would include the safe operation of N Reactor as a prime consideration.

2.4.11.5 Plant Requirements

The minimum river flow as stated in Section 2.4.11.1 will not be less than 36,000 cfs. This is the minimum design flow for normal operation. The reactor will be shut down when flow is below 36,000 cfs. The primary reason the river flow would drop below this level would be a planned reduction for some purpose.

The plan would provide sufficient flow and time for orderly reactor shutdown and time to prepare for emergency or alternate shutdown coolant flow. Although there are no known natural events to cause flow rates lower than 36,000 cfs, safe shutdown could be maintained with an estimated river flow rate of 12,000 cfs. 21

2.4.11.6 Heat Sink Dependability Requirements

At the minimum allowable river flow of 36,000 cfs, as described in Section 2.4.11.3, there is sufficient depth at the water intake facility to continue normal full power operation of N Reactor. The raw water system is described in Section 9.2.

A procedure has been established to provide a warning of stoppage at Priest Rapids Dam, at which point the reactor operator would be alerted and prepare for shutdown as a precautionary measure if the condition persists. However, emergency procedures at Priest Rapids, described in Section 2.4.14.3, ensure that the situation would be corrected before having any significant impact on river flow. 2

Minimum design water elevation for the emergency coolant system pump intakes is 370 ft above MSL. This value is based on the lowest elevation practical that would still allow water to reach the intake considering the river bottom profile in this area and allow for pump drawdown. Extrapolating the curve on Figure 2.4-12 and allowing for conservatism, 370 ft corresponds to an estimated flow of approximately 12,000 cfs. Flows this low are extremely improbable and, in any event, would be caused at Priest Rapids Dam or further upstream which would allow sufficient warning time to achieve a safe shutdown condition. 21

The Columbia River is a large river with a minimum flow of 36,000 cfs which provides a reliable heat sink as required by Regulatory Guide 1.27. 21

2.4.12 Dispersion, Dilution and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters

Accidental release of radioactive effluent directly to the river could result from a minor leak within an auxiliary system heat exchanger. Such a discharge would contain very small quantities of radioactivity. All effluent streams with the potential for discharging minor amounts of radioactivity directly to the river are monitored to limit any accidental release to an insignificant quantity. A discussion of release pathways and the design basis for the affected systems is provided in Section 11.2. Any major release would occur within the 105-N and 109-N buildings where the drainage system would route the release to the emergency disposal basin. A discharge to this basin would enter the groundwater and reach the river via springs. Discharge to groundwater is discussed in Section 2.4.13.

Approximately 30 years of measuring routine discharges to the river and measurements in the river show that, at the plant boundary 25 miles downstream and at the first downstream water user, essentially complete mixing-dilution has occurred. There are no downstream water users within the controlled Hanford Reservation. The first downstream water user (consumer) is located about 28 River Miles from the reactor.

The Columbia River is a large, swift flowing river providing good mixing and large dilution factors even at its minimum flow of 36,000 cfs (16,200,000 gpm).

The final environmental statement on Waste Management Operation for the Hanford Reservation (Ref. 26) describes the dispersion, dilution and travel times for radionuclides released from N Reactor to the Columbia River. The pathways of radioactivity releases to the river from N Reactor which contribute to human exposure are discussed in Section 11.2.

2.4.13 Groundwater

2.4.13.1 Description and Onsite Use

The N Reactor project is located on glacial outwash gravels which are about 50 feet thick. The gravels contain sand, cobbles and boulders, are relatively clean and overlay approximately 60 feet of Ringold sands and gravels which contain scattered cobbles. Locally, the Ringold sands and gravels may be cemented by calcium carbonate and iron oxides. The Ringold sand and gravel are underlain by interbedded clay, silt and sand which forms the bottom of the unconfined aquifer. This interbedded clay, silt and sand unit, which overlays the Columbia River basalt flows, is 400 feet thick at the site. The unconfined aquifer in this portion of the Hanford Reservation is bounded by the Columbia River on the north and east and the Gable Butte and Gable Mountain basalt anticlines on the south and west (Figure 2.4-13). The aquifer bottom is at an elevation of 335-350 ft MSL beneath the project site giving a saturated aquifer thickness of from 35-65 feet in this area.

Over the life of the Hanford Project, numerous geologic and hydrographic groundwater studies have been made (Ref. 20-25).

The unconfined aquifer is hydraulically connected to the Columbia River, and seasonal variations in the groundwater table are evident. Other variations in the water table are caused by disposal of water to the ground from DOE facilities such as the N Reactor and the chemical processing areas south of Gable Butte. Figure 2.4-13 shows the groundwater contours over the entire Hanford Reservation.

The region across the Columbia River in the vicinity of the plant consists of glacial outwash gravels extending about five miles landward from the river where basalt outcroppings form a series of bluffs. Although little information concerning the groundwater in this area exists, it can be reasonably assumed that its characteristics are similar to that on the plant side of the river. Hence, the Columbia River will form a hydrologic boundary, and some bank storage and release will occur as the flow in the river increases and decreases.

In wells, more than five miles north of the river, groundwater found in basalt is primarily in confined aquifers.

The plant does not use any groundwater supply. All of the water required for both consumptive and process purposes is taken directly from the Columbia River.

