WHC-SP-0001

N Reactor Limited-Scope Probabilistic Risk Assessment

DO NOT MICROFILM
COVER

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

Westinghouse
Hanford Company Richland, Washington

WHC-SP--0001

DE90 009493

N Reactor Limited-Scope Probabilistic Risk Assessment

Westinghouse Hanford Company

Science Applications International Corporation

Date Published
July 1987

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

Westinghouse Hanford Company

P.O. Box 1970
Richland, Washington 99352

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

ACKNOWLEDGMENTS

This document had many contributors who provided written input; management and coordination; and consultation, review, and expert opinion.

Project Management

D. H. Walker, Project Manager
Science Applications International Corporation (SAIC)

Principal Contributors

Science Applications International Corporation

D. C. Aabye
A. M. Kolaczkowski
C. L. Mason
Z. T. Mendoza

Westinghouse Hanford Company

E. E. Leitz
M. D. Zentner

The following is a list of individuals who, in addition to the principal contributors, contributed to document development.

Westinghouse Hanford Company

M. J. Anderson
G. R. Armstrong
K. R. Birney
F. G. Buck
P. A. Carlson
G. A. Coles
F. C Han
J. T. Long
K. R. Mikkelsen
R. D. Morrison
L. D. Muhlestein
D. M. Ogden
P. R. Patterson
T. B. Powers
W. J. Quapp
S. P. Roblyer
R. C. Ross
H. Toffer
R. E. Whiteside
A. D. Wilcox

Science Applications International Corporation

J. E. Kelly
J. R. Penland
B. J. Shapiro
National Technical Systems (Subcontractor)

Consultants/Reviewer

P. R. Davis (PRD Consulting)
A. L. Camp (Sandia National Laboratories)
D. M. Kunsman (Sandia National Laboratories)
N. J. McCormick (University of Washington)
D. F. Paddleford (Westinghouse - Pittsburgh)
B. D. Sloan (Westinghouse - Pittsburgh)

The technical drafts of the document were assembled and largely written by SAIC. Final editing and document production were performed by WHC.

PREFACE

This report has been prepared as part of the N Reactor Accelerated Safety Enhancement Program to support review and assessment of the N Reactor Confinement System ability to protect public health and safety during a reactor accident.

Companion reports to this volume are:

UNC, 1987, N Reactor Confinement System Review, UNI-4420, Impell Corporation, Los Alamos Technical Associates, Inc., Mollerus Engineering Corporation, and UNC Nuclear Industries, Inc., Richland, Washington.

WHC, 1987, Executive Summary, N Reactor Confinement System Review and Limited-Scope Probabilistic Safety Assessment, WHC-SP-0003, Westinghouse Hanford Company, Richland, Washington

N REACTOR LIMITED-SCOPE PROBABILISTIC RISK ASSESSMENT

CONTENTS

CONTENTS (Cont.)

FIGURES

TABLES

1.0 SUMMARY

This document reports the results of a Limited-Scope Probabilistic Risk Assessment (PRA) of N Reactor operation. The study has a three-fold purpose:

- o Support the overall N Reactor Confinement Effectiveness Study intended to provide a basis for comparison between a standard nuclear plant with containment and the N Reactor with confinement

- o Provide early and conservative estimates of radiological risk from operation of N Reactor, using the best information available currently to assist decision makers as they examine questions pertaining to continued N Reactor operation

- o Identify the accident sequences that are principal contributors to radiological risk so that resources for the current full-scope PRA can be focused on analysis of these sequences.

N Reactor is a unique graphite-moderated, horizontal pressure-tube, pressurized light-water-cooled reactor designed and operated for co-production of special nuclear materials and up to 860 MW of electrical power. This reactor, which has a National Defense mission, has not always been measured against commercial standards. Current studies attest to the safety of continued N Reactor operation with additional confirmation provided by this Limited-Scope PRA.

The Limited-Scope PRA used the basic framework of a Level 3 PRA.[1] A brief description of PRA methodology is provided in Section 2.0. Simplified event trees and fault trees are used to model accident progression paths, and plant and confinement systems responses. Existing information from commercial reactor PRAs is used, as well as information available from past and present N Reactor studies in developing and quantifying the trees. The preliminary models and data from the Level 1 PRA were utilized, as updated in July 1987. Radionuclide releases to the environment are estimated, based largely on information developed during commercial reactor safety studies. Radiological consequences and risks are calculated with the CRAC2 computer code developed for the U.S. Nuclear Regulatory Commission (NRC).

There are some limitations associated with a limited-scope study of this type that contribute to the uncertainty of the results. Though the uncertainty was not quantified, the results are intended to be conservative and appropriate for fulfilling the study objectives.

These limitations are:

- o Common-mode failures within a system were modeled even though common-mode failures involving more than one system are not accounted for generally; an effort was made to identify common-mode failures involving multiple systems due to external events

- o Operator actions to recover from equipment failure are not considered; this adds conservatism to the analyses

- o The response of structures and components to seismic events is based on limited, best-estimate evaluations of fragility parameters

- o The seismic hazard curve is based on evaluation of data used for the WPPSS Nuclear Plant-1 and -2 seismic hazard study; it does not take advantage of a potential favorable ground structure at N Reactor

- o Aging of plant components is not considered in developing failure rate data; the effect of this limitation is judged to be minor, due in part to numerous plant upgrade programs, both past and ongoing.

The Limited-Scope PRA risk results for N Reactor are presented in Figures 1-1 and 1-2, respectively. Seismic events are the dominant accident initiators leading to fuel damage. No early fatalities are predicted for credible accident sequences leading to fuel damage at N Reactor. Calculated latent fatalities are low compared to those calculated in WASH-1400[2] and in the Indian Point-3 PRA.[3] A comparison is made to Indian Point because it is the only nuclear power plant for which an NRC public hearing board based its findings on a PRA. In the Indian Point public hearing, it was concluded that the risk to the public health and safety is acceptably low for plant operation to continue. The lower radiological risks calculated for N Reactor result from the combination of plant design features and the remote location of the plant.

Table 1-1 compares N Reactor risk and fuel damage frequency values with the NRC safety goals.[4,5] The calculated values are well within the safety goals.

A bounding analysis was performed in which the entire core inventory of volatile fission products (noble gases, iodine, cesium, tellurium) and 1% of other species was assumed to be released to the environment. Risk was estimated by associating a frequency to the bounding release equal to the total calculated frequency of 100% fuel damage (4 E-05/yr). This frequency was selected to ensure conservatism rather than representing a realistic estimate for a release of this magnitude. The results of these bounding risk calculations are shown as Figures 1-3 and 1-4 for early and latent fatalities, respectively. Even with bounding releases assumed, the estimated N Reactor early fatality risks at higher consequence levels are lower than those for the WASH-1400 and Indian Point-3 plants. Latent fatality risks are comparable to the WASH-1400 plants and are lower than Indian Point-3. Neither WASH-1400 results nor Indian Point results include the assumption of a similar bounding radionuclide release.

The bounding analysis provides a reasonable upper bound limit for the risk from N Reactor operation. This, in turn, bounds the potential effects of study limitations such as limited consideration of common mode failures.

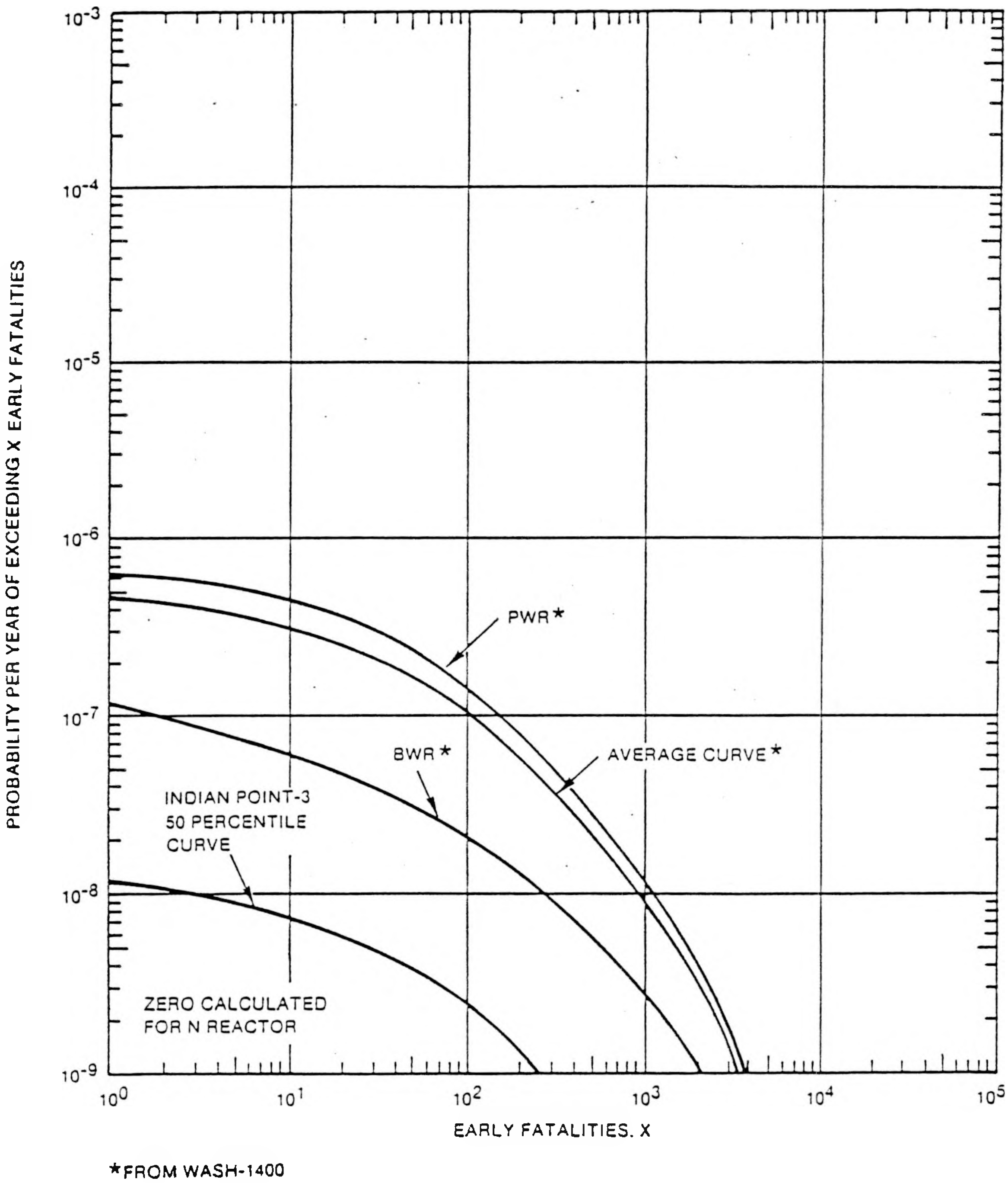


Figure 1-1. Comparative Risk for Early Fatalities

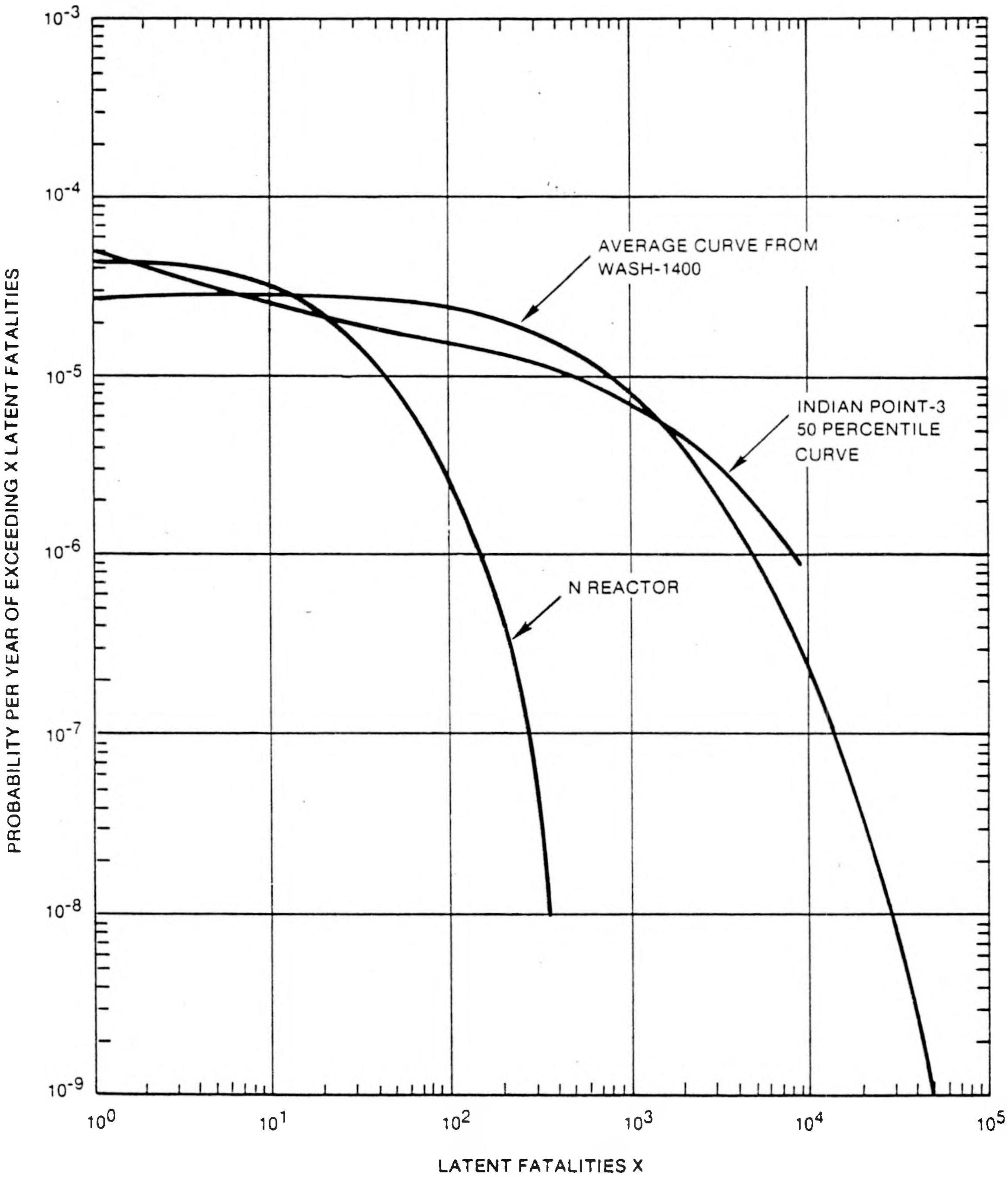


Figure 1-2. Comparative Risk for Latent Fatalities

Table 1-1. Comparison of N Reactor Limited-Scope Results with Nuclear Regulatory Commission Safety Goals.

Goals	Quantified design objective	N Reactor result
Primary Safety Goal[1]		
Prompt fatality risk in reactor vicinity	Limit increase in individual annual risk of accidental death to an increment of 4 E-07/yr. Apply to circular area within 1 mi from plant boundary.	No early fatalities outside plant boundary.
Latent fatality risks in reactor vicinity	Limit increase in individual annual risk of cancer death to an increment of no more than 2 E-06/yr. Apply to offsite population within 10 mi from plant.	3 E-11/yr
Secondary Draft Safety Goal[2]		
Likelihood of large-scale core melt	Likelihood of large-scale core melt should be less than 1 E-04/yr.	7 E-05/yr[3]

[1]From Reference 5
[2]From Reference 4
[3]Total calculated frequency of $\geq$30% fuel damage

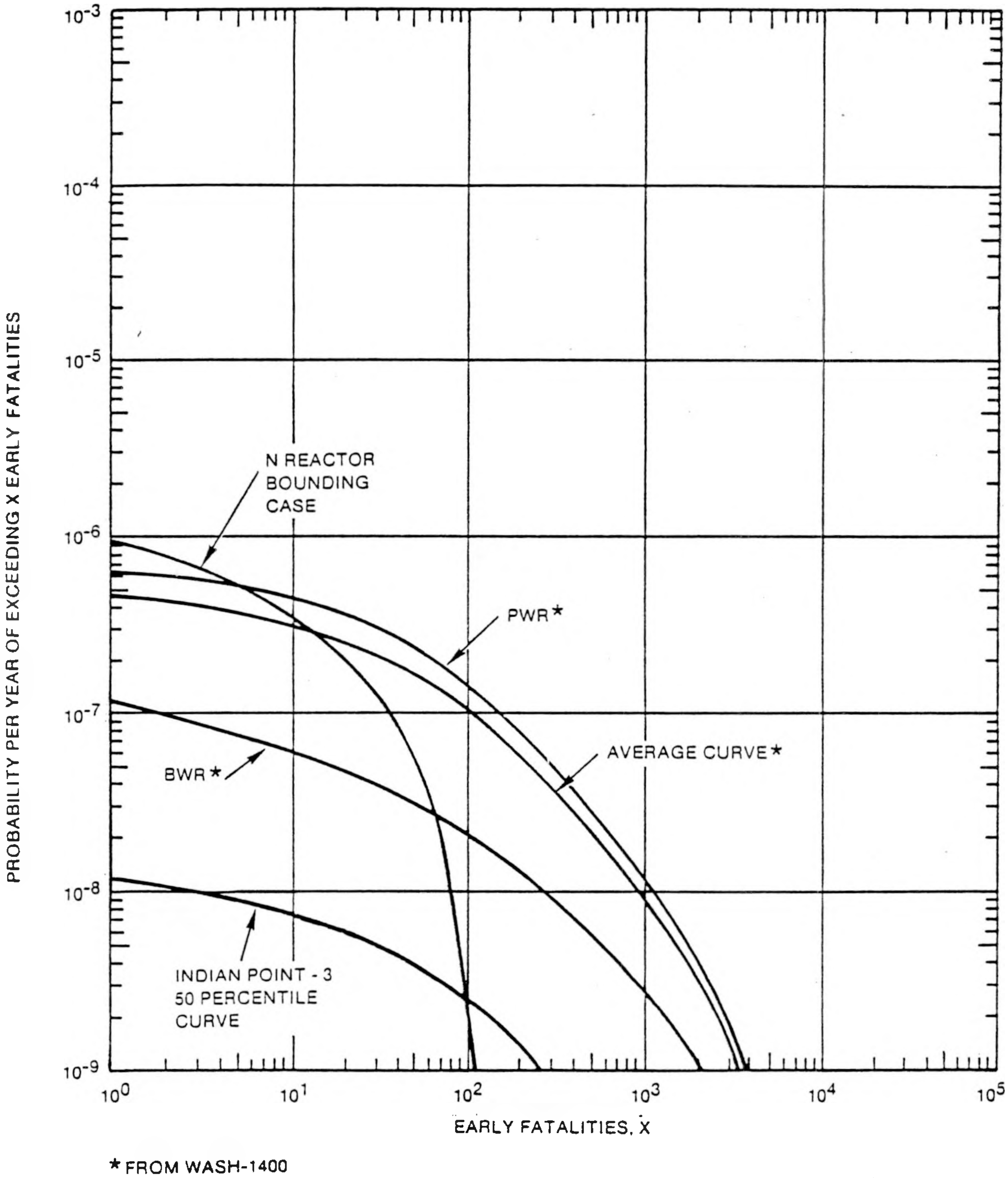


Figure 1-3. Comparative Risk for Early Fatalities - Maximum Bounding Release

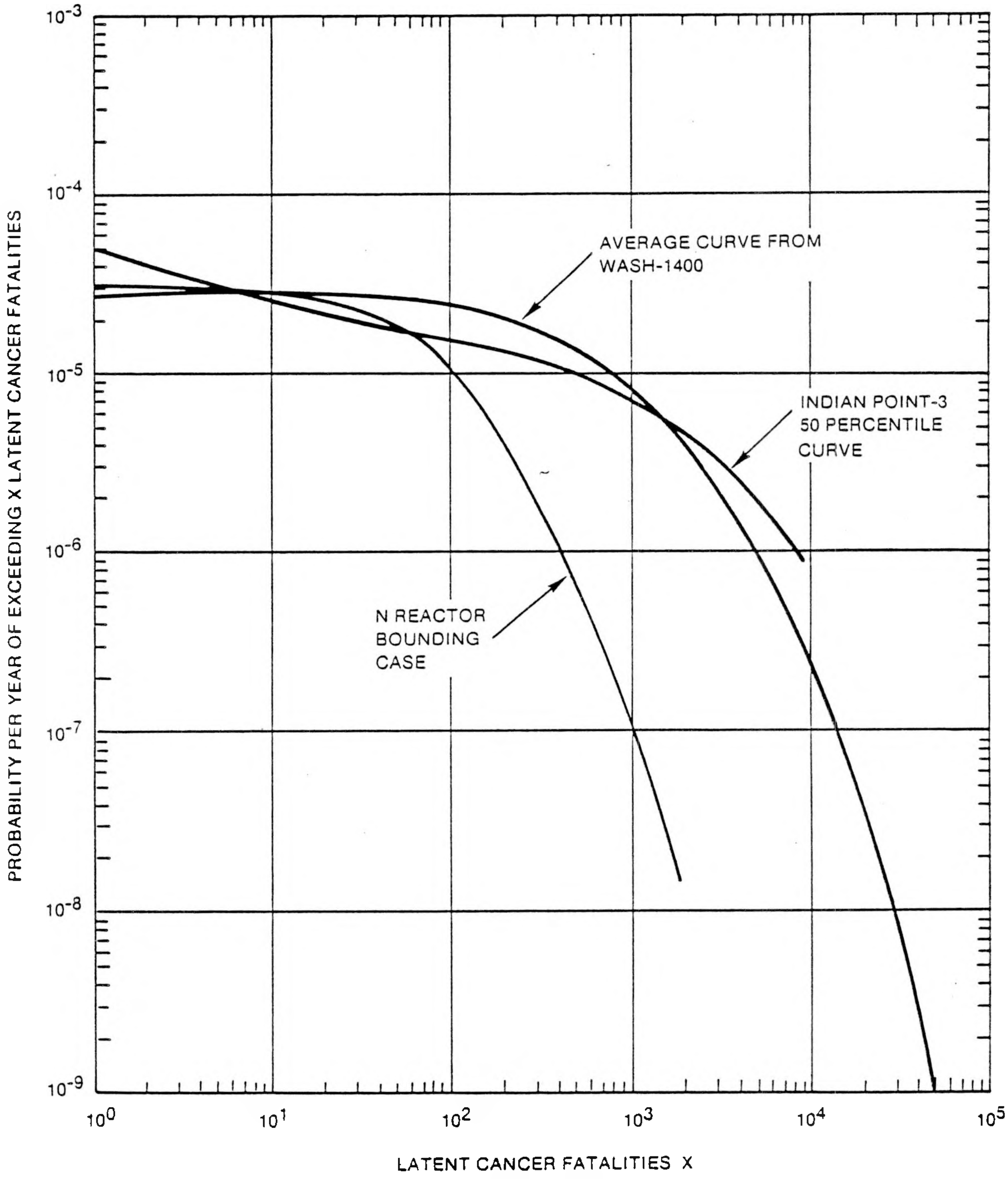


Figure 1-4. Comparative Risk for Latent Fatalities - Maximum Bounding Release

1.1 References

1. NRC, 1983, PRA Procedures Guide, NUREG/CR-2300, Nuclear Regulatory Commission, Washington, D.C.

2. NRC, 1975, Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, Nuclear Regulatory Commission, Washington, D.C.

3. Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., 1982, Indian Point Probabilistic Safety Study, New York, New York.

4. NRC, 1983, Safety Goals for Operation of Nuclear Power Plants, Policy Statement, Nuclear Regulatory Commission, Washington, D.C.

5. NRC, 1983, Safety Goals for Nuclear Power Plant Operations, NUREG-0880, Rev 1, for comment, Nuclear Regulatory Commission, Washington D.C.

2.0 INTRODUCTION

N Reactor is a graphite-moderated, pressure-tube reactor, designed and operated for both production of special nuclear materials and for the production of up to 860 MW of electric power. The fuel is low enrichment uranium and the coolant is pressurized light-water.

N Reactor is the only large water-cooled, graphite-moderated reactor in operation in the U.S. In April 1986, there was an accident at the Chernobyl nuclear power station in the United Soviet Socialist Republic (USSR) that resulted in severe core damage and a large fission product release to the atmosphere. The reactor involved in the accident was also a graphite-moderated, water-cooled, pressure-tube reactor, although of significantly different design than the N Reactor. As a result of that accident, the design and operation of the N Reactor have been investigated and scrutinized by a number of investigating teams under the direction of the U.S. Department of Energy (DOE). While all reviewers agree that the Chernobyl scenario could not occur in N Reactor, these investigations have resulted in the definition of a number of safety upgrades, many of which are being accomplished as part of the reactor standdown performed during 1987.

At the time of the Chernobyl event, a PRA for the N Reactor for the purpose of improving plant safety and availability was in-progress. The objective of the PRA was to develop a detailed model of the plant which could be used to improve plant availability, assess the responses of the plant and its associated systems to upset conditions, and to estimate the likelihood that such upset conditions could progress to the point of damage to the nuclear fuel. The PRA has recently been expanded: (1) to assess the response of the confinement systems to accident conditions involving radionuclide release to the confinement; (2) to estimate the magnitude and frequency of radionuclide releases to the environment; (3) to estimate the health effects of such releases; and (4) to estimate risk to the public. In addition to the comprehensive PRA, WHC decided that a limited-scope PRA should be performed.

2.1 Purpose and Objective of the Limited Scope Probabilistic Risk Assessment

In early February 1987, WHC decided that a Limited-Scope PRA was appropriate for the purpose of: (1) providing early estimates of the risk associated with continued operation of N Reactor; and (2) supporting an ongoing N Reactor confinement effectiveness study, which is intended to provide a basis for comparison between a standard nuclear plant with containment and the N Reactor with confinement. In addition, the limited-scope PRA would identify accident sequences that are the principal risk contributors, so resources could be focused on examining these sequences in-depth in the full-scope PRA.

The general intent in this study is to provide conservative estimates, and in many instances, conservative assumptions are made. There are, however, a number of limitations associated with studies of this type that may be inherently nonconservative. An example is the limited treatment of common-mode failures. This and other limitations of the Limited-Scope PRA are noted in Section 2.5.

2.2 Earlier Limited-Scope PRA Studies

Limited-Scope PRAs have been used in the past to provide early, conservative risk profiles and to guide the more detailed PRA, which usually follows. Most often the scoping studies are not reported formally as part of the study documentation because the results are supplanted by the results of the more detailed study.

One specific limited-scope study that was reported was performed as part of the Zion/Indian Point PRAs.[1] In late 1979, the NRC staff reported to the NRC that (on the basis of preliminary calculations) the reactors located at these two sites represented 90% or more of the risk from operating commercial nuclear power plants. The preliminary NRC calculations were based on radionuclide releases and accident frequencies calculated for the WASH-1400 Surry plant, assuming the Surry plant was located at the Indian Point and the Zion sites. A limited-scope study was performed over a period of a few months, in which the results reported in WASH-1400 were extrapolated to the Zion and Indian Point plant. The purpose of the study was to calculate conservative estimates for radionuclide release, accident frequencies, and risk for these specific plants at their sites. The calculations showed the accident risks for these plants to be significantly lower than the risk estimates for the WASH-1400 Surry plant when actual plant design features were accounted for.[2] A full-scope PRA followed, which confirmed the results of the limited-scope study.

2.3 Overview of Probabilistic Risk Assessment Methodology

In 1975, the NRC published the Reactor Safety Study (WASH-1400). This document provided the results of the first serious attempt in the U.S. to quantify the risks from the operation of nuclear power plants. The study employed a relatively new technical discipline referred to as PRA whose advent represented a major advancement in the assessment of nuclear reactor safety. Prior to 1975, no realistic and comprehensive estimate of the risk of nuclear power generation existed. Nuclear safety was based on designing the plants to withstand a spectrum of deterministic accidents that were thought to encompass the most important accidents. The PRA approach, on the other hand, provides a systematic and integrated assessment of conceivable accident initiators coupled with a comprehensive evaluation of plant system responses. The result is a numerical estimate of public risk, rather than a safe versus nonsafe judgement.

In the PRA methodology, accident initiating events are postulated and their frequencies are estimated based on actual plant operating experience or application of other data. These initiating events are used as starting events on "event trees," which are basically logic diagrams that consider the plant response to various combinations of safety system success or failure, following the postulated initiating event. Figure 2-1 provides an example of an event tree. The headings on the event tree represent safety functions supplied by systems within the plant. The horizontal lines across the event tree represent the various possible accident progression paths or

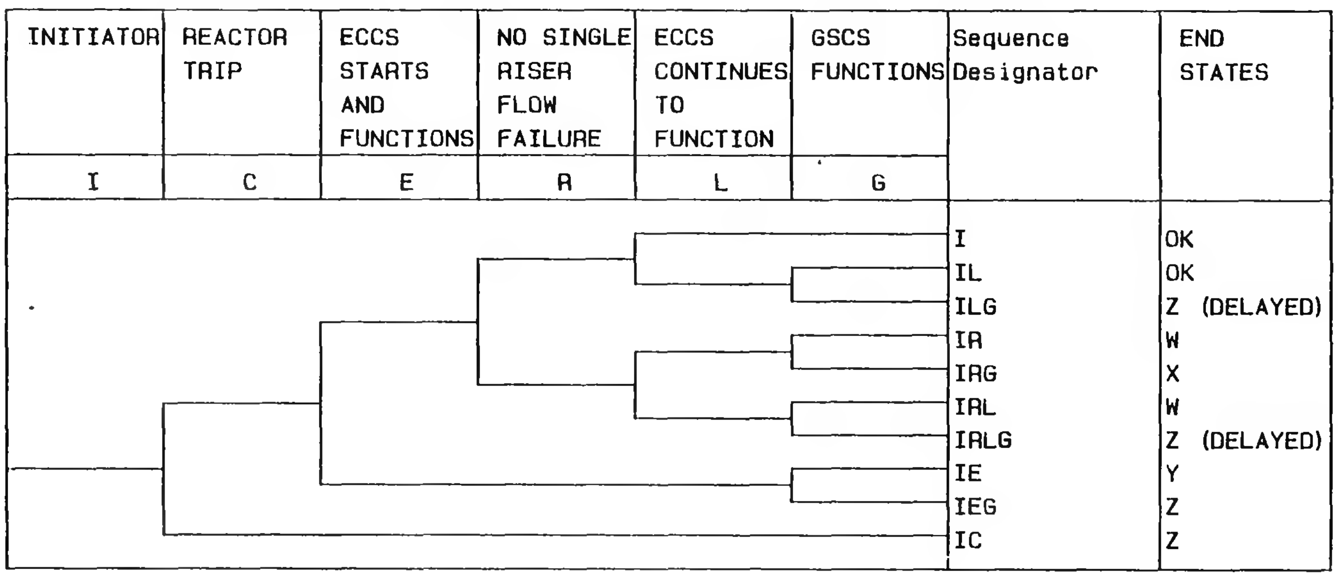


Figure 2-1. Example Event Tree.

sequences that can occur following the initiating event. The lines entereach event tree heading from the left and branch into two segments. The branches represent success or failure of the function/system represented by the event tree heading, with the top branch indicating success and the bottom branch indicating failure. In some instances, there is no branching, which indicates the system has failed as a consequence of an earlier failure of a different system. This may also indicate that the system failure has no effect on the sequence. The far right column indicates the end-state of each sequence, in this case either successful cooling of the reactor fuel or a fuel damage state.

The next step in the PRA process is to quantify the frequency of the accident sequences delineated on the event trees. This is done by combining the frequency of the initiating event with the success or failure probabilities, as appropriate, for each branch point of the individual accident sequences. The failure probability for each of the event tree headings is generally determined by one of two methods. If the event tree heading represents a system for which sufficient operating data exist, the failure probability may be obtained from this source. If insufficient data exist, the failure probability is established by the fault tree technique. The fault trees are also logic diagrams that subdivide a system into its mechanical, electrical, and hydraulic components; the components are arranged through the use of logic gates so the manner that individual components contribute to system failure is depicted. An example of a fault tree is shown in Figure 2-2. By assigning each individual component a failure rate, which is obtained from applicable databases, the failure rate of the system can be calculated.

After the frequency of each accident sequence has been computed, the probabilities of confinement success or failure are estimated for each dominant plant sequence.

For each important combination of plant and confinement failures, a radioactive source term is estimated. This source term defines the fraction of the core inventory of each important radioactive species estimated to be released to the environment. This source term is then used to calculate the public health effects, given a specific confinement failure mode.

By combining the accident sequence probabilities with the associated radiological public health effects, an estimate of radiological risk can be obtained. This risk is generally expressed as either early fatalities that result from short-term exposure to large doses of radiation, or latent fatalities (cancers) which may result from exposure to lower doses of radiation over a long period of time.

This description is a very simplified overview of the PRA process. The process is described in more detail in the joint nuclear industry/NRC PRA Procedures Guide.[3] When applied in detail, the process can be extremely complex and exhaustive. A full-scope PRA may evaluate millions of accident sequences. Such an effort may cost several million dollars and require up to 40 man-yr of effort. The N Reactor Limited-Scope PRA does not include and is not intended to include, all of the detailed analysis normally found in a full-scope PRA.

Since the completion of WASH-1400 in 1975, there have been approximately 35 additional PRAs completed for a variety of plants in the

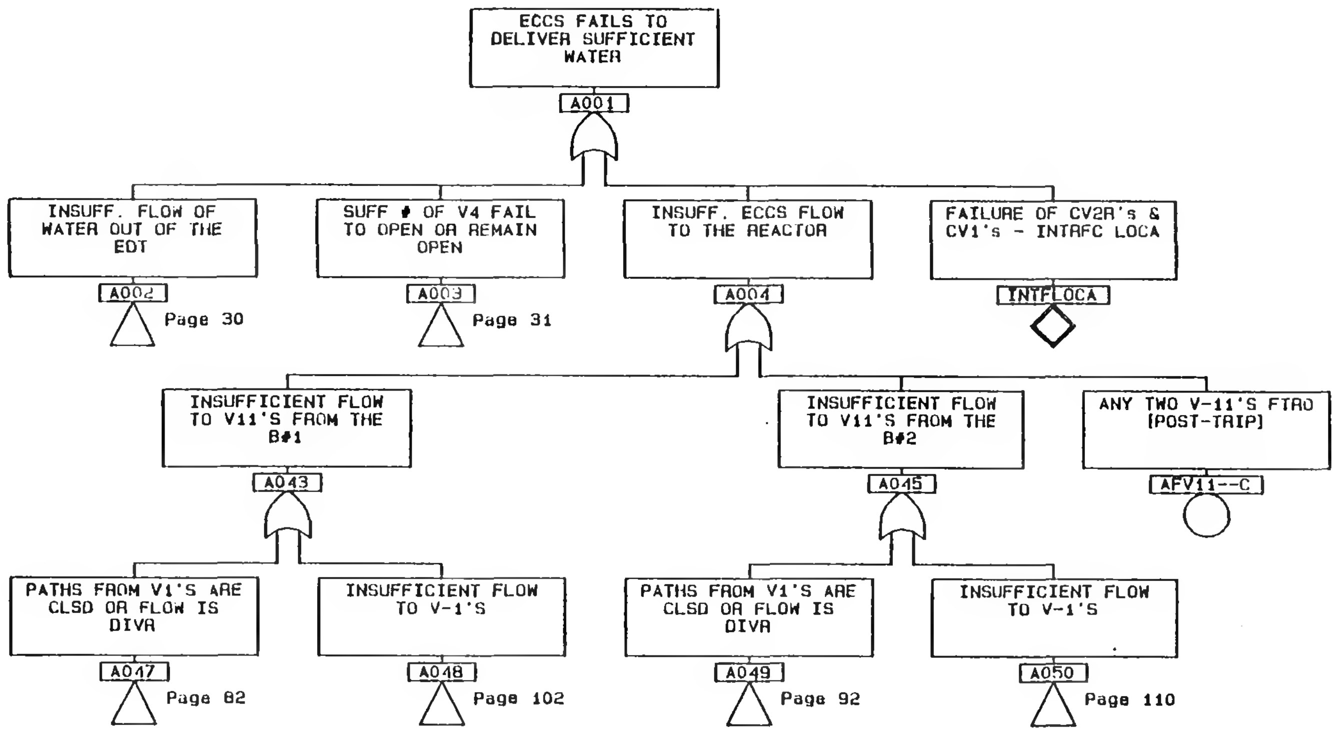


Figure 2-2. Example Fault Tree.

U.S., and several PRAs have also been completed for foreign plants. Ingeneral, these studies have estimated low public risks from the operation of nuclear power plants. Typically, the probability of a serious (core damage) accident has been found to be on the order of one chance in 10,000/yr (usually written as 1 E-04/yr) or less, and early fatality risks have generally been estimated at less than one chance in a million/yr (1 E-06/yr) for a person residing within one mi from the plant. Similarly, latent cancer fatalities have been generally estimated at less than one chance in a million/yr. For comparison, the average risk of death from accidents and other adverse effects for an individual in the U.S. is approximately one chance in 2,500 (4 E-04/yr) and the average risk of death from cancer is 1.9 E-03/yr based on statistics for the year 1982. A further discussion of PRA methods, applications, and limitations can be found in NUREG 1150.(4)

2.4 Application of Probabilistic Risk Assessment Methodology to this Study

The Limited-Scope PRA uses the basic framework of a full scope Level 3 PRA as outlined in Section 2.3.(3) The basic difference is that the limited-scope PRA uses available information, together with simplified event trees and fault trees, to develop the risk estimate. The approach taken in this study for each of the analysis steps is described in the following.

2.4.1 Accident Initiators

The first step is to establish the accident initiators for analysis. Two categories were evaluated, those associated with internal plant upsets (including loss of power) and those generated external to the plant. Internal initiators were derived in part from existing evaluations, including the ongoing full-scope PRA and a recent study by the Los Alamos National Laboratory (LANL), discussed in Section 2.7.(5) In addition, discussions were held with plant operations personnel to determine other possible internal accident initiators. External events were considered based on previous PRA studies, as well as events unique to the Hanford site. Of the external initiators, only seismic activity was analyzed in-detail. Section 4.0 presents the results of the initiating event evaluations.

2.4.2 Plant Event Trees

The first step in the plant response analysis is to identify the principal safety functions accomplished by plant systems in preventing an initiating event from progressing to fuel damage. These functions become the top events of simplified plant event trees that are constructed for each of the accident initiators. The trees are quantified using available data to determine the success/failure probabilities at each branch. Preliminary fault trees being developed for the Level 1 PRA were used in this quantification process. Data for the quantification of fault trees and event trees were taken from the database being prepared for the Level 1 PRA, from the database in the LANL study, and from other generic data sources. Plant event tree end-states are grouped into magnitude of fuel damage bins ranging from a single pressure tube to 100% of the fuel. The plant response analysis is discussed in Section 5.0 of this report.

2.4.3 Confinement Analysis

The confinement analysis parallels the plant analysis in its basic method. Confinement response is analyzed for each plant accident sequence judged to be a significant contributor to risk. Critical safety functions are identified and become the top events on a simplified confinement event tree. The event tree branches are quantified using either available data, simplified fault trees, or results from existing phenomenological studies. End-states in the confinement event tree are grouped into release categories for which the magnitudes of radionuclide release to the environment are similar. In this study, it is assumed the response of confinement to a plant accident does not alter the progression of that accident. Fuel damage progression and confinement event tree analysis are discussed in Sections 6.0 and 7.0 of the report.

2.4.4 Radionuclide Releases

Radionuclide release values for each release category are derived from radionuclide penetration fraction estimates for the various radionuclide transport steps from the fuel to the environment. The penetration fractions are based mainly on existing studies of radionuclide transport in light water reactors. Estimates of radionuclide retention from these studies were used, along with the N Reactor system configurations and expected radionuclide transport times to derive the source term values for consequence calculations. Some N Reactor studies were also available for estimating source term magnitudes. The radionuclide release analysis is presented in Section 8.0.

2.4.5 Consequence and Risk Analysis

Consequence calculations for each of the release categories were performed by the U.S. Department of Energy's Pacific Northwest Laboratory (PNL) operated by Battelle Memorial Institute and Sandia National Laboratories using the CRAC2 Code (described in Section 9.0). Consequence indices reported in this study are early fatalities, population exposure, and latent fatalities. Consequence calculations were also performed for a bounding release directly to the environment of 100% of the core inventory of volatile fission products. This release case bounds the possible consequences of a fuel damage accident at N Reactor. Consequences are estimated only for the offsite population, and consequence calculations are decribed in Section 9.0.

One output of the plant and the confinement event trees is a set of frequency estimates for each of the release categories. Calculated consequences for each release category and frequencies for the release category are combined to produce the overall risk estimates presented in Section 10.0. Bounding case risk results are also included in Section 10.0.