2.4.13.2 Sources

At the present time, there are no groundwater users on either side of the river in the vicinity of the project. A number of wells in the vicinity of the Columbia River exist far downstream of the project. These wells are supplied with groundwater which originates outside of the influence of the plant.

At the present time, the land south of the Columbia River and in the vicinity of the plant is owned and controlled by DOE. There are no plans at this time for withdrawal and use of groundwater. The area across the river between the river and State Highway 24 and extending east to the bluffs is controlled by the Bureau of Sport Fisheries and Wildlife and operated presently as a game refuge. No active withdrawal of groundwater occurs in this area at the present time.

Figures 2.4-13 to 2.4-16 show the seasonal variation of the water table in the vicinity of the project. A number of wells adjacent to the project have been monitored as shown in Figure 2.4-17. The hydrographs for some of these wells are shown in Figure 2.4-18. Figure 2.4-19 provides a map of wells located at the N Reactor site.

The local groundwater mound under the 100-N area was created by liquid effluent discharges to the 1301-N ground disposal facility and trench, and maintained at a new lower level by discharges to the 1325-N basin. These discharges average about 2000 gpm during reactor operation and 1000 gpm during reactor shutdown periods. A similar groundwater mound existed under the 100-K area trench until the K reactors discontinued operation in January 1971, and disposal of water to the ground ceased. It took 1 to 2 years for the 100-K area mound to disappear after reactor shutdown. It is reasonable to assume that the 100-N area mound will also recede within 1 to 2 years after discharges stop.

2.4.13.3 Accident Effects

Accidental release of hazardous liquid to the ground could result in some of the spilled materials eventually reaching the Columbia River.

During the life of N Reactor, several studies (Ref. 21-25) have been made of the travel times to the Columbia River of routine waste discharges to the 1301-N facility and to an emergency disposal basin. Estimates of minimum travel times from 1301-N to the river determined from numerical modeling, analog modeling and field measurements varied from 5 to 8 days for transport. Transmissivities determined from measured travel times and two pumping tests in the area range from 5500 to 50,000 ft3/ft/day. The most probable transmissivity value for the area is between 6000 and 10,000 ft3/ft/day. Effective porosities assumed for this area vary from 0.02 to 0.3.

These studies (Ref. 21-25) assumed that relatively high flow rates over long time periods had created saturated flow conditions where travel times (routes) are short. The routine release to 1301-N has been terminated. Thus saturated conditions will exist for the 1325-N operational and accident releases. The saturated mound for the 1325-N facility is at a lower level and further distance from the Columbia River. 21

Travel time to the river allows for decay of short half-lived radionuclides, and the ion exchange properties of the soil removes to varying degrees other nuclides from the effluent. Measurements made routinely under the Effluent Monitoring Program for the 1301-N, at its peak mound level indicated decontamination factors from these phenomena of from 1 to 2,500,000 for the radionuclides involved. Table 2.4-4 shows these decontamination factors and measured release data to the 1301-N facility and at the river springs. In addition, the minimum dilution factor of the river is about 8,000 at low river flow. Accidental releases from the 1301-N would be expected to experience the same decontamination factors and similar dilution. 21

21

The calculated annual whole body dose rate to the average individual from radionuclides in past routine liquid release from the 1301-N is 0.00005 mrem compared to the DOE limit of 170 mrem per year. An operational spill could 21

have caused a total accidental dose at the low population boundary zone of .022 rem. Hypothetical large accidental spills and consequences are discussed in Chapters 11.2 and 15.

The 1301-N facility has been retired and replaced by a new facility of similar design (1325-N) for operational releases and an additional facility (1312-N) for accident mitigation releases such as Emergency Core Cooling System releases. The description of these systems, their design basis involving effects of releases is provided in Section 11.2.

2.4.13.4 Monitoring Requirements

Routine discharges of radioactive effluents are routed to the 1325-N facility. Accidental releases would be redirected to the 1312-N Liquid Effluent Retention Facility (LERF). The objective is to allow the absorptive properties of the soil to remove radioactive isotopes from the routine effluent and to contain accident releases. The routine operational effluent will enter the groundwater which is connected to the Columbia River. A monitoring program measures the routine discharge to the 1325-N facility for water quantity and radionuclide concentration. Radionuclides are routinely measured at the river bank. This monitoring program will continue and be intensified by additional sampling for any accidental discharges to either the 1325-N facility or 1312-N Liquid Effluent Retention Facility. In addition, monitoring wells in the vicinity of the 100-N area are monitored on a routine basis for water level and radioactivity measurements. This groundwater monitoring would be intensified for any accidental release to the basin. See Sections 11.2 and 11.5.

Since there are no users or planned users of the groundwater, a system to warn users of accidental releases to groundwater does not exist.

2.4.13.5 Design Bases for Subsurface Hydrostatic Loadings

All foundations except 181-N are above maximum groundwater level; therefore, hydrostatic considerations were not necessary. No dewatering was required for construction except for the 181-N river pump house structure. Design bases for all structures are discussed in Section 3.8.5.

There is no permanent dewatering system in place.