2.5 Study Limitations

The limited-scope PRA is neither intended to possess the rigor of a detailed PRA (one is in-progress for N Reactor) nor to include an analysis of uncertainty in the calculated values. The principal limitations of this study are discussed in the following.

2.5.1 Common-Mode Failures

Common-mode failures of redundant components within the same system are considered to a limited extent in this study. Potential common-mode failures and dependencies involving more than one system are generally not taken into account in this study. This limited treatment of common-modes is a potential nonconservatism. The seismic analysis identified a number of common-mode system failures that could result from a large earthquake and these were accounted for. A more thorough treatment of common-mode failures is included in the ongoing Level 1 PRA.

2.5.2 Operator Errors/Recovery Actions

The Limited-Scope PRA includes operator errors of omission (i.e., failure to take specified actions necessary for success of a safety function). Errors of commission are not modeled, and human errors are not considered as potential accident initiators. These factors can lead to nonconservative estimates. However, operator actions to recover from equipment failures are also not considered, and this is believed to be conservative.

2.5.3 Seismic Evaluations

The Limited-Scope PRA identifies seismic events as the principal contributor to overall risk. The following two limitations are associated with the seismic evaluations.

- Seismic Frequency-Intensity Curve. An estimate of frequency of occurrence for a range of peak ground accelerations was developed for United Engineers and Constructors by Woodward-Clyde. The curve, which is presented in Section 4.4.2, is a best-estimate, point value curve based on conservative interpretations of geologic structures which are not as yet characterized fully.

- Equipment Fragility. Identification of equipment judged to be vulnerable to damage in a seismic event and estimates of failure parameters are based on a plant inspection with subsequent review of design drawings and stress analyses.

2.5.4 Plant Design Modifications

This study is based on the plant design as it existed on January 1, 1987. A number of plant upgrades are in-progress. However, with a few exceptions for some critical components, these upgrades were not considered; this approach results in an element of conservatism in the overall analysis.

2.5.5 Source Term Estimates

Magnitudes of various radionuclide species releases resulting from fuel overheating are based on limited data for metallic fuel. Estimates of radionuclide retention during transport from failed fuel through the confinement spaces to the environment are based on tests and calculations for light-water reactors, with appropriate allowances for conservatism in the form of lower assumed retention. Limited calculations for the N Reactor confinement geometry and associated post-accident environment were available for this study. Additional work is planned and as part of the full-scope PRA.

Source terms were estimated only for radionuclide releases to the atmosphere. Potential releases to liquid pathways are judged to be unlikely and would result in substantially lower radiation exposure to the offsite population than releases to atmospheric pathways. Exclusion of liquid pathways from this study is discussed in Section 9.0.

2.5.6 Aging

N Reactor has been operating for a number of years. Component failure rates were derived, in part, from the plant operating history, as well as generic data. No trend analysis was performed to predict future failure rates. Therefore, component aging is not considered in this study. There are, however, a number of N Reactor programs that monitor and counteract the effects of aging. These include:

- Equipment maintenance standards prescribe preventive maintenance of safety-related structures, systems, and components

- An inservice inspection program, conforming generally to nuclear plant requirements established by the American Society of Mechanical Engineers, is in-place to ensure reactor coolant system pressure boundary integrity

- Engineered safety feature upgrades have been made as industry standards have evolved

- The recent plant-wide programs were designed to restore plant components to their original conditions; these are the Restoration Program (1980-1986) and the Productivity Retention Program begun in 1986.

2.6 Uncertainties

There has been no attempt to quantify uncertainties in this study. Both input data and results are point value estimates. Uncertainty has been bounded by calculating consequences for a maximum bounding fission product release to the environment. These consequences are expressed in terms of risk by assigning what is judged to be a conservatively high estimate of fuel damage frequency. This bounding estimate is further discussed in Sections 8.0, 9.0, and 10.0.

2.7 Relationship to Los Alamos National Laboratories Probabilistic Risk Assessment Preliminary Results

A Level 2 PRA is being performed by LANL under DOE sponsorship. The purpose of the LANL study is to perform an independent review of the N Reactor Updated Safety Analysis Report (NUSAR). Similarities and differences between the preliminary results of the LANL study[5] and this study are summarized in the following.

- The LANL study will, but has not yet, addressed the confinement system.

- The LANL study has not yet considered different fuel damage end-states; therefore, the preliminary frequencies reported by LANL are for fuel damage in more than one pressure tube.

- The preliminary LANL study results include operator recovery actions, so the frequency estimate for most sequences is correspondingly reduced compared to those contained herein.

- Both the preliminary LANL study and this study include only limited consideration of common-cause failures.

- The scope of the LANL study does not include external events such as earthquakes; in this study, earthquakes are found to be the dominant risk contributors (as has been the case for many plants for which full-scope PRAs have been performed).

- This study considered station blackout as an initiator and evaluated the outcome, and LANL considered single bus failures as initiators; because these two initiators result in different sequences, it is difficult to compare the results. The station blackout sequence was considered in this study as encompassing all complete primary coolant flow loss sequences other than those caused by pipe breaks (Table 4-2).

- This study quantified the individual system fault tree models and then quantified the accident sequences by multiplying the individual system unavailabilities for each sequence. The LANL study formed the sequences by linking the fault tree models and then quantifying the linked sequence model. Although this difference probably does not affect the results significantly, it makes it more difficult to compare the results of the two analyses.

2.8 References

1. 1980, An Evaluation of the Residual Risk from the Indian Point and Zion Nuclear Plants, Off-Shore Power Systems Report 36A75.

2. Early, P. J., Power Authority, State of New York, and W. J. Cahill, Consolidated Edison, Dockets 50-247 and 50-286, (letter to H. R. Denton, U.S. Nuclear Regulatory Commission staff, May 23, 1980).

3. NRC, 1983, PRA Procedures Guide, NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. NRC, 1987, Reactor Risk Reference Document, NUREG 1150, U.S. Nuclear Regulatory Commission, Washington, D.C.

5. Stack, D. W., et al., 1987, Preliminary Results of a Probabilistic Risk Assessment for the N Reactor, UNI-I-111 (Unclassified Controlled Nuclear Information), UNC Nuclear Industries, Inc., Richland, Washington.

3.0 PLANT OVERVIEW

This section presents an overview of the N Reactor site, major structures, confinement design concept, and the reactor. Also included are discussions of the major plant systems. Most of the information is summarized from the NUSAR.[1] The system one-line drawings reflect the reactor systems as they are modeled in the Level 1 PRA as of July 1987. The models reflect the plant configuration as of January 1987.

3.1 Site

N Reactor is located in the north central part of the DOE Hanford Site. The Hanford Site occupies parts of Benton, Franklin, Adams, and Grant Counties in Southeastern Washington State as shown in Figure 3-1.

The Hanford Site also shown in Figure 3-1, is composed of 570-mi^2 of semi-arid land. The Columbia River flows through the north and northeastern part and forms approximately half of the eastern boundary of the site. The ground cover is principally sagebrush and rabbit brush. The terrain is generally flat and open with gently rolling hills except for the Gable Mountain and Gable Butte formations in the center, the east river bank bluffs, Rattlesnake Hills on the west, and Saddle Mountains to the north.

The site contains the following major facilities in addition to N Reactor:

- The five reactor areas designated 100-B, 100-K, 100-D, 100-H, and 100-F, which contain eight shutdown special nuclear materials production reactors
- The 300-Area, which contains a fuel fabrication facility and laboratory facilities supporting all of the DOE Hanford programs
- The two areas for fuel reprocessing, waste processing, and waste storage designated 200-E and 200-W areas
- A commercial nuclear waste burial operation on land leased to the State of Washington
- The Fast Flux Test Facility (FFTF)
- The three Washington Public Power Supply System (WPPSS) nuclear power plants, one operating, one mothballed, and one terminated.

The Hanford Site is in a region of low- to moderate-seismicity. On the basis of the slight damage that has been experienced since 1840, the U.S. Coast and Geodetic Survey designates the area as seismic Zone II, implying the potential for only moderate damage from earthquakes. The sand and gravel underlying the Hanford Site provide excellent protection against such damage. Earthquake intensities greater than four on the Modified Mercalli Scale (MM-IV) have not occurred in the Hanford area in recent

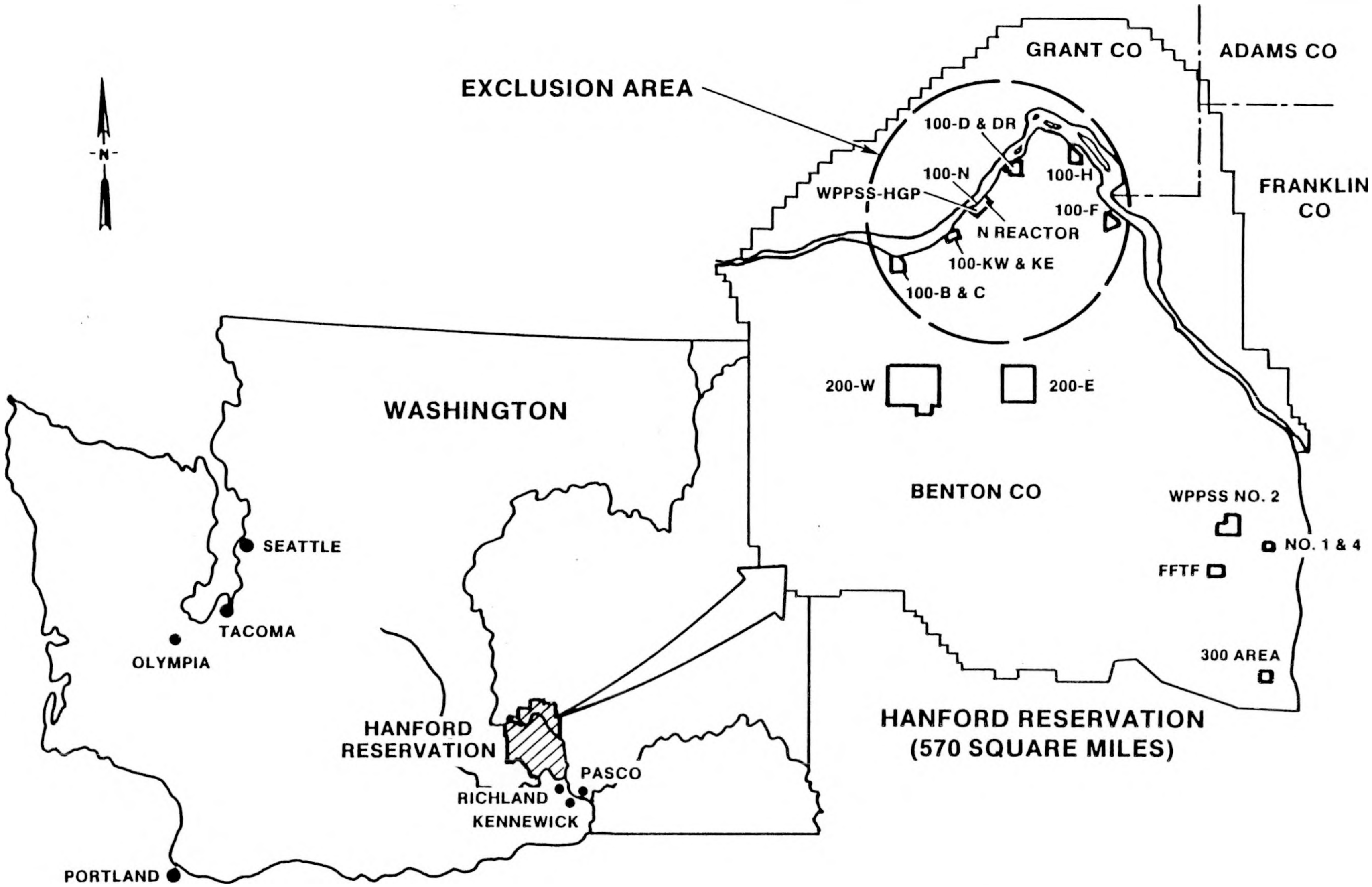


Figure 3-1. Location of N Reactor.

times. An earthquake of MM-IV intensity corresponds to a minimum horizontal ground acceleration of 0.007g.

The regional climate is mild and dry with occasional periods of high wind. The average temperature is approximately 54 ^{O}F with a record high of 115 ^{O}F and a record low of -27 ^{O}F. The annual precipitation averages 6.28 in. with a record high of 11.45 in. and a record low of 3.26 in. The mean wind speed averages 7.6 mi/h with a record high of 80 mi/h.

The location of N Reactor is a 90-acre site (designated 100-N Area) on the right descending (south) bank of the Columbia River in the northcentral part of the reservation. The plant is approximately 27 air mi and 38 river mi from the city limits of Richland, Washington. N Reactor has an administratively established 5-1/2 mi exclusion boundary and a 20 mi, low population zone boundary. The offsite population within 20 mi is approximately 6,500 persons. There are approximately 340,000 persons living within a 50 mi radius from the plant. Land use within 50 mi of N Reactor includes homes, business outlets, light industry, other nuclear facilities, and farming.

3.2 Major Structures

A plot plan of N Reactor is shown in Figure 3-2. The buildings of principal importance in the limited-scope risk assessment are 105-N, 109-N, 181-N, and 182-N.

The reactor and its cooling systems are housed in two adjoining buildings, 105-N and 109-N. An isometric view of these buildings is shown in Figure 3-3. The reactor, including its control and trip systems, is located in 105-N. Reactor coolant pumps, steam generators, pressurizer (located in a penthouse), graphite and shield cooling components (for recirculating mode operation), and other auxiliary equipment are located in 109-N. Portions of 105-N and 109-N have been analyzed for safe shutdown earthquake (SSE) loads. These include the foundations, walls, roofs and tunnels that enclose the reactor, the reactor coolant system, Emergency Core Cooling System (ECCS) piping and other components, horizontal control rods/drives, ball system components, and the main control room. Reactor confinement consists of the central portion of 105-N and the portion of 109-N which houses reactor coolant system components. The confinement envelope is shown in bold outline in the plan view shown in Figure 3-4.

The confinement structure is constructed of reinforced concrete, except for eight steel roof trusses and two steel box girders located near the top of 105-N. The principal structural walls vary in thickness from 3 ft 0 in. to 5 ft 9 in. and provide shielding as well as pressure confinement capability. The 105-N confinement structure is enclosed on three sides and the top by 105-N and is contiguous with the 109-N confinement structure on the south.

Also included in 105-N, but outside the confinement envelope, 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,

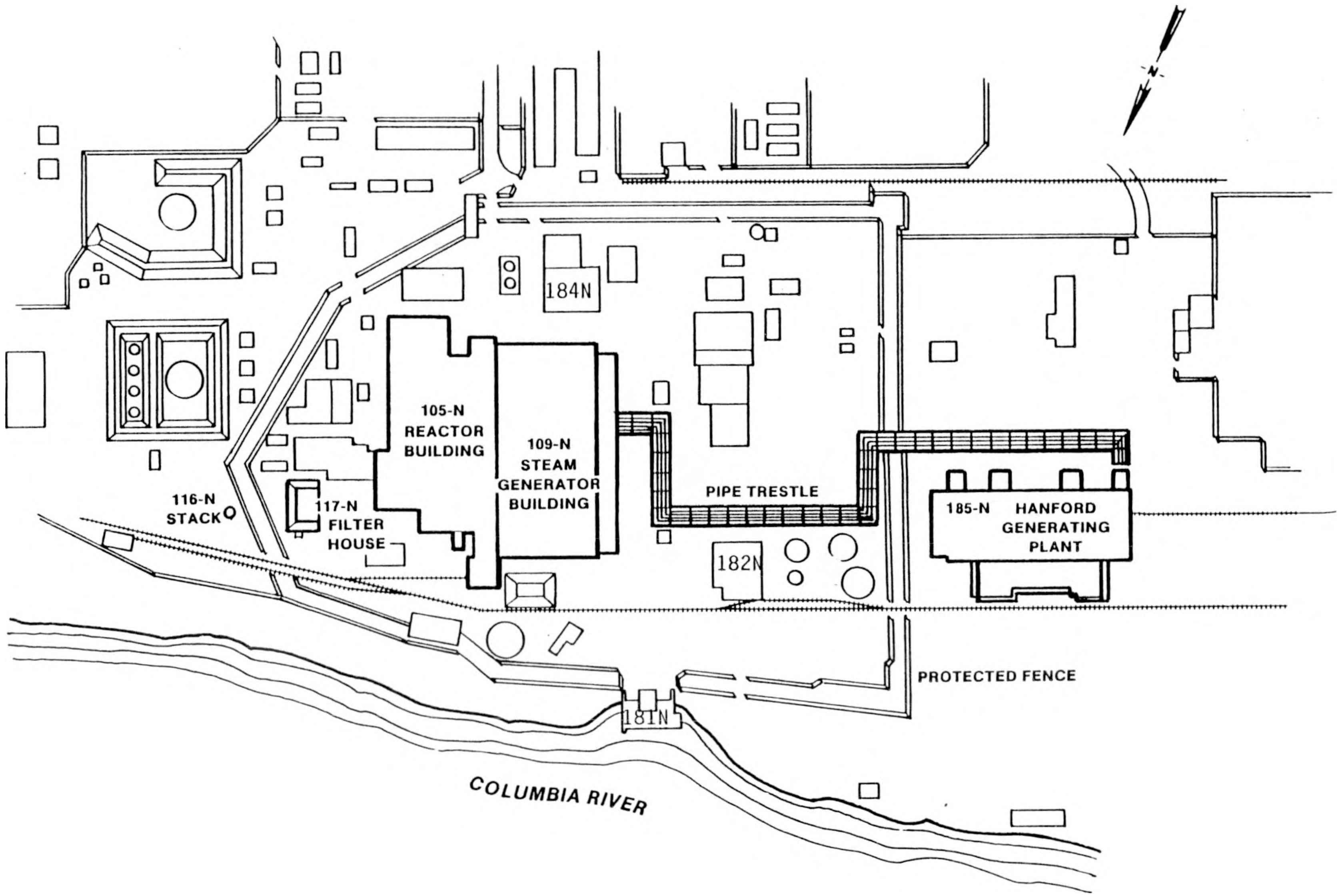


Figure 3-2. N Reactor Plot Plan.

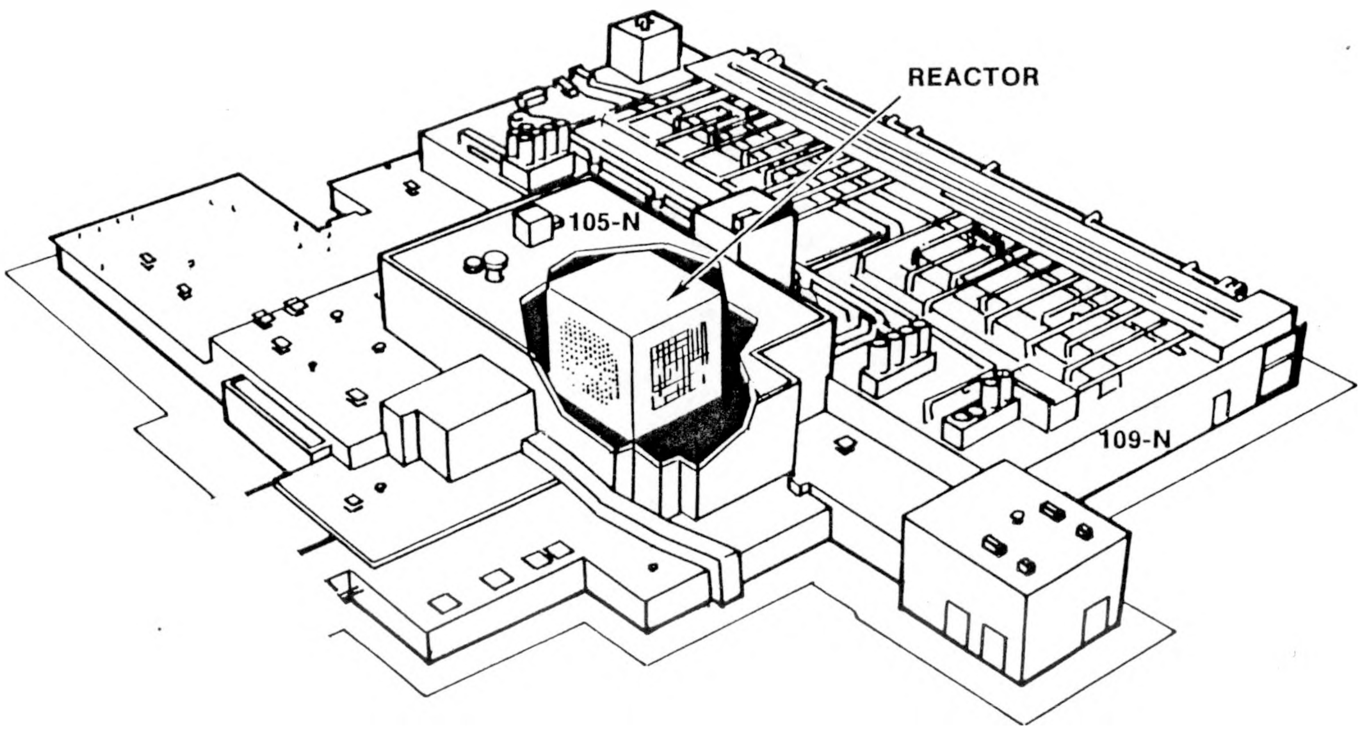


Figure 3-3. Buildings 105-N and 109-N.

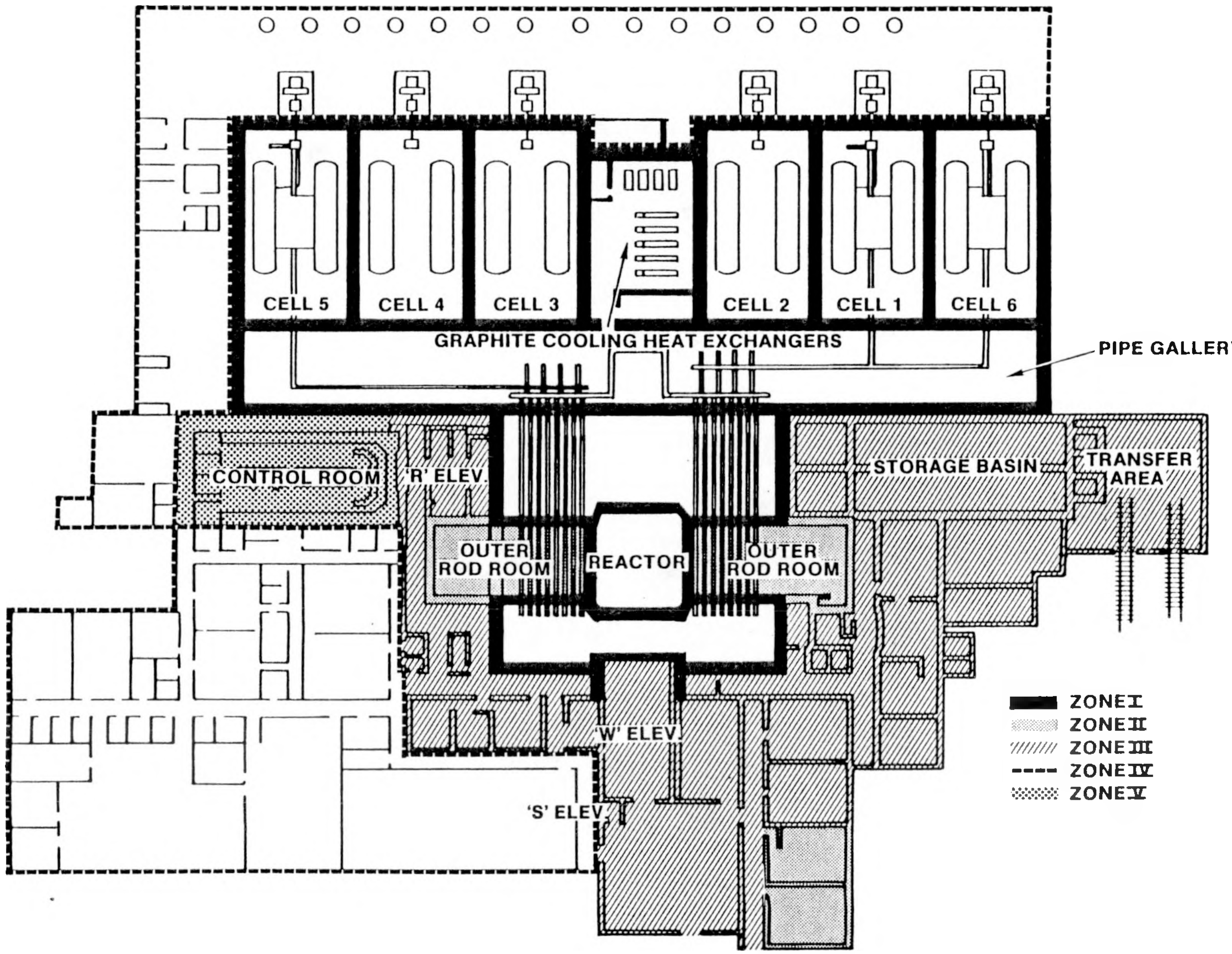


Figure 3-4. 105-N/109-N Plan View Looking Down from 60-Foot Elevation.

motors, fans, compressors, pumps, valving, and piping; and the maintenance shops and offices. Personnel access to the 105-N confinement from adjacent building areas is through shielding doors located for convenience at each floor level. Inflatable seals, designed to withstand 5 $lb/in.^2$, are provided on these doors.

Within the 109-N confinement structure, the steam generators and RCS pumps are installed in six isolated rooms, or cells, and 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. As with 105-N, this concrete structure provides shielding and missile protection, in addition to serving as the confinement barrier.

The confinement envelope is designed for an internal pressure of 5 $lb/in.^2$ (gage), except for the six steam generator cells which are designed for an internal pressure of 12 $lb/in.^2$ (gage).

The river pump house (181-N) is constructed of reinforced concrete and provides support and suction wells for seven deep-well turbine pumps. Three low-lift diesel pumps supply water to the emergency core cooling system (ECCS). Four electrical pumps supply river water to N Reactor (via the circulating raw water system) during normal operation. The ECCS pumps, diesel engines, and auxiliary equipment are housed in a concrete superstructure located on the main floor of the river pump house. The superstructure walls are constructed of hollow concrete blocks and are reinforced with vertical steel rods embedded in concrete fill and with horizontal layers of steel mesh. The roof slab is cast integrally with a series of beams and girders. The 181-N walls have been analyzed for Safe Shutdown Earthquake (SSE) loads of 0.25g.

The 182-N houses five diesel-driven pumps serving the ECCS and confinement fog spray. High-pressure injection pumps, low-pressure raw water and the high-pressure raw water pumps are also located in 182-N. Safety-related pumps are located in the basement. 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 that support the main operating floor are constructed of reinforced concrete. In addition, there are a number of structural steel columns that provide independent support of piping systems located in the basement. Above-ground walls are made of hollow concrete blocks, anchored to a structural steel framework. In addition, a series of bolts and square washers connect the block walls directly to peripheral wall columns. Concrete blocks are reinforced with vertical steel bars and horizontal steel mesh located between each row of blocks. The 182-N walls have been analyzed for SSE loads of 0.25g.

3.3 Confinement

The confinement design concept is to vent the initial steam release resulting from a large reactor coolant pipe break directly to the atmosphere. Following this initial mass/energy release and prior to fission product release into the confinement from damaged fuel, the steam vent valves are closed and a controlled release path to the atmosphere is established through High-Efficiency Particulate Air (HEPA) filters, charcoal filters, and a 200-ft vent stack.

The following N Reactor design features differ from commercial nuclear plants, which use containment rather than confinement systems. These features enable N Reactor, with its confinement system, to achieve protection of the surrounding population from the potential consequences of a severe accident. These features include:

- Metallic uranium fuel with its relatively high thermal conductivity in comparison to oxide fuel
- Approximately 365 MT of fuel resulting in unusually low specific power and a comparatively long time (420 S) to reach fuel cladding failure temperature in the unlikely event that all core cooling is lost
- Maintenance of low radioactivity in the reactor coolant by promptly shutting the reactor down and discharging any fuel element that experiences cladding failure
- Graphite moderator (640 tons) and reflector (1,160 tons) that act as a heat sink
- A separate moderator cooling system, which limits potential fuel melting if all direct cooling water supplies to the pressure tubes are lost
- A remote site with low close-in population.

The confinement envelope surrounds the reactor coolant system and is designed to maintain structural integrity with internal pressures ranging from +5 $lb/in.^2$ (+12 $lb/in.^2$ for the steam generator cells) to -2 $lb/in.^2$. Additional structural description is provided in Section 3.2. Post-accident confinement system operation is summarized in the following. Important confinement system features are shown in Figure 3-5.

Confinement isolation is initiated automatically by an increase in confinement pressure to 2 in. water gauge (w.g.), by actuation of the ECCS, or by remote manual actuation from the control room. The 105-N and 109-N each have separate circuits for both confinement and fog-spray actuation. The confinement sequence that establishes the filtered release mode is initiated by two redundant circuits. Cross vent doors between 105-N and 109-N ensure the sequence control circuits would be tripped for both buildings by a pipe break occurring in either building.

Regardless if the initial pressure surge is in 105-N or 109-N, the sequence of events is generally the same. Confinement heating and ventilating (H&V) pathways are isolated immediately by closing the 105-N and 109-N Zone I confinement H&V isolation valves and stopping the Zone I supply and exhaust fans. One cross-vent door is open normally between 105-N and 109-N to provide pressure equalization between the buildings. If the pressure differential between 105-N and 109-N is not equalized by the one open normally cross- vent door, one or more of the remaining seven cross-vent doors automatically open to equalize the pressure differential.

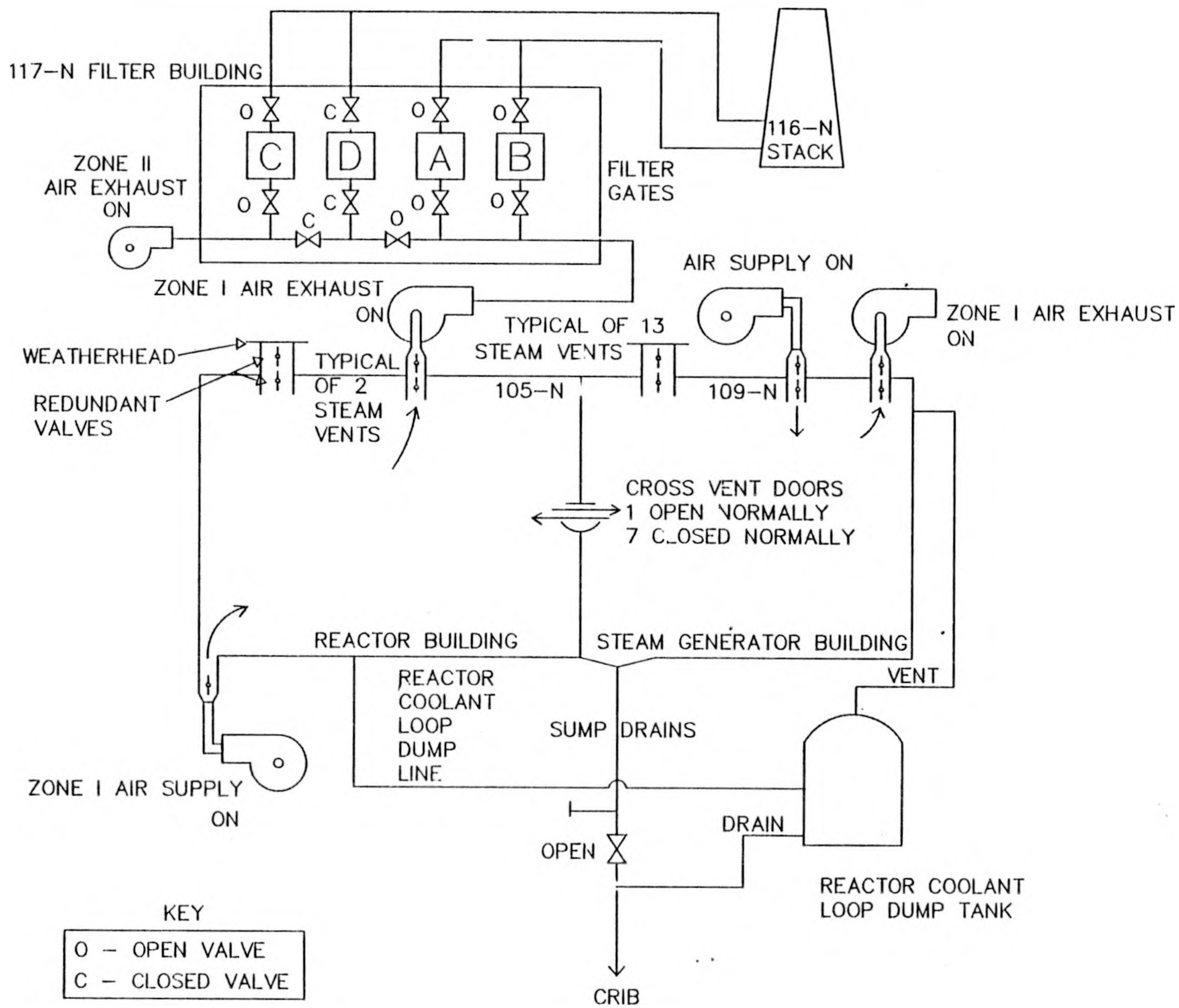


Figure 3-5. N Reactor Zone I Confinement Components.

If confinement pressure reaches 10 in. of water, the fog-spray stations in 105-N and in the 109-N pipe gallery are actuated automatically. The fog-spray stations in the 109-N steam generator cells are manually actuated. All fog-spray stations may be initiated by remote manual actuation from the control room. The fog-spray system provides pressure control and washes fission products and aerosols from the confinement atmosphere. In an emergency, water for the fog-spray system is supplied by two diesel-driven fog-spray pumps which receive suction from the low-lift diesel-driven ECCS pumps (See Section 3.8.2).

If pressure in 105-N and 109-N continues to rise, the weatherhead caps, which cover the top of the steam vents, will blow-off and pressure will be released through the open steam vent paths; redundant steam vent valves in each steam vent are normally in the open position. In all design basis events, the initial pressure relief occurs before any fuel damage occurs. Therefore, the vented steam has no appreciable radionuclide content. Once the pressure in confinement decreases, a filtered vent path to the environment is opened and the 105-N and 109-N steam vent valves are closed to prevent an unfiltered release. The filtered vent path is established by opening at least one of the three 105-N exhaust paths to the filters and by realigning filter gates in 117-N. Normally, 105-N Zone I exhaust is released through filter cells A and B. Following an accident, Zone I exhaust is routed through 117-N filter cell D by remote-manual filter gate realignment. The filtered release path contains HEPA and charcoal filter elements that remove most of the fission products (except for noble gases) before the air is released out of the stack to the environment. The motive force for filtered release is provided by natural draft from confinement to the plant stack.

3.4 Reactor

The graphite-moderated reactor utilizes a horizontal pressure-tube array cooled by light-water. It is designed for production of special nuclear materials at power levels up to 4,000 MWt. The reactor core (shown in Figure 3-6) is housed in an 1,800-ton graphite cuboid approximately 33 ft by 33 ft at the face and 39 ft long. A total of 1,003 horizontal Zircaloy-2 pressure tubes, designed for 1,825 $lb/in.^2$ and 600 ^{o}F penetrate the graphite moderator. The reactor contains approximately 365 MT of low-enrichment metallic uranium fuel. Fuel rods are of a coaxial, tube-in-tube design, and are inserted into the cooling water pressure tubes. Perpendicular to the pressure tubes are 84 horizontal control rods containing boron carbide. Control rods enter the reactor from both sides and provide operating reactivity control, flux shaping, and primary emergency shutdown (trip) capability.. A second backup trip system is provided by a ball safety system. A total of 107 vertical channels penetrating the core can be gravity-filled with special neutron absorbing balls to provide rapid shutdown of the reactor. The horizontal control rods and ball safety system each operate with sufficient speed to prevent fuel damage in an accident and provide sufficient shutdown capacity to prevent a subsequent return to criticality.

In addition to the control and reactor shutdown systems, there are inherent safety features in both the core design and operating mode. These features include a negative power Doppler coefficient, a negative void

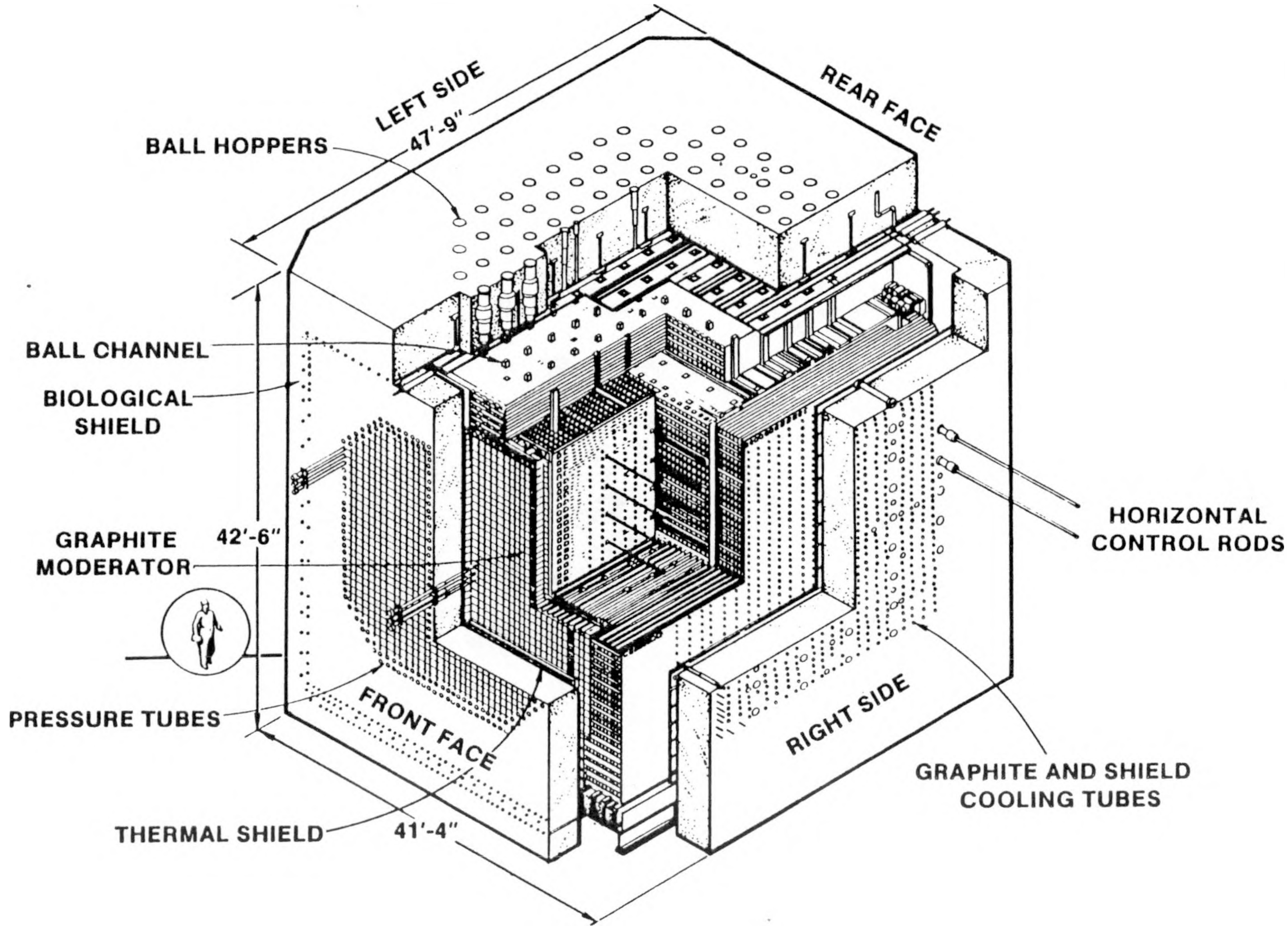


Figure 3-6. N Reactor Core Components.

coefficient, and a large delayed neutron fraction.[1] Operation throughout the years has demonstrated that N Reactor is very stable and xenon oscillation is not a concern.

The reactor core is a structure of interlocking graphite bars, with overall dimensions of 39 ft 5 in. long; 33 ft wide; and 33 ft 3-1/2 in. high. The moderator is penetrated by 1,044 horizontal channels on an 8-in. horizontal and 9-in. vertical lattice spacing; however, only 1,003 channels contain the Zirconium alloy pressure tubes and fuel elements. Venting provisions are included in the graphite stack to permit the escape of reactor coolant that would be released in the unlikely event of a pressure tube break. These vent channels allow dissipation of the steam-water mixture in all three directions and have sufficient cross-sectional area to limit potential damage to the single bar of graphite in which a tube break might occur.

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.

A thermal shield surrounds the graphite moderator and reflector and intercepts most of the radiation heat energy that escapes from the reactor system. The function of the shield is to reduce the heat load on the biological shielding. A 1-in. thick boron steel plate is used as the thermal shield material except at the inlet and outlet ends of the reactor where 8-in. thick cast iron blocks are used. Cooling tubes are welded to the boron steel plate to facilitate the removal of absorbed heat. The cast iron inlet and outlet thermal shield is cooled by direct contact with the exterior surface of the pressure tubes through which the reactor coolant flows. Concrete biological shielding prevents activation of equipment and components immediately adjacent to the reactor. The biological shield supports the thermal shield, pressure tube assemblies, horizontal control rods, and ball hoppers. It is also the pressure envelope for the helium cover gas surrounding the reactor block.

N Reactor operates with concentric tube-in-tube Zircaloy-clad fuel elements. The metallic uranium fuel is slightly enriched to various levels of U-235 (0.94 to 1.25 percent). In the past, other materials have been used as a target in conjunction with an enriched uranium driver fuel element to produce isotopes such as tritium and plutonium-238. The fuel cladding is Zircaloy-2, metallurgically bonded to the uranium by a coextrusion process. Individual fuel element lengths range from 15 to 26 in.

3.5 Reactor Control and Shutdown Systems

3.5.1 Horizontal Control Rods

The horizontal control rod system consists of 84 rods with hydraulically operated drive mechanisms. Forty-five rods enter from one side of the reactor block and 39 rods enter from the other side. The rod system provides both operating control and rapid shutdown (scram) of the reactor. The scram mode achieves 75% rod insertion within 1.5 s. The horizontal control rods are water-cooled and have individual rod drives capable of scramming from any position, as well as slower drive speeds for fine control of reactor flux distribution and power level. Individual rods are set up for control or scram use by means of selector switches in the reactor control room; a specified number of fully withdrawn safety control rods are always reserved for the scram function. Interlocks prevent control rod withdrawal unless the specified number of safety rods are withdrawn properly and are ready to scram. Each hydraulic cylinder rod drive unit has its own independent accumulator to provide energy for scram. Control equipment is fail-safe in that loss of electric power will de-energize the control solenoid valves and accumulator pressure will scram the rods. The horizontal control rod system is activated automatically to shut-down the reactor during appropriate offnormal conditions.

3.5.2 Ball Safety System

The ball safety system provides a second scram system with the required shutdown speed. The control elements for this system are 0.425-in diameter neutron absorbing balls (boron carbide, B_4C). Ball hoppers are imbedded in the top biological shield above 107 vertical channels. The hopper gates, held closed by DC-powered solenoids, are designed with spring loading to open in case of control power failure, as well as on appropriate safety-circuit signals, to allow the balls to enter the channels by gravity.

3.6 Reactor Coolant System

The Reactor Coolant System (RCS) is located in 105-N and 109-N. The system flow diagram is shown in Figure 3-7. Isometric drawings of the RCS are presented in Figures 3-8, 3-9, and 3-10. 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. These lines are carbon steel, sized to accommodate the required reactor coolant flow rate. Design basis conditions are 1,825 $lb/in.^2$ (gage) and 600 ^oF.

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. Piping from the reactor outlet manifold to the steam generators is 26-in. carbon steel, and piping from the steam generators to the reactor inlet manifold is 24-in. carbon steel. 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 online with the sixth isolated for maintenance.

The reactor coolant recirculating pump for each loop is a single-stage, horizontal-centrifugal unit utilizing injection type seals to minimize outleakage. Each pump is located inside the shielded cell, which houses the associated steam generator units.

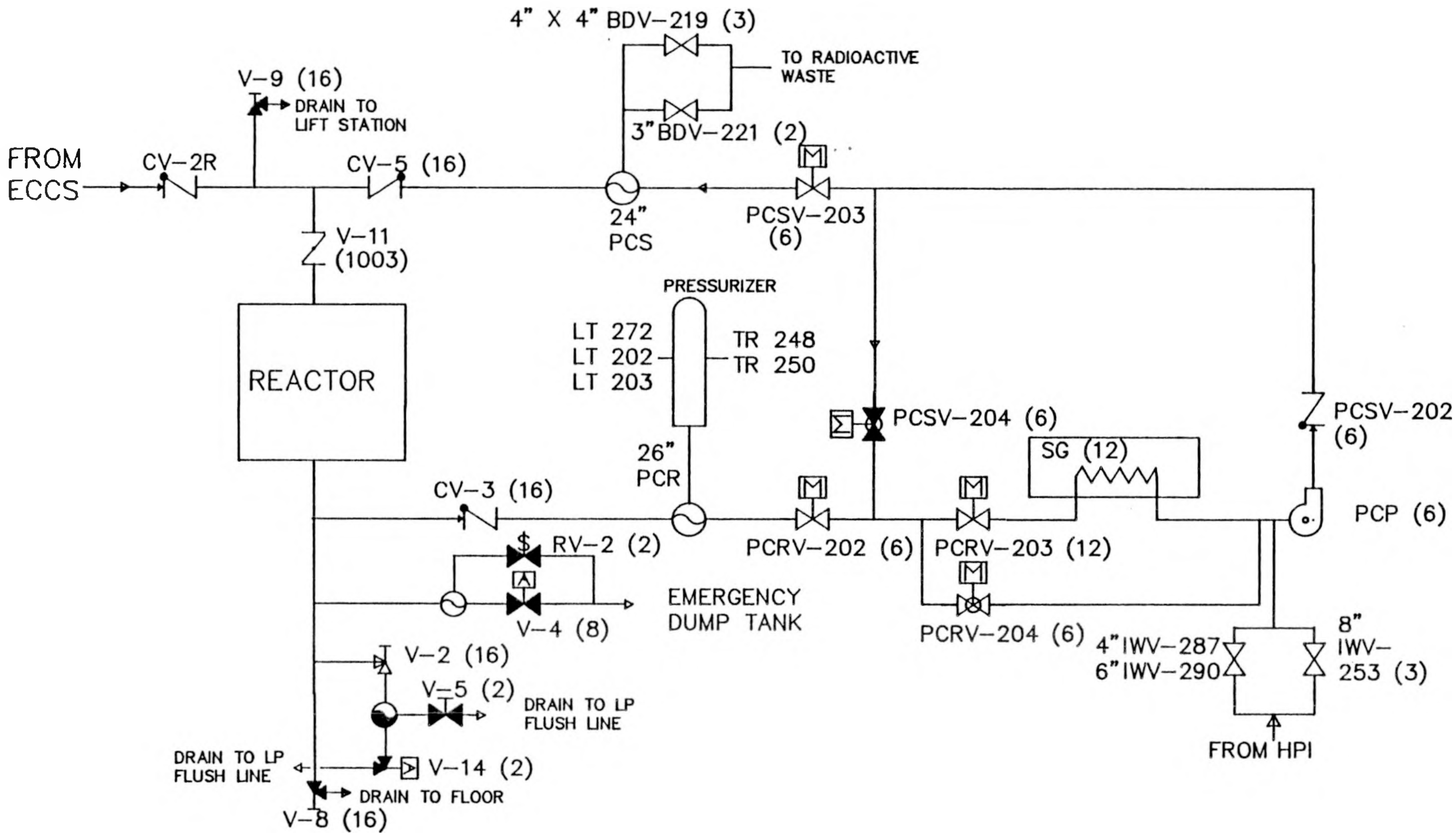


Figure 3-7. Primary Coolant System Flow Diagram.

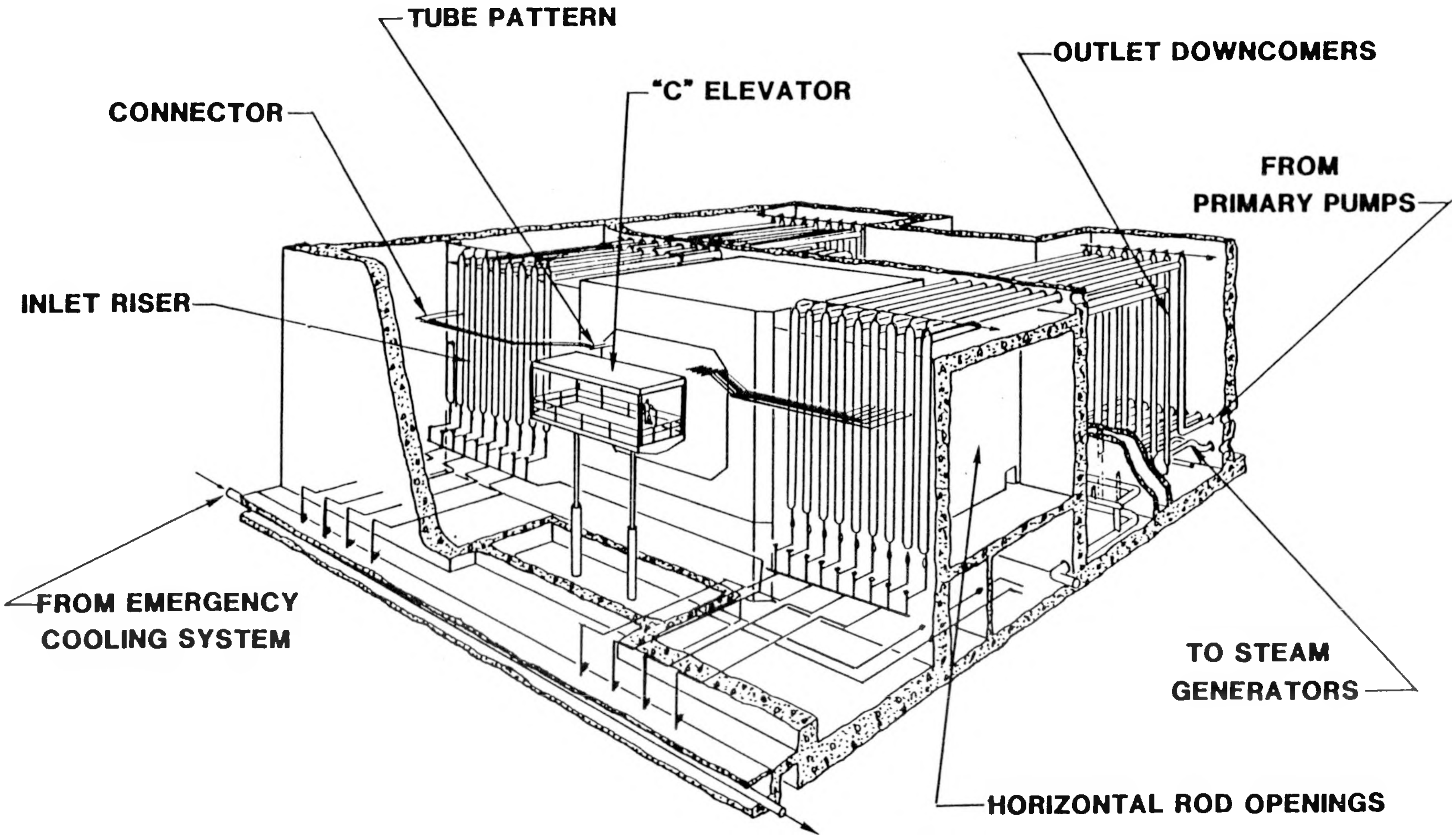


Figure 3-8. Reactor Inlet Piping.

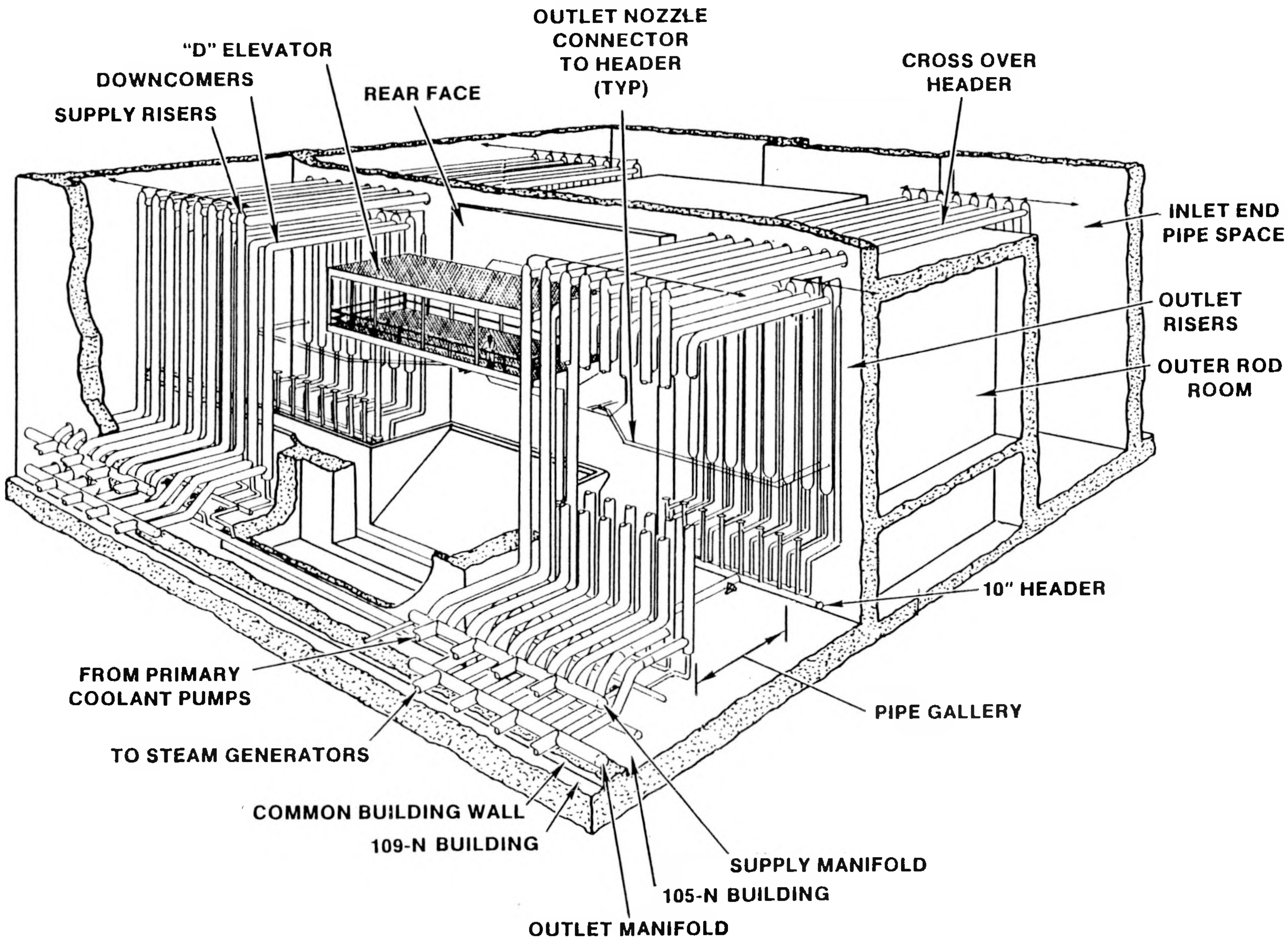


Figure 3-9. Reactor Outlet Piping.

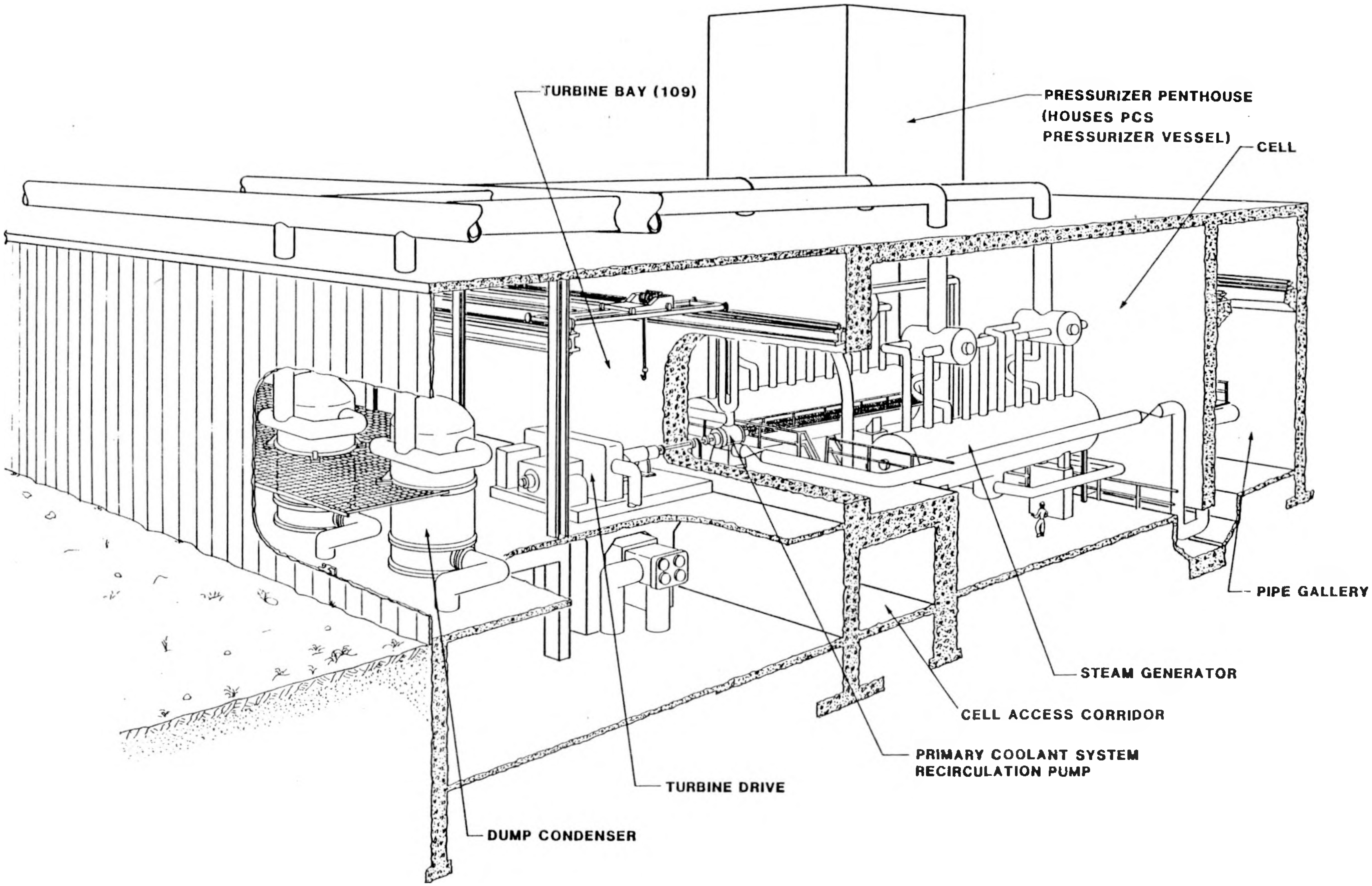


Figure 3-10. Typical Steam Generator Cell.

Steam-turbines are the main drivers for the reactor coolant recirculating pumps. AC electric power is required for turbine operation because of the dependency on auxiliaries such as lubricating oil and cooling water supply to the drive turbine surface condenser. Each drive turbine has a design rating of 9,000 hp at 3,600 rpm. Steam for the RCS pump drives is supplied by the main steam generators during normal reactor operation and by plant service oil-fired boilers when steam from the main steam generators is not available. Steam to the onsite 15,625-kW turbine-generator is supplied in the same manner.

A 400-hp electric pony motor capable of driving the reactor coolant pump at about 900 rpm is mounted on each turbine drive shaft. These motors provide sufficient water recirculation 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 and are connected to the pump by an extension shaft through the cell wall.

Reactor coolant pressure and temperature are controlled to prevent boiling at any point in the system. A pressurizer is provided to control system pressure and to accommodate volume changes resulting from normal coolant density changes during transient conditions. The pressurizer is a 1,200-ft^3 cylindrical vessel and is connected directly to the reactor outlet piping. These two separate, electric immersion-heater systems having a total capacity of 2,350-kW maintain the water in the pressurizer at the desired saturation conditions. Under normal operating conditions, approximately 40% of the useful surge vessel volume contains saturated steam.

3.7 Secondary Cooling System

The secondary steam system removes heat from the reactor coolant system by boiling secondary water in the shell side of the steam generators. A basic diagram of the secondary system is shown in Figure 3-11. During previous operation solely for production of special nuclear materials, the major fraction of this steam was routed to 16 dump condensers arranged in-parallel and cooled by Columbia River water; thermal discharge limits presently prevent operation of this many dump condensers. These condensers operate at a pressure near that of the steam generators. Condensate is pumped from the dump condensers back to the steam generators for recycling. A portion of the steam generated is utilized by the coolant pump drive turbines and by the plant turbine-generator, which provides a part of the plant's AC power requirements.

The condensate pumps for half of the dump condensers are driven by power from the offsite AC power supply and the other half by power from the in-plant turbine-generator (onsite AC electrical power supply). Plant service boiler auxiliaries, such as feed water pumps, fuel oil pumps and draft fans, are likewise duplicated with one unit powered by the offsite system and the other by the onsite turbine-generator. This provides partial capacity operation of the secondary steam system and full-plant service boiler operation in the event of a complete failure of either the offsite AC power supply or the onsite AC power supply.

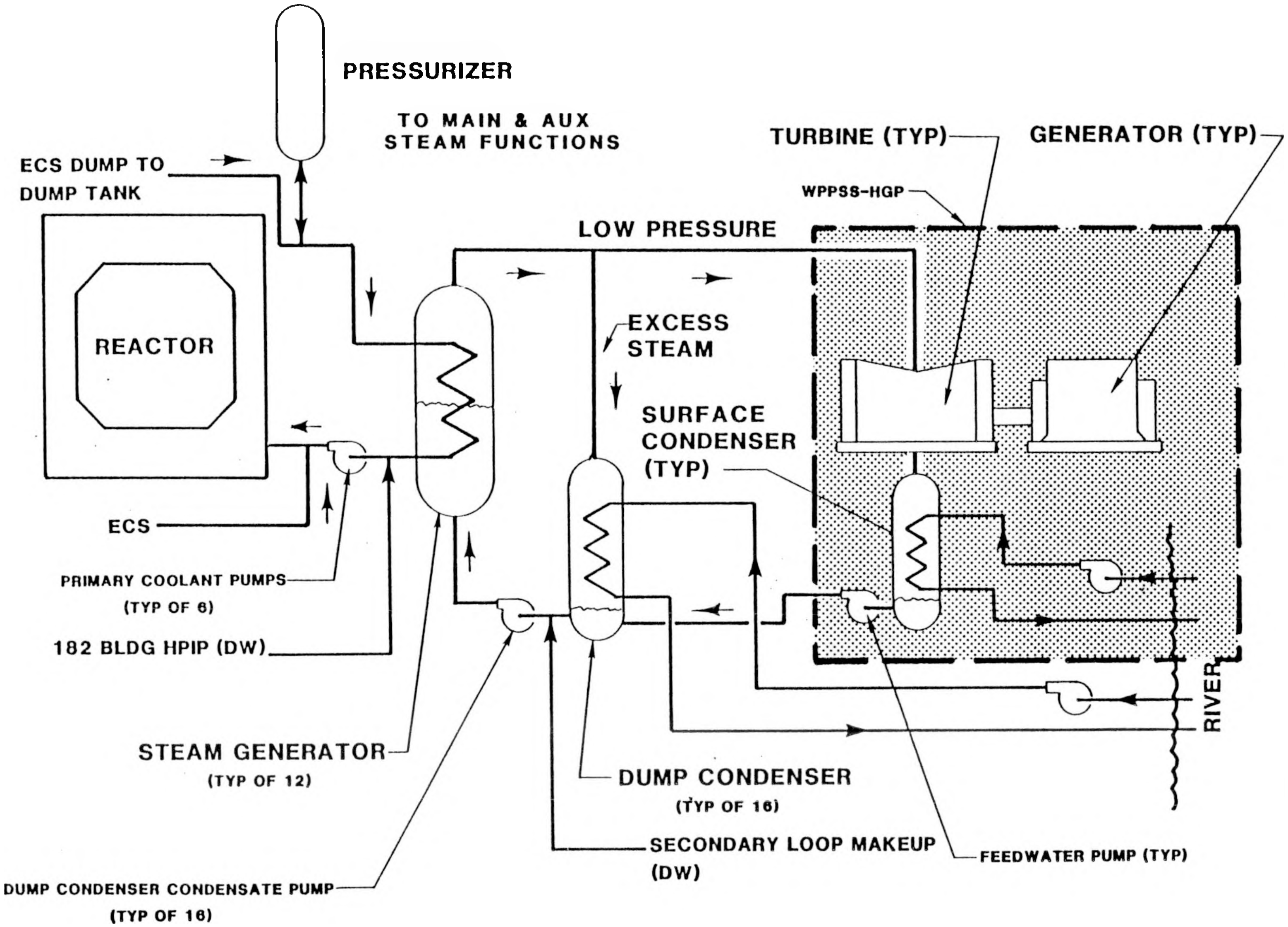


Figure 3-11. N Reactor Dual Purpose Operation.

During dual-purpose reactor operation, the major fraction of the steam generated is routed to the WPPSS Hanford Generating Project. 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 up the full steam load in the event of a rapid load rejection. Loss of feedwater supply, steam-line rupture and loss of river cooling water are all accidents that can defeat heat transfer from the reactor to the secondary system. However, the normal feedwater inventory in the steam generators is capable of cooling the reactor for approximately five min at full-power. Following reactor trip, adequate cooling can be provided for approximately 1 hr without further feedwater addition, provided the reactor coolant recirculation pumps remain in-operation.

3.8 Major Safety Systems

The following subsections provide summary descriptions of the safety systems which are most significant in accident mitigation. These are the plant systems whose response to an accident is directly modeled in the event trees presented in Section 5.0.

3.8.1 Reactor Trip System

The reactor trip system (RTS) limits the consequences of an abnormal condition by initiating trip (scram) of the horizontal control rod systems and, if required, the ball safety system. The RTS includes all electronics, signal processing equipment, and cabling from system sensors to the hydraulic rod drive mechanisms and the ball hopper latching mechanisms.

The RTS is energized during normal plant operation and fails safe as the result of failures such as loss of AC or DC power to the system.

The RTS detects abnormal conditions by monitoring a large number of plant parameters.

The RTS provides the following control signals in addition to reactor scram actuation:

- Start ECCS diesels (high and low lift)
- Open steam generator bypass (primary side) valves
- Start the standby HPI pumps
- Shut dump condenser condensate outlet valves
- Start standby boiler feed pumps
- De-energize pressurizer heaters
- Start afterheat removal pumps
- Reset pressurizer pressure and level setpoints

o Reset reactor recirculating cooling pumps to 30% flow.

These RTS actions are referred to as the Discontinuous Actions or the DAs. The DAs perform the following principal functions: (1) limit reactor coolant cooldown rate (shrinkage and thermal stress); (2) activate partially the ECCS in anticipation of the need for coolant addition; (3) enhance the ability for makeup to the primary loop; and (4) partially activate the standby boilers in anticipation of the need for additional steam supply.

An analysis is documented which concludes the DAs are not required to function to achieve safe shutdown and continued cooling of the fuel, without ECCS actuation.[2]

3.8.2 Emergency Core Cooling System

The ECCS provides a separate, independent water source for once-through cooling of the reactor. The ECCS provides core cooling for accidents that involve loss of coolant inventory, loss of flow through the reactor, or loss of normal heat sink. The system flow diagram is shown in Figure 3-12. Three diesel-driven high-lift pumps are provided to deliver water to the bottom of the 16 inlet risers where it enters the reactor coolant system through check valves when the system pressure falls below approximately 300 lb/in.^{2} (gage). The suction supply for the high-lift pumps consists of:

- o Demineralized water tank with a gross capacity of 900,000 gallons, and a minimum reserve of 412,500 gallons for ECCS use only (plant design changes are being considered to increase the ECCS reserve to extend time available for operator action)

- o The filtered water tank with a gross capacity of 855,000 gal (that may be brought online via a remotely operated valve).

After the supply from the demineralized water tank is exhausted, suction will normally transfer to three low-lift diesel-driven pumps located in the river pump house. The switchover is accomplished automatically through an air-water displacement system in the ECCS Silo that normally diverts flow from the low-lift pumps back to the river. Eight redundant air-operated valves (V-3 valves) in the supply system open in order for water to reach the core. Four V-3 valves serve each side of the reactor.

The ECCS flow is discharged via eight redundant air-operated valves (V-4 valves) connected to the reactor outlet piping. Four V-4 valves normally serve each side of the reactor. Because the V4-2 valve is currently removed (blanked off), this value is modeled as failed closed to reflect current reactor configurations. These valves function to depressurize the reactor coolant system following pipe breaks, loss of flow accidents, and loss of heat sink accidents thereby allowing flow of the low-pressure emergency cooling water through the core before fuel failure occurs. The ECCS discharge flow is routed to the emergency dump tank which normally contains sufficient cool water to quench the entire coolant system inventory. If continued ECCS flow is required, the dump tank will overflow to the liquid effluent retention facility. Control solenoids for the V-3 and V-4 valves

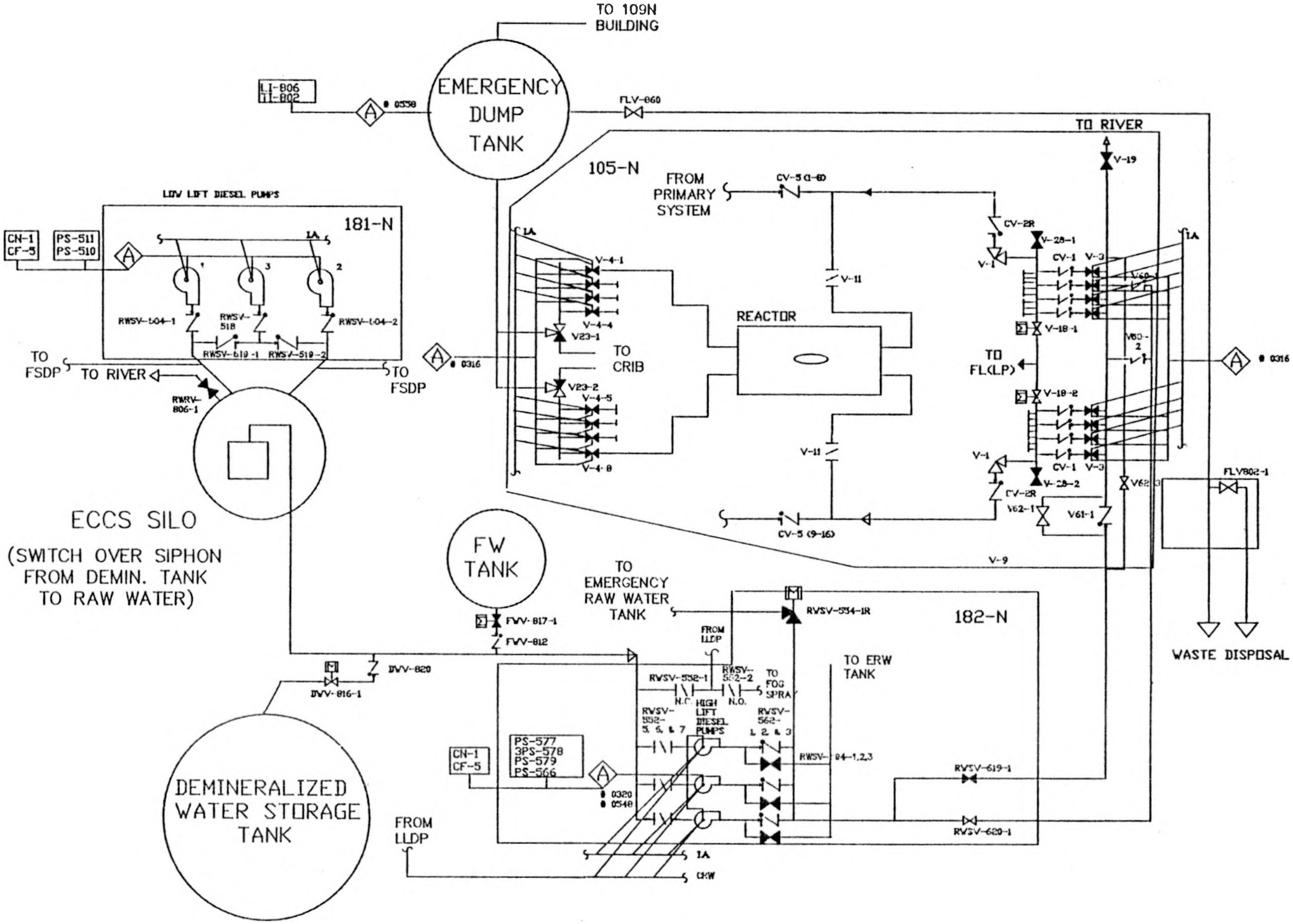


Figure 3-12. Emergency Core Cooling System.

fail as-is on loss of DC control power. Two independent DC power supplies are provided to ensure a high level of system reliability.

ECCS actuation and operation do not require AC power. The ECCS pump diesels are automatically air-started either on receipt of a reactor trip or by the engineered safety features actuation system (ESFAS). The engines may also be started manually. The V-3 and V-4 valves are opened automatically by the ESFAS. An ESFAS signal will result from any of the following conditions:

- Any two of the following trips:

  Primary coolant low-flow (2/4 trip logic)

  Primary coolant high temperature (2/3 trip logic)

  Primary pump low-speed (4/6-trip logic)

  Primary coolant low- or high-pressure (2/3-trip logic)

- Confinement high-pressure coincident with reactor coolant high- or low-pressure,

- Manual actuation.

Emergency procedures require reduction of ECCS flow to the reactor approximately five min following reactor trip.(3) The degree of reduction depends upon whether or not a loss of coolant has occurred. Procedures require the operator to place the GSCS in the once-through cooling mode rather than start the ECCS if the ECCS is not actuated within seven min following an ECCS flow demand. The seven min time limit is currently being reevaluated and will likely be extended considerably. The current basis is to avoid introducing water into the core after the first fuel element failure has occurred. More recent analysis indicates the prescribed operator action does not minimize the extent of accident progression nor does it minimize total risk.

3.8.3 Graphite and Shield Cooling System

The Graphite and Shield Cooling System (GSCS) (Figure 3-13) provides cooling for the graphite moderator, reactor thermal shields, and primary shields. Following reactor shutdown, and with no other core cooling available, the GSCS provides sufficient heat removal capacity to limit fuel damage to approximately 30% of the core.(1) If ECCS functions until the decay heat load has decreased to the point that GSCS can adequately cool the core and then fails, fuel damage can be avoided. For this study, it is assumed that 2 h is an adequate time for this reduction in heat removal requirement to occur. Ongoing analyses are expected to shorten this time. If so, the 2 h assumption will be revised in the full-scope Level 1 PRA.

During normal operation, the GSCS operates in a recirculation mode with three pumps, four heat exchangers, and a pressurized surge tank. The recirculation pumps are AC powered and the heat exchangers are cooled by the circulating raw water system. A once-through mode is also provided for the GSCS, with automatic transfer on either low GSCS flow or high GSCS

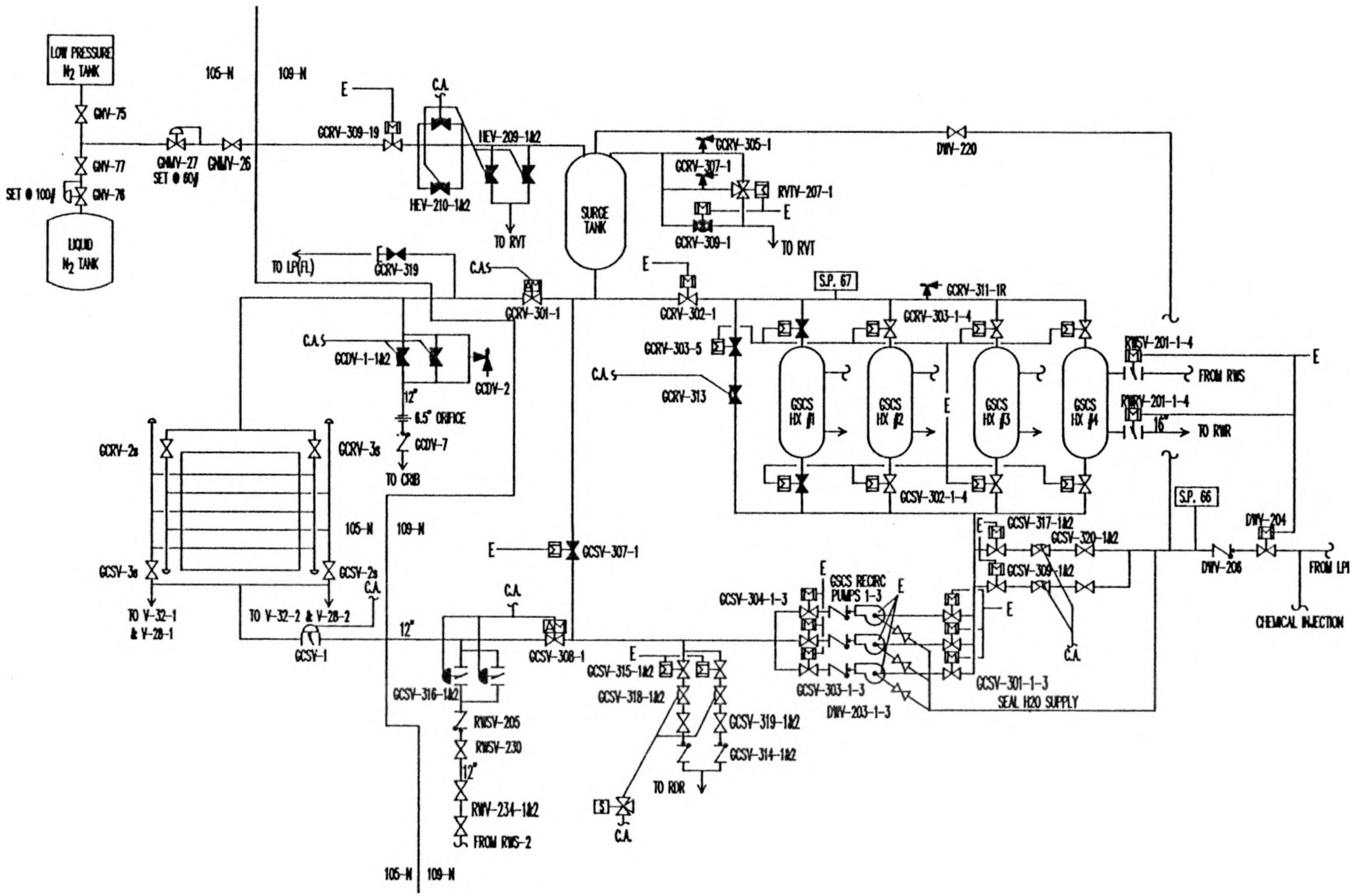


Figure 3-13. Graphite and Shield Cooling System.

temperature. Switchover to once-through operation is initiated by opening redundant supply valves from the high-pressure raw water system and redundant discharge valves to the radioactive drain system, which discharges the effluent to the Liquid Waste Disposal Facility (1325-N). The high-pressure raw water system is supplied by the circulating raw water pumps and the high-pressure raw water pumps, both dependent on AC power. An alternate supply is available from the two diesel-driven, fog-spray pumps, which in a loss of AC power, would receive suction supply from the low-lift ECCS pumps. The fog-spray diesels start automatically on low raw water header pressure and also on a high confinement pressure signal from the ESFAS. The AC power is not required for the GSCS to shift to the once-through mode, nor is AC power required to supply water to the GSCS from the low-lift diesels and the fog-spray diesels. A small tanked source of water to the fog-spray diesels (maximum of approximately 120,000 gal) is available normally from the emergency raw water tank. Additional water is available from the filtered water tank (855,000 gal capacity) by manual valve operation if this supply is not required by the ECCS.

3.8.4 High Pressure Injection System

The High Pressure Injection (HPI) System is the normal source of all makeup water for the reactor coolant system. During normal operation, HPI compensates for both random and planned leakage from the reactor coolant system, provides seal injection water for the reactor coolant recirculating pumps and is one of the sources for pressurizer spray. The system flow diagram is shown in Figure 3-14.

Principal HPI system components are five motor-driven pumps with variable speed fluid drive couplings and the 900,000-gal demineralized water storage tank. There are two HPI pumps powered from A Bus, two powered from B Bus, and one is an installed spare (powered from either A Bus or B Bus). During normal operation, two pumps are in-operation, one powered from each of the two buses. The remaining two pumps are in-standby, ready for automatic start. The standby HPI pumps are both started by the reactor trip signal. One standby pump is started if the total HPI demand reaches 1,300 gal/min; both are started at 2,600 gal/min. Total HPI capacity (at normal system pressure) is 6,000 gal/min.