2.4.14 Technical Specification and Emergency Operation Requirements

The PMF does not create an adverse hydrological condition on safety-related facilities with the exception of the 181-N river pump house structure. While this flood would be above the 181-N operating deck, the emergency pump diesel drives are 5 feet above the deck. See Section 2.4.3 for further discussion on the safety of this facility with respect to flooding. | 1

Low river flow rates are limited by regulation to 36,000 cfs. This minimum flow rate allows for normal operation. Should the flow rate fall below this rate, a notification from Grant County PUD and/or the low river level alarm would cause the shutdown of the reactor. Technical Specification 3/4.20 specifies a minimum level of 378 ft, 4 in. to ensure adequate accident flow during plant operation. Low flow perturbations at Priest Rapids Dam are reflected at N Reactor in 1 to 3 hours (Ref. 2.7). Emergency procedures are in place for extreme low river flow rates or a total loss of flow at Priest Rapids Dam which would cause a river level drop below pump intake needs in about 90 minutes. Thus there would be extensive cooldown time before initiating once-through cooling with emergency pumps using stored water. Stored water would provide adequate cooling for a substantial time period which would permit Priest Rapids Dam personnel time to reestablish flow. | 21 | 21

There is a reporting procedure through Bonneville Power Administration for notifying downstream water users of abnormal water release increases at Priest Rapids Dam or any other dam on the Columbia River. This reporting procedure may require up to 10 minutes from incident to reactor scram. For decreases in flow rate from Priest Rapids Dam, the PUD operations notify the 105-N control room by direct phone call when the flow rate drops or is expected to drop below 45,000 cfs. In addition, a low river level alarm at the river gauging station one mile below Priest Rapids Dam will annunciate in the 105-N control room.

The warning procedures for floods and unusual releases allow sufficient time for orderly shutdown and emergency action. Requirements for mandatory shutdown

and securing of the reactor are included in the Technical Specifications. Advance emergency actions are included in the Emergency Procedures, which are directed at securing the reactor plant for the expected duration of high or low river level events.