The HPI system capacity establishes the upper limit for a small-break loss of coolant accident (i.e., a coolant loss that can be compensated for by the HPI system). During a non-LOCA event, the HPI system is required to operate to prevent the pressurizer from emptying, following a reactor trip. Where applicable, the event trees in Section 5.0 are constructed on the assumption that loss of HPI will result in a primary system failure to remove heat and ECCS actuation signal is required.

3.9 Major Support Systems

The following subsections provide summary descriptions of the major support systems that are important to the operation of the reactor and safety systems.

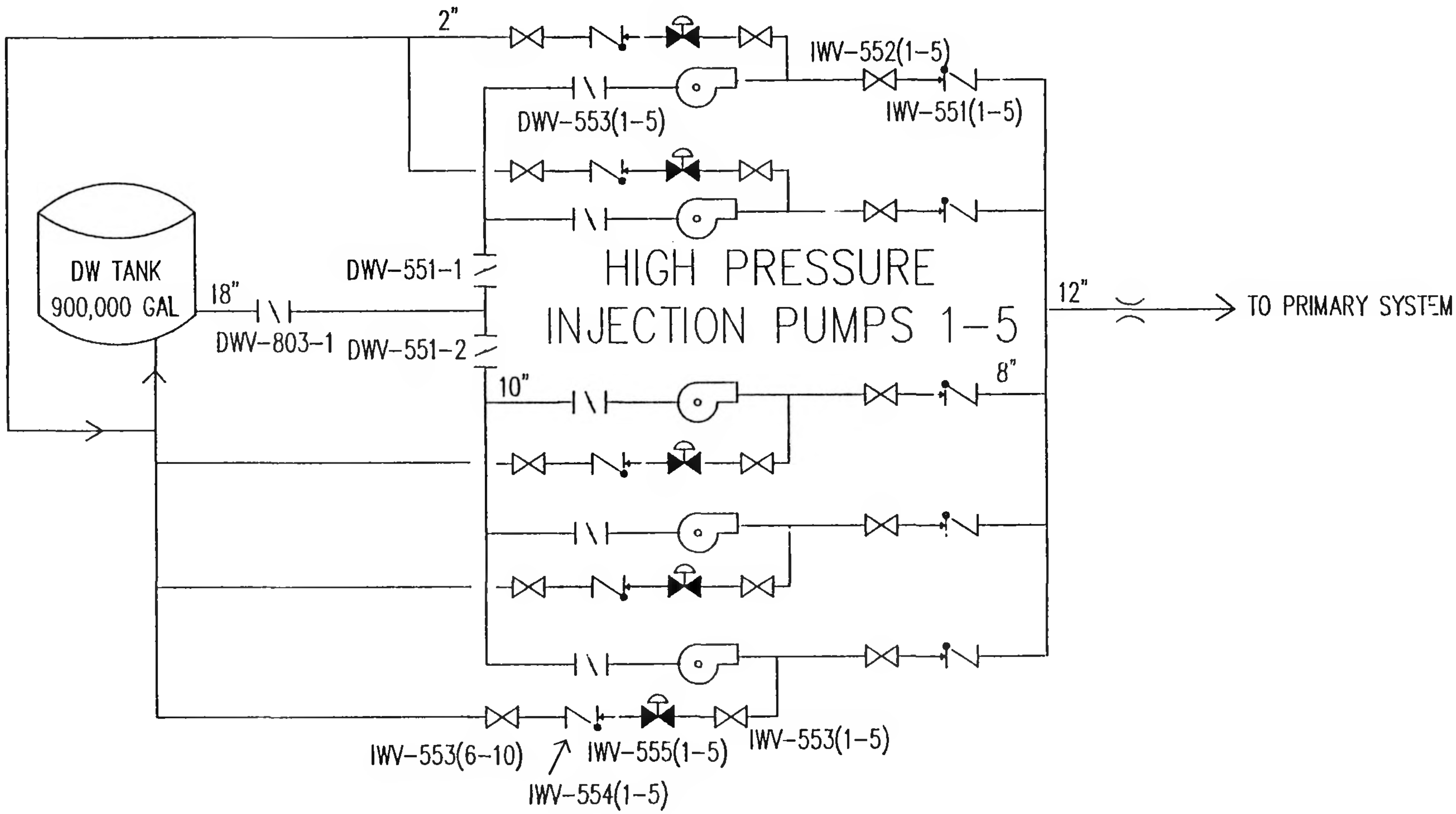


Figure 3-14. High Pressure Injection System.

3.9.1 Electric Power

With the exception of the steam-turbine-driven reactor coolant pumps, all equipment for normal operation of the reactor and its cooling system is electrically powered. This power is supplied by two independent systems: (1) the offsite Hanford, 230-kv Loop and (2) the onsite 13.8-kv turbine-generator in the 184-N power house. The onsite turbine-generator is rated at 22 MVA. Electrical loads for the various functions are distributed between A and B Buses.

The A Bus is supplied from the Hanford 230-kv loop, via two physically independent tie circuits. The Hanford loop is supplied from two independent Bonneville Power Administration substations with multiple sources to each substation. The B Bus is supplied from the turbine-generator in the 184-N power house, which receives steam either from N Reactor or from a redundant set of onsite, oil-fired boilers.

The A and B Buses are isolated from each other electrically during operation. This separation is maintained at all user voltages (13.8-kv to 120-v). Both buses are required for normal reactor operation, but either bus can support the necessary loads during a reactor scram and while the reactor is shutdown. In the event of failure of one system, interconnection is possible by manual switching, once the reactor is in a safe shutdown condition. This arrangement ensures independence and precludes failure in one system inducing failure in the other, or a single fault causing failure of both systems. The offsite Hanford 230-kv loop system and the onsite 13.8-kv power system are independent from the power generated by the WPPSS Hanford Generating Project (using steam from N Reactor in dual-mode operation).

Essential AC instrument power (120-v) is backed up by the DC power system through an uninterruptable power supply system and is therefore independent of the normal sources of A and B Bus AC power.

The DC power systems provide a continuous source of power to critical instrument and control components and provide status indication of N Reactor facilities.

The 24-v DC power system supplies power for status indication of operating systems and standby equipment needed for safe shutdown, following an accident.

The 125-v DC power systems in each building provide power for instrumentation that regulates, controls, and monitors operation. Major systems served include the rod and ball safety circuits; the emergency cooling and graphite cooling water control systems; the confinement and fog-spray control systems; critical primary system control valve circuits; the reactor coolant pump circuits; turbine-generator controls; and 13.8-kv, 4.16-kv, and 480-v switchgear. There are two independent 125-v DC power supply systems in the 109-N building and three in the 105-N building. Some of the ECCS diesel engine starting solenoids and governors are also supplied by the 182-N and 181-N DC power supply systems.

3.9.2 Raw Water Systems

The principal raw water systems are circulating raw water (CRW), low-pressure raw water supply (LPRW), and high-pressure raw water supply (HPRW). The raw water supply headers are designated RWS-1 (low pressure) and RWS-2 (high pressure).

The CRW system (see Figure 3-15) consists of four pumps (located in the 181-N pumphouse) and appropriate headers and lines that supply Columbia River water for a variety of plant uses including:

- Water supply to the filter plant
- Cooling for the GSCS heat exchangers
- Cooling for the reactor coolant pump drive turbine condensers and oil coolers
- Cooling for the 16 secondary cooling system dump condensers
- Service water pump supply
- Water supply for the low-pressure and high-pressure raw water pumps.

The three low-pressure raw water pumps (see Figure 3-16) take suction from the CRW system and provide cooling water for the HPI pump oil coolers, the high-pressure air compressors, and the auxiliary jacket cooling supply for the high-lift diesels and the fog-spray diesels.

The high-pressure raw water system (see Figure 3-17) consists of three high-pressure raw water pumps and two fog-spray diesel pumps. These pumps take water from the CRW pumps. In addition, the fog-spray diesel pumps can be supplied by the low-lift diesel pumps (LLDP) as well as the emergency raw water tank (normally aligned) and the filtered water tank (via a manual isolation valve). The high-pressure raw water system supplies water to:

- The GSCS in the once-through cooling mode
- Confinement fog-sprays
- Horizontal rod cooling system (backup).

With the exception of the fog-spray diesel pumps and ECCS diesel pumps, all of the raw water pumps require AC power for operation.

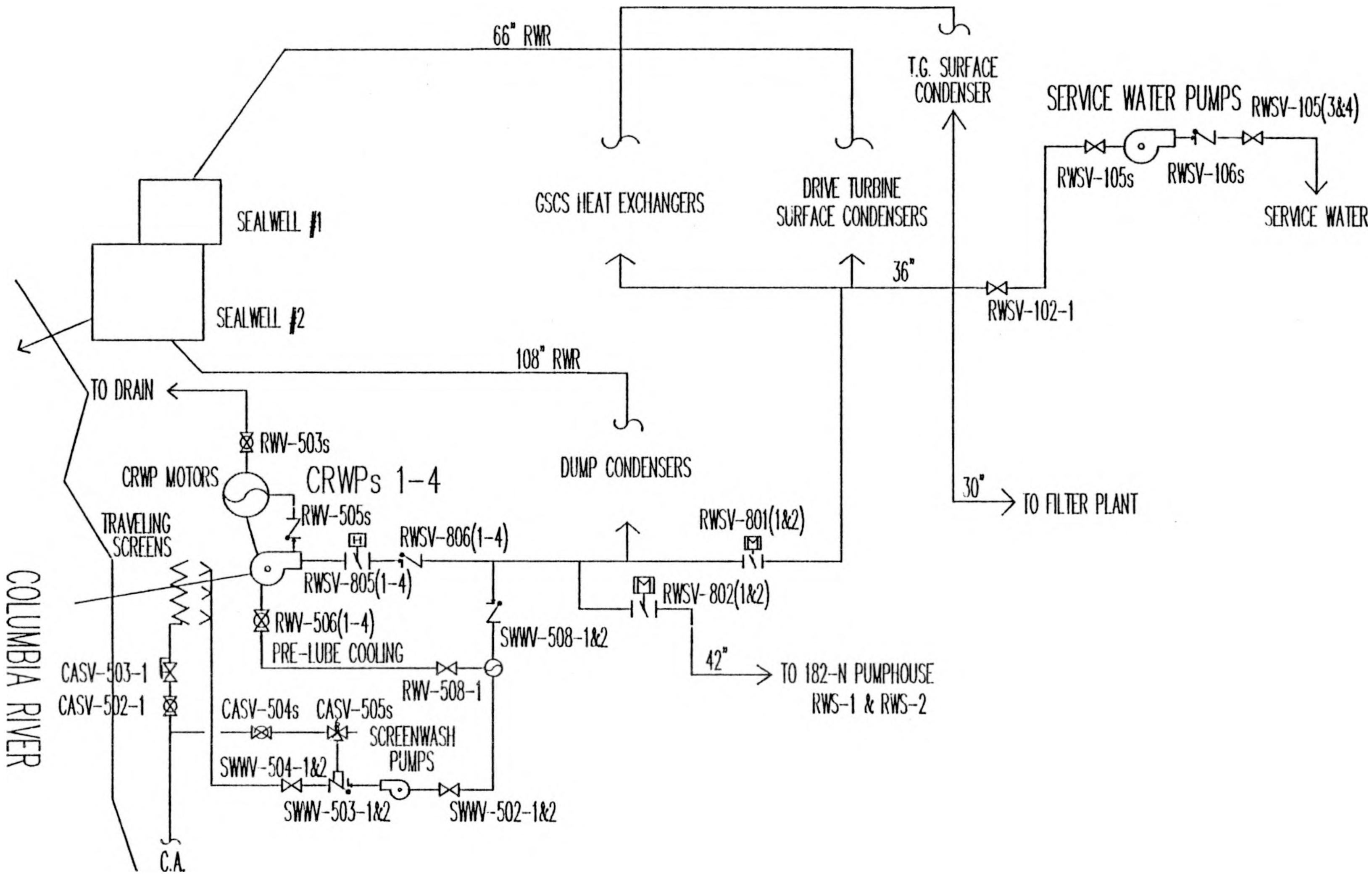


Figure 3-15. Circulating Raw Water System.

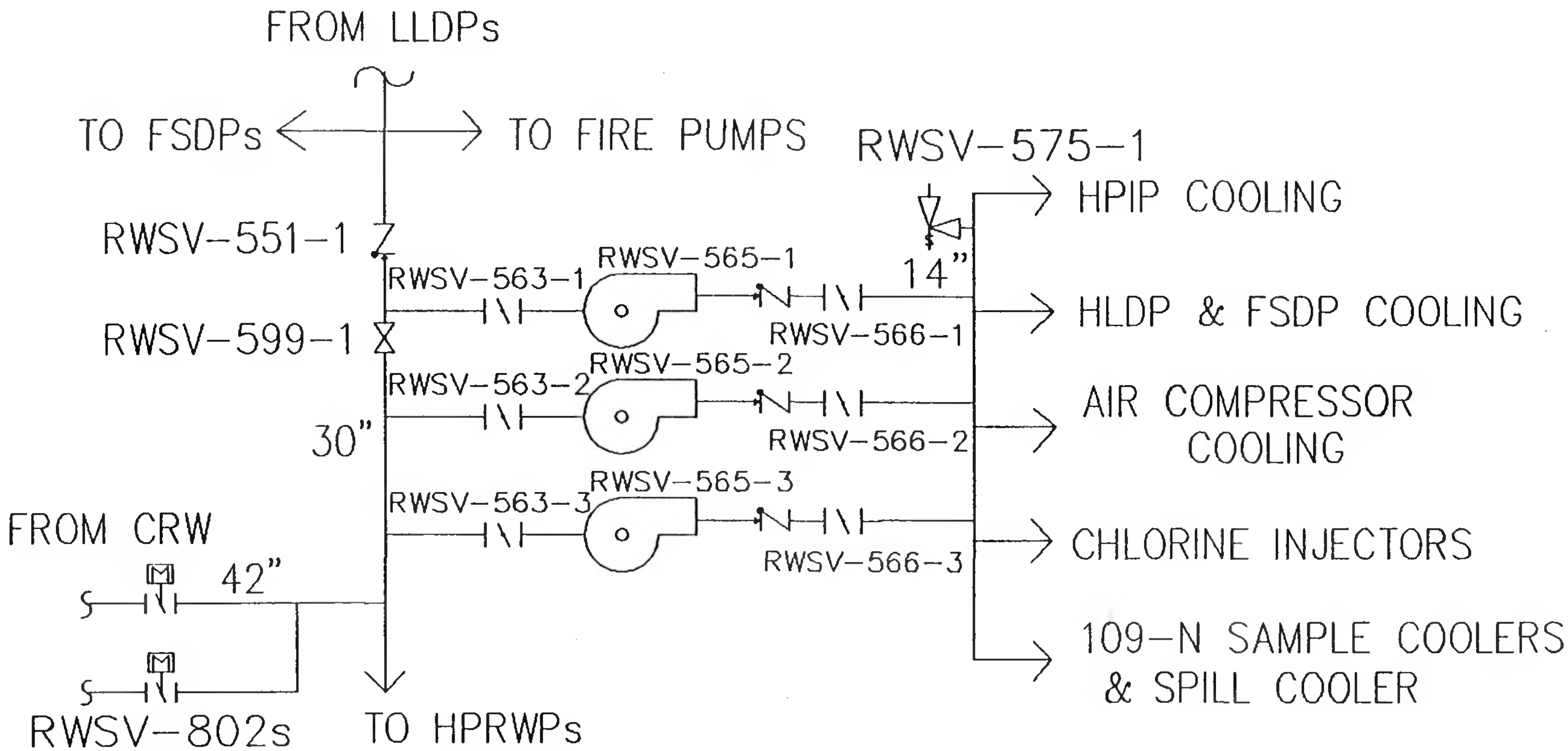


Figure 3-16. Low Pressure Raw Water System.

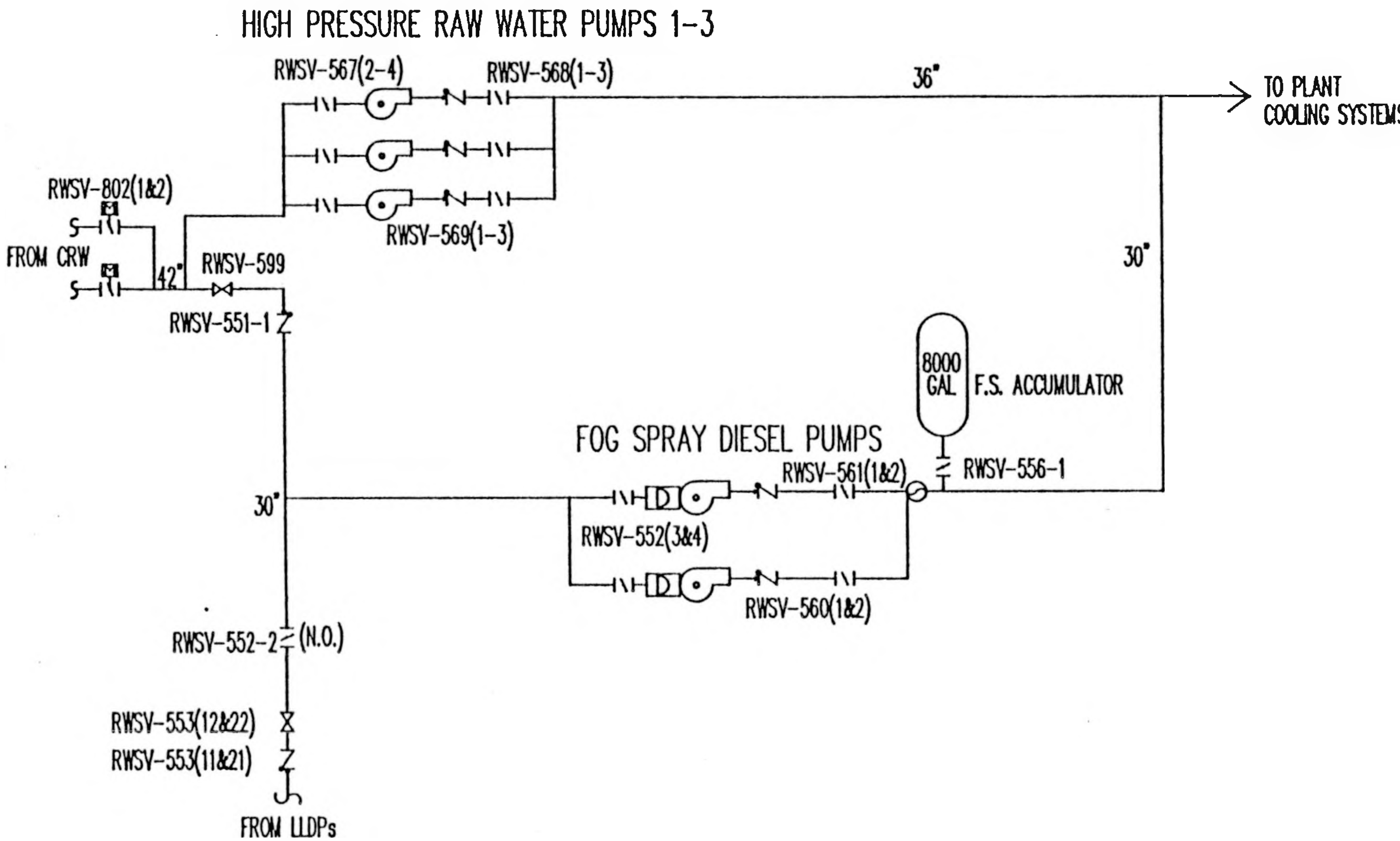


Figure 3-17. High Pressure Raw Water System.

3.10 References

1. N Reactor Updated Safety Analysis Report, UNI-M-90, UNC Nuclear Industries, Inc., Richland, Washington.

2. S. W. Heaberlin, 1987, Success Criteria for Primary Discontinuous Action, Technical Memorandum (TM)-87-ESA-08, (letter to P. A. Carlson, UNC Nuclear Industries, Inc., Richland, Washington, February 2, 1987).

3. UNC, 1986, Emergency Response Guide B, UNI-M-2, VOL 2, UNC Nuclear Industries, Inc., Richland, Washington.

4.0 INITIATING EVENTS

4.1 Summary of Initiating Events

A summary of the initiating events and frequencies developed for this study is provided in Table 4-1. The selection of internal and external events is discussed in Sections 4.3 and 4.4, respectively.

4.2 Scope of Events Considered

Internal initiating events considered in this study are based on the initiator groups identified in Phase I of the full-scope Level 1 PRA.[1] These initiators were identified from a review of plant historical data and from the Phase I functional analysis of systems responses to accident conditions. Initiators resulting from support system failures were also considered based on the historical data analysis. These results will be further reviewed in the full-scope Level 1 PRA and additional accident initiators may be identified.

External initiators were identified for consideration based on other nuclear reactor PRAs and on the analyses contained in the NUSAR.[2] These events were reviewed, largely on the basis of engineering judgement, to identify those expected to be risk-significant.

4.3 Internal Events

This section summarizes the analyses of internal accident initiators included in this study.

4.3.1 <u>Preliminary Internal Event List</u>

A preliminary internal initiating event list was generated by integration of the Phase 1 PRA functional analysis and the historical data analysis. These analyses developed a functional relationship between the causes of plant trips or shutdowns, and the functions and associated systems called on to mitigate plant damage that could result from the initiating events. The initiators identified from the safety functional analysis and plant historical data were then collected into more general groupings, based on the following common characteristics:

- o Initiators that result in the response of the same mitigating systems with the same success criteria were grouped together
- o Initiators for which the physical progression of the accident activates mitigating functions in the same sequence were grouped together.

Table 4-2 lists the initiator groups, group descriptions, and corresponding initiators.

Table 4-1. Summary of Initiating Event Frequencies

Event	Frequency/Yr
Large Loss of Coolant Accident	1.0 E-3
Small LOCA	1.2 E-2
Local Loss of Flow	2.5 E-2
Rapid Shutdown from Power	11
Transients	5
Overpower	2.7 E-1
Large Steam Line Break	1.3 E-3
Station Blackout	5.2 E-5
Interfacing LOCA	1.0 E-9
Seismic	See Figure 4-2

Table 4-2. Accident Initiators and Initiator Groups. (sheet 1 of 4)

Group number	Initiator Group	Group Description	Corresponding Initiator
1	Large LOCA	Events causing openings in the primary system boundary such that ECCS pumps are called-on to supply makeup adequate to prevent fuel overheating (i.e., break sizes that drop the primary pressure to the 200-lb/in^2 range).	• Failure of primary system valve (large, rupture) • Pipe leak/break in primary system (large)
2	Intermediate LOCA	Events where the break does not depressurize the primary system and the HPI system cannot adequately supply makeup; thus, V-4 actuation to depressurize is required.	• Failure of primary system valve (intermediate, rupture) • Pipe leak/break in primary system (intermediate) • Spurious relief valve opening (multiple valves)
3	Small LOCA	Primary system leaks that maintain primary system pressure above ECCS actuation pressure, and successful HPI operation provides makeup water to replace leakage.	• Failure of primary system valve (small, seal/packing) • Pipe leak/break in primary system (small) • Spurious relief valve opening (single valve) • Pump seal failure
4	Leak between primary and interfacing system	Primary system leaks outside confinement boundary	• Leak between primary and interfacing system (ECCS)
5	Primary system local loss of flow	Partial flow loss in the primary coolant system.	• Flow blockage in one tube • Valve closure
6	Primary system global loss of flow	Events causing complete primary coolant flow loss.	• Loss of make-up by HPI systems • Failure of all reactor coolant pumps • Closure of all reactor coolant valves • Loss of steam to reactor coolant pump drive turbines and coincident pony motor failure • Loss of AC power to A Bus and B Bus
7	Steam generator tube rupture	Events involving breach of the interface between the primary and secondary systems	• Steam generator tube rupture

Table 4-2. Accident Initiators and Initiator Groups. (sheet 2 of 4)

Group number	Initiator group	Group description	Corresponding initiator
8	Small steam line break inside confinement	Any steam leak from the secondary system within the confinement boundary not large enough to cause initial primary system depressurization (by overcooling)	• Steam line break inside confinement • Steam blowdown (inadvertent opening of relief valve)
9	Large steam line break inside confinement	Steam line breaks inside confinement large enough to depressurize primary system such that ECCS is initiated	• Steam line break inside confinement (large)
10	Steam line break outside confinement	All excessive steam flow events outside confinement; different from breaks inside confinement due to confinement loading and heat removal capacity	• Steam line break (small or large, outside confinement) • Increased steam flow
11	Large feedwater line break inside confinement	Large feedwater line breaks inside confinement results in confinement loading and loss of heat removal capacity	• Feedwater line break (large)
12	Other feedwater line breaks	Events resulting in loss of feedwater other than large breaks inside confinement	• Feedwater line break (small inside confinement, any size outside confinement) • Inadvertent actuation/failure of surge tank high level spill (system)
13	Loss of feedwater	Events preventing flow of secondary coolant to one or more steam generators (i.e., loss of heat sink for primary system (steam generators) and secondary systems (dump condensers, etc.))	• Loss of HGP turbine generator system • Failure of dump condenser • Failure of circulating raw water system • Loss of steam generator (not tube rupture) • Loss of feedwater flow • Loss of flow from HGP turbine (condensate return from HGP turbine) • Loss of condensate steam
14	Turbine trip	Events where normal plant heat sink/load is lost	• Loss of steam flow/steam pressure control valve closure (all valves) • Loss of condenser vacuum • Condenser leakage • Turbine trip • Generator trip

Table 4-2. Accident Initiators and Initiator Groups. (sheet 3 of 4)

Group number	Initiator group	Group description	Corresponding initiator
15	Partial loss of main steam flow	Considered separately from turbine trip due to lower consequences and less need for dump condensers	• Failure of steam pressure steam flow regulator • Loss of steam flow/steam pressure control valve closure (single valve)
16	Overpower transient	Events causing an increase in reactor power; all reactivity insertion events	• Uncontrolled HCR withdrawal (slow) • Uncontrolled HCR withdrawal (fast) • Loss of HCR cooling • Leakage of HCR coolant into core • Inadvertent discharge of balls • Fuel element enrichment errors • Fuel reseats at operating pressure • Loss of moderator (graphite) cooling • Leakage of GSCS coolant into core • Increased reactor coolant flow/excessive make-up • Inadvertent cold water injection into the primary system • Reflector/shield shift (structural failure) • Control rod position error
17	Spurious ECCS initiation	Events where an inadvertent actuation signal starts ECCS in absence of any safety safety problem.	• Spurious ECCS signal initiation
18	Reactor trip	Events involving reactor trip in absence of any safety problem	• Automatic plant trip • Manual plant trip
19	Excessive feedwater	Events where excess heat is removed from steam generators due to excessive feedwater addition	• Increased feedwater flow
20	Loss of moderator cooling	Events unique to graphite-moderated N Reactor	• Loss of moderator (graphite) cooling • Leakage of GSCS coolant into core

Table 4-2. Accident Initiators and Initiator Groups. (sheet 4 of 4)

Group number	Initiator group	Group description	Corresponding initiator
21	Loss of HCR cooling	Events resulting from unique N Reactor design where control rods require cooling	• Loss of HCR cooling • Leakage of HCR coolant into the core • Loss of high-pressure filtered water • Loss of low-pressure filtered water
22	Support system failure	Events initiated by failure of a support system	• Loss of offsite power • Loss of an AC bus • Loss of a DC bus • Loss of instrument air • Loss of circulating raw water

ECCS = Emergency Core Cooling System
HCR = Horizontal Control Rod
HGP = Hanford Generating Plant
HPI = High-Pressure Injection System
LOCA = Loss-of-coolant accident.

PST87-1159-2

The following internal initiator groups are analyzed in detail in this study:

- Large LOCAs (Groups 1,2, and 17)
- Small LOCAs (Group 3)
- Primary System Global Loss of Flow (Group 6)
- Primary System Local Loss of Flow (Group 5)
- Reactor Trip From Power (Group 18)
- Overpower Transients (Groups 16, 19, and 21)
- Large Steam Line Break Inside Confinement (Group 9)
- Support System Failure (Group 22)
- Interfacing LOCA (Group 4).

4.3.2 Basis for Exclusion

Internal initiating groups that were determined to be either of lesser probability or unlikely to lead to fuel damage are the secondary system upsets (Groups 7, 8, 10, 11, 12, 13, 14, and 15). Because N Reactor has a much larger secondary coolant inventory than commercial nuclear power plants, and ten normally operating steam generators (arranged in five cells of two steam generators each), it is considerably less sensitive to changes in the secondary system than commercial power plants. The large steam-line break is considered to bound the consequences of any other secondary system initiator.

Loss of moderator cooling (loss of GSCS) was not considered as a separate initiator since it is a branch in every event tree and its immediate effect, a very small reactivity change, is implicitly included in the overpower transient event tree.

A study has been performed[3] that investigated the potential for graphite oxidation and the potential contribution of such an event to the progression of severe accidents. The study concluded that, for postulated core degradation scenarios, the graphite oxidation would not be a significant contributor to the heat balance.[3] It was also concluded the stored energy (Wigner) is insignificant, relative to the ability of the graphite to accommodate such an energy release in its thermal mass. The overall conclusion was the graphite oxidation does not represent an accident initiator, nor would it aggravate accidents initiated by other causes. It was also recently determined that a graphite fire was not a contributing factor at the Chernobyl accident.[4]

4.3.3 Quantification

This section discusses the quantification of the frequency for each of the internal initiators identified in Section 4.3.1. The frequencies are listed in Table 4-1.

4.3.3.1 Large Loss of Coolant Accidents. The overall large LOCA frequency was estimated based on a combination of total piping length above 2.0 in. inner diameter (ID) and the number of valves in the primary system[5] and the related failure probabilities, using data from WASH-1400.[6] The results of the calculations are as follows.

	Length/ Number	Failure Frequency/Yr
Piping (above 2.0 in. ID):	12,597 ft	1.3 E-05
Valves:	90	9.9 E-04
Pumps:	5	5.2 E-06
Total Primary Large LOCA Frequency:		1.0 E-03

Where basic failure rates used in this calculation are:

- o Piping: 1.0 E-9 failures/ft/yr
- o Valves: 1.1 E-5 failures/valve/yr
- o Pumps: 1.0 E-6 failures/pump/yr.

The failure mode of concern is catastrophic loss of pressure boundary integrity.

4.3.3.2 Small Loss of Coolant Accidents. Small piping LOCA (defined in Section 5.3.2) frequencies were estimated in a manner similar to that used for large LOCAs (i.e., determining the piping length, number of valves, and multiplying these values by relevant failure frequencies/valve or /ft of piping. The basis for process tube break frequency is contained in a probabilistic fracture mechanics study of the reactor process tubes completed in 1986.[7] The results of the calculations are as follows.

	Length/ Number	Failure Frequency/Yr
Piping (less than 2 in. ID):	164,661 ft	1.6 E-04
Valves:	1100	1.2 E-02
Frequency of a Pressure Tube Break Within N Reactor:		1.0 E-05

Total RCS Small LOCA Frequency:	1.2 E-02

4.3.3.3 Primary System Global Loss of Flow. The failure mechanism of concern is a total loss of reactor coolant flow due to the simultaneous loss of the primary coolant pump drive turbines and pony motors. The direct initiator would be a drive turbine trip or loss of steam combined with a loss of offsite power to A Bus. Analysis of the primary system fault trees shows the only significant common-cause failure that would result in simultaneous loss of all primary coolant pumps is the simultaneous loss of both A and B Bus. Because this scenario is analyzed in more detail in the station blackout event tree, a separate event tree is not developed for global loss of primary system flow.

4.3.3.4 Primary System Local Loss of Flow. The preliminary frequency for a complete pressure tube blockage (2.5 E-02/yr) is based on an historical data analysis of partial blockages performed by LANL.[8] More detailed analysis during the full-scope Level 1 PRA is expected to reduce this estimate.

4.3.3.5 Rapid Shutdown from Power. Rapid shutdown from power includes normal shutdowns initiated by manual scram, trips due to instrument anomalies, and transients (defined as conditions in the reactor or reactor coolant systems having the potential to result in fuel failure if the reactor scram systems fail to function). A review of the last 10 yr of reactor operation showed there have been an average of 11 trips/yr in this category.[9] Of these, an average of 5/yr are considered transients with potential for fuel damage if the reactor scram (rod and ball systems) functions fail.[10]

4.3.3.6 Overpower. The frequency for this initiator was determined by analysis of plant outage data.[11] There have been six overpower transient events in 22 yr of operation for a frequency of 2.7 E-01/yr. These transients were caused by horizontal rod erratic movement (three), GSCS dump (one), a fast period during rod calibration tests (tripped at a power level of 0.5 MW), and one rate of rise trip caused by a drive turbine speed increase.

4.3.3.7 Large Steam Line Break Inside Confinement. The large steam line break frequency was calculated based on a combination of total piping length and the number of valves in the large steam line portion of the secondary system.[12] The results of the calculations are as follows:

	Length/ Number	Failure Frequency/Yr
Piping:	7,228 ft	7.2 E-06
Valves:	130	1.3 E-03
Total Steam Line		

Break Frequency: 1.3 E-03

4.3.3.8 Support System Failures. Two support system failures were identified as potential accident initiators: (1) total loss of AC power (a station blackout event); and (2) loss of circulating raw water. Other support system failures were considered to be less significant and will be analyzed in the full-scope PRA.

- Station Blackout. This event is particularly significant because it may result in a global loss of flow. Additionally, many safety-related systems have AC power dependencies.

  Because N Reactor has two independent AC power sources, a station blackout would require the simultaneous loss of both. Based on historical data[13] there have been 6 loss of A Bus events over the past 23 years. Although the longest time power outage was 9 min, it is conservatively assumed that the frequency of a 1-h power loss is 6 events in 23 yr or 2.6 E-01/yr. For a station blackout to occur, B Bus must also be lost during this 1-hr period. Calculations based on data from the preliminary LANL Level 1 PRA[8] indicate the dominant B Bus failure mode to be a loss (failure to continue to run) of the turbine-generator, at a frequency of 2.0 E-04/h. Combining the frequency of these two failures results in a frequency of a station blackout lasting 1 h of 5.2 E-05/yr.

- Loss of Circulating Raw Water. Total CRW loss is significant because many safety-related systems are dependent on CRW for cooling and/or water supply. Another essential function is to provide reliable cooling to the turbine generator and the primary cooling pump surface condensers on loss of A Bus. The system design ensures the failure of any single active component will not result in the failure of the total system. This is accomplished by the use of dual, electrical power supplies, split supply-headers with isolation valves, standby equipment, and a system designed to supply adequate coolant under both normal and emergency conditions.

  An operational analysis of the CRW system[14] has shown it is a major contributor to plant outage time and has caused unscheduled outages. Most of these outages were due to CRW pump motor problems; however, human error, debris, and procedural control have also been contributors.

  The CRW requirement during plant operation is that one A Bus and both B Bus CRW pumps must be operating. One pump must continue to operate during any transient until cold shutdown is reached. Thus, the total loss of CRW initiator is assumed to include the loss of three operating CRW pumps and the failure to start the backup pump. The only credible initiator identified for this event is a total loss of AC power and thus, this event is not considered as a separate initiating event.

4.3.4 Interfacing Loss of Coolant Accident

One potentially important interfacing system LOCA sequence was identified. This sequence involves the potential combination of a failure of any one of four CV-1 check valves, and a failure of 1 of 8 CV-2R check valves to properly reseat on either side of the reactor, so high-pressure leakage is possible past these valves.

The interface between the low-pressure ECCS and the high-pressure primary system is shown in Figure 4-1. On an ECCS demand, the V-3 water inlet valves to the core and the V-4 depressurization valves open simultaneously. This implies if a leak path is available past at least one set of CV-1/CV-2R valves, a high-pressure failure of the low-pressure ECCS is possible, when the V-3 valves open. For purposes of this analysis, it is assumed such a high-pressure surge will cause ECCS failure (due to failure of pump seals) and coincidentally provide a fission product release path to 182-N, thereby bypassing confinement.

In analyzing the importance of this scenario, the frequency of occurrence must be estimated. It is judged that the interfacing LOCA can occur (given ECCS demand under high, primary-system pressure conditions) if any one CV-1 and any one CV-2R valve (on a given side of the reactor) are failed and the RV-1 relief valve in the ECCS line fails to open <u>or</u> if any two CV-1 and two CV-2R valves are failed (RV-1 opening is assumed to be insufficient in this case).

The CV-2R valves are checked for proper seating at least once every six mos. The CV-1 valves are checked at least annually. While the CV-1 check tests both the CV-2R and CV-1 valves together, small holes in the outboard CV-2R valves on both sides of the reactor allow for pressure buildup (but not high flow capability) up to the CV-1 valve during the tests. Failure of a CV-1 to reseat should therefore become apparent in the test even if all the CV-2R valves properly reseat. Thus, the semi-annual test is assumed to be adequate to check the valid closure of the CV-1 valves as well.

Given the 6 mo. check of the CV-2R valves and the annual check of the CV-1 valves, and using 3.75 E-07/hr (Reference 6) as a mean value for a significant reverse leak failure rate for the check valves, the frequency of a single CV-2R failure is 8.2 E-04/yr and of a single CV-1 failure is 1.6 E-03/yr. Thus, the frequency of any of the 64 possible combinations of check valves being failed is 8.4 E-05. Using a failure on demand of the appropriate RV-1 valve (depending on which side of the reactor the failed check valves exist) of 3 E-04[15] results in an interfacing LOCA probability of approximately 2.5 E-08. Combinations of 2 sets of CV-2R and CV-1 valves thereby already failing the RV-1 function contributes less than 1 E-09 so the overall interfacing LOCA probability given ECCS demand under high-pressure remains about 2.5 E-08.

The number of ECCS high-pressure demands/yr is quite low. Based on the other event tree sequences and frequencies discussed in Section 5.0, the ECCS demand rate under high-pressure is estimated to be approximately 2 E-02/yr. This is supported by the fact that no ECCS demand has occurred during power conditions in the plant's history. This implies the overall frequency of the interfacing LOCA sequence is less than 1.0 E-09/year. This estimate suggests the interfacing LOCA is not a significant risk

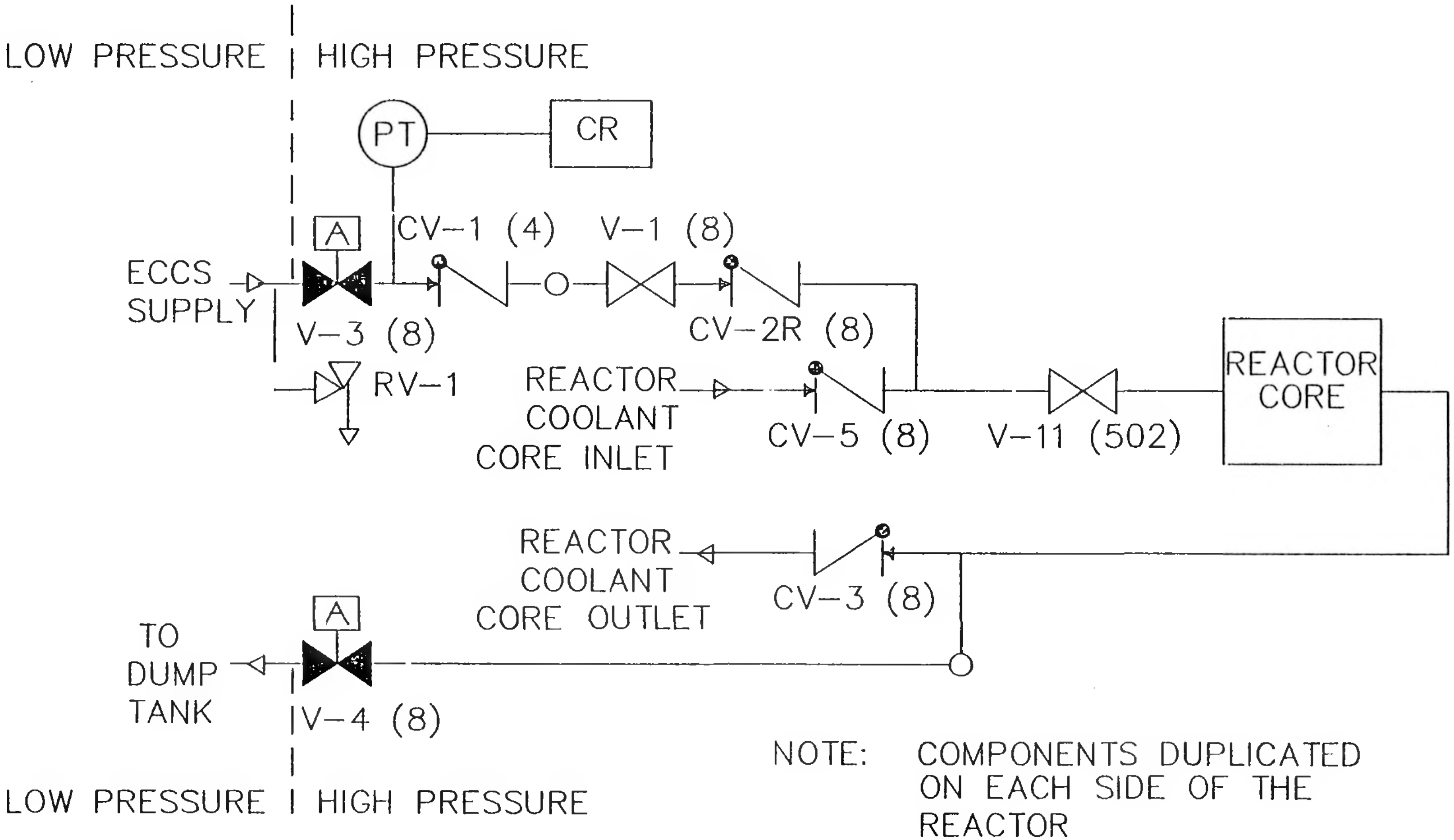


Figure 4-1. Emergency Core Cooling System/Primary System Interface for Interfacing Loss of Cooling Accident.

contributor. However, it is addressed in the risk calculations because confinement bypass results and because of its significance in previous power reactor PRAs.