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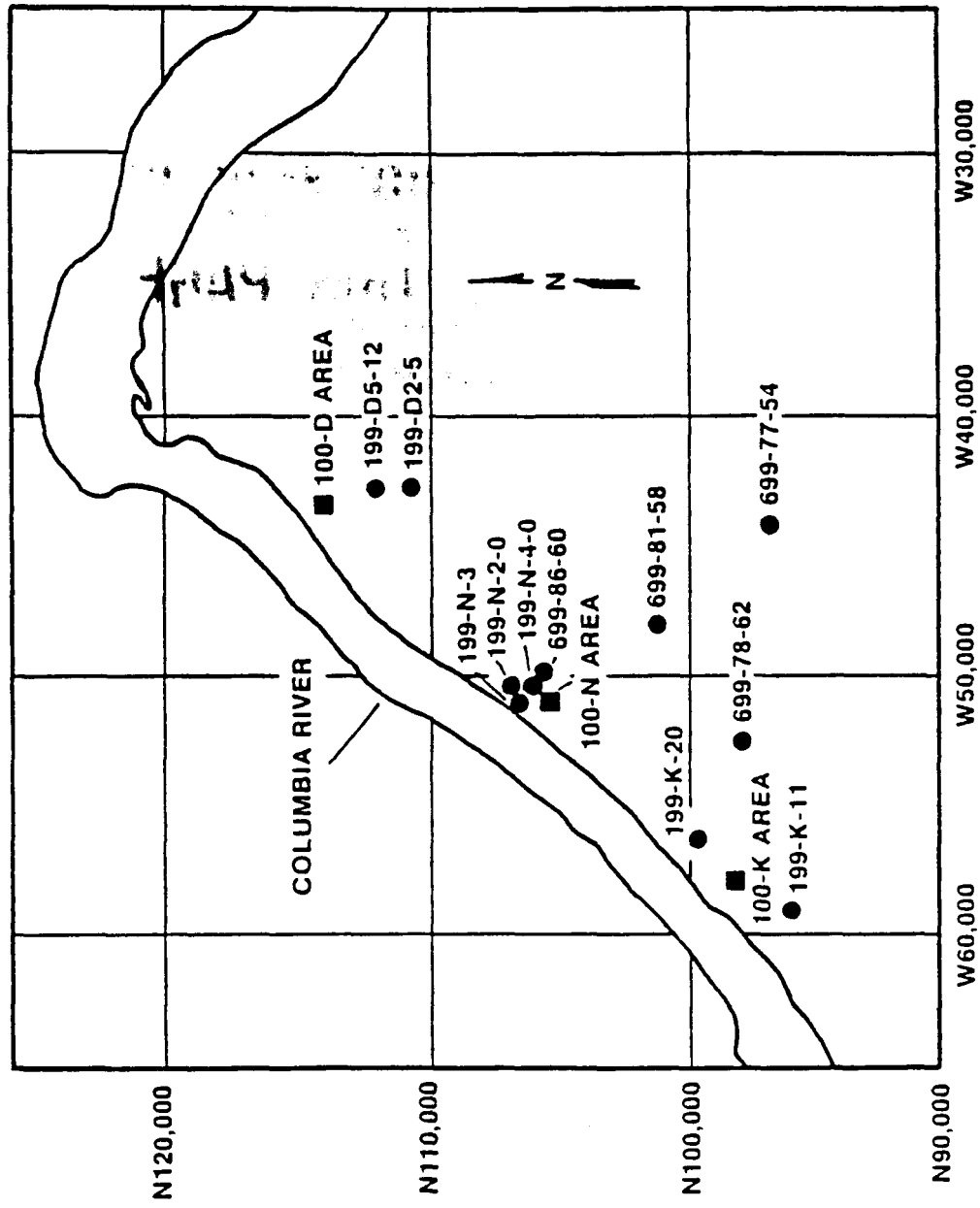
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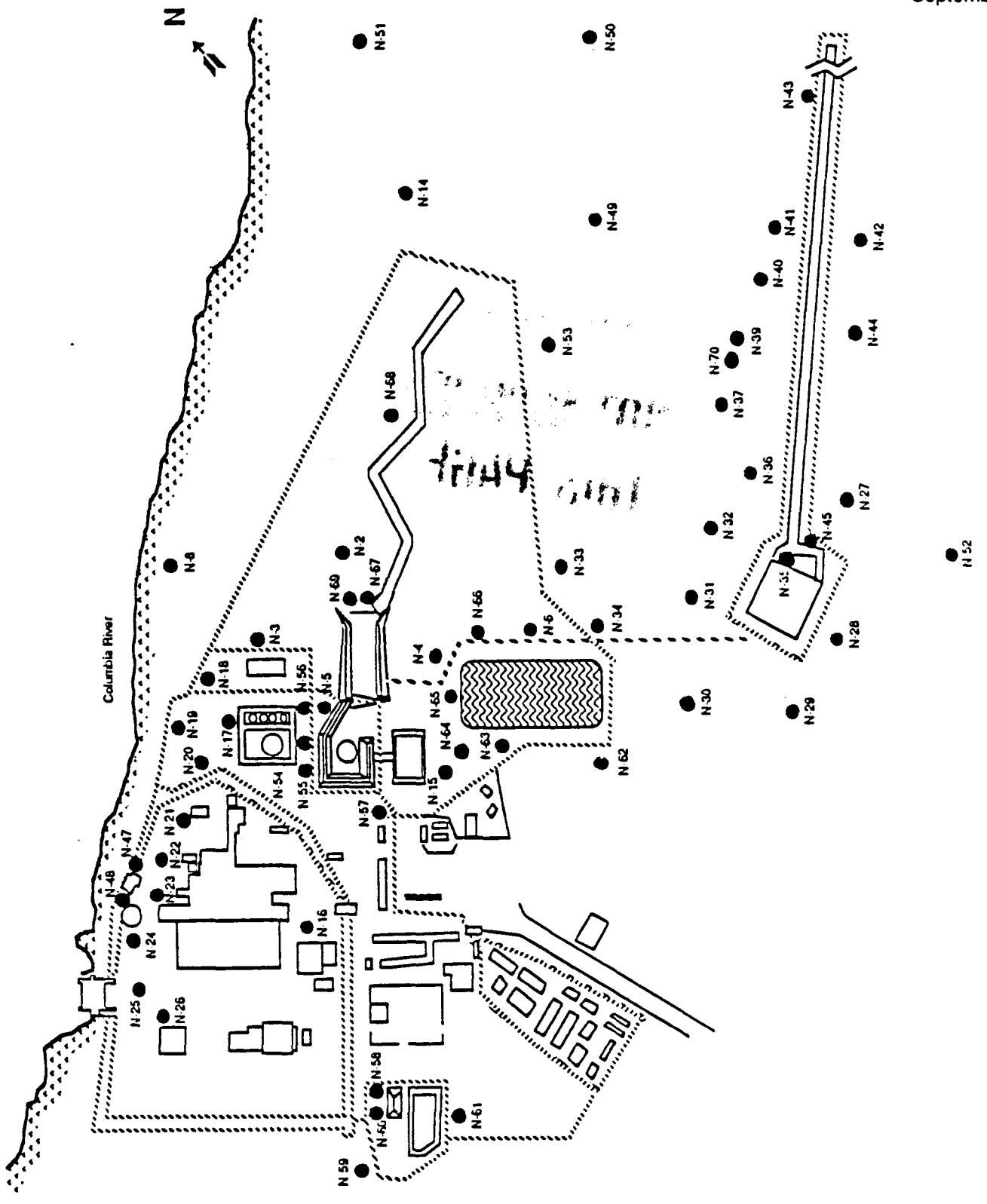


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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	Well Location in Vicinity of N Reactor Site (1975)
	Figure 2.4-17

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Westinghouse Hanford Company N REACTOR SAFETY ANALYSIS REPORT	N Reactor Monitoring Well Locations
	Figure 2.4-19

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APPENDIX 15.6C MELT COMPUTER CODE

Description of Melt Code

The MELT code analyzes the N Reactor fuel temperature transient during RELAP 4 based analyses of postulated single pressure tube events. The code calculates fuel temperature, Zircaloy oxide thickness, hydrogen generation, fraction of fuel cladding failed, and the fraction of uranium which melts. 21

Method of Solution

The MELT code uses a forward, explicitly-finite, differencing, transient, thermal analysis. At the beginning of the time step, the net heat transfer into each node is determined. Added to this heat flow is fission and decay nuclear heating and chemical heat due to the aqueous reaction of the Zircaloy-2. The summation of all heat flux into the node is multiplied by the time step and divided by the thermal capacitance to determine the increase in the node temperature.