4.4 External Events

4.4.1 General Initiator Review

A review was conducted of external events that might disrupt normal operation leading to the need for emergency heat removal. Seismic activity has been found to be a significant risk contributor in a number of commercial reactor PRAs and was, therefore, quantitatively analyzed to a degree of detail consistent with the limited scope of this study. Section 4.4.2 describes the analysis of N Reactor response to a seismic event. Other potential hazards (for example, external flood, missiles, transportation accidents, and toxic gas) were analyzed in the NUSAR(2) and were judged not to be significant risk contributors.

During the course of this limited-scope study, a plant walkthrough was conducted, of which one purpose was to identify sources of fire and internal flooding appeared to be capable of interfering with safe shutdown of the reactor and continued core cooling. During the walkthrough, virtually all areas of 105-N, 109-N, 181-N, and 182-N were visited, as well as surrounding areas of the plant.

The fire investigation took into account fire prevention and mitigating features, as well as potential sources of fire. Two potential fire sources were identified: (1) bulk fuel storage tanks; and (2) two large, dry transformers in the instrument cabinet and cable spreading room beneath the main control room (room 6, 105-N). On further investigation, both of these were eliminated as credible hazards. A bulk oil fire would be well-separated from plant vital equipment and control areas and would not immediately affect normal plant operation emergency systems operation. A foam fire suppression system is provided for the storage tanks and a well-equipped, highly trained fire-fighting force is continuously available within 4 min of receiving an alarm. (A larger backup force is also available, if needed.) Investigation of the room 6 transformers revealed they are dry-type with a low probability of fire or explosion. Even in the event of an energetic fault (such as a dead-short) it was determined that sufficient force to penetrate the casing would not be generated.

A more complete fire analysis will be performed as part of the full-scope PRA; however, it is not expected that any risk dominant fire initiators will be identified.

Only one internal flood hazard was identified that was considered to warrant further analysis in this Limited-Scope PRA. The postulated event is catastrophic rupture of a 30-in raw water line adjacent to room 6, 105-N. A flood inside room 6 could lead to loss of A and B Buses, failure of the V-3 and V-4 ECCS valves to open automatically, as well as other possible mitigating system failures and spurious actuation. This scenario is judged to be nonrisk significant because of design features that would delay or possibly prevent entry of water into room 6. These features include adjacent corridors leading to floor drains, 18-in plates welded to the bottom of door frames, and the ease that the raw water pumps can be deenergized to terminate break flow. In addition, ECCS pumps can easily be started from local controls in 181-N and 182-N. Although V-3 and V-4 controls could be impacted by flood, they are located high above the floor in their respective cabinets and are likely to remain functional. As is the case for fire, a more comprehensive treatment of internal flooding will be included in the full-scope PRA. Based on the plant walkthrough conducted during the course of this study, it is not expected that any risk-significant flood initiators will be identified.

4.4.2 Seismic Activity

4.4.2.1 Plant Design Background. The original seismic design of N Reactor critical structures, systems, and components preceded, by approximately 10 yr, the development of the design criteria and guides that were applied to commercial nuclear power plants. N Reactor was designed in accordance with the earthquake requirements in the 1955 and 1958 editions of the Uniform Building Code (UBC), except where greater conservatism was deemed appropriate for a nuclear power plant. Seismic design followed the methods and requirements for Seismic Zone II, except for those structures, systems, and components required for safe shutdown, emergency core cooling, and confinement. The latter were either designed or qualified using UBC Zone III horizontal force accelerations of typically 0.2g for most rigid structures and 0.5g for flexible structures, such as emergency coolant piping. The design earthquake for Zone II is usually considered to be 0.16g, which corresponds to approximately intensity VII on the MM scale (MM-VII).

Extensive studies of the Hanford Site seismology were initiated in 1965 and in 1967. Results were used to establish an acceleration of 0.25g as the SSE.[2] A continuous program has since been underway to upgrade and validate critical N Reactor systems to this SSE. Critical systems include those structures, systems, and components necessary to maintain the plant in a safe shutdown condition with emergency cooling. This class of structures, systems, and components is identified as Seismic Category I and a complete listing is presented in the NUSAR.[2]

4.4.2.2 Seismic Hazard Characterization. A seismic exposure analysis was conducted to estimate the probabilities of free-field horizontal ground motions exceeding a spectrum of horizontal ground motions at the N Reactor site. The analysis is described in detail in UNI-4426.[16] The report defines the analytic methodology; inputs to the analysis, including those associated with seismic source definition; maximum earthquake magnitude; earthquake recurrence intervals; and attenuation between the earthquake source and structure.

In general, the probability of exceeding a certain level of ground motion during a specified interval at a site depends on: (1) the location and geometry of earthquake sources relative to the site; (2) the recurrence of earthquakes of various magnitudes up to the maximum magnitude for each source; and (3) the attenuation of ground motions from the sources to the site. For this Limited-Scope PRA, the source models and attenuation relationships developed in 1982 for the WNP-1 and 2 projects (located 14 miles from the N Reactor site) were used.

Because geologic structures are not generally known with certainty, the probability of a given geologic structure being a seismic source is included in this analysis. Assuming certain geologic structures are earthquake sources, tectonic models have been proposed for the Columbia plateau that result in a range of possible fault geometries for each potential seismic source. The fault geometry affects the seismic exposure in two ways: (1) it defines the distance between an earthquake source and the site; and (2) it constrains the maximum magnitude of an earthquake that can be produced by a given fault. In the seismic exposure analysis, ranges of possible fault geometries are probabilistically incorporated.

The results of the seismic exposure analysis are shown in Figure 4-2. The analysis indicates a return period of approximately 12,800 yr for a peak ground acceleration of 0.25g. At this level of ground motion the analysis indicates that the Umtanum Ridge - Gable Mountain source provides 55% of the overall exposure contribution. In 1981, studies by Golder and Associates showed that secondary faults associated with the anticline experienced displacement during the late Pleistocene Era. During the Seismic Exposure Study conducted in 1982 for the WNP-1 and -2 projects, it was therefore concluded that there was a significant possibility the anticline was associated with a reverse fault capable of producing significant earthquakes. Resulting from these interpretations, the Umtanum Ridge-Gable Mountain-Southeast Anticline structure produces most of the seismic exposure for the N Reactor site.

Small magnitude earthquakes have limited engineering significance because of their short duration and low energy content. Therefore, the annual mean exceedance curve shown in Figure 4-2 is based on earthquakes of MM-V and greater. To provide perspective to this magnitude, Table 4-3 shows the relationship between the MM scale, the Richter scale, and the approximate range of accelerations associated with each of these scales.[17] It can be seen that the MM V intensity is equivalent to Richter level 4.6, and to accelerations ranging from 0.015g to 0.030g. Comparison of these seismic levels to the following widely known earthquakes shows the N Reactor exposure curve (Figure 4-2) includes earthquakes that are orders-of-magnitude lower in destructive capacity than those that have caused severe damage.

Event	Date	Richter	Modified Mercalli
San Francisco	1906	8.2	XI
Japan	1933	8.9	XII
Mexico	1957	7.8	X-XI
Alaska	1964	8.5	XI-XII
Mexico	1985	8.0	X-XI
Idaho	1985	6.0	VII-VIII

The MM scale is based on levels at which physical occurrences are observed (e.g., windows shaking, chimneys falling, etc.). The Richter Scale is directly related to energy, with each level equivalent to an energy release.

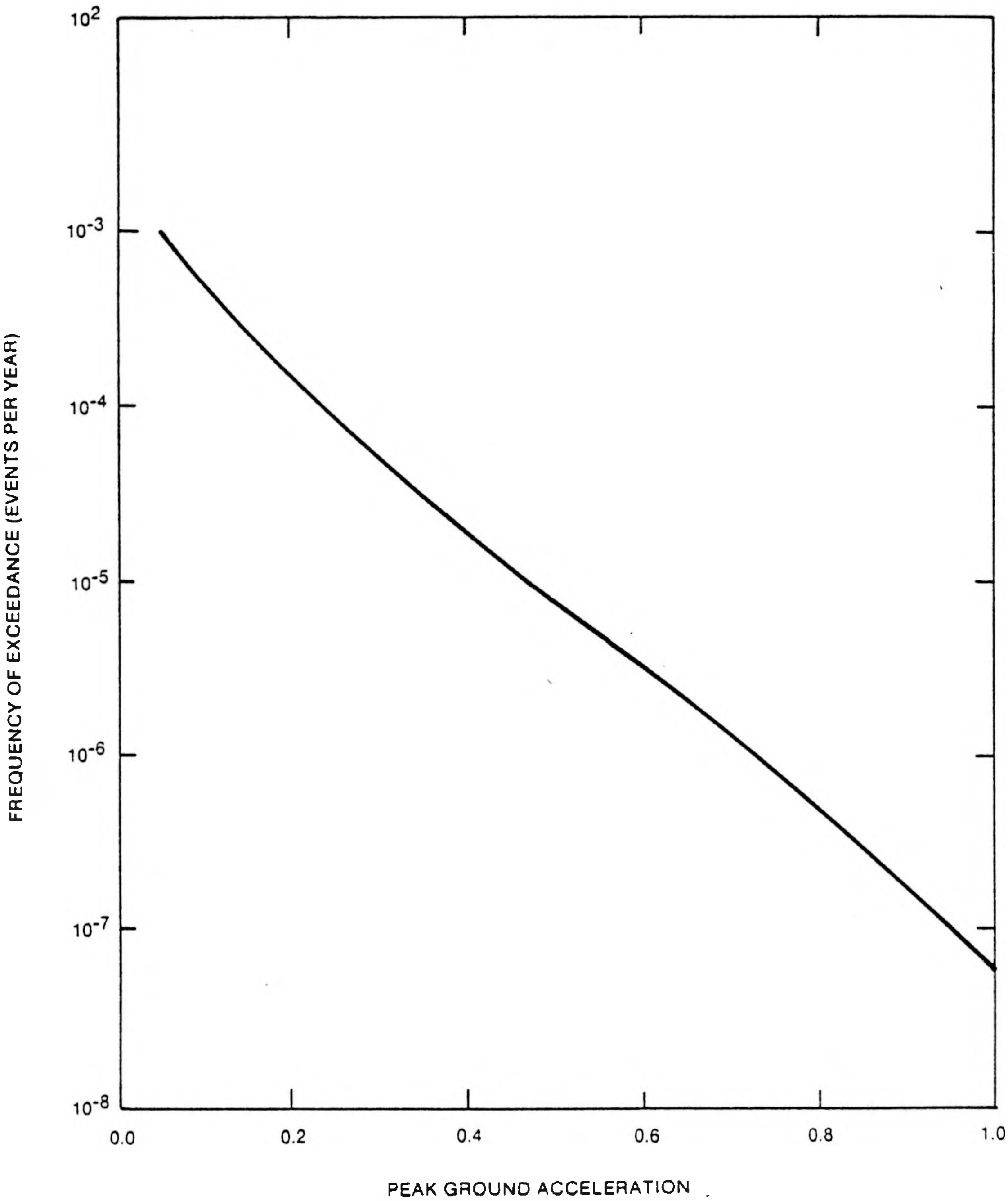


Figure 4-2. Seismic Exposure Curve.

Table 4-3. Seismic Terminology Relationships

Range of Accelerations (g)	Approximate Richter at Midscale	Modified-Mercalli Intensity
0.003 - 0.007	3.4	III
0.007 - 0.015	4.0	IV
0.015 - 0.030	4.6	V
0.030 - 0.090	5.2	VI
0.070 - 0.220	5.8	VII
0.15 - 0.30	6.4	VIII
0.30 - 0.70	7.0	IX
0.45 - 1.5	7.6	X
0.5 - 3.0	8.2	XI
0.5 - 7.0	8.9	XII

Component failure analyses (discussed in Section 5.0) were conducted for 10 acceleration intervals within the range 0.05g to 1.0g. The midpoint acceleration of each interval and the corresponding frequency estimate (based on Figure 4-2) are listed in Table 4-4.

Existing mapping by geologists for the Basalt Waste Isolation Program suggests that Gable Mountain consists of a group of left-stepping echelon domes. This kind of structure is associated normally with strike slip faults, rather than reverse faults. If this characteristic is confirmed, it is likely recalculation of the seismic exposure curve would result in lower frequencies of occurrence at all accelerations used in this study.

4.4.2.3 Plant Seismic Inspection. A plant walkthrough was conducted on March 4, 1987 to identify safety-significant components with potentially high vulnerability to seismic damage. The structures and components identified during this inspection, along with estimated fragility parameters, are listed in Table 4-5. Details of the fragility parameter estimates are provided in Appendix A.

The tabulated fragility parameters are defined as follows:

- A-hat -- The estimated median ground acceleration capacity (g) of the structure or component (i.e., the best estimate of the acceleration where failure would occur)
- Beta-R -- The logarithmic standard deviation describing the random variation in the estimated capacity
- Beta-U -- The logarithmic standard deviation describing the uncertainty in the estimated capacity.

These parameters define a lognormal probability distribution for failure of the particular component as a function of acceleration.

A brief summary of each structure and component evaluated during the walkthrough follows.

- Primary Shield. Primary shield uplift is postulated to result in multiple pressure tube failures with little or no opportunity for the ECCS to provide cooling. In the seismic event tree it is assumed conservatively that acceleration sufficient to fail pressure tubes will also fail GSCS water lines.
- Graphite Stack. Movement of the graphite stack can cause misalignment of the control rod and/or ball insertion channels, resulting in reduced scram capability. The plant design includes initiation of an automatic ball system trip at an acceleration level of 0.007g, but credit for this trip is not taken because it is possible that peak acceleration can be reached within the ball system response time. As discussed in the NUSAR,[2] balls will begin exiting the hoppers 2 s following occurrence of an earthquake, and all balls will be out of the hoppers within 50 s.

Table 4-4. Seismic Acceleration Intervals and Frequencies.

Interval No.	Interval Bounds (g)	Midpoint Acceleration (g)	Interval Frequency/yr)*
1	0.05 - 0.1	0.075	6.6 E-04
2	0.1 - 0.2	0.15	3.3 E-04
3	0.2 - 0.3	0.25	8.6 E-05
4	0.3 - 0.4	0.35	2.9 E-05
5	0.4 - 0.5	0.45	1.1 E-05
6	0.5 - 0.6	0.55	4.3 E-06
7	0.6 - 0.7	0.65	1.7 E-06
8	0.7 - 0.8	0.75	7.1 E-07
9	0.8 - 0.9	0.85	3.0 E-07
10	0.9 - 1.0	0.95	1.3 E-07

* Each interval frequency is determined from Figure 4-2 by subtracting the frequency of exceeding the upper interval bound acceleration from the frequency of exceeding the lower bound acceleration.

Table 4-5. Seismically Sensitive Structures and Components.

Item	Consequence of failure	Estimated fragility parameters		
		A-Hat	Beta-R	Beta-U
1. Primary shield uplift: Process tube failure	Large LOCA	0.9	0.25	0.5
2. Graphite stack: Rod channel misalignment	Partial trip failure	0.6	0.2	0.25
3. Graphite stack: Ball channel misalignment	Partial trip failure	0.4	0.2	0.25
4. Steam generator supports(1)	Large LOCA	0.7	0.2	0.35
5. 182-N: Masonry wall collapse	Loss of ECCS, GSCS, fog spray	0.5	0.2	0.4
6. 182-N: Structural steel collapse(1)	Loss of ECCS, GSCS, fog spray	0.6	0.25	0.3
7. 181-N: Collapse	Loss of ECCS, GSCS, fog spray	(2)		
8. ECCS silo: Pipe failures	Loss of ECCS	0.55	0.2	0.35
9. Pressurizer supports(1)	Large LOCA	(3)		
10. ECCS water tanks(1)	Partial ECCS failure	(3)		
11. Core riser connections(1)	None	(3)		
12. Long-term oil storage tanks(1)	Loss of diesels	(3)		
13. 181-N batteries(1)	Loss of diesels	(3)		

NOTE:
(1) Not explicitly modeled in this study. See Sections 4.4.2.3 and 5.4 for discussion
(2) Fragility parameters were not specifically estimated, but are assumed to be the same as 182-N.
(3) Fragility parameters were not estimated in this study.
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System
LOCA = Loss-of-coolant accident

- Steam Generator Supports. Failure of the steam generator supports could result in excessive loads on connected piping and pipe breaks in both the primary coolant and steam systems. The pressurizer supports were also identified as a potential failure point but were judged to be significantly less sensitive than the steam generator supports and were not investigated further.

- 181-N and 182-N Buildings. These buildings house the high-lift ECCS diesels (182-N), the fog spray diesels (182-N), and the low-lift ECCS diesels (181-N). The 182-N has a structural steel frame with external walls of reinforced concrete below grade and masonry block above-grade. Gross building damage from collapse of masonry walls or steel frame overstressing is assumed to disable the emergency diesels by damage to the engines, fuel lines, control panels, or other components such as air lines. Fragility estimates for 182-N are based on block wall collapse. It is assumed the 181-N has the same seismic fragility as 182-N.

- ECCS Silo. The ECCS silo is a concrete cylindrical structure which passively transfers the source of water for the high lift ECCS pumps from storage tanks to the discharge of the low lift ECCS pumps. Structural failure of this tank will result in loss of ECCS due to loss of water supply to the high-lift pumps.

- ECCS Water Storage Tanks. These tanks are believed to have low seismic resistance, but were not considered further because: (1) tank capacity represents only a short-term ECCS supply; (2) tank failure will not disable the ECCS, provided suction is supplied by the low-lift diesels; and (3) the ECCS fault tree success criteria require the low-lift diesel pumps operate to supply water to the suction of the high-lift diesel pumps.

- Miscellaneous

1. Connections (approximately 4-in. dia.) between the core inlet risers appeared prone to failure and were noted during the walkthrough. These were determined to be either solid bars or blanked sections of pipe (i.e., not flow passages between the risers). Their failure would therefore not result in a LOCA, and they are not considered further.

2. Long-term oil storage tanks were considered to have low resistance but were not considered further because underground diesel tanks are considered to have high resistance and can supply fuel for approximately 7 d. Resupply by tank truck or other means would be highly likely within that period of time.

3. The 181-N batteries were considered vulnerable to seismic motion due to lack of shimming. This is an easily rectified problem, and will be corrected prior to reactor restart.

4. Because the AC power distribution system is not seismically qualified to 0.25g, an earthquake is assumed to result in a total loss of AC power to A and B Buses. Loss of AC power

results in a loss of primary coolant flow and a small LOCA due to loss of the high-pressure injection pumps. Loss of AC power also results in a reactor trip and an ECCS initiation.

The seismic event tree is presented and its quantification discussed in Section 5.4.

4.5 References

1. Zentner, M.D., J. E. Kelly, et al., 1986, Phase I N Reactor Safety and Reliability Evaluation Program Final Report, UNI-3665, UNC Nuclear Industries, Inc., Richland, Washington.

2. UNC, 1987, N Reactor Updated Safety Analysis Report, UNI-M-90, UNC Nuclear Industries, Inc., Richland, Washington.

3. Quapp, W. J., et al.,1987, A Preliminary Assessment of N Reactor Graphite Combustion Risks, WHC-IP-0022, Westinghouse Hanford Company, Richland, Washington.

4. Henri, J. O. (Rockwell) and A. K. Postma (Benton City Technology), 1987 Annual Meeting, Hydrogen Generation During the TMI-2 and the Chernobyl Events, ANS, Dallas, Texas, June 7-11, 1987.

5. Leitz, E. E., 1987, Primary System Piping, (memorandum to M. D. Zentner, UNC Nuclear Industries, Inc., Richland, Washington, February 11, 1987).

6. NRC, 1975, Reactor Safety Study, WASH-1400 (NUREG-75/104), U. S. Nuclear Regulatory Commission.

7. Scott, K.V., et al., 1986, Probabilistic Failure Assessment of the N Reactor Pressure Tubes, UNI-3655, UNC Nuclear Industries, Inc., Richland, Washington.

8. Stack, D.W., et al., 1987, Preliminary Results of a Probabilistic Risk Assessment for the N Reactor, UNI-I-111, UNC Nuclear Industries, Inc., Richland, Washington.

9. Leitz, E. E., 1987, N Reactor Operating Limits,(memorandum to M. D. Zentner, UNC Nuclear Industries, Inc., Richland, Washington, February 2, 1987).

10. Leitz, E. E., 1987, N Reactor Scrams Which Prevented Plant Damage, (memorandum to file, UNC Nuclear Industries, Inc., Richland, Washington, July 1987).

11. Leitz, E. E., 1987, Analysis of N Reactor Unscheduled Shutdowns, UNI-3052, UNC Nuclear Industries, Inc., Richland, Washington.

12. Leitz, E. E., 1987, Secondary System Piping, (memorandum to M. D. Zentner, UNC Nuclear Industries, Inc., Richland, Washington, February 11, 1987).

13. Leitz, E.E., 1984, Operational Analysis of the N Reactor Electrical Power Supply System, UNI-3442, UNC Nuclear Industries, Inc., Richland, Washington.

14. Leitz, E.E., 1986, Operational Analysis of the Circulating Raw Water System, UNI-3608, UNC Nuclear Industries, Inc., Richland, Washington.

15. SNL, 1983, Interim Reliability Evaluation Program Procedures Guide, Sandia National Laboratory, NUREG/CR-2728.

16. United Engineers and Contractors, 1987, Evaluation of Seismic Exposure for the N Reactor Plant, Hanford, Washington.

17. Lewis, E. E., 1977, Nuclear Power Reactor Safety, John Wiley and Sons, New York, New York.

5.0 PLANT ACCIDENT SEQUENCES

5.1 Summary of Dominant Accident Sequences

Dominant accident sequences that may result in fuel damage are summarized in Table 5-1. These sequences are based on the internal and external initiator event trees described below. The tabulated sequences are those with an estimated frequency of 1 E-07/yr and greater. Frequencies for all evaluated sequences are shown in the individual event trees.

5.2 Event Tree End-states

Accident sequence end-state descriptions (X, Y, Z, etc.) are shown in Table 5-2 and are based on the amount of fuel that may be damaged during the accident. No intent is made to assess the corresponding radionuclide releases. These estimates are provided in Section 8.0. Where accident timing is important, the time estimate is also given in Section 8.0.

5.3 Internal Initiator Event Trees

Based on the internal accident initiators identified in Section 4.3, event trees were developed that identify potential accident sequences and end-states for each initiator. The end-states are defined in Table 5-2. The event trees and the descriptions of the top events are included in the following sections. Although the structures for several of the event trees are similar, the quantification of the branches may be different, depending on the initiator. Thus an event tree is included for all identified internal initiators.

Event probabilities are estimated, based on quantification of the preliminary detailed system fault tree models currently being developed for the full-scope Level 1 PRA. The detailed system fault tree models, including the failure data for the basic events and the quantified cutsets, are provided in WHC-SP-0021.[2] Support system failure probabilities were also estimated from the fault tree models and are included.[2] Failure data include plant specific component failure rates currently being developed and various generic PRA databases.[3,4, and 5] Specific references for each basic event failure rate are included in WHC-SP-0021.

5.3.1 <u>Large LOCA Event Tree</u>

These LOCAs are defined to occur in the Primary Coolant System (either the inlet or the outlet side) and result in leakage that cannot be compensated for by the High Pressure Injection System. The event tree for large LOCAs is shown in Figure 5-1.

Table 5-1. Dominant Accident Sequence Frequencies.

Fuel Damage End-state	Initiator	System Failures	Sequence Freqency per year
Z: 100% fuel damage	Seismic	The 181-N/182-N buildings fail to remain intact, resulting in total ECCS and GSCS failure	3.7 E-05
	Seismic	Primary shield uplift (pressure-tube failures)	2.5 E-06
	Rapid shutdown	Reactor trip fails	2.0 E-06
	Seismic	Reactor trip fails	9.6 E-07
	Overpower	Reactor trip fails	1.1 E-07
Y: ≤30% fuel damage	Rapid shutdown	HPI fails to function after trip; ECCS fails to start and function	8.0 E-06
	Seismic	ECCS fails to start and function	7.6 E-06
	Rapid shutdown	Primary/secondary fails; ECCS fails to start and function	7.0 E-06
	Large LOCA	ECCS fails to start and function	2.1 E-06
	Overpower	HPI fails to function after trip; ECCS fails to start and function	2.0 E-07
	Overpower	Primary/secondary fails; ECCS fails to start and function	1.7 E-07
W: ≤2% fuel damage	Rapid shutdown	HPI fails to function after trip; Single-riser ECCS failure	5.1 E-06
	Rapid shutdown	Primary/secondary fails; Single-riser ECCS failure	4.5 E-06
	Seismic	Single-riser ECCS failure	5.0 E-07
	Large LOCA	Single-riser ECCS failure	4.7 E-07
	Overpower	HPI fails to function after trip; Single-riser ECCS failure	1.3 E-07
	Overpower	Primary/secondary fails; Single-riser ECCS failure	1.1 E-07
Single tube: ≤0.1% fuel damage	Local loss of flow	None (*)	2.5 E-02

*There are single-tube failure sequences that also involve mitigating system failures. Some of these sequences have frequencies greater than 1 E-07/yr, but in all cases, frequencies are much less than 2.5 E-02/yr.

ECCS = Emergency Core Cooling System
GSCS = Graphic and Shield Cooling System
HPI = High Pressure Injection

Table 5-2. Plant Accident End-States.

End State	Percent Fuel Damage	Description
Z	100	Coincident ECCS and GSCS failure or failure to scram (regardless of ECCS and GSCS success). More analysis may be needed in some cases to determine the precise end-state.
Y	≤30	All direct cooling to the fuel via ECCS is unavailable. Fuel damage is limited to 30% due to the heat removal capacity of GSCS, which continues to operate. This end-state is the NUSAR hypothetical accident.[1]
X	≤6	Fuel damage in 64 (out of a possible 1,003) process tubes due to failure to deliver ECCS flow to any one of the 16 core inlet risers. All other ECCS functions are successful; however, GSCS fails to provide a heat removal capability from damaged fuel. Cooling from adjacent tubes supplied to nonaffected risers is conservatively neglected.
W	≤2	Same as X, but GSCS succeeds and limits fuel damage within the affected riser to 30% of that which could occur in-end state X.
Single tube	≤0.1	Fuel damage in a single process tube.
Uncertain	--	Some damage may result, but further analysis will be required to determine the extent.
OK	None	No fuel damage results.

ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling Systems
NUSAR = N Reactor Updated Safety Analysis Report
[1]Reference 1

PST87-1159-5

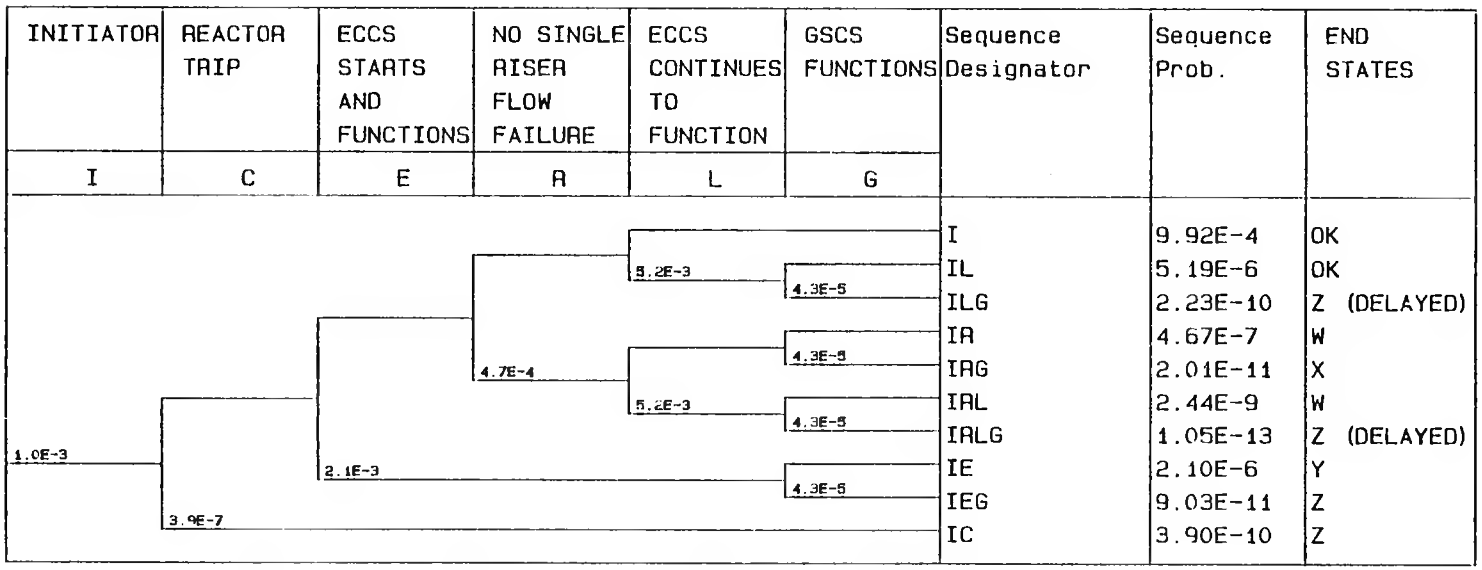


Figure 5-1. Large LOCA Event Tree.

1. Reactor Trip: Event C. Success for this state is a complete and timely reactor scram on an appropriate scram signal. A failure is assumed to result in end-state Z, with response of ECCS and GSCS having no effect. Branch probability is 3.9 E-07.

2. ECCS Starts and Functions: Event E. Success for this state is the ECCS initiates on-demand, at least two V-3s and two V-4s on each side of the reactor open, two high-lift diesels and two low--lift diesels start and run, and all required functions operate for 2 h. The assumption is if ECCS functions until the decay heat load has decreased to the point the GSCS can adequately cool the core and then fails, fuel damage can be avoided. For this study, it is assumed that 2 h is an adequate time for this reduction in heat removal requirements to occur. Subsequent analysis may shorten this time-frame; if so, it will be reflected in the full-scope Level 1 PRA failure models. If ECCS fails before the end of the 2 h period, fuel damage is assumed to occur. Branch probability is 2.1 E-03.

3. No Single Riser Flow Failure: Event R. Success for this state requires flow to the core through all ECCS flow paths. Failure implies the flow through one of 16 core inlet risers has failed, probably because a closed V-1 valve, a CV-2R check valve failure to open, or a CV-5 valve failure to remain closed. This will affect 1/16th of the core (64 tubes), and may result in some fuel damage. The failure probability for this event is based on an ongoing analysis of plant-specific failure data for all ECCS check valves that is being conducted as part of the full-scope Level 1 PRA. Preliminary results indicate a failure probability for loss of emergency cooling to any one of the 16 risers is 4.7 E-04. This is the branch probability value.

4. ECCS Continues to Function: Event L. Success for this state is the ECCS continues to function beyond 2 h after initiation. This event assumes a single high-lift diesel provides sufficient cooling. Branch probability is 5.2 E-03.

5. GSCS Functions: Event G.Success for this state is the GSCS continues to function in the recirculation mode (two of the three pumps, three of the four heat exchangers) or operates successfully in the once-through mode (flow from the RWS-2 system from either the high-pressure raw water pumps or the fog-spray diesel pumps). If ECCS fails after 2 h, but GSCS continues to operate, no fuel damage will occur. If the ECCS fails before the decay heat removal requirement drops to the capacity of the GSCS, some fuel damage will occur. A GSCS failure given early ECCS actuation, but subsequent failure, will likely result in delayed but complete fuel failure. Branch probability is 4.3 E-05.

5.3.2 Small LOCA Event Tree

This tree describes accident sequences initiated by a small-break LOCA. The definition of a small-break LOCA is one that can be mitigated by the High Pressure Injection system. Mitigation includes being able to provide adequate makeup water in the event of a reactor scram and consequent coolant shrinkage (assuming loss of the DAs which are described in Section 3.8.1). The small LOCA event tree is shown in Figure 5-2.

1. HPI Functions: Event H. Success for this state is that HPI provides sufficient coolant makeup to mitigate the leakage as described above. The two operating HPI pumps must continue to run and a third pump must start for success. A failure of HPI leads to a reactor trip and an ECCS initiation signal. Branch probability is 1.0 E-03.

2. Reactor Trip: Event C. Success for this state is a complete and timely reactor scram on an appropriate scram signal. A failure of the trip with the HPI operational may allow continued operation without fuel damage, as long as makeup and recirculation are adequate, although a controlled shutdown should result. Operator actions may be required to bring the plant to a safe shutdown state. Failure of a scram without HPI is defined to result in end-state Z, with response of ECCS and GSCS having no effect. Branch probability is 3.9 E-07.

3. HPI Functions After Trip: Event D. This event is included to provide for the possibility that a reactor scram may provide an upset to the HPI system because it must change its operational mode to accommodate loop shrinkage. Successful operation of HPI after a scram is continued operation of three pumps. A success results in normal shutdown. A failure will result in an ECCS demand. The probability of this event is approximately the same as Event H because the HPI system failures are dominated by loss of HPI pump cooling and speed control failures, which are not dependent on the number of pumps required to be in operation. Branch probability is 1.0 E-03.

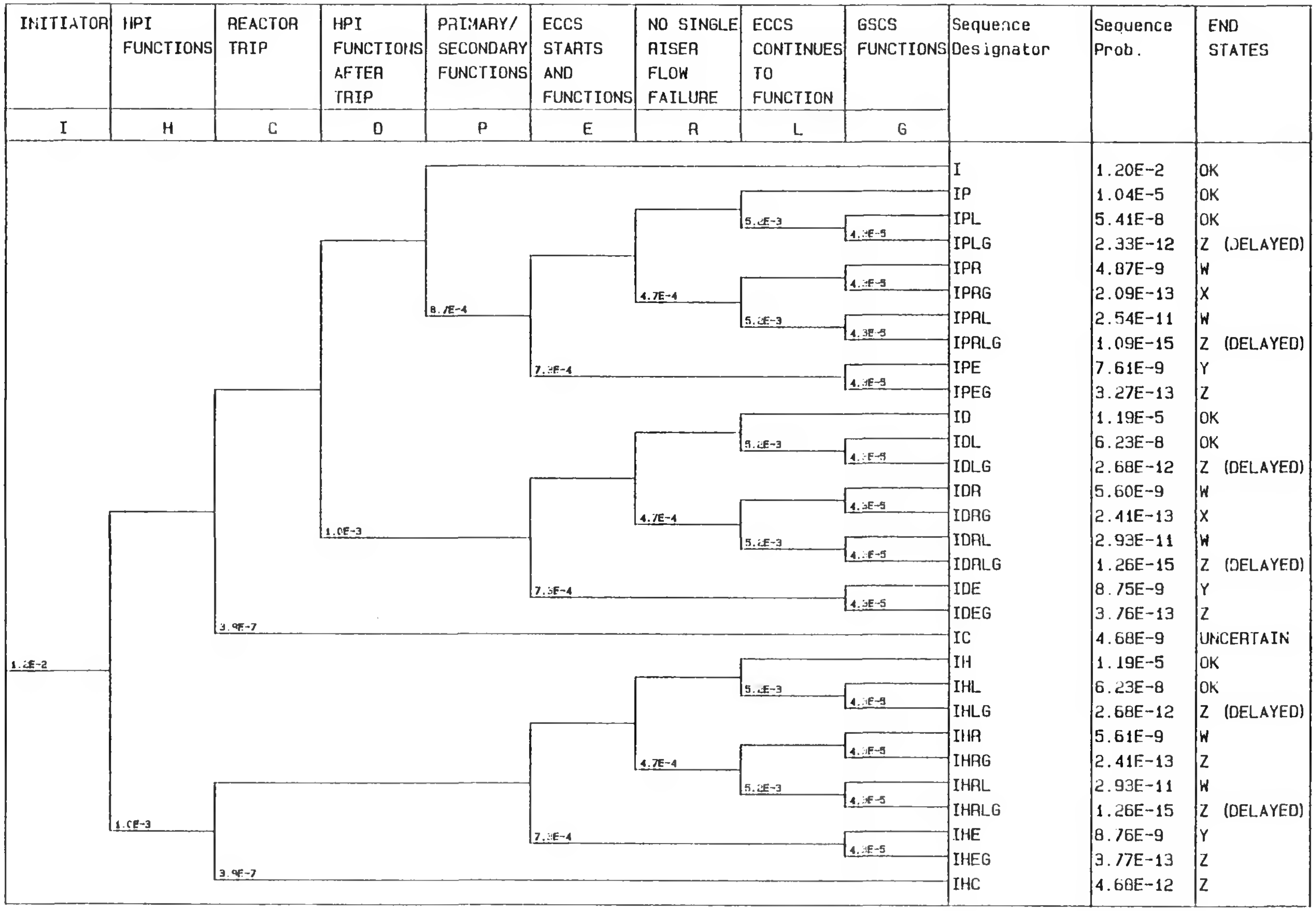
4. Primary/Secondary Functions: Event P. Success for this state requires the primary system remains able to remove decay heat. The success criteria for operation of the primary system, such that ECCS does not actuate, specifies that three of five steam-generator cells (two on one side and one on the other side) and three of five reactor coolant pumps (RCPs) must operate. Branch probability is 8.7 E-04.

5. ECCS Starts and Functions: Event E. See Section 5.3.1; however, only one high-lift diesel is required to start and run. Branch probability is 7.3 E-04.

6. No Single Riser Flow Failure: Event R. See Section 5.3.1.

7. ECCS Continues to Function: Event L. See Section 5.3.1.

8. GSCS Functions: Event G. See Section 5.3.1.



HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-2. Small LOCA Event Tree.

5.3.3 Local Loss of Flow Event Tree

This tree describes the accident sequences initiated by a local loss of flow, essentially blockage of a single pressure tube. Since the flow blockage will result in a loss of cooling to that single tube even if all cooling functions operate as intended, all end-states will include single-tube damage. The extent of the damage reported in the NUSAR is being verified using state-of-the-art computer codes. The local loss of flow event tree is shown in Figure 5-3.

1. Reactor Trip: Event C. Success for this state is a complete and timely reactor scram on an appropriate scram signal. If the reactor fails to scram, it is assumed the pressure tube will fail, resulting in a small LOCA. However, the probability of this event is small, relative to the other events and thus, these branches are not shown on the event tree. Branch probability is 3.9 E-07.

2. HPI Functions After Trip: Event H. Success for this state is HPI functions normally (two operating HPI pumps continue to run or a failed pump is replaced by the standby pump) after a reactor scram. This function is required to ensure adequate makeup water is available to account for loop shrinkage after a scram. Failure will result in an ECCS demand. Branch probability is 1.0 E-03.

3. Primary/Secondary Functions: Event P. See Section 5.3.2.

4. ECCS Starts and Functions: Event E. See Section 5.3.3.

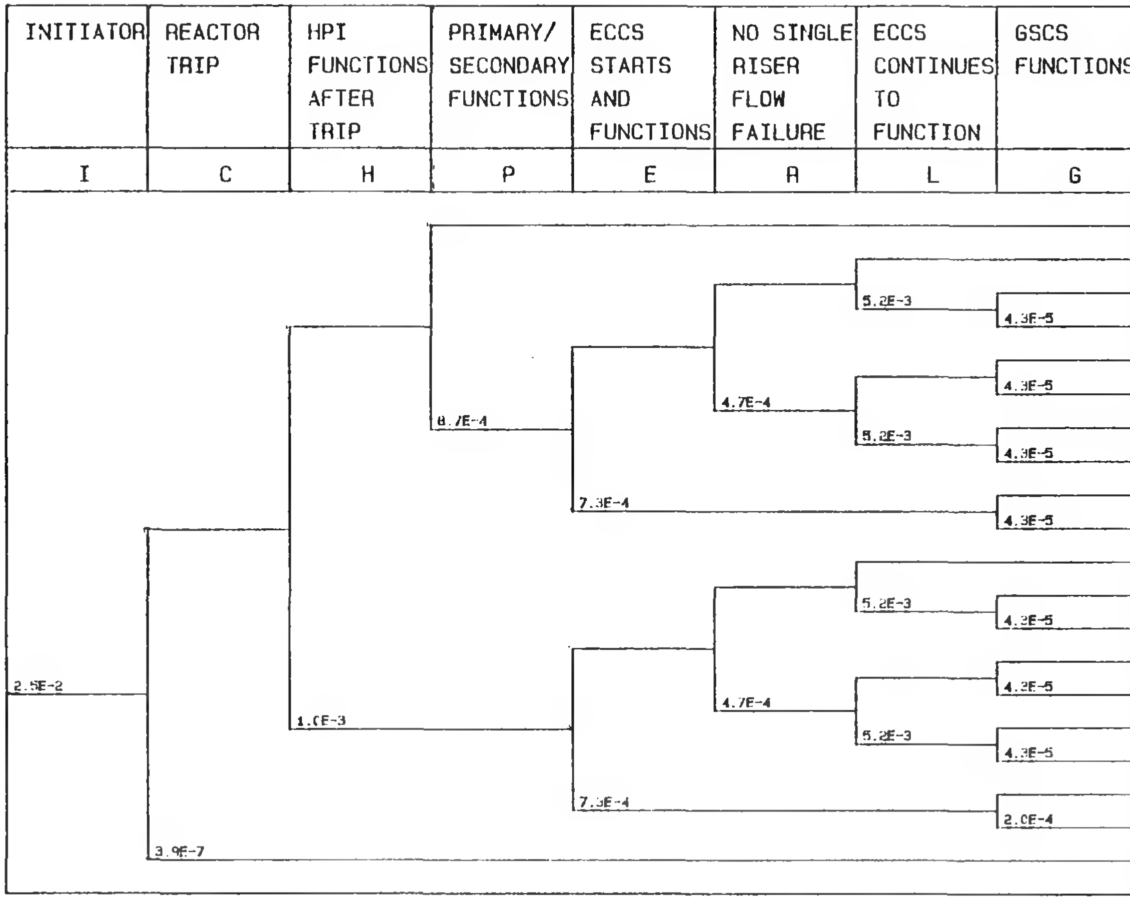
5. No Single Riser Flow Failure: Event R. See Section 5.3.1.

6. ECCS Continues to Function: Event L. See Section 5.3.1.

7. GSCS Functions: Event G. See Section 5.3.1.

5.3.4 Rapid Shutdown from Power Transient Event Tree

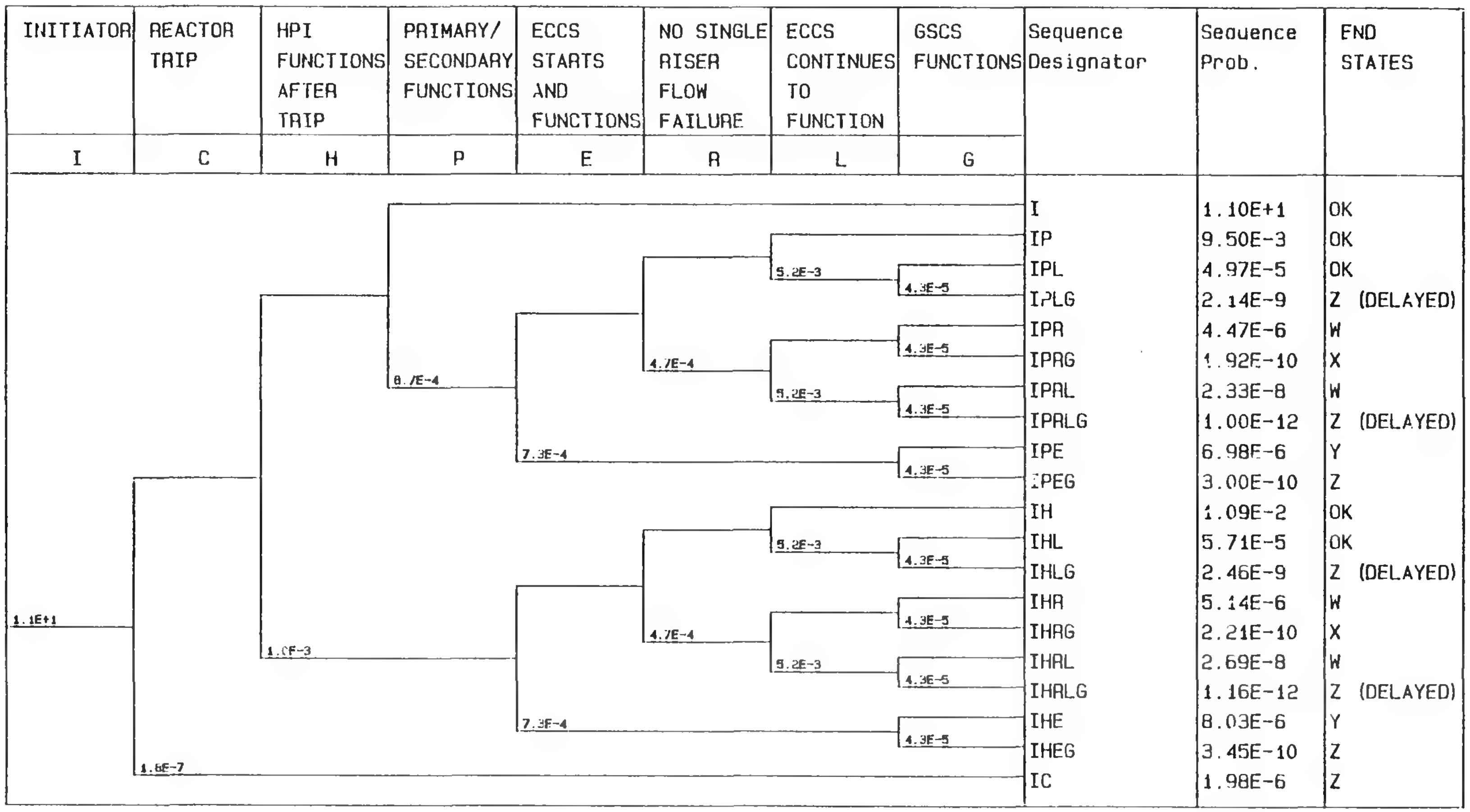
The initiating event in this tree is a reactor trip with the reactor at power. This is the normal rapid shutdown path the reactor experiences an average frequency of 11 times/yr. Of these, an average of 5/yr result from transients potentially requiring the trip system to function for core protection. The key failure of significance for this set of sequences is failure of normal cooling methods to function; otherwise, an ECCS initiation signal will occur, resulting in essentially a large-break LOCA (after the V-4s open). If the V-4s do not open, the potential exists for a high-pressure boiloff, with possible unfiltered release. This is discussed further in Section 5.5. The rapid shutdown event tree is shown in Figure 5-4. Reactor trips for cause, such as in response to a LOCA, are considered in other event trees.



Sequence Designator	Sequence Prob.	END STATES
I	2.50E-2	1 TUBE
IP	2.16E-5	1 TUBE
IPL	1.13E-7	1 TUBE
IPLG	4.85E-12	Z (DELAYED)
IPR	1.02E-8	W + 1 TUBE
IPRG	4.37E-13	X + 1 TUBE
IPRL	5.31E-11	W + 1 TUBE
IPRLG	2.28E-15	Z (DELAYED)
IPE	1.59E-8	Y + 1 TUBE
IPEG	6.82E-13	Z
IH	2.48E-5	1 TUBE
IHL	1.30E-7	1 TUBE
IHLG	5.58E-12	Z (DELAYED)
IHR	1.17E-8	W + 1 TUBE
IHRG	5.02E-13	X + 1 TUBE
IHRL	6.11E-11	W + 1 TUBE
IHRLG	2.63E-15	Z (DELAYED)
IHE	1.82E-8	Y + 1 TUBE
IHEG	3.65E-12	Z
IC	9.75E-9	UNCERTAIN

HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-3. Local Loss of Flow Event Tree.



HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-4. Rapid Shutdown from Power Transient Event Tree.

1. Reactor Trip: Event C. Success for this state is a complete and timely reactor scram given an appropriate scram signal. Of the 11 trips/yr, five have been transient scram initiators resulting from transients potentially requiring scram for core protection. Thus, the reactor trip branch failure probability has been adjusted to reflect the fraction of trips caused by transient scram initiators. Branch probability is 1.8 E-07.

2. HPI Functions After Trip: Event H. Success for this state is that HPI functions normally (two operating HPI pumps continue to run or the failed pump is replaced by the standby pump) after a reactor scram. This function is required to ensure adequate makeup water is available to account for loop shrinkage after a scram. Failure will result in an ECCS demand. Branch probability is 1.0 E-03.

3. Primary/Secondary Functions: Event P. See Section 5.3.2.

4. ECCS Starts and Functions: Event E. See Section 5.3.3.

5. No Single Riser Flow Failure: Event R. See Section 5.3.1.

6. ECCS Continues to Function: Event L. See Section 5.3.1.

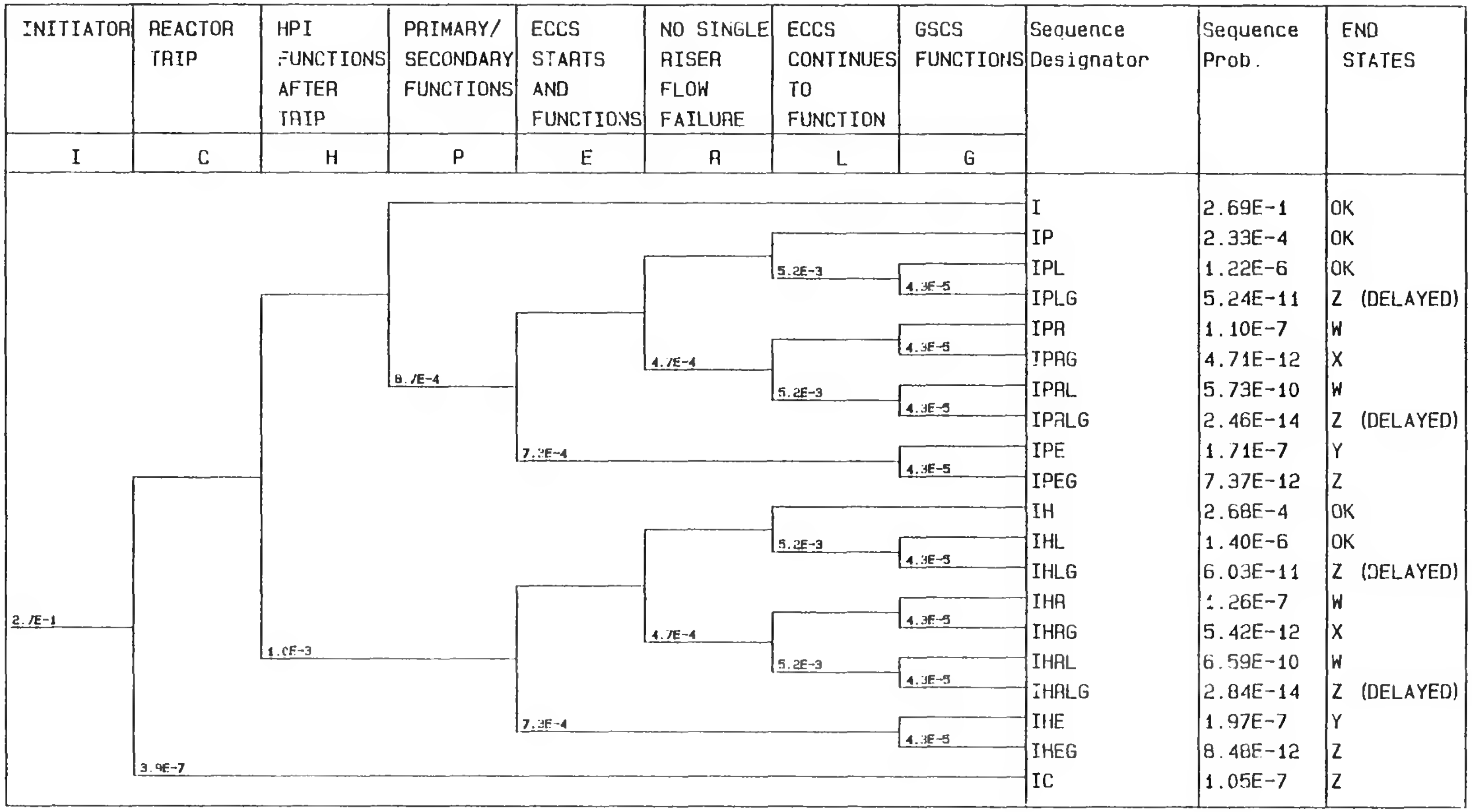
7. GSCS Functions: Event G. See Section 5.3.1.

5.3.5 Overpower Event Tree

This tree describes the accident sequences caused by a reactivity insertion, resulting in an overpower transient. The key event is that the reactor must trip, otherwise severe fuel damage will result. If the reactor trip occurs, the shutdown path is the standard one that occurs on every trip; HPI and the Primary/Secondary systems must function to remove decay heat, otherwise an ECCS demand will occur. If the ECCS demand does occur, the V-4s must open to allow the reactor to depressurize and prevent a high-pressure boiloff. Further analysis may identify a common-cause failure (as yet undefined) that could result in failure of all V-4s to open; if so, this scenario will be analyzed in the full-scope Level 1 PRA. The overpower transient event tree is shown in Figure 5-5. The events are similar to those described for the rapid shutdown event tree in Section 5.3.4 with a branch probability of the failure to trip branch (Event C) of 3.9 E-07.

5.3.6 Large Steam Line Break Event Tree

This tree describes the accident sequences caused by a large steam-line break in the secondary system. Because secondary steam is important to the plant, both as a source of electricity (B Bus) and as a source of motive power (primary coolant pumps), a large steam-line break will result in significantly different responses than other reactivity insertion events. It is assumed conservatively the boiler systems cannot respond to steam demand after a large steam-line break (they must pick up steam load from hot standby). If an ECCS demand occurs, the V-4s must open to allow the reactor to depressurize and prevent a high-pressure boiloff. As noted in Section



HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-5. Overpower Event Tree.

5.3.5, high-pressure boiloff is the subject of ongoing analysis in the full-scope Level 1 PRA. The large steam-line break event tree is shown in Figure 5-6.

1. Reactor Trip: Event C. Success for this state is a complete and timely reactor scram on an appropriate scram signal. A failure is defined to result in end-state Z, with response of ECCS and GSCS having no effect. Branch probability is 3.9 E-07.

2. HPI Functions After Trip: Event H. Success for this state is the HPI functions normally (two pumps continue to operate) after a reactor scram. This function is required to ensure that adequate makeup water is available to accommodate for loop shrinkage after a scram. One significant difference in this sequence is the secondary side will initially be removing more heat than normal (due to the energy lost through the steam-line break) so there will be increased primary loop shrinkage and thus, a greater demand on the HPI system. Another difference is that the steam-line break will likely cause a loss of B Bus, resulting in a loss of the B Bus powered HPI pumps. The coincident failure of the operating A Bus pump and failure of the A Bus standby pump to start will result in an ECCS demand. Branch probability is 1.1 E-02.

3. Primary/Secondary Functions: Event P. Success for this state is the primary system remains able to remove decay heat. A key result of the steam-line break is the possible loss of steam supply to the primary coolant pumps, leading to a need for at least three of the five A Bus powered pony motors. Failure of this function will result in an ECCS demand. Branch probability is 2.1 E-03.

4. ECCS Starts and Functions: Event E. See Section 5.3.3.

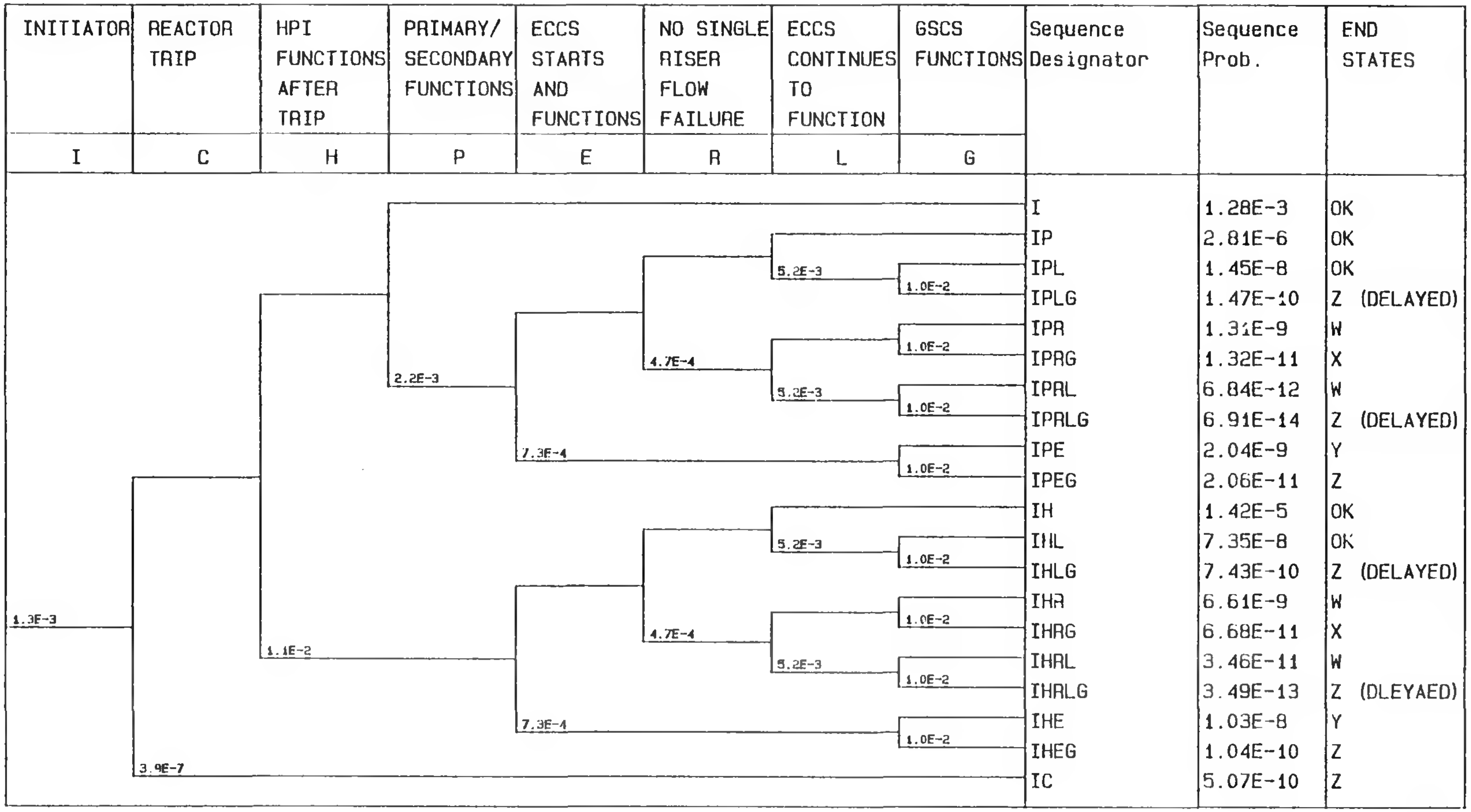
5. No Single Riser Flow Failure: Event R. See Section 5.3.1.

6. ECCS Continues to Function: Event L. See Section 5.3.1.

7. GSCS Functions: Event G. See Section 5.3.1; however, loss of B Bus is assumed to require GSCS operation in the once-through mode with flow from the fog-spray diesel pumps. Branch probability is 1.0 E-02.

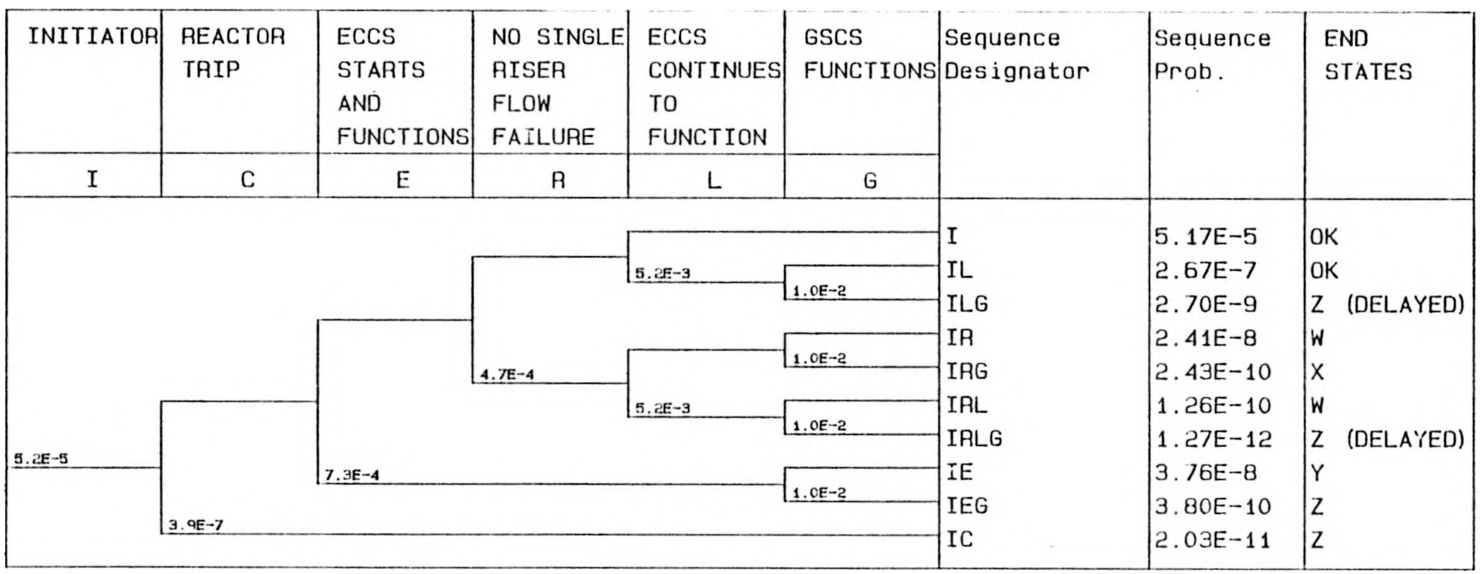
5.3.7 Station Blackout Event Tree

This tree describes the sequences resulting from a concurrent loss of AC power to both A and B Bus. This initiator has special significance because many safety-related systems are dependent on AC power. This tree (Figure 5-7) also applies to a global loss of flow and the loss of the circulating raw water system (CRW)(Section 4.3.3.8). Although the tree appears similar to the normal transient tree, there are some key differences, primarily the change in failure probability for the GSCS function because loss of AC power requires system operation in the once-through cooling mode. Because the top events are essentially the same as for the normal transient event tree, they will not be described further.



HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-6. Large Steam Line Break Event Tree.



HPI = High Pressure Injection System
ECCS = Emergency Core Cooling System
GSCS = Graphite and Shield Cooling System

Figure 5-7. Station Blackout Event Tree.

5.4 External Initiator Event Tree

An event tree was developed, based on the seismic event characterized in Section 4.4. The seismic event tree is described below. Fuel damage end-states are those defined in Section 5.2.

Seismic events are characterized in Section 4.4, with respect to peak ground acceleration, magnitude (energy content), and frequency of occurrence. The acceleration range 0.05g to 1.0g is considered in developing and quantifying the composite seismic event tree shown in Figure 5-8. End-state frequencies in the composite tree are the totals from each of the ten acceleration interval event trees. The composite tree does not show branch probabilities because branch probabilities could only be shown for each acceleration interval.

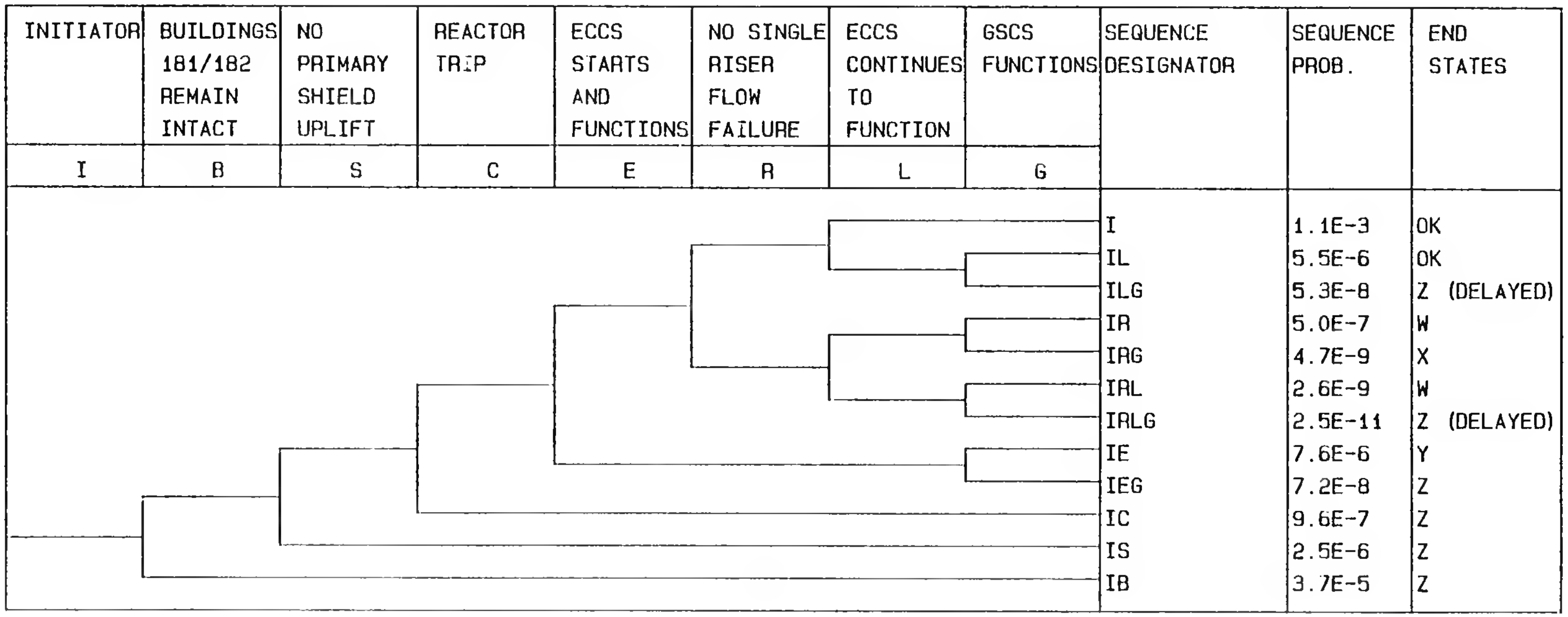
A seismic event is conservatively assumed to cause a complete loss of AC power. This, in turn, results in a total loss of reactor coolant flow and a loss of reactor coolant makeup from the HPI system. The ECCS actuation is required because of the loss of flow and (possibly) because of the net loss of coolant caused by loss of the HPI pumps. As noted in Section 4.4.2.3, a large LOCA could result from seismically induced failure of the steam generator supports. However, a large LOCA is not included as a separate event in the seismic event tree because it requires the same response and results in the same fuel damage states as the total loss of reactor coolant flow (which is included as part of the initiating event).

The seismic event tree structure is similar to the loss of AC power event tree, but with branches added for pressure-tube rupture and structural failure of 181-N or 182-N. The latter results in loss of ECCS, GSCS, and confinement fog spray. Building failures were included in the event tree, rather than in the respective system fault trees, as a convenient way to depict the dependency of the three systems on a single event.

Branch descriptions for the seismic event tree are as follows.

1. Primary Shield Uplift: Event S. Failure at this branch is defined as multiple pressure-tube and graphite cooling system line ruptures at the reactor faces due to primary shield uplift. Such ruptures would defeat the ability of both the ECCS and GSCS to provide cooling to the affected fuel assemblies. Failure at this branch, therefore, leads directly to end-state Z. The probability of failure is calculated from estimated fragility parameters and the seismic hazard curve.

2. Reactor Trip: Event C. This branch is the same as Branch C in other trees, except for its quantification. In the seismic tree, failure to trip includes misalignment of both the control rod and the ball insertion channels, in addition to seismically independent failures from the trip system fault tree. Channel misalignment probabilities are determined from the seismic hazard curve and estimated fragility parameters. Credit is not taken for the ball system trip at an acceleration of 0.007g, because the time to peak acceleration could be less than the ball system response time (see Section 4.4).



*Sequence frequencies are the totals for the ten acceleration intervals. Initiator frequencies and branch failure probabilities are defined for each interval tree but not for the composite tree.

Figure 5-8. Seismic Initiator Event Tree.

3. Buildings 181/182 Remain Intact: Event B. This branch accounts for dependent failure of ECCS, GSCS, and confinement fog spray that could result from failure of either of these buildings. The causative mechanism is structural collapse, which disables the diesels. Building collapse probabilities are estimated from the seismic hazard curve and estimated fragility parameters. Sufficient design information was not available during this study to estimate fragility parameters for 181-N. It is assumed therefore, the parameters estimated for 182-N also apply to 181-N.

4. ECCS Starts and Functions: Event E. This branch is the same as Branch E in other trees, except for its quantification. In the seismic tree, the branch probability includes structural failure of the ECCS silo (see Section 4.4.2.3) in addition to the events modeled in the ECCS fault tree. Silo failure probability is determined from the seismic hazard curve and estimated fragility parameters.

5. No Single Riser Flow Failure: Event R. This branch and its quantification are the same as Branch R in other trees.

6. ECCS Continues to Function: Event L. This branch is the same as Branch L in other trees. Some of the bulk fuel storage tanks are considered vulnerable to seismic damage; however, seismic-resistant tankage provides approximately a 7-day supply of fuel for each ECCS and fog-spray diesel. Tank truck resupply is judged to be sufficiently reliable within a 7-day period so as not to significantly reduce long-term ECCS success probability.

7. GSCS Functions: Event G. This branch is the same as Branch G in other trees; however, due to the loss of AC power, operation is possible only in the once-through mode with flow from the fog-spray diesel pumps.

To calculate risk over the complete spectrum of seismic events, the ground acceleration range from 0.05g to 1.0g is divided into 10 intervals. The seismic event tree is quantified separately for each interval, and the results for each interval are combined to give the sequence probabilities shown in Figure 5-8. The choice of 10 intervals balances the greater accuracy, which results from more intervals against the desire to reduce calculational burden. Calculated sequence frequencies decrease as the number of intervals is increased.

The first step in quantification is to estimate the probability of individual component failure at the midpoint acceleration of each interval. To do this, the acceleration capacity of each structure or component (the acceleration at which failure occurs) is assumed to be lognormally distributed with median capacity equal to A-hat and logarithmic standard deviation equal to the square root of Beta U squared plus Beta R squared. These parameters are listed for each component in Table 4-5 and are briefly described in Section 4.4.2.3. The lognormal distribution of acceleration at failure results in a fragility curve for each component similar to the example curve shown in Figure 5-9. The fragility curve is a plot of failure probability versus acceleration. The effect of varying the logarithmic standard deviation (Beta R and/or Beta U) is shown in Figure 5-9. The

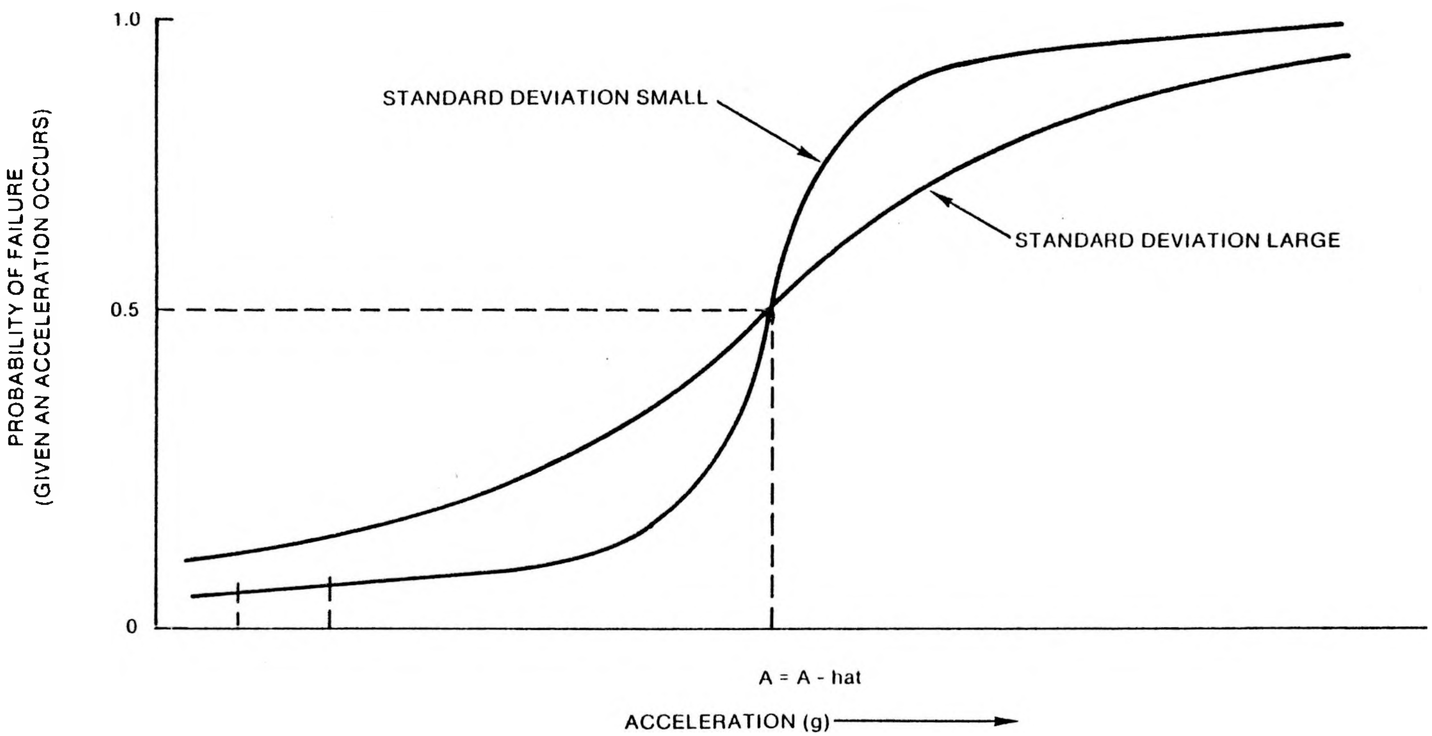


Figure 5-9. Typical Component Fragility Curve.

fragility curve gives the probability of failure, given that a specified acceleration occurs. The probability of a failure does not include frequency of occurrence of the acceleration itself.

For each of the 10 acceleration intervals, the probability of component or structure failure is estimated from the fragility curve as a function of the midpoint acceleration. (Failure probabilities are actually calculated from a close numerical approximation to the fragility curve.) Individual component failure probabilities from seismic excitation are combined with nonseismic failure probabilities from system fault trees to arrive at the failure probability for each branch in the seismic event tree. The logic combinations for each branch are as follows:

- Buildings 181-N/182-N -- Failure of 181-N from seismic excitation or failure of 182-N from seismic excitation
- Primary Shield Uplift -- Uplift from seismic excitation
- Reactor Trip -- (Displacement of rod channels from seismic excitation and displacement of ball channels from seismic excitation) or (nonseismic reactor trip system failure from RTS fault tree)
- ECCS Starts and Functions -- Failure of ECCS silo from seismic excitation or nonseismic failure of ECCS from ECCS fault tree
- Other Branches -- From system fault trees with no seismically induced failures.

The seismic event tree is quantified for each interval by applying the initiator frequency and calculating individual sequence frequencies. The initiator is an earthquake with peak ground acceleration falling in the interval being quantified. The frequency of an acceleration within an interval is equal to the frequency of accelerations equal to, or greater than, the lower interval bound minus the frequency of accelerations equal to, or greater than, the upper interval bound.

This procedure, consisting of the following steps, is repeated for each of the remaining intervals:

- Calculate individual component/structure failure probabilities from seismic excitation
- Combine seismic and nonseismic failure probability, as appropriate, to determine each event tree branch probability
- Quantify the event tree by applying the initiator (interval) frequency and calculating sequence frequencies.

The total frequency of each sequence is obtained by summing the frequency for the sequence from each of the 10 intervals. For example, the total frequency of sequence IB is the sum of the sequence IB frequencies for all of the 10 intervals.

5.5 High Pressure Boiloff

High-pressure boiloff is postulated as a possible mechanism for unfiltered release of fission products from confinement. In such a scenario, ECCS is required because of a nonLOCA event, such as total loss of reactor coolant flow or loss of heat sink. Failure of all ECCS dump valves (V-4s) prevents reactor coolant system depressurization, emergency coolant cannot reach the core, and heatup of both fuel and coolant results. As heatup progresses, coolant pressure increases and the RV-2 relief valves will begin to cycle open and closed. Eventually some process tubes will lose cooling, resulting in fuel melting and pressure-tube failure. At the point of pressure-tube failure, steam release will increase confinement pressure. If confinement pressure reaches the level at which the steam vent weatherheads will lift, an effective bypass of confinement will occur. Because fuel failure will have already occurred, the initial release of steam from confinement could have a significant fission product content. In addition, potential exists for increased hydrogen production because more water might be available to react with overheated fuel and cladding.

The probability of independent failure of all V-4 valves to open on an ECCS demand has been determined to be less than 1.0 E-10, and independent failure is therefore not a significant contributor to overall risk. Analyses to-date have not identified a common-cause failure mode; however, ongoing analysis as part of the full-scope Level 1 PRA may identify a credible common-cause resulting in failure of all V-4s to open. In that event, and if the common-mode cause cannot be eliminated, the appropriate high-pressure boiloff sequences will be analyzed in detail.

5.6 References

1. Westinghouse, 1987, N Reactor Limited-Scope Probabilistic Risk Assessment, System Fault Tree Model Quantification, WHC-SP-0021, Westinghouse Hanford Company, Richland, Washington.

2. Stack, D. W., et al., 1987, Preliminary Results of a Probabilistic Risk Assessment for the N Reactor, UNI-I-111, UNC Nuclear Industries, Inc., Richland, Washington.

3. SNL, 1983, Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, Sandia National Laboratory.

4. SNL, 1984, Summary of Accident Sequence Evaluation Program Work for Senior Consultant Group Review, Sandia National Laboratory.

6.0 FUEL DAMAGE PHENOMENOLOGY

6.1 Fuel Characteristics

N Reactor fuel elements (see Figure 6-1) consist of two concentric tubes of metallic uranium clad in Zircaloy-2. The elements are contained in a Zircaloy-2 pressure tube. The metallic uranium is an alloy containing small (ppm) levels of silicon, iron, and aluminum to reduce its growth characteristics in a high-temperature radiation environment. There are significant differences between the metallic uranium fuel elements used in N Reactor (which have an average linear power of 114.6 kw/ft for the two annular fuel elements) and the uranium oxide fuel pins in a light-water reactor that run at approximately 7 kw/ft, average. However, the average heat flux is approximately the same, ~240,000 BTU/ft^2/h in N Reactor, and ~210,000 BTU/ft^2/h in a light water reactor.

6.2 Fuel Damage Progression

Fuel damage progression in N Reactor involves approximately 16 to 19 fuel elements in a pressure tube, separated by the graphite moderator from the adjacent fuel columns. The pressure tubes run through N Reactor horizontally. Therefore, any metallic uranium must first escape from its cladding, and second, penetrate its pressure tube and third penetrate the graphite block before it can possibly mix with any metallic uranium from another fuel column. Both the pressure tube (Zircaloy-2) and the graphite represent formidable physical barriers to melt progression at the melting temperature of N Reactor metallic uranium fuel (relevant melting points are: uranium - 1090 ^{o}C, Zircaloy - 1850 ^{o}C, graphite - 3700 oC).

Because the N Reactor pressure tubes are approximately 53-ft long and the fuel column is 34.8-ft long, there is sufficient length of relatively cool fuel and spacer material and pressure tube at each end to justify the assumption the molten uranium will solidify before escaping out the end of a pressure tube.(1) The only other escape paths are axial expulsion as a result of a pressure pulse, or penetration of the pressure tube wall and surrounding graphite. Axial expulsion of fuel material would not be expected as part of a fuel melt progression.