The thermal capacitance calculations include the sensible heat and the latent heat of the α-β and β-γ transformation as well as latent heat of melting.

Conduction

The thermal resistance from the center of the inner fuel to the surface is calculated as follows:

$$R = \ln \left(\frac{MD_u}{ID_u} \right) / 2\pi \ K_u L + \ln \left(\frac{ID_u}{ID_z} \right) / 2\pi \ K_z L$$

where MD is the diameter based on the center of gravity of the cross-section of uranium.

$$MD = SQRT \ \ 0.5(ID^2 + OD^2)$$

The thermal resistance from the inner and outer portions of the outer fuel are calculated similarly using the proper diameters.

The thermal resistance from the center of the tube block to the outer surface is:

$$R = \ln (1.08 \frac{b}{CD})/2\pi K_g L$$

where:

R = Thermal resistance, Btu/hr-°F
L = Length of section, ft
K_g= Thermal conductivity of graphite, Btu/hr-ft-°F
b = Base width, ft
ID= Diameter of inner hole, ft
CD= Central diameter of block, ft

While the thermal resistance from the center of the tube block to the surface is:

$$R = \frac{x}{A K_g}$$

where:

x = conduction path distance, ft
A = cross-sectional area of conduction path, ft^2

The small dimensions of the gas gaps, between the fuel elements and the pressure tube and tube block, combined with the fact that the tube is horizontal, results in negligible natural convection. The heat transfer in these regions is considered to be pure conduction. The conductivity of the steam is computed from the expression:

$$k = 8.64 \times 10^{-4} (1 + 3.778 \times 10^{-4} T)(T)^{.5}/(1.0 + 6.93.7/T)$$

The conductance of the helium gap is:

$$C_{He} = kA/\Delta x = 22.8343 + 2.3683 \cdot \left(\frac{T}{100}\right) - .001404 \cdot \left(\frac{T}{100}\right)^3 \text{ Btu/hr/°F}$$

This expression accounts for the temperature dependence of the helium thermal conductivity.

where:

T is the mean gap temperature, and

Δx is the mean gap width.

The hot diameters of the fuel elements, pressure tube and graphite tube block are calculated based on the mean temperatures. In the fuel element it is assumed that the Zircaloy cladding has a negligible effect on the total thermal expansion. Therefore, the expansion of the fuel is based entirely on the uranium.

$$D' = D \quad (1.0 + \alpha\,(T-70))$$

where:

D' = hot diameter, inches

D = input diameter at 70°F, inches

α = mean coefficient of linear expansion, inch/inch-°F

T = temperature, °F

The heat transfer across the gaps is based on the hot diameters rather than the input diameters.

During the first seconds of the LOCA transient, the water in the tube boils, after which some of the expanding water in the cold-leg riser enters the tube and is converted to steam. This process results in heat transfer between the steam and the surfaces integral to the pressure tube. As the steam flows through the tube, the heat transferred by convection is not directly transferred to the adjacent surface as is the case in conduction across

the gap. MELT vectorally adds the heat loss to the steam as calculated in RELAP4 from each affected node. Either the heat transfer rate vs time or an input value of the film coefficient and steam temperature may be input to calculate the heat transfer. This is a user option.

Another mode of heat transfer across the gaps is thermal radiation:

$$Q_r = F\ A_i\ \sigma\ (T_i^4 - T_o^4)\ \text{Btu/hr} \qquad \text{Ref. (1)}$$

where:

Q_r = Rate of heat transfer by radiation from the inner surface to the outer surface, Btu/hr

F = Gray-body shape factor based upon the area of the inner surface Dimensionless

A_i = Area of the inner surface, ft^2

σ = Stefan-Boltzman constant $.1714(10^{-8})$ $Btu/hr/ft^2/R^4$

T_i = Temperature of the inner surface, °R

T_o = Temperature of the outer surface, °R

The emissivity of the graphite, which is included in the grey body shape factor, is assumed to be 0.8, and the emissivity of the Zircaloy-2 ranges from 0.66 at 1472°F to 0.800 at 2372°F (Ref.2).

This forms the thermal resistance integral to the tube block. The resistances external to the tube block are calculated and input by the user.

Capacitance

The thermal capacitance of all the nodes are based on the volumetric mass times specific heat of the material.

$$C = M\,C_p$$

The code provides three options:

1. normal capacitive block where:

$$T = T + \Delta t \quad \frac{\Sigma Q}{C}$$

2. surface node having no thermal capacitance where:

$$T = \frac{\Sigma(Tn \cdot 1/R)}{\Sigma 1/R}$$

3. constant temperature nodes where the temperature remains constant at the initial input value.

Node Internal Heat

The internal heat of the nodes originates from two sources.

1. Internal Nuclear Heat Generation or Decay Heat

   Separate nuclear power generation decay relationships are used for each region. It is assumed that the heat decay rate is uniform throughout the graphite adjacent to the pressure tube. Power decay data are of the form

$$P = \exp(A\,\ell n\,\theta + B) \quad \cdot$$

   where:

   P = normalized specific power, Btu/sec
   θ = time from scram, seconds
   A = constant
   B = constant

The power decay functions have been normalized by dividing the power per foot in a given region by the total lattice power prior to shutdown. Power generation for a region at a given time is the power decay function multiplied by the initial lattice power.