Results of fuel melting tests for typical N Reactor irradiated elements have been documented.(1) Test fuel pieces were irradiated as special production test material to obtain relatively accurate irradiation exposure levels. During the melting experiments, the fuel showed some degree of separation at the cladding-uranium interface, starting at temperatures ranging from 790 ^{o}C to 1080 ^{o}C. Rupture was characterized by a forceful smokey gas expulsion. At the lower fuel burnup (approximately 2,000 MWD/MTU), the initial gas burst was followed immediately by an expulsion of a mass of plastic uranium through the rupture. Little or no uranium was expelled from the fuels with the highest burnup (above approximately 4,000 MWD/MTU). At the higher burnups, the fission gases tended to separate the cladding from the fuel; whereas at the lower burnup, it tended to cause the uranium to form a foamy mass that was expelled.

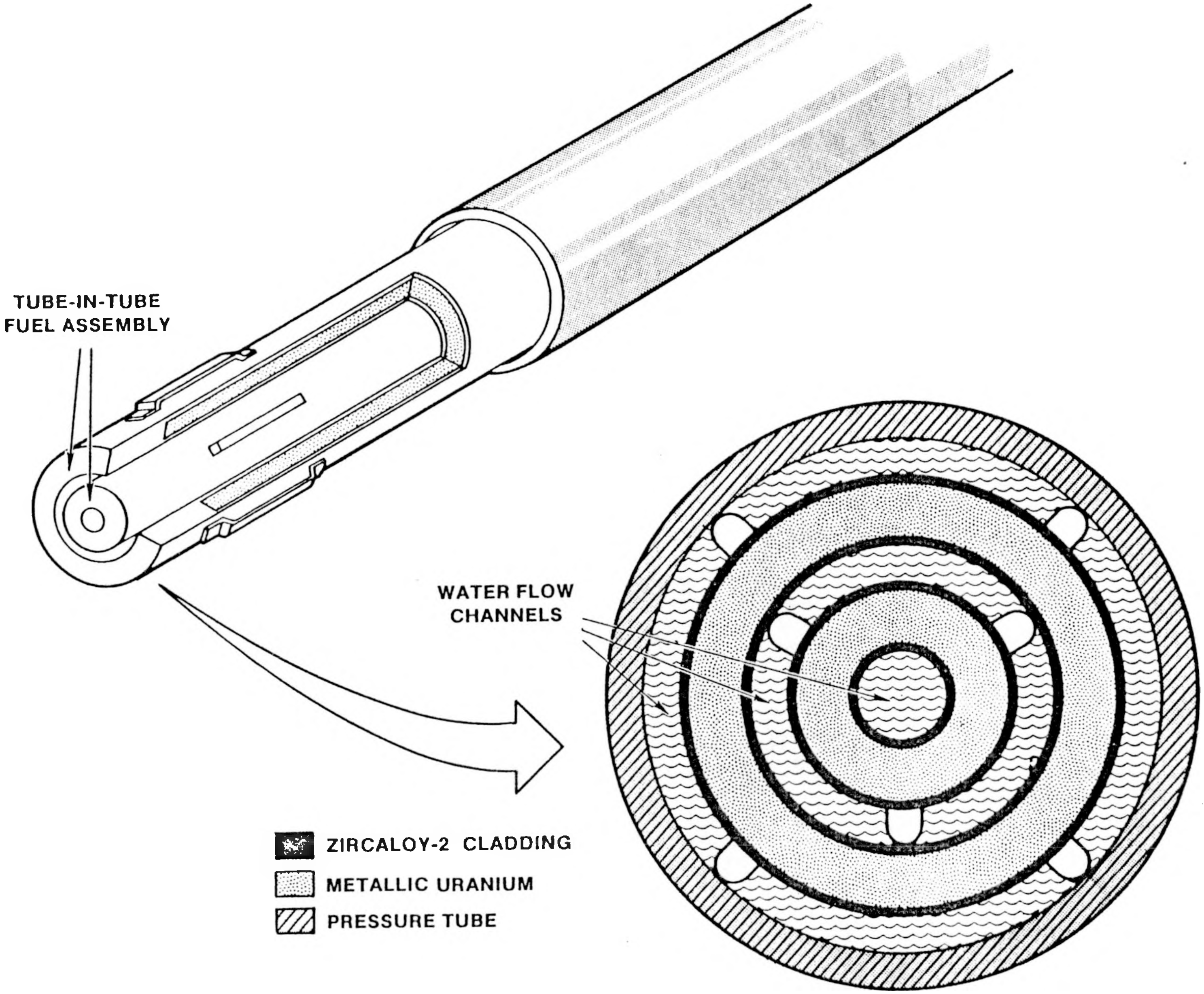


Figure 6-1. N Reactor Fuel Element.

There were nine fuel melting tests (five outer and four inner elements) conducted and documented. The documentation states that "Of the twenty outer fuel elements ruptured to-date, only one produced a secondary rupture that expelled uranium to the annular space between the fuel and the process tube (SNH-11)."[1] Two definite failure patterns are reported:

- Outer fuel elements ruptured initially at the interface between the inner cladding and the end-cap
- Inner fuel elements ruptured initially near the longitudinal center of the outer cladding.

With both the inner and outer fuel elements expelling fuel to the inner coolant annulus, that channel will most likely be blocked by the failure. Any coolant or steam circulation would be limited to the central hole and the outer coolant subchannel, formed by the fuel element and the pressure tube. Secondary ruptures of the inner element observed during the testing would add to the plug in the inner subchannel and subsequent collapse into the central hole would be expected. This leaves only the outer subchannel with a reasonable likelihood of remaining open after rupture and melting of the fuel.

With severe fuel damage, oxidation of the fuel may occur following a pressure-tube rupture, if the fuel is exposed to an oxidizing (steam or air) atmosphere. If plugging prevents coolant from reaching a surface, then oxidation would be reduced significantly. After the initial fuel rupture, it was suggested[1] the principal source of hydrogen would be from oxidation of Zircaloy, with some hydrogen added from oxidation of the exposed uranium beyond the upstream and downstream plugs. This condition limits the surface available for oxidation and thus, reduces the heat load to be dissipated. It also reduces the hydrogen generation rate.

As the temperature continues to rise, the situation changes and is described in HW-70652,[2] for temperatures of 1200 ^{o}C and 1300 ^{o}C in an air atmosphere. The fuel elements in the tests were unirradiated.[2] Zircaloy oxidation rates in steam are expected to be approximately a factor of three lower than in air at these temperatures. Because of the relative ease, testing was completed in an air atmosphere rather than in steam. The fuel elements were of the inner variety with outer cladding, 0.040-in nominal thickness, and inner cladding, 0.025-in nominal thickness. Three elements were subjected to testing. They were supported in a stainless-steel trough with a layer of mica between the trough and the fuel element. This served to protect the adjacent surface of the cladding from oxidation and gave a comparison of the rate of attack of molten uranium on Zircaloy with that from oxidation. One element was heated to 1200 ^{o}C and held for 1 h. The other two were heated at 1300 ^{o}C and held at temperature for 1 h. The reported penetration by the uranium was at the bottom, where oxidation by the air was minimal. The penetration depth was approximately 4 and 8 mils at 1200 oC and 1300 ^{o}C, respectively. The average penetration due to oxidation by the air was 31 mils on the outer surface (initially 40 mil cladding thickness) and the inner cladding was completely destroyed in 1 h at 1100 ^{o}C to 1200 ^{o}C. At 1200 ^{o}C to 1300 ^{o}C large areas of the outer cladding were completely destroyed. The fuel pieces were approximately 8-

in. long, 1-1/4 in. outer diameter (OD) and 7/16-in. ID. The stainless steel trough was semicircular (OD = 1-1/2 in.), indicating a close fit between the trough, mica, and fuel.

Once fuel begins to oxidize, it expands significantly. If all the fuel in an outer element were completely oxidized, the volume change would be larger than the space available in the outer coolant subchannel, even at 100% density for the uranium oxide. This provides a means for blocking a channel, but a significant amount of time at high-temperature in an oxidizing atmosphere is required.

There is also the potential for slumping into the inner coolant subchannel, which would be partially filled by fuel from the inner fuel element.

An estimate of the time required for oxidation of metallic uranium as a function of temperature in the 1100 °C to 1600 °C range was calculated.(3) The analysis(3) considers the following:

- Loss of coolant and subsequent loss of ECCS produce the potential for core damage
- Core heatup is determined by decay heating and local oxidation effects, and depends on the status of the GSCS
- Fuel failure occurs in the range of 1000 °C to 1050 °C and uranium is expelled into the inner coolant subchannel blocking that subchannel; fuel melting may follow
- Oxidation of the uranium proceeds as a function of temperature and time and tends to reduce the coolant flow area
- At some point in time, enough uranium is oxidized to cause a volume change sufficient to block the pressure tube. Hydrogen production is reduced at that time, even if steam is available.

6.3 Fission Product Release from Fuel

The following discussion deals primarily with the temperature range above the melting point of metallic uranium (approximately 1100 °C). It is concluded that the complete oxidation of metallic uranium would release 100% of the iodine, tellurium, noble gases, and cesium.(3) The relative release magnitudes for these and several other fission product species are reported to be the following, based on the fraction of uranium oxidized:

- Noble Gases are R = 1.0x
- Iodine is R = 1.0x
- Tellurium is R = 1.0x
- Cesium is R = 1.0x

- Ruthenium is $R \leq 0.05x$
- Strontium is $R \leq 0.01x$
- Zirconium is $R \leq 0.01x$
- Cerium is $R \leq 0.01x$
- Barium is $R \leq 0.01x$

Where R = fraction of the fission product inventory released.
x = fraction of uranium oxidized.

Although 100% oxidation is not expected to accompany fuel failure, the source terms generated in Section 8.0 are based on the assumption of complete oxidation.

At the fuel rupture temperature, noble gases are released to the pressure tube and may be swept out to the RCS piping, depending on flow conditions in the RCS. Failures are expected only in the event of complete loss of normal and ECCS flow. Under such conditions, preliminary calculations indicate core voiding in 700 to 800 s. In this case, diffusion would be the primary transport process. In the early stage of the accident, all fission products will traverse the same path from fuel to confinement. At a later time, escape through a rupture in the pressure tube may be possible, if conditions leading to pressure tube failure exist.

6.4 Fission Product Transport Consideration

Fuel damage accidents involve loss of cooling to the fuel in one or more pressure tubes. For the Z or Y end states (see Table 5-2), cooling is lost for sufficient time for damage to occur in 30% or more of the fuel. For the Y end-states, the GSCS remains operable and limits the extent of fuel damage; for the Z end-state, GSCS also fails, and most of the fuel in the core is damaged. For both cases, the ECCS has failed to provide cooling to the fuel.

For those LOCA sequences involving loss of ECCS cooling flow, there is an open flow path from the fuel to the confinement via the pipe break and through opening of the V-4 valves to the dump tank. The water in the primary system will flash to steam during the depressurization process. Depending on break location and size, differing fractions of the primary system inventory will pass through the core on its way out of the system. For the cold leg manifold break (large LOCA), the system depressurizes rapidly over the first 150 s and somewhat slower thereafter out to approximately 1,200 s. The mass flow through the core and out the break and the open V-4 valves is continually decreasing and reaches essentially zero at approximately 1,200 s.

The primary mechanism for fission product transport following fuel failure is the flow of steam through the system. Because fuel failures for this accident are calculated to occur at approximately 1,000 s and shortly thereafter, the diminishing steam flow through the core will provide the only motive force for transport of the noble gases out of the system. Little, if any, of the volatile fission products will have been released from the fuel and transported out of the core in the approximately 200 s following fuel failure and prior to the steam mass flow decreasing to essentially zero.

Consequently, for LOCA accidents ending in the Z or Y end-states, fission product transport out of the primary system is limited. An analytical study of N Reactor fuel was performed reviewing the thermodynamically preferred chemical forms of the fission products and their physical forms.(4) The study found that the thermodynamic conditions would not favor aerosol formation, and the plating and chemisorption processes in the colder end regions of the pressure-tubes occupied by spacers (approximately 7- to 10-ft on the ends of the core) would remove essentially all of the volatile fission products.

Therefore, the combined effects of minimal steam flow, and the chemical/physical behavior of the fission products will retain essentially all of the fission products, other than noble gases, within the N Reactor core. In this document, it is conservatively assumed that fission product release from the fuel is transported from the core and is not reduced in the pressure-tube or piping. Exceptions to this are discussed in Section 8.2.3.

6.5 References

1. Reid, D. L., Hesson, G. M., and Hammond, J. E., 1970, Interim Report - The Metal-Water Reactions of Irradiated N Reactor Fuels at 1025°C to 1080 °C, BNWL-1361, Battelle Northwest Laboratories, Richland, Washington.

2. Reid, D. L. and Hilliard, R. K., 1961, Cladding Integrity of NPR Fuel Elements Above the Uranium Melting Point, HW-70652, Richland, Washington.

3. Birney, R. K., 1987, Fission Product Release Considerations for N Reactor Fuel, (memorandum to W. J. Quapp, UNC Nuclear Industries, Inc., Richland, Washington, February 2, 1987; detailed in UNC memorandum, K. R. Birney to W. J. Quapp, Suggested Prescriptive Fission Product Release Parameters, April 2, 1987).

4. Cronenburg, A. W., 1987, Scoping Analysis of Fission Product Behavior and Transport in the Primary Coolant System for Postulated N Reactor Severe Accidents, WHC-SP-0071, Westinghouse Hanford Company, Richland, Washington.

7.0 CONFINEMENT RESPONSE

7.1 Summary of Confinement Response Analysis

Section 3.0 of this report discusses the major features of the confinement for N Reactor. The confinement serves as the final barrier to fission product release to the environment, should fuel damage occur. Therefore, success of the confinement systems and the integrity of the confinement have a direct effect on the resulting fission product release and subsequent consequences of any fuel damage accident.

This section discusses the event tree modeling and quantification of the possible sequences that describe the confinement response to various accidents. This modeling considers the response and impact on radionuclide release of the major confinement systems, including confinement isolation, confinement fog sprays, the building steam vents, and the filtered exhaust system. The possible phenomenological effects of a hydrogen burn or steam explosion are also considered. These all affect the magnitude of the air release path that is available for fission product release. Liquid can also be released from the plant, but this pathway is not developed in the confinement response analysis. The consequences of such a release would be relatively small and are discussed in Section 9.0. The sequences described by the confinement event tree models depict the potential scenarios, which generally range from success or negligible releases to unfiltered air releases. Some noble gas releases are expected for all fuel damage events, even with confinement success. The latter cases are considered in Section 9.0 and shown to be insignificant.

Based on the estimated frequency of each plant damage state (see Section 5.0) and the probability of each corresponding confinement scenario, the frequencies of various source terms are identified. Each source term represents a level of fission product release as a result of the accident. The frequency for each source term and the accompanying consequences collectively describe the risk from fuel damage accidents at N Reactor.

Confinement event trees are quantified considering important interactions among confinement features, and between confinement features and core cooling systems. Based on the small number of shared components or common support equipment among these systems, it is possible to quantify the confinement responses as independent events with only a few probabilities being dependent on the specific fuel damage scenario. The quantification results reported in this section are conditional probabilities given the fuel damage accident. Quantification results are presented in Section 7.4.

7.2 Confinement Response Events

7.2.1 Top Event Selection Process

The selection of the top events to be included in the confinement response model was based on plant documentation and on information obtained during discussions with plant personnel. Much of this information came from the NUSAR,[1] preliminary Sandia National Laboratory work on the full-scope Level 2/3 PRA, and ongoing work being performed for the full-scope Level 1 PRA on the confinement systems. Additional insight was gained during the plant walkdown conducted in March 1987.[2] Another significant source of information was a week-long meeting with plant operations and engineering staff during February 1987, when the basic confinement structure and systems were discussed in detail.

The top events in the tree were also chosen so the following major distinctions could be made in the tree end-states and the source term magnitudes:

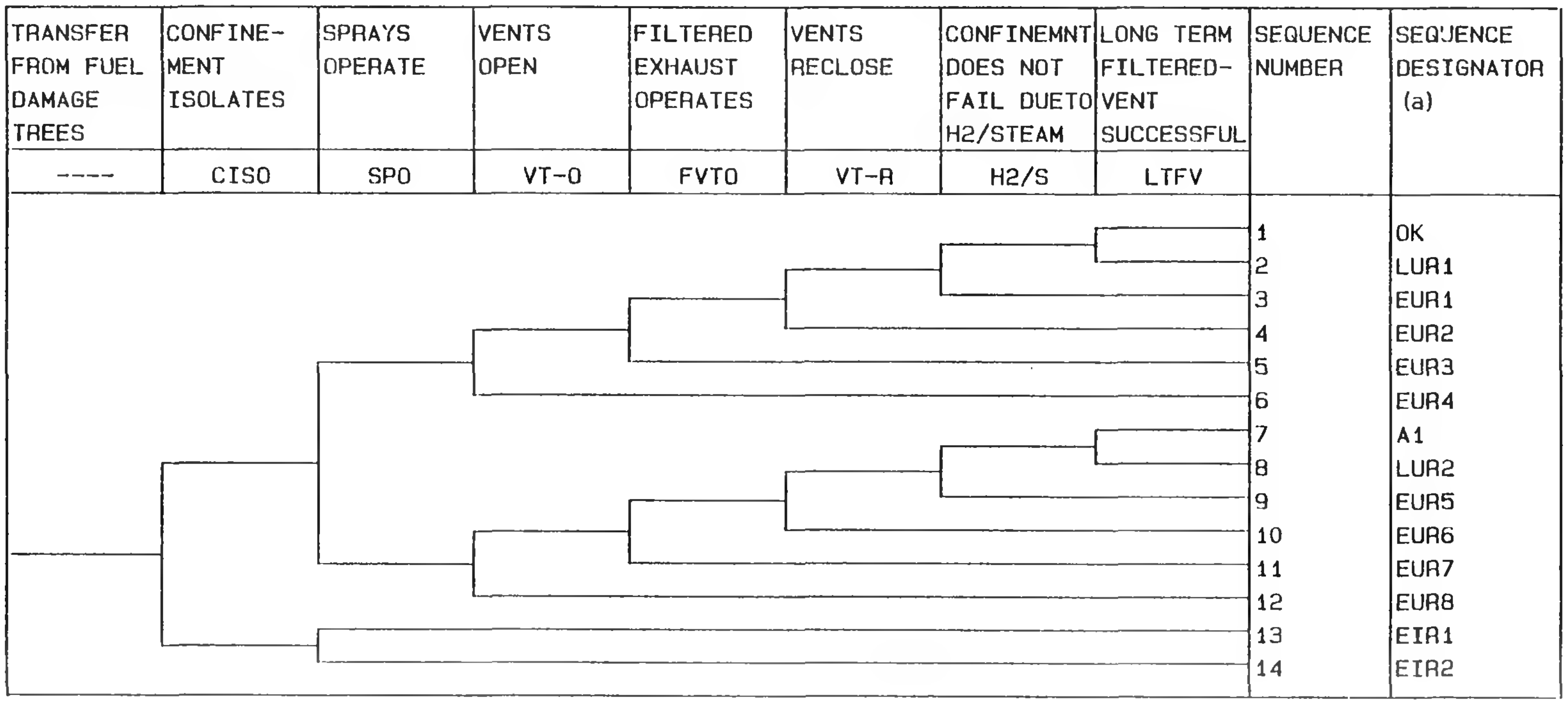
- A noble gas release only
- An unfiltered air release, with or without confinement fog sprays, for fission product removal
- A delayed or late unfiltered air release, with or without confinement fog sprays.

These general distinctions are used to define the source term categories presented in Section 8.0.

7.2.2 Overview of Event Descriptions and Event Trees

The confinement system and its operation are described in more detail in Section 3.3. Regardless of the type of fuel damage accident, the confinement response is nearly the same. On the initial pressure surge, confinement isolation first takes place followed by (as necessary) confinement fog-spray initiation and opening of the 105-N/109-N steam vents and cross-vent doors between the two buildings to relieve the initial pressure rise (such as for a LOCA). In all design basis events (except for the single-tube loss of flow) and in all known events beyond the design basis, the initial pressure relief occurs before any fuel damage has resulted and the vented material contains no appreciable radionuclides. In the single-tube loss of flow, a small quantity of fission products from a single fuel assembly will be present in the steam that is vented initially. Once the pressure in confinement decreases, the filtered exhaust path becomes operable and the building steam vent valves which previously were open, are closed. This directs any further air release (which would contain fission products if fuel damage results) through a series of filters that remove most of the fission products before the air is released out the stack to the environment. Fans are not operating at this time so exhaust flow is the result of stack draft and displacement by water flowing into confinement. The filtered exhaust flow path must continue to operate over the long-term (hours to days). Therefore, an additional event was added to the event tree that considers delayed filtered exhaust system failure from such effects as overheating.

Figure 7-1 presents the general confinement event tree model resulting from the above considerations. Because of the virtually identical confinement response to all postulated accidents, this single event tree can be linked to the outcome of each plant event tree end-state. With one exception (to be discussed later), this event tree is used to investigate the confinement response for each accident sequence. The event tree is



Note (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

Figure 7-1. General Confinement Event Tree.

structured so that the events of concern summarized previously are shown in approximate chronological order following an accident. Event tree end-states are OK; unfiltered air release, with or without fog sprays; and delayed unfiltered air release, with or without fog sprays.

The following event descriptions apply to the general confinement event tree shown in Figure 7-1.

1. Confinement Isolates: Event CISO. Success for this event is complete isolation of all the normally operating (H&V) paths that communicate with Zone I of either 105-N or 109-N. Confinement doors are required to be closed, with their seals properly inflated. In this way, there is no fission product bypass path. Failure is defined as one or more of these pathways remaining open.

2. Fog Sprays Operate: Event SPO. Success for this event requires operation of at least one fog-spray pump and adequate automatic spray stations in 105-N and/or 109-N, as determined by the location of a steam release. Failure results in no fission product removal from the confinement atmosphere by fog sprays and delayed depressurization.

3. Vents Open: Event VT-0. The 105-N/109-N steam vent paths open at design pressures (one at 1.25 ± 0.25 $lb/in.^2$ and fourteen at 2.0 ± 0.25 $lb/in.^2$) due to weatherheads blowing off to relieve confinement pressure to avoid exceeding the design value. Failure results in confinement overpressure from an insufficient number of open steam vent paths. (The number of steam vents required is dependent on the specific sequence.)

4. Filtered Exhaust Operates: Event FVTO. Success is defined as initial operation of the filtered exhaust system. This includes opening of at least one vent path to the 117-N filter building, switching to the D filter cell, and proper filter performance (i.e., at least 95% fission product trapping efficiency of particulates and halogens) before air is released from the stack. Failure is considered to result if any of these conditions is not met. Note that fans do not operate so flow is the result of stack draft and air displacement by water from the sprays or other flows into the confiner.

5. Vents Reclose: Event VT-R. Success is defined as closure of the steam vent isolation valves associated with the weatherheads opened in event VT-0. This ensures air exiting confinement passes through the filtered exhaust path. Failure to close any one of the fifteen steam vent pathways (containing redundant valves in series) results in a fission product bypass path directly to the environment.

6. Confinement Does Not Fail Due to H_2/Steam: Event H_2/S. Success means that a damaging overpressure condition does not occur, such as might result from hydrogen burn, steam release, etc. Overpressure can fail the confinement

structure, filter media, or other equipment, resulting in a direct release path to the environment.

7. Long-Term Filtered Exhaust Successful: Event LTFV. Success is defined as continued filter effectiveness. The principal failure mechanisms are clogging, overpressure, and overheating. Failure results in a release of fission products that had previously been collected on the filter media.

The general confinement event tree includes 14 sequences or end-states. The description of each follows.

- Sequence 1. Any appreciable unfiltered air release paths are avoided. The confinement properly isolates, fog sprays operate, building steam vent paths open and reclose, and filtered exhaust operation is successful including long-term operation.

- Sequence 2. All operations are successful, except that long-term operation of the filtered exhaust fails (e.g., due to overheating and ultimately failing). It is likely that some of the trapped fission products escape to the environment. Fog sprays operate to lessen the overall fission product release.

- Sequence 3. Steam or noncondensible gas formation or other phenomena such as explosions or hydrogen burn, cause sufficient damage to confinement to create unfiltered air release paths that bypass or otherwise defeat the filtered exhaust path. Confinement fog sprays operate as a fission product removal mechanism.

- Sequence 4. At least one of the 105-N/109-N steam vent paths that opened fails to properly reclose, creating an unfiltered air release path. Success requires reclosure of all steam vent paths that opened. Confinement fog sprays operate as a fission product removal mechanism.

- Sequence 5. The filtered exhaust path fails to operate because none of the H&V exhaust dampers reopen after initial isolation or the filters themselves fail to perform their function (e.g., early overheating or filter failures due to seismic effects). Confinement fog sprays operate as a fission product removal mechanism.

- Sequence 6. The steam vent paths fail to open with the potential for confinement overpressure failure and unfiltered environmental release.

- Sequences 7 through 12. These sequences are similar to Sequences 1 through 6, except there is no fission product removal/scrubbing because confinement fog sprays fail to function.

- Sequences 13 and 14. At least one of the normal H&V paths fails to isolate such that at least one unfiltered air release path exists. Confinement fog sprays are successful in Sequence 13 and fail in Sequence 14.

The simplified confinement tree shown in Figure 7-2 is used for a local (single-tube) loss of flow accident. The fuel damage end-state is a single-tube failure with fuel melting and release of fission products before pressure-tube failure. An early release of fission products would occur if the confinement failed to isolate or if the fog sprays fail to operate. If fog sprays operate, confinement pressure would remain below that required to lift the weatherheads. If fog sprays fail to operate, the weatherheads will lift and fission products will be released with the vented steam, regardless of whether confinement isolation has occurred. The following sequence (end-state) descriptions pertain to the simplified confinement event tree for the single tube failure shown in Figure 7-2. Sequence numbering is the same as in the more general confinement event tree.

- Sequence 1. The confinement isolates and the fog sprays operate as designed.
- Sequence 7. Same as Sequence 1 except the fog sprays are inoperable. This results in weatherheads opening and an environmental release.
- Sequences 13 and 14. Environmental releases result from failure of confinement isolation. In Sequence 13, fog sprays operate. In Sequence 14, they do not, and environmental releases also occur via the steam vent when the weatherheads blow.

Section 7.4 addresses more specifically how each confinement event tree is used in conjunction with the important fuel damage sequences and the resulting plant damage states. The next section summarizes fuel damage states that serve as inputs to the confinement and source term analyses.

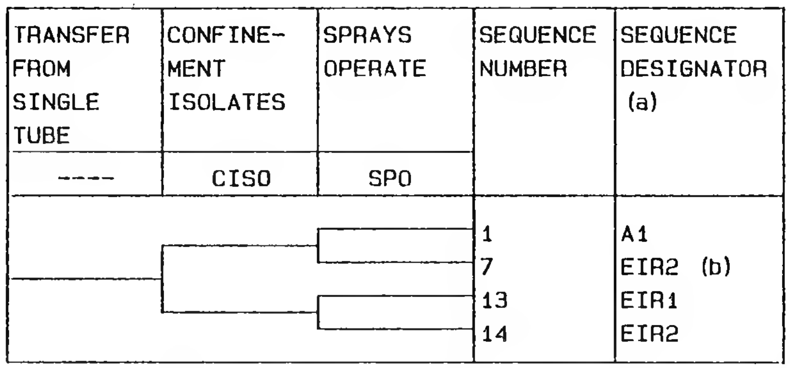
7.3 Plant Damage State Groupings

The plant damage states that categorize the degree of fuel failure for each fuel damage accident sequence are discussed in Section 5.2. Each different category or plant damage state represents a different level of fission products released to the confinement atmosphere and thus, available for release if confinement failure occurs. In summary, five major fuel damage categories were identified:

- W: $\leq$2% Fuel Damage
- X: $\leq$6% Fuel Damage
- Y: $\leq$30% Fuel Damage
- Z: 100% Fuel Damage
- Single Tube: Fuel Damage in a single process tube, equivalent to approximately 0.1% fuel damage.

In addition, a subcategory of the Z damage state is also defined:

- Z (Delayed): 100% Fuel Damage occurring at time >>2 hours.



Note: (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.
(b) With the fog spray failure, the weatherheads will lift and fission products will be vented with the steam release. This has the same characteristics as a confinement isolation failure.

Figure 7-2. Single Tube Confinement Event Tree.

The confinement response (unfiltered air release with no fog sprays, etc.), coupled with the accident sequence and fuel damage state, provides the basis for estimating radionuclide release to the environment (source term). Other factors considered in evaluating source terms are discussed in Section 8.0.

7.4 Confinement Response Sequence Analysis

Based on the plant response analysis in Section 5.0, 12 of the highest frequency sequences were selected for consideration in the confinement response analysis. These sequences (listed in Table 7-1 and numbered for reference purposes) include a wide range of sequence types and fuel damage end-states. Sequences represented include transients, transients without scram, loss of coolant, and loss of flow. The interfacing system LOCA sequence has a low-frequency, relative to the other sequences, but is included because of its prior significance in several light-water reactor PRAs.

Separate quantification efforts were required to describe the confinement response to the 12 dominant accident scenarios. Each is discussed in the subsections follow. The detailed confinement system fault tree models, including the failure data for the basic events and the quantified cutsets, are provided in WHC-SP-0021.[3] Support system failure data was also estimated from the full-scope Level 1 fault tree models.[3] Component failure rates are obtained from plant specific data currently being developed for the full-scope Level 1 PRA and from various generic PRA databases.[4,5, and 6] Specific references for each basic event failure rate are included in WHC-SP-0021.[3]

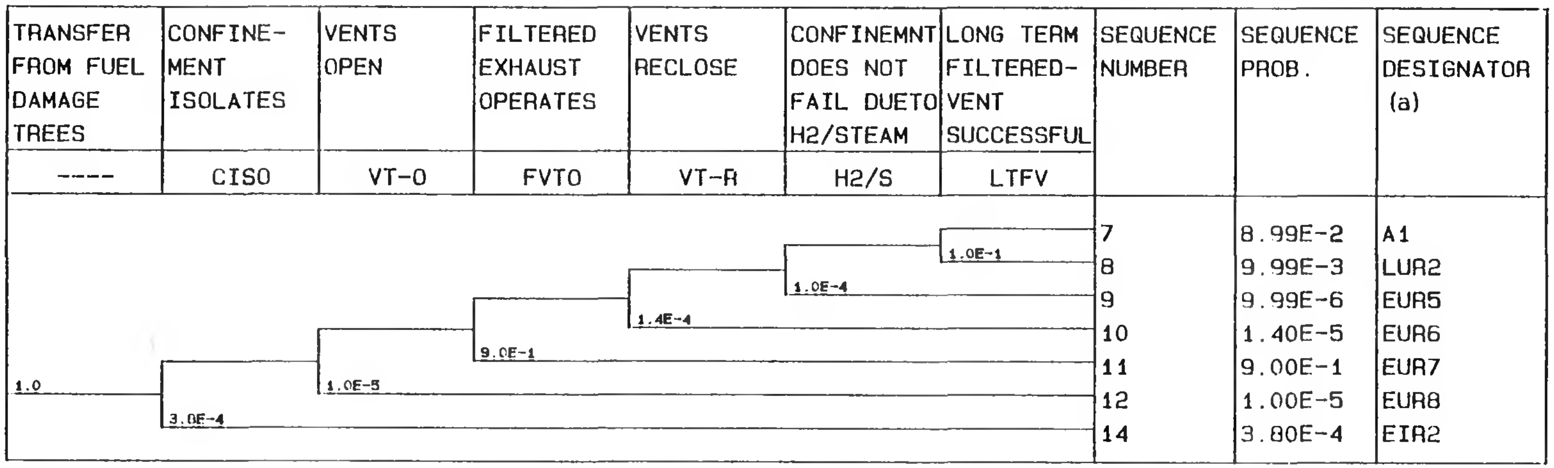
7.4.1 Quantification of Confinement Response for Sequence 1

Figure 7-3 shows the quantification of the confinement event tree for the seismic event with failure of 181-N/182-N. Events CISO, VT-O, and VT-R are quantified, using fault trees for these confinement features.

Confinement steam vents and other isolation features were reviewed and judged not to be susceptible to failure in a seismic event. Recent seismic testing of the steam vent valves has verified this judgement.[7] In an earthquake, event SPO results from failure of 182-N, which leads to fog-spray diesel failure. (The 182-N response is described in Section 5.4.) Therefore, end-states 1 through 6 and end state 13 on the general confinement tree are excluded. Filter bypass (Event FVTO) is assigned a high probability of failure because it is assumed that filter integrity and proper filter operation are unlikely following a seismic event. The anticipated failure mode is leakage past the filter housing frames. The H2/S event is judged to have a low probability of failure since the core will be steam-starved in this accident (no ECCS, GSCS, or HPI). Even if the filtered exhaust path were to operate early, subsequent long-term operation of the filter is considered relatively unlikely, due to failure of 117-N filter cooling sprays. Failure probabilities for these events were developed on the basis of engineering judgement. By a significant margin, early failure of the filtered exhaust path represents the most risk significant confinement response (EUR7). Because of the common-cause

Table 7-1. Plant Sequences Analyzed for Confinement Response

Plant Sequence Number	Fuel Damage End State	Initiator	System Failures	Sequence Frequency
1	Z	Seismic	Buildings 181-N/182-N fail to remain intact	3.7E-5
2	Z	Seismic	Primary shield uplift (pressure tube failures)	2.5E-6
3	Z	Seismic	Reactor trip fails	9.6E-7
4	Z	Transient	Reactor trip fails	2.0E-6
5	Y	Seismic	ECCS fails to start and function	7.6E-6
6	Y	Rapid Shutdown	HPI fails after trip, ECCS fails to start and function	8.0E-6
7	Y	Rapid Shutdown	Primary/Secondary fails, ECCS fails to start and function	7.0E-6
8	Y	Large LOCA	ECCS fails to start and function	2.1E-6
9	Y	Interfacing System LOCA	None	1.0E-9
10	W	Rapid shutdown	HPI fails after trip, Single riser ECCS failure	5.1E-6
11	W	Rapid shutdown	Primary/Secondary fails, Single riser ECCS failure	4.5E-6
12	Single Tube	Local loss of flow	None	2.5E-2



Note: (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

Figure 7-3. Confinement Event Tree for Plant Sequence 1.

seismic event, there is no confinement fog-spray operation to remove fission products.

7.4.2 Quantification of Confinement Response for Sequence 2

Figure 7-4 shows the quantification of the confinement event tree for the seismic event with failure of the pressure tubes. In this case, the fog sprays are not failed by the seismic event and event SPO was quantified using the fault tree. (No fog-spray system components were judged to have a high susceptibility to seismic damage.) In the last end-state depicted on the event tree, SPO failure is high due to the relative contribution of common-cause DC power failures affecting both CISO and SPO, when AC power is lost (assumed to occur in the seismic event). Otherwise, the same considerations apply as described in Section 7.4.1. With the confinement fog sprays functioning, long term filtered exhaust failure is considered less likely because (1) 117-N filter spray operation may not be required because the best engineering judgement is that air flow through the filters will be sufficient to provide the required cooling; and (2) designed filter capacity is greater than the filter load that would result from a fuel damage accident.

Early filtered exhaust bypass, with and without confinement fog-spray operation, are the dominant confinement failure responses (EUR3 and EUR7). In the more likely case, confinement fog sprays are operable to remove fission products.

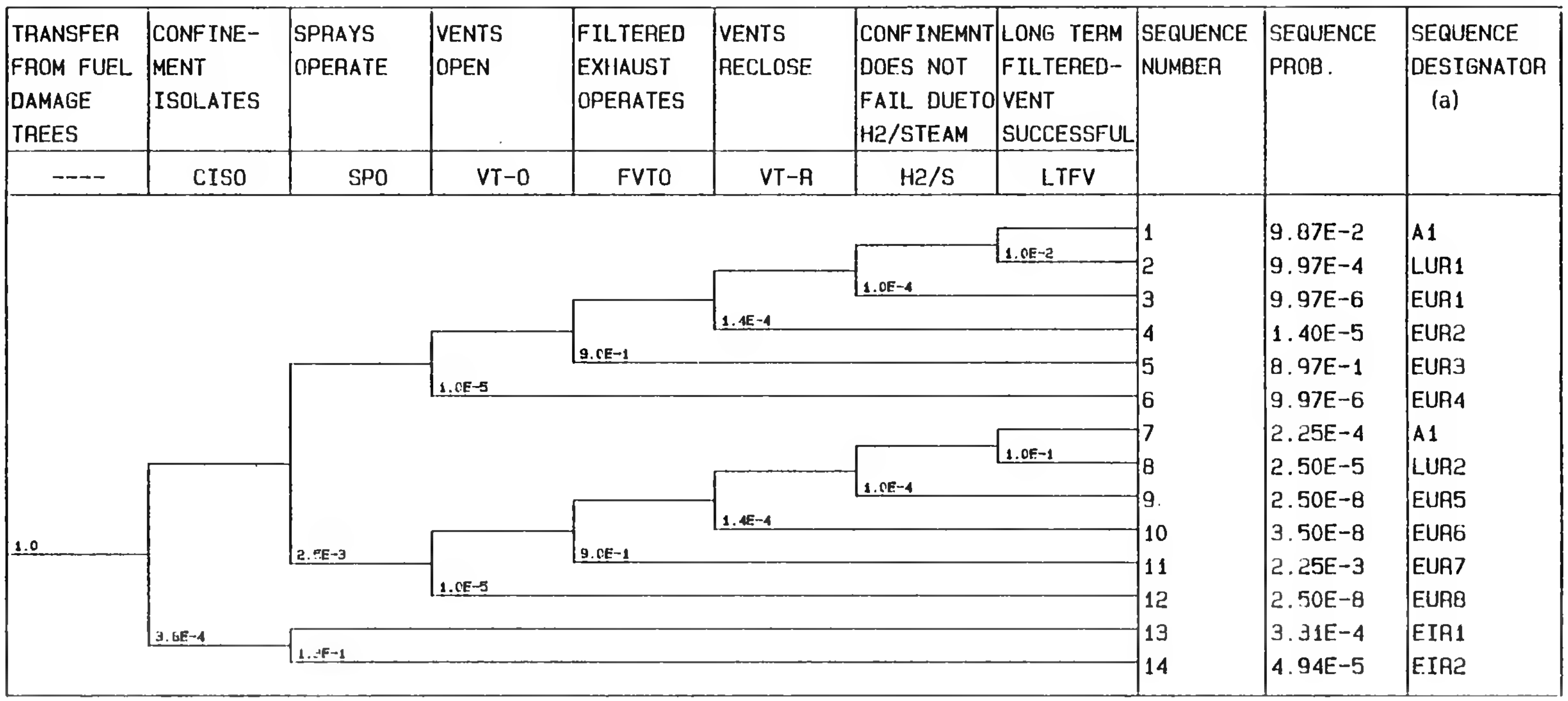
7.4.3 Quantification of Confinement Response for Sequence 3

Figure 7-5 depicts the quantification of the confinement event tree for the seismic event with failure of reactor trip. The considerations for this sequence are the same as for the previous accident sequence, except for the potential hydrogen burn. Due to voiding and partial shutdown reactivity insertion, reactor power will be decreased, but may remain at a level where the ECCS cannot provide adequate cooling. Because ECCS water is potentially available for steam production, a significant confinement failure probability due to hydrogen burn has been included. The probability of hydrogen burn is based on engineering judgement.

The dominant confinement failure responses include filtered exhaust bypass with confinement fog sprays operable (EUR3) and an energetic release as a result of an H_2 burn, also with fog sprays available (EUR1). The non fog-spray cases are almost a factor of 1,000 less probable than these dominant responses.