2. Chemical Power Generation

Oxidation of zirconium by steam is assumed to take place on zircaloy surfaces inside the pressure tube. The outside of the pressure tube is assumed free of chemical reaction. The reaction products are described by the following chemical balances:

$$2H_2O \rightarrow 2H_2 + O_2 + 2\ (56.93) \qquad \text{kcal/gm-mole-}H_2O$$
$$Zr+2H_2O \rightarrow ZrO_2 + 2H_2 + 261.5 \qquad \text{kcal/gm-mole-Zr}$$

Combining these balances:

$$Zr+2H_2O \rightarrow ZrO_2 + 2H_2 + 147.6 \quad \text{kcal/gm-mole-Zr}$$

or

$$Zr+2H_2O \rightarrow ZrO_2 + 2H_2 + 265{,}700 \text{ Btu/lb-mole-Zr}$$

The equation for chemical power generation used in MELT has the form:

$$P_c^1 = 69560 \cdot D \cdot \dot{W}_Z \text{ Btu/inch/foot/pound-mole-Zr}$$

where:

P_c^1 = Rate of chemical heat release per foot axial foot, Btu/ft

D = Sum of Zircaloy surface diameters for the component in question (target, driver or pressure tube). Outer surface of pressure tube is helium blanketed and assumed not to react

$\dot{W}_Z$ = Rate of reaction, lb-mole-Zr/ft^2/sec

The rate of reaction is determined from the parabolic rate law of Baker and Just (Ref. 3) for a constant temperature reaction.

$$W^2/t = 33.3(10^6) \exp(-E^*/RT)$$

A dimensional conversion yields the relation used in MELT.

$$\dot{W} = 0.00835(1/W) \exp -413.5/(T/100)$$

where:

W = Weight of Zircaloy reacted per unit area, mg/cm^2
$\dot{W}$ = $lb/moles/ft^2$-sec
E^* = Activation energy
= 45,500 gm-cal/gm-mole
R = Perfect gas constant
= 1.1 gm-cal/gm-mole - °R
T = Surface Temperature, °R
t = Time from beginning of reaction, minutes

Energy Storage

Energy storage terms appear in the equations of temperature change as products of mass and specific heat. Storage terms for all solid and liquid phases are handled in this fashion. The density and lattice dimensions are referenced to room temperature to provide a consistent calculation of mass. Energy absorbed in phase transformation is accounted for by assigning a small temperature interval to the transformation and treating the energy absorption mechanism as a specific heat, thus eliminating the requirement for mass balances. The effective specific heat for a phase transformation is determined from the following relationship:

$$(C_p) \text{ effective} = H/\Delta T$$

where:

H = Heat of transformation per unit mass, Btu/lb_m
ΔT = Temperature interval assigned to the transformation

Heat capacities are evaluated individually for the phases tabulated below:

Material	Phase	Temperature Range °F
Zircaloy-2	α	Below 1580
	α-β	1580-1590
	β	Above 1590
Uranium	α	Below 1242
	α-β	Tranformation 1242-1252
	β	1252-1420
	γ	Molten Transformation 1994°F - 2034°F
	γ - liquid	Above 2034
Graphite Moderator	Solid	All temperatures of interest

APPENDIX 15.6D
RELAP 5 MODELS FOR ANALYSIS OF THE N REACTOR LIMITING LOCA

Introduction

This Appendix provides a summary description of the average and hot channel RELAP5 models used for N Reactor application in the limiting LOCA reanalyses discussed in Section 15.6.5.3.4. A detailed, component-by-component description of the models, and the changes to RELAP5 necessary to support horizontal core reflood analysis, are provided in Reference 1.

In the N Reactor, coolant is routed to the 1003 horizontal pressure tubes via inlet riser standpipes. Each inlet riser is 15.75 in. inside diameter, and is connected to approximately 63 process tubes (two columns). During normal operation, coolant enters the sixteen risers from the cold leg manifold. After leaving the manifold, the coolant flows up a vertical section of the riser piping, across the inlet crossover piping, through the CV-5 check valves, into the inlet riser piping, and through the pressure tubes. 21

During a LOCA, ECCS signals which open the V-3 valves are generated that initiate ECCS coolant flow. ECCS coolant enters each of the inlet risers at the bottom, through the four-inch diameter CV-2R check valves. The coolant then flows through the pressure tubes and out through the V-4 valves to the dump line.

Analytical Models

RELAP5 simulates thermal-hydraulic systems through the use of generic component models from which general systems can be simulated. These models represent such elements as pumps, valves, pipes, heat structures, turbines, steam separators, and control systems. As was also true for RELAP4, a very large number of nodes or components, limited only by economic considerations associated with the computer memory and running time required for the simulation, can be used in a problem. RELAP5 also includes special process models that consider such effects as flow form loss, flow branching, choked flow, boron tracking, and the presence of noncondensible gas.

Average Channel RELAP5 System Model

The average channel RELAP5 system model represents the reactor core, primary coolant loop and associated systems [e.g., high pressure injection (HPI), emergency core cooling (ECCS)] and the steam generators. The primary side model consists of the reactor core, hot and cold leg piping, pressurizer, HPI system, letdown system, ECCS, and depressurization system (i.e., V-4 valves).

The N Reactor contains 6 steam generator "cells", each consisting of hot and cold leg piping, one primary coolant pump, and two steam generators. One of the 6 steam generator cells is normally out of service. Therefore, during normal operation, there are 10 active steam generators. The secondary side model lumps the 10 steam generators together into a single entity.

Figure 15.6-177 is a pictorial, schematic layout of the N Reactor on which the model is based. Figures 15.6-176, 15.6-212, and 15.6-213 shows the node arrangement of the RELAP5 N Reactor average channel LOCA model. Figure 15.6-176 shows the layout of the primary coolant and steam generator systems. The core node arrangement shown in Figure 15.6-213 is inserted into the Figure 15.6-176 coolant loop node scheme. Component numbering is shown on these figures as three-digit numbers.

If a component has multiple volumes, it is indicated by separate boxes and single digits (e.g., the recirculation pump discharge piping Component 154 is divided into 4 volumes). Heat structures associated with components are shown as "dotted" or shaded areas adjacent to the component. There may be more than one heat structure per component (e.g., the cold leg crossover piping Component 158 has 3 heat structures). The average channel LOCA model shown in Figures 15.6-176 and 15.6-213 has a total of 242 volumes, 271 junctions, and 192 heat structures.

The model illustrated by Figure 15.6-213 shows the detailed node arrangement of the core region, which includes the vertical inlet and outlet risers, the horizontal inlet and outlet connector piping, the inlet and outlet spacer sections, and the core section of the pressure tubes.

The core was modeled as eight separate flow channels, each of which represents an equal-height subdivision of four horizontal rows of process tubes in the reactor core. Channels are numbered 1 through 8, where Channel 1 refers to the top four rows and Channel 8 represents the bottom four rows of pressure tubes.

Within each of the eight core channels, three components model the fueled region of the core. These three components represent the coolant flow paths through the pressure tubes and the tube-in-tube fuel elements. Heat structures representing the pressure tube/graphite arrangement, and the inner and outer fuel elements are modeled.

The pressure tube heat structure represents the pressure tube, the graphite-coolant gas gap, and the graphite. The fuel element heat structures represent the inner and outer Zircaloy cladding and the metallic-uranium fuel. Heat structures associated with the inlet/outlet connectors and the inlet/outlet spacers represent piping walls and the unheated graphite reflector regions. Heat structures associated with the connectors, and inlet and outlet riser pipes represent the piping walls.

The model in Figure 15.6-176 shows the node arrangement for the loop piping, which includes the hot and cold legs, steam generator U-tubes, the steam generator bypass, the pressurizer, the HPI system, and the ECCS. Three components are modeled that represent the sixteen core outlet crossover pipes from the core outlet risers to the hot leg manifold. The CV-3 check valves are also modeled. Heat structures associated with these and other subsequent loop piping components represent the pipe walls.

The hot leg manifold is a component, and the associated heat structure represents the manifold shell. The piping and steam generators from the five operating cells are lumped into a single flow path in the RELAP5 model. Two components represent the lumped hot leg piping from the hot leg manifold to the steam generator inlets. Six components represent the steam generator inlet, U-tube, and steam generator outlet regions of the primary coolant system. Two-sided heat structures represent the steam generator U-tube walls.

The steam generator bypass path is modeled as two components. One component models the 5 primary coolant pumps, which are modeled actively (i.e., pump performance is calculated using homologous curves, and is based on pump speed, fluid conditions, and the status of the driving motor). One component models the makeup to the primary system due to pump seal leakage. Two components represent the cold leg manifold and the cold leg crossover piping leading to the core inlet risers. The CV-5 check valves are modeled between the inlet piping and the core inlet risers. The tops of the inlet risers are also shown in Figure 15.6-213.

Two components represent the pressurizer and the surge line. The cycling spray line from the cold leg manifold to the pressurizer is modeled, as well as the associated pressurizer spray valve. Heat structures are used to represent the walls of the pressurizer and the pressurizer heaters.

The HPI system modeling is shown in the lower left of Figure 15.6-176. HPI pumps 1 and 2, and HPI pumps 3 and 4 are modeled, respectively, as one component. These pumps are actively modeled (i.e., pump performance is calculated based on homologous curves and fluid conditions). The HPI supply and the pump suction piping are represented by two components. The HPI injection line is modeled as three components. Primary coolant pump seal injection flow is extracted from the HPI line by a time-dependent junction. The cold spray line from the HPI pump discharge to the pressurizer, and the pressurizer spray valves that control flow from the HPI system are modeled. The IWV-287, -290 and -253 valves, and the check valves where the HPI system injects into the primary coolant pump suction are modeled.

The spill valves on the cold leg manifold are modeled as three components. Letdown is modeled as a time dependent junction. The volumes that these flow paths discharge into are represented by three components. The ECCS is modeled and is activated by an ECCS trip, which causes ECCS fluid to be pumped into the bottoms of the inlet risers. The ECCS rate is a function of inlet riser pressure.

The emergency dump system is modeled to represent the V-4 valves and piping. The high pressure relief system is modeled to represent the RV-2 valves and associated piping. Both of these systems route flow from the bottom of the core outlet downcomer to the emergency dump tank; this path being modeled by four components. Code safety valves on the pressurizer and the hot legs are represented, and connect to flow paths that discharge to the volumes represented by two components. Check valves in the primary coolant and HPI pump discharge lines are represented.

Details of the steam generator region node arrangement is shown in the upper left of Figure 15.6-212. A description of steam generator primary side modeling was previously given. On the secondary side, feedwater enters the bottom of the steam generators. Feedwater temperature is defined, and the feedwater flow rate is controlled based on the indicated steam generator level. One component represents the feedwater inlet regions on the bottoms of the steam generators. Five components represent the boiler regions of the steam generators.

Heat from the primary coolant system is transferred across the U-tubes and enters the secondary system in these components. Four components represent the outer regions of the steam generator secondary side. The boiler and outer regions of the secondary system are coupled with cross-flow junctions at each level. One component represents the steam generator separator region.

In the RELAP5 model, an ideal separation process is modeled (i.e., 100% or complete separation). All liquid entering the separator from the boiler is returned to the outer region and all steam is passed to the steam line. During power operations, a recirculation is established with fluid flowing upward through the boiler region and liquid flowing downward in the outer region. Because of the cross-flow paths between boiler and outer regions, the recirculation is non-uniform.

Portions of the downward flow in the outer region is returned to the boiler region at each of the elevations where the two regions are coupled. The steam lines are represented, as well as the steam stop valves. One component represents the steam pressure condition downstream of the steam stop valves.

In addition to the U-tubes, heat structures are used in the model to represent the piping walls of the steam lines and the cylindrical shell of the horizontal steam generators.

The RELAP5 model of the N Reactor includes simulation of the following five major plant control systems: pressurizer pressure control, pressurizer level control, HPI pump control, steam generator pressure control, and steam generator feedwater control. In addition, steady-state controllers are incorporated to establish initial plant conditions at their desired steady-state values.

The steady-state plant controllers used were a pump speed controller (to attain desired core mass flow), a primary system pressure controller (to attain desired pressurizer pressure), and a secondary system pressure controller (to attain the desired steam generator heat removal rate).

Hot Channel RELAP5 Model

The hot channel RELAP5 model (Figure 15.6-288) represents either a 5000 kw Mark IV-fueled pressure tube with a standard fuel column of sixteen fuel elements, 26.1 inches long, or a 4700 kw Mark IA-fueled pressure tube with a fuel column of 19 elements. A Mark IA-fueled pressure tube consists of two 26.1 inch long standard Mark IV fuel elements at each end, with fifteen 20.88 inch long standard Mark IA fuel elements between them.

The maximum lattice power, represented as 5000 KW, is based on a reactor nominal full power level of 4000 MWt with a conservative Hot Channel Factor of 1.25 (see Figure 4.4-25). 4700 kw represents 94% of the reactor nominal full power with the hot channel factor of 1.25 applied. These differences are discussed in detail in Section 15.6.5.1.2.

The hot channel analyses are based on the approach used for LOCA hot channel analysis of the Vermont Yankee BWR plant (Ref. 2). The hot channel models are very similar to the average channel system model of a pressure tube shown in Figure 15.6-213, except that each fuel element is individually modelled, so the active hot channel zone is comprised of 16, rather than 3, nodes as discussed in Reference 3.

The three flow channels associated with each fuel element are also individually modeled. Hydrodynamic conditions external to the hot channel pressure tube and associated elements are modeled as time-dependent boundary conditions, which are derived from RELAP5 average channel LOCA analysis.

The physical elements considered in the hot channel model are also shown schematically in Figure 15.6-288. Heat transfer occurs from the inner and outer fuel elements (which represent Zircaloy cladding and metallic uranium) to the inner/outer annuli and center hole flow paths. Heat is also transferred across the process tube, through the graphite coolant-gas gap, and to the graphite. 21

GSCS cooling is considered as heat transfer to a constant temperature heat sink using a constant heat transfer coefficient. No GSCS cooling is assumed at the front and rear of the active zone graphite (i.e., for the first two and last two fuel regions). In RELAP5, heat flow is calculated on a one-dimensional basis, and radiative heat transfer is not explicitly modeled.

The RELAP5 node configuration for the hot channel model is shown in Figure 15.6-288. The same convention, previously explained for the average channel RELAP5 model, is used for numbering components, heat structures, and volumes. Heat structures are again illustrated by shaded areas.

LOCA blowdown and reflood hydrodynamic conditions are simulated in this RELAP5 model by specifying time-dependent boundary conditions in terms of inlet and outlet riser pressures. Heat and fluid flow conditions within the hot channel are then computed by RELAP5 based on the local thermal and

hydrodynamic conditions appropriate for the inner/outer fuel element, pressure tube and graphite arrangement.

REFERENCES - 15.6D

1. EGG-TFM-7891, "Simulation of Cold Leg Manifold Break and Station Blackout Sequences in the N Reactor," M.A. Bolander and C.D. Fletcher, dated February 1988.

2. YAEC-1547, "Vermont Yankee BWR Loss-of-Coolant Accident Licensing Analysis Method," Dr. R. Thomas Fernandez and Dr. Hugo C. DaSilva, Jr.

3. WHC-SP-0285, "Evaluation Model Analysis of Cold Leg Crossover Break for N Reactor," B.E. Schmitt and D.M. Ogden, dated May 1988.

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