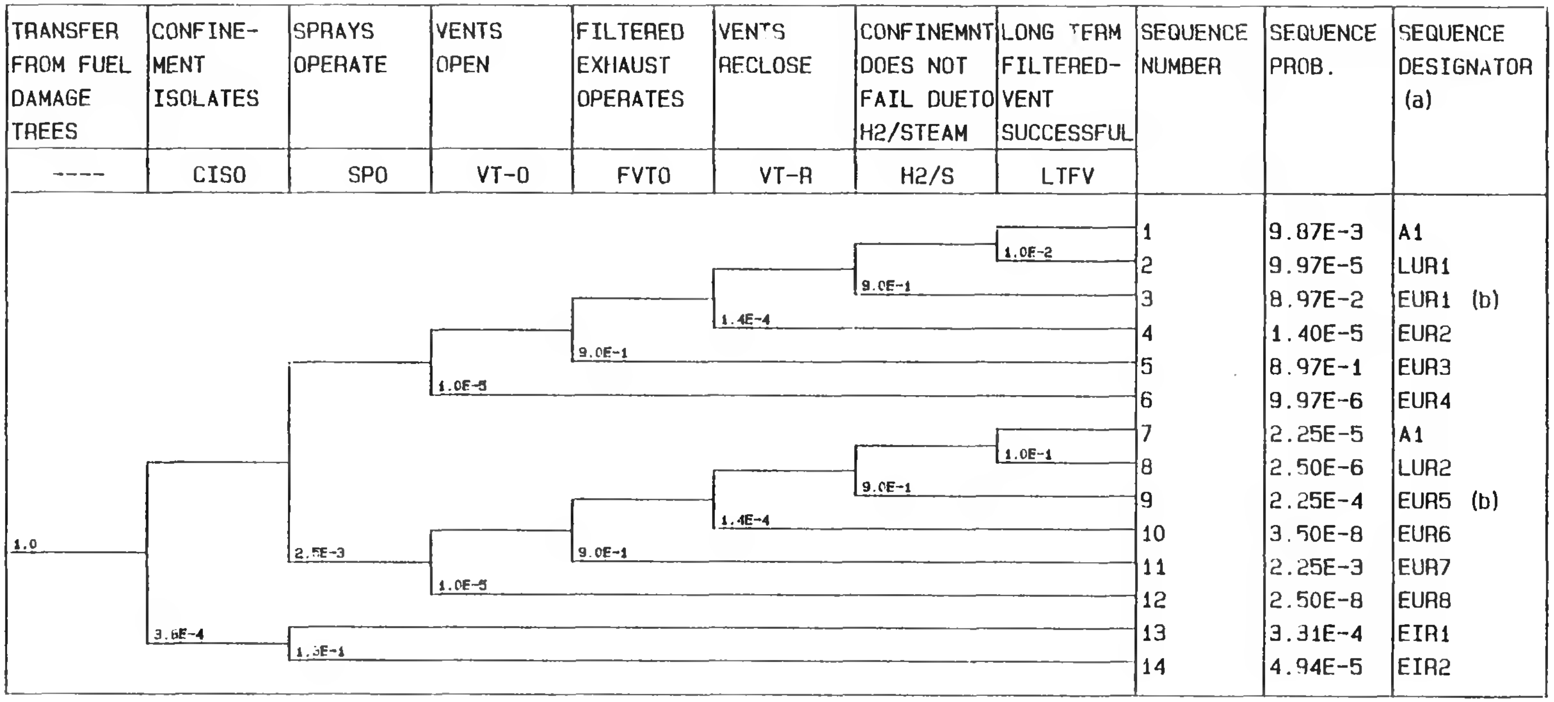
7.4.4 Quantification of Confinement Response for Sequence 4

This sequence is a rapid shutdown transient with failure of reactor trip. The confinement response is the same as in Sequence 3, except the filtered exhaust system is not assumed to be bypassed as a consequence of the seismic initiator common to Sequences 1 through 3. The confinement event tree for Sequence 4 is shown in Figure 7-6. The dominant confinement failure responses are EUR1 and EUR3.



Note: (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

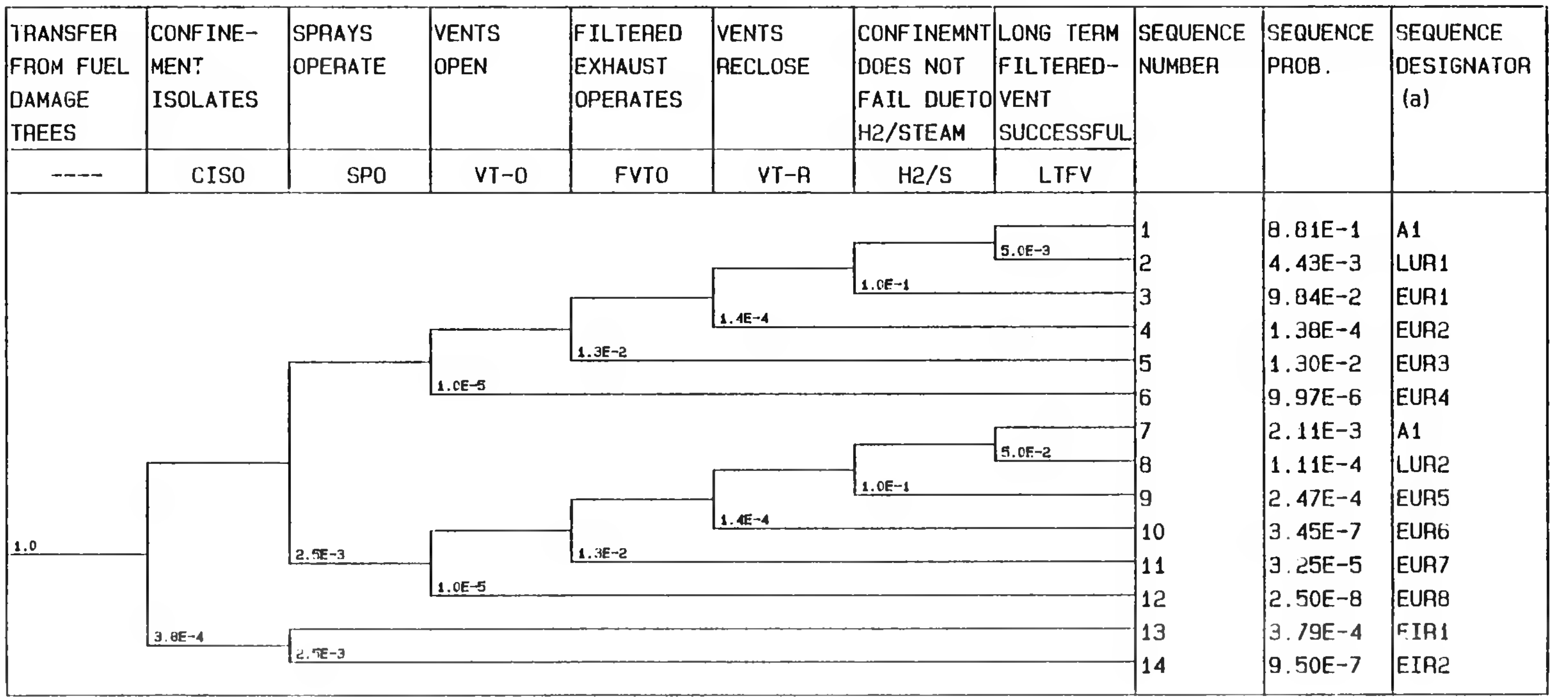
Figure 7-4. Confinement Event Tree for Plant Sequences 2 and 5.



Note: (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

(b) The estimated probability of confinement failure at branch H2/S is 0.1. To use the general confinement tree for Sequence 3, a probability of 0.9 is entered for the H2/S branch so the correct probability results for the risk-dominant end-states EUR1 and EUR9.

Figure 7-5. Confinement Event Tree for Plant Sequence 3.



Note: (a) Al = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

Figure 7-6. Confinement Event Tree for Plant Sequence 4.

7.4.5 Quantification of Confinement Response for Sequence 5

This sequence involves a seismic event with failure of ECCS. In this sequence, ECCS failure results from seismic loading of the silo or from seismically independent causes. The GSCS remains operable, thereby limiting the fuel damage to the Y damage state. The confinement response is expected to be the same as described in Section 7.4.2 and Figure 7-4. The dominant confinement failure is therefore bypass of the filtered exhaust pathway with confinement fog sprays operable (EUR3).

7.4.6 Quantification of Confinement Response for Sequence 6

Figure 7-7 depicts the quantification of the confinement event tree for the reactor trip event with loss of primary/secondary cooling caused by a loss of HPI system function and the coincident failure of ECCS.

Similar considerations apply to the quantification of this tree as for the previous accident scenarios. However, the probability of LTFV failure is lowered somewhat, considering the potential for operator initiation of the filter cooling sprays (if needed). Also, the failure probability for event FVTO is based on fault tree results and is dominated by failure of only the D filter cell (this is the cell used during accident conditions). Therefore, some filtering would still be performed by the cell in operation. The limited fuel damage (Y damage state) for this sequence and the failure of ECCS suggests that little hydrogen will be produced, resulting in a low probability for the H2/S event. The H2/S event probability value is based on engineering judgement.

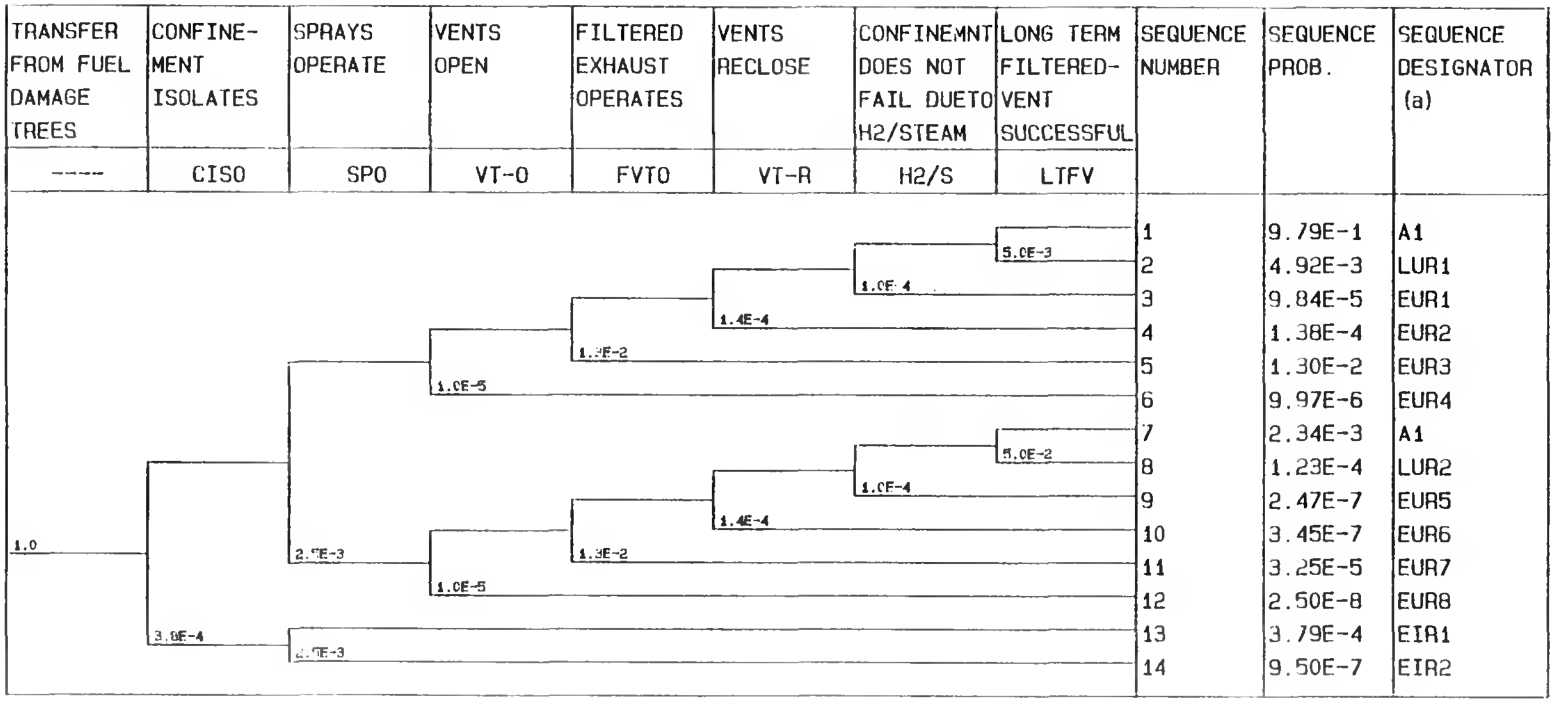
The results show that early failure of D cell is the dominant confinement failure response. In this case, some filtering is being performed by the operating filter cell and the confinement fog sprays are operable as a scrubbing feature (EUR3). With confinement fog sprays operable, the other potentially important confinement failure paths are long-term failure of the filtered exhaust (LUR1), confinement isolation failure (EIR1) and failure to reclose the exhaust gates (EUR2). Without confinement fog sprays, long- and short-term filtered exhaust failures appear to be most important (LUR2 and EUR7). Again, the short-term failure is only failure of D cell (EUR3, LUR1, EIR1, EUR2, LUR2, EUR7).

7.4.7 Quantification of Confinement Response for Sequence 7

This sequence is the same as Sequence 6, except the loss of primary/secondary cooling is caused by failures such as loss of coolant pumps or loss of feedwater supply, rather than loss of the HPI system. The confinement response is the same as for Sequence 6 and is shown in Figure 7-7.

7.4.8 Quantification of Confinement Response for Sequence 8

This is a large LOCA with failure of the ECCS; however, the GSCS operates to limit the fuel damage to a Y damage state. The confinement response is the same as that described in Section 7.4.6 and illustrated in Figure 7-7.



SEQUENCE NUMBER	SEQUENCE PROB.	SEQUENCE DESIGNATOR (a)
1	9.79E-1	A1
2	4.92E-3	LUR1
3	9.84E-5	EUR1
4	1.38E-4	EUR2
5	1.30E-2	EUR3
6	9.97E-6	EUR4
7	2.34E-3	A1
8	1.23E-4	LUR2
9	2.47E-7	EUR5
10	3.45E-7	EUR6
11	3.25E-5	EUR7
12	2.50E-8	EUR8
13	3.79E-4	EIR1
14	9.50E-7	EIR2

Note: (a) A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.

Figure 7-7. Confinement Event Tree for Plant Sequences 6, 7, 8, 10, and 11.

The dominant confinement failures are early failure of D cell, or long-term failure of the filtered exhaust, with or without confinement fog sprays (EUR3, LUR1, EUR7, LUR2). Confinement isolation failure or failure to reclose all the exhaust gates, both with successful operation of the fog sprays (EIR1, EUR2), may also be important.

7.4.9 Quantification of Confinement Response for Sequence 9

This sequence involves an interfacing system LOCA, which is a LOCA in the low-pressure portion of the ECCS. Two fission product release paths exist from the fuel to the environment. One is a direct result of the failure of the ECCS injection line so the fission products bypass the confinement and are released into the 182-N where the ECCS pump seals are assumed to be failed. The competing pathway is out through the ECCS V-4 valves, through the dump tank, and back into confinement through the tank vent line. No special confinement tree has been constructed because part of the release (assumed to be 50%) bypasses confinement with a probability of 1.0 (V-SEQ, See Chapter 8.0).

7.4.10 Quantification of Confinement Response for Sequence 10

The confinement response probabilities and dominant failures are the same as described in Section 7.4.6, and Figure 7-7 applies. The only difference is the fuel damage is limited to a W damage state.

7.4.11 Quantification of Confinement Response for Sequence 11

The confinement response probabilities and dominant failures are the same as described in Section 7.4.6, and Figure 7-7 applies. The only difference is the fuel damage is limited to a W damage state.

7.4.12 Quantification of Confinement Response for Sequence 12

The plant sequence is a single-tube loss of flow. Confinement response is quantified using Figure 7-8. In this tree, events CISO and SPO are quantified from system fault trees, as before. Fog-spray failure, with fission product release via the weatherheads, is the risk dominant failure path (EIR2). Fuel damage is limited to a single pressure tube.

7.5 Dump Tank Considerations

For transient sequences where there is a demand for ECCS, the V-4 valves open and reactor coolant is vented to the dump tank. Because transient sequences do not involve a breach of the coolant pressure boundary, radionuclides released during fuel failure flow from the core region to the dump tank before being vented back to the confinement. The

TRANSFER FROM SINGLE TUBE	CONFINE-MENT ISOLATES	SPRAYS OPERATE	SEQUENCE NUMBER	SEQUENCE PROB.	SEQUENCE DESIGNATOR (a)
----	CISO	SPO			
			1	9.97E-1	A1
1.0		2.5E-3	7	2.50E-3	EIR2 (b)
	3.8E-4		13	3.79E-4	EIR1
		2.5E-3	14	9.50E-7	EIR2

Note: (a) Al = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release
Confinement Release Modes (Sequence Outcomes) are further described in Section 8.0, Table 8-2.
(b) With fog-spray failure, weatherheads will lift and fission products will be vented with the steam release. This has the same characteristics as a confinement isolation failure.

Figure 7-8. Confinement Event Tree for Plant Sequence 12.

water in the dump tank will retain a significant portion of the fission products that enter the tank.

The dump tank function is part of the overall confinement system, and radionuclide removal in the dump tank might have been included as a top event on the confinement event trees (CETs), but was not. Fission product removal in the dump tank was, however, considered in the source term analysis in Section 8.0. Expected radionuclide penetration fractions for the dump tanks are listed in Section 8.0 for transient events.

An analysis of the dump tank reliability during seismic events was not available for use in this Limited-Scope PRA. Therefore, no credit was taken for fission product scrubbing in the dump tank for seismically initiated sequences without loss of reactor coolant system integrity.

Neglecting fission product removal in the dump tank for seismic sequences represents a potential conservatism in the confinement/source term analysis. This potential conservatism is important because seismic sequences are found to dominate risk. Dump tank availability will be evaluated in-detail in the full-scope PRA currently being performed.

7.6 References

1. UNC, N Reactor Updated Safety Analysis Report, UNI-M-190, UNC Nuclear Industries, Inc., Richland, Washington.

2. Kolaczkowski, A.M., (SAIC), to G. R. Armstrong (WHC), 1987, N Reactor Plant Walkdown for Mini-PRA Fire/Flood Analysis, (letter, March 6, 1987).

3. WHC, 1987, N Reactor Limited-Scope Probabilistic Risk Assessment, System Fault Tree Model Quantification, WHC-SP-0021, Westinghouse Hanford Company, Richland, Washington.

4. Stack, D. W. et al., 1987, Preliminary Results of a Probabilistic Risk Assessment of the N Reactor, UNI-I-111, UNC Nuclear Industries, Inc., Richland, Washington.

5. SNL, 1983, Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, Sandia National Laboratory.

6. SNL, 1984, Summary of Accident Sequence Evaluation Program Work for Senior Consultant Group Review, Sandia National Laboratory.

7. Hamilton, D. W., to be issued, Seismic Quantification of Confinement Steam Vent Valve Assembly, UNI-4474, Westinghouse Hanford Company, Richland, Washington.

8.0 SOURCE TERMS

8.1 Fission Product Inventory

The source terms available for release from the confinement atmosphere to the environment depend on the core fission product inventory. This, in-turn, is a function of the total power and the operating history. For all accident sequences evaluated in this study, the core was assumed to be at the end-of-cycle configuration with 69 full-power operating days (full-power equals 4,080 MWt). The fission product inventories were calculated by using the RIBD code.[1,2] The results of this calculation are shown in Table 8-1. End-of-cycle cesium inventories are lower than in power reactors because of the much lower burnup in N Reactor and the long time needed for cesium to reach equilibrium. This is significant because cesium is a principal contributor to latent health effects.

8.2 Atmospheric Radionuclide Releases

8.2.1 Background

The evaluation of accident sequences at N Reactor has identified a spectrum of fuel damage sequences and confinement functional failure modes. The dominant (in terms of frequency) plant damage sequences are summarized in Table 5-1 and a reduced set (numbered 1 through 12) is listed in Table 7-1. It is the reduced set in Table 7-1 that was used as input to the confinement analysis and as input to this analysis. Each of these 12 plant damage sequences could have been coupled with each confinement failure mode, leading to hundreds of unique accident sequences, each with its own radionuclide release. However, it is generally adequate to reduce this complexity by generating a reduced set of radionuclide release bins called release categories. This limited set of release categories contains, within the individual category, sequences with generally similar release magnitudes. In this study, release categories were collapsed so sequences with releases at the mid-point of a category were roughly separated (in terms of release magnitude) by a factor of three or more (about 1/2 decade) from the neighboring categories.

This section discusses the approach used to estimate source terms and presents the results of that process. The approach relies on studies specific to N Reactor,[3,4,5,6, and 7] as well as existing fission product transport calculations for light-water reactors[8] in deriving removal estimates for each transport step. Each transport step, including release from the fuel, transport through the reactor piping, transport through the confinement system, removal by confinement and its subsystems, and release to the environment, has an associated penetration factor, which is the ratio of radionuclide magnitude exiting the process to that entering the process. The penetration factors are used to calculate a source term magnitude for each accident sequence along the sequence pathway identified in the confinement event trees in Section 7.0. The transport pathways and removal mechanisms associated with the confinement release modes are described in Table 8-2.

Table 8-1. N Reactor End Of Cycle Fission Product Inventory.

Radionuclide	Inventory at Reactor Shutdown (Curies)
KR-85	1.113 E+05
KR-85M	4.27I E+07
KR-87	8.229 E+07
KR-88	1.164 E+08
RB-86	8.564 E+03
SR-89	9.672 E.07
SR-90	9.170 E+05
SR-91	1.927 E+08
Y-90	8.906 E+05
Y-91	1.095 E.08
ZR-95	1.138 E+08
ZR-97	2.033 E+08
NB-95	5.865 E+07
MO-99	2.109 E+08
TC-99M	1.820 E+08
RU-103	8.103 E+07
RU-105	4.933 E+07
RU-106	2.431 E+06
RB-105	4.261 E+07
SB-127	5.564 E+06
SB-129	2.448 E+07
TE-127	4.830 E+06
TE-127B	2.700 E+05
TE-129	2.200 E+07
TE-129M	4.607 E+05
TE-131M	1.340 E+07
TE-132	1.483 E+08
I-131	9.979 E+07
I-132	1.493 E+08
I-133	2.340 E+08
I-134	2.613 E+08
I-135	2.184 E+08
XE-133	2.332 E+08
XE-135	3.641 E+07
CS-134	3.853 E+04
CS-136	4.710 E+05
CS-137	9.395 E+05
BA-140	2.103 E+08
LA-140	2.103 E+08
CE-141	1.548 E+08
CE-143	2.011 E+08
CE-144	2.864 E+07
PR-143	1.946 E+08
ND-147	7.670 E+07
PU-239*	5.380 E+04

*Value shown is the total actinide inventory in terms of dose equivalent Pu-239 and is not the Pu-239 inventory.

Table 8-2. Summary of Potential Confinement Release Modes

Release Mode*	Description and Qualitative Attributes of Release Mode
A1 = CET 1 (Sprays) A1 = CET 7 (No Sprays)	Confinement Systems Work Significant fission product attenuation occurs before release to the environment. The filtered vents function. This release mode corresponds to end-states for CET Sequence numbers 1 and 7.
LUR1 = CET 2 (Sprays) LUR2 = CET 8 (No Sprays)	Late Unfiltered Atmospheric Release These release modes result from failure of the filter system after radionuclides have been deposited on the filters. For conservatism, burning is the assumed failure mode. For the LUR1 mode, sprays function; for the LUR2 mode, they do not. The radionuclides associated with this release mode include iodine and particulate fission product species resuspended from filters due to burning. The 80% of airborne fission products escaping the confinement are assumed to be collected by the filter prior to burning. Filter inventories would be less for sequences with sprays, than for those without. A resuspension factor of 90% is used for the burning failure mode. Release modes LUR1 and LUR2 correspond to end-states for CET Sequence numbers 2 and 8, respectively.
EUR1 = CET 3 (Sprays) EUR5 = CET 9 (No Sprays)	Early Unfiltered Atmospheric Release Due to Overpressure These release modes result from overpressure failure of the confinement due to effects such as excess steam generation and potential hydrogen burning. Radionuclides released during initial core heatup would be filtered. For conservatism, it is assumed that no more than 10% of the release from the fuel takes place during the early release phase when filtration is effective. The overpressure event occurs late in the accident sequence after significant radionuclide deposition. A pulse release of the remaining airborne fission products would occur on confinement overpressure failure. For events with more energetic fuel damage progression, such as might occur for large LOCAs or transients without scram, no credit for filtration is taken. EUR1 sequences have sprays operating and EUR5 are sequences without sprays. For conservatism, the releases for mode EUR1 are taken to be the same as for EUR5. Release modes EUR1 and EUR5 correspond to end-states for CET Sequence numbers 3 and 9, respectively.

Table 8-2. Summary of Potential Confinement Release Modes. (Cont.)

Release Mode*	Description and Qualitative Attributes of Release Mode
EUR2 = CET 4 (Sprays) EUR6 = CET 10 (No Sprays)	<u>Early Unfiltered Release Through the Steam Vents Which Fail to Reclose</u> In these release modes the steam vents that are initially open to vent steam from 105-N or 109-N, do not reclose after the weatherheads open. It is conservatively assumed that the radionuclides would flow directly to the 105-N steam vents bypassing deposition sites in 109-N. EUR2 has fog sprays in operation, while for EUR6 sprays are not available to scrub airborne fission products from the confinement. Initially, 10% of the release is assumed to be filtered (as for EUR1). The later release, following pressure tube penetration, is assumed to preferentially leak through the 105-N vent, thereby bypassing the filters. Release modes EUR2 and EUR6 correspond to end-states for CET Sequence numbers 4 and 10, respectively.
EUR3 = CET 5 (Sprays) EUR7 = CET 11 (No Sprays)	<u>Early Unfiltered Release Due to Filtered Vent Path (Initial Unavailability)</u> Unlike EUR1 and EUR5, the filters are not available for either the early or late release. The 109-N deposition sites are conservatively neglected. Leakage at relatively high rates is assumed through 105-N failure locations. For release mode EUR3, sprays function; for release mode EUR7, they do not. Release modes EUR3 and EUR7 correspond to end-states for CET Sequence numbers 5 and 11, respectively.
EUR4 = CET 6 (Sprays) EUR8 = CET 12 (No Sprays)	<u>Early Unfiltered Releases Without Significant Radionuclide Retention</u> These modes are of low probability and do not appear as a separate release category. However, they are treated for completeness. Qualitatively, these release modes involve failure of the weatherheads to open with early confinement overpressure failure and unfiltered release larger in magnitude than EUR3 and EUR7. For EUR4, sprays function; for EUR8, they do not. Most of the airborne fission products are assumed to escape the confinement building due to significant gas flow rates that provide the driving force for transport, thus minimizing the effectiveness of natural deposition processes. Release modes EUR4 and EUR8 correspond to end-states for CET Sequence numbers 6 and 12, respectively.

Table 8-2. Summary of Potential Confinement Release Modes. (Cont.)

Release Mode*	Description and Qualitative Attributes of Release Mode
EIR1 = CET 13 (Sprays) EIR2 = CET 14 (No Sprays)	<u>Early Isolation Failures Involving Transport Paths Through the HVAC and Possibly a Forced Flow Through the Ventilation System</u> EIR1 is expected to be less severe than EIR2, due to the sprays scrubbing effect on airborne fission products in confinement. Release modes EIR1 and EIR2 correspond to end-states for CET Sequence numbers 13 and 14, respectively.

*CET = Confinement Event Tree (See Section 7.0)

8.2.2 Overview of Core Melt Progression

In calculating fission product escape to the environment, it is necessary to estimate the magnitude of the release from the fuel and track the transport of the radionuclides through the reactor coolant system and confinement so the removal mechanisms are accounted for. A brief overview of the fuel damage progression is presented in this section to provide perspective for the assumptions made in calculating penetration factors.

N Reactor fuel is contained in horizontal pressure tubes, which are separated by the graphite moderator from adjacent pressure tubes and fuel. During a severe accident with loss of ECCS and successful reactor shutdown, the fuel heats up, causing fuel clad failure, consequently, volatile fission products diffuse from the fuel and are released as the fuel melts within the pressure tube. If the pressure tube is breached as a result of fuel melting, released fission products escape to the cover gas system and then directly to the confinement. Pressure-tube integrity is expected to be maintained during heatup and initial fuel melting for most accident sequences due to the availability of the GSCS, the significant heat capacity of the graphite, and the highly reliable depressurization of the reactor coolant system. As a result of the latter, ambient pressure on the pressure tube is very low as tube heatup occurs.

For most of the sequence types considered in this study, the pressure tubes will not be breached. In this condition fission products can only escape to the confinement system through existing flow paths in the reactor coolant system. For transient events, the reactor coolant system is initially intact, and fission products are transported to the confinement through the dump tank (quench tank) where scrubbing and removal of radionuclides from the carrier gas is substantial (except for noble gases). As long as coolant system integrity is maintained, the release of fission products to the confinement system would be significantly mitigated. Unmitigated releases to confinement would occur only if the pressure tubes were penetrated.

Because of the relatively low melting point of the metallic uranium metal fuel, only the more volatile fission product species are released during initial melting and prior to potential pressure-tube penetration. If fuel heating progresses to the point of pressure-tube penetration, lower volatility fission product species can be released. Such release is enhanced both by the higher fuel temperatures and possible exothermic reactions. The formation of more volatile forms of tellurium and ruthenium is likely if the fuel is oxidized.

During a LOCA event, a similar fuel damage progression is expected. However, the transport path would not necessarily involve scrubbing through the dump tank. Therefore fission product release directly to the confinement through the pipe break is assumed in the analysis.

For events involving significant fuel melting and penetration of the pressure tubes (such as sequences with loss of ECCS and GSCS cooling), heating of the molten fuel and surrounding graphite can continue. The graphite pile represents substantial additional heat capacity so the heatup rate for the fuel/graphite material will be substantially lower than the heatup rates calculated for light-water reactors.

Studies of fuel heatup and damage progression after pressure-tube penetration were not performed as part of this study. However, to provide a bound for consequences and risk, it was assumed all of the core inventory of volatile fission products (noble gases, iodine, cesium, tellurium) are released to the environment without any retention by the confinement structures or confinement subsystems. Results of the consequence and risk calculations for the bounding case are presented in Sections 9.0 and 10.0. A 1% release for less volatile fission product species was assumed in the bounding case. The assumption of no retention by confinement structures or subsystems is believed to adequately compensate for any possible higher releases of the less volatile species.

8.2.3 Basis of Radionuclide Penetration Fractions

The approach used in estimating radionuclide retention during transport and the magnitude of release to the environment is to develop a set of penetration factors for each transport step. The penetration factors used for the various steps in the fission product transport path are summarized in Table 8-3 and are discussed in the following. A penetration factor is defined as the ratio of the quantity of a radionuclide exiting a transport step to the total that entering the process or step. It expresses the fraction of the radionuclide that remains available for further transport.

The fractional release of fission product material from the fuel is also shown in Table 8-3. Plant damage states involving less than full core damage are scaled down to reflect the actual fission product inventories as follows: Fuel damage state Y would involve release of 50% of the fission product inventory (involving failure of 30% of the core containing 50% of the fission product inventory). The X damage state involves release of 6% of the fission product inventory and the W damage state involves release of 2% of the inventory.

The bases for the fractional fission product releases from the fuel and the penetration factors listed in Table 8-3 are described in the following.

o General Considerations

- Radionuclide Release during Fuel Damage. Fuel damage progression and radionuclide release from fuel during fuel heatup are discussed in Section 6.0 and earlier in this section. For overheating and oxidation of the metallic fuel, the fission product release values were derived from tests on N Reactor fuel. For the early stage of fuel melting before pressure-tube penetration, tests indicate release values of 100% for noble gases, 50% for iodine, and 30% for cesium. For the continuing heatup and a possible oxidation phase following pressure-tube penetration, release of the remainder of the iodine and cesium, 100% of the tellurium, 5% of the ruthenium, and 1% of other less volatile fission products is assumed. These values are believed to be conservative for metallic fuels.[9]

Table 8-3. Fuel Release Fractions and Fission Product Penetration Factors.

Fission Product Group	Fuel Release Factor[1]: Prior to Pressure-Tube Penetration	Fuel Release Factor[1]: Following Pressure-Tube Penetration	Pressure-Tube Penetration Factor	Dump Tank Penetration Factor[2]	Fog-Spray Penetration Factor	Filter Penetration Factor	Internal Deposition Penetration Factor: Delayed Failure[3]	Internal Deposition Penetration Factor: Early Failure[4]
NOBLE GASES								
(Kr-Xe)	1.0 E+00		1.0 E+00	1.0 E+00	1.0 E+00	1.0 E+00	1.0 E+00	1.0 E+00
IODINE								
(Elemental)	9.5 E-01	5 E-02[5]	1.0 E+00	5.0 E-02	1.0 E-01	2.5 E-01	1.0 E-01	2.0 E-01
(Particulate)	5.0 E-02	---	1.0 E+00	5.0 E-02	1.0 E-01	1.0 E-03	1.0 E-01	2.0 E-01
(Organic)	(6)	---	1.0 E+00	1.0 E+00	1.0 E+00	1.0 E+00	1.0 E-01	1.0 E+00
OTHER SPECIES								
Cesium	3.0 E-01	7.0 E-01	1.0 E+00	5.0 E-02	1.0 E-01	1.0 E-03	1.0 E-01	2.0 E-01
Tellurium	---	1.0 E+00	1.0 E+00	1.0 E+00	1.0 E-01	1.0 E-03	1.0 E-01	2.0 E-01
Ruthenium	---	5.0 E-02	1.0 E+00	1.0 E+00	1.0 E-01	1.0 E-03	1.0 E-01	2.0 E-01
Other	---	1.0 E-02	---	5.0 E-02	1.0 E-01	1.0 E-03	1.0 E-01	2.0 E-01

Notes:
1. Normalized fission product escape fraction from the fuel, given fuel damage state Z.
2. Elemental iodine and particulates released prior to pressure-tube penetration are scrubbed for transient events. For seismic events no credit for retention in the dump tank is taken.
3. Late failure modes (which do not involve substantial driving forces due to hydrogen and steam generation) are principally driven by the displacement of the confinement gases by the fog-spray water.
4. Early isolation failures of the confinement would result in higher escape fractions than late loss of integrity due to natural removal processes. The listed factors are for short residence times.
5. 5% of total iodine is assumed released during the later phases of fuel damage following pressure-tube penetration. Late release is taken in the ratio 0.95 for elemental and 0.05 for particulate.
6. 1.6% of the iodine entering confinement and not scrubbed by sprays, is assumed to be converted to organic iodide.

- Distribution of Radioiodine among Chemical and Physical Forms. The chemical form of radioiodine released from fuel during accidents has been a matter of extensive study for light-water reactors. Generally, it has been concluded that gas phase reactions at the high temperatures in the core region rapidly convert most of the released radioiodine to CsI if it is initially released as I_2.(8) For metallic fuels, which melt at significantly lower temperatures, the form of iodine released and transported in the reactor system has not been investigated to the same extent as for light-water reactor fuels. It is assumed in this study 95% of the iodine is released and transported as elemental iodine (I_2) and 5% as particulate. Penetration factor estimates for these two forms of iodine (I_2 and particulates) are the same except for the greater removal of particulates by filters (see Table 8-3). Thus, the assumption that a large fraction of the radioiodine is in the elemental form, is conservative.

- Formation of Organic Iodides. With respect to formation of organic iodides, which are more penetrating than either I_2 or particulate iodine, it is assumed 1.6% of the total iodine reaching the confinement and not scrubbed by sprays is converted to methyl iodide (CH_3I).(4) More recent evaluations by the NRC staff as part of fission product source term reassessments indicate the maximum conversion to methyl iodide to be in the range of 0.02% to 0.04%.(10) A conservative methyl iodide conversion fraction of 0.5% was used in the recently issued NUREG-1150.(11) Values based on commercial light-water reactor technology appear to be appropriate for N Reactor, because for the majority of events, there is no opportunity for interaction between the released radioiodine and very hot graphite. Thus, the value used in this study is believed to be conservative.

o Pressure Tube and Piping Penetration Factors. With two exceptions, no credit is taken in this study for fission product retention in the pressure tubes and piping during transport from the fuel region through the pressure tubes and piping to the break location or, alternatively, to the confinement via the V-4 valves and the dump tank. For most sequences, water will not be available at the time of radionuclide release to generate steam or hydrogen for radionuclide transport; thus, flow rates will be low. The assumption of no radionuclide retention is regarded as very conservative for N Reactor.

One of the exceptions, with regard to pressure-tube retention, is a slowly developing transient sequence where penetration factors of 0.9 and 0.5 are applied for radioiodine and other volatile fission products, respectively. The other exception is the accident sequence involving failure of the interfacing valves in the ECCS system when there is an ECCS demand at high pressure. Such sequences can lead to pressurization of the low-pressure portion of the ECCS system and probable failure at the ECCS high-lift diesel pump expansion joints. Transported fission products have a flow path of substantial length through the cooler portions of the ECCS system (see the discussion on release category N-3 in Section 8.3). Based on the Light-Water

Aerosol Containment Experiments conducted at the Hanford Engineering Development Laboratory (specifically for such sequences), a penetration factor of 0.3 is applied for both radioiodine and other volatile fission products.[12]

- <u>Dump Tank Penetration Factors</u>. For transient events and for seismic events without a concurrent pipe rupture, the vent path for released fission products is from the pressure tubes, to the reactor system, through the V-4 valves, through the dump tank and associated piping, and back to the confinement. Based on tests of fission product retention in water pools, significant scrubbing of radioiodine and aerosols is expected.[8] To account for potential dump tank bypass during transient events, an equivalent retention factor of 95% (or a penetration factor of 0.05) was estimated for iodine and cesium. No retention is assumed for other particulates, ruthenium, and tellurium. Other nonvolatile fission products are assumed to be released after pressure tube failure and therefore bypass the dump tank. For seismic initiated events, no dump tank retention is assumed and a penetration factor of 1.0 is utilized.

- <u>Fog Spray Removal</u>. Spray removal tests and spray technology evaluations are summarized in ORNL-TM-2412.[13] Spray decontamination factors of 100 or more were generally observed for radioiodine by alkaline sprays. Decontamination factors are the reciprocal of the penetration factors. Measured decontamination factors fall to the range of 10 to 100 for radioiodine by sprays without such additives. Decontamination factors in the range of 100 are generally observed for particulate materials. For the purpose of this study, a conservative decontamination factor of 10 is assumed for radioiodine and particulate material (penetration factor of 0.1). For noble gases and methyl iodide, no credit is taken for radionuclide removal from the confinement atmosphere by sprays.

- <u>Failure to Isolate Confinement</u>. For sequences involving failure to isolate confinement, a penetration factor of 0.2 is assumed for all species, except noble gases and methyl iodide. The 80% retention is based on the low flow rates of material from the fuel melt region (fission products, steam, noncondensible gases). This material vents to the confinement and is the primary driving force for releases from a confinement with a large overall volume of approximately 3,000,000 ft^3. Thus, low venting rates and substantial deposition in confinement are expected, even for failure to isolate sequences. Confinement ventilation fans are assumed to be deenergized. This is reasonable because continued air flow through exhaust fans would require a fan to remain energized and two redundant dampers in the fan inlet to fail to close. Note the same penetration factor is applied for confinement failure with large fission product releases.

- <u>Confinement Failure from Energetic Events</u>. Postulated energetic events in the confinement include hydrogen deflagration or explosion and steam explosion. As discussed in Section 6.0, these events are regarded generally as having low probability, even though a relatively high probability is conservatively assumed for the seismic event with failure of reactor trip. Energetic failures would occur late in the accident sequence (if they occur at all), after substantial time for fission product deposition. These energetic events have the potential for releasing much of the inventory in the confinement atmosphere, were they to occur, but deposition would have reduced the available inventory to a low value. Therefore, a penetration factor of 10% is applied for such events.

- Delayed Confinement Failure. For delayed confinement failure modes, there is little physical driving force to move fission products from the confinement to the environment. For these sequences, it is assumed 10% of all fission products escape largely as a result of displacement of the confinement atmosphere by fog spray water accumulation.

- Filtration and Venting. Retention of fission products by filter media is accounted for (except for noble gases which are not retained) for sequences where the filter system remains functional. The thickness of the charcoal beds in the filter systems is one inch. As a result, the retention of I_2 and methyl iodide is limited. Based on extrapolation of conservative guidance in NRC Regulatory Guide 1.52[14] to a 1-in. thick charcoal bed, penetration factors of 0.25 for elemental iodine and 1.0 for methyl iodide are used.

8.2.4 Results

The overall escape fractions of fission products (to the environment) for each confinement release mode described in Table 8-2 are estimated based on the assumptions, guidelines, and penetration factors discussed above. These escape fractions are summarized in Tables 8-4 and 8-5 for LOCAs and transient events, respectively.

In sequences such as fast-developing transients without scram, there is the potential for early pressure-tube penetration, in which case early fission product escape would largely bypass the dump tank. This type of scenario could also occur in sequences where substantial amounts of steam are available for fuel oxidation. For this type of sequence, it is assumed that 10% of the iodine and 70% of cesium would be released subsequent to pressure tube penetration and bypass the dump tank. The environmental release fractions are listed in Table 8-6 for such sequences.

The source terms presented in the tables are those involving full core damage (fuel damage state Z). For lesser fuel damage states (e.g., Y or W), fission product environmental release magnitudes are scaled down by the lesser fuel damage fraction to reflect the smaller amounts of fission products released from the fuel.

As a first step in arriving at a reduced set of source terms for consequence calculations, the risk-dominant confinement release mode(s) were identified for each of the plant accident sequences listed in Table 7-1. Source terms were then matched to each combination of plant accident sequence and risk-dominant confinement release mode based on

Table 8-4. Environmental Release Fractions For Loss of Coolant Accident (Fuel Damage State Z).

Release Mode (CET End-State)	Fission Product Escape Fractions Kr-Xe	Iodine	CH_3I	Cesium	Ruthenium	Tellurium	Others
A1 (1,7)	1.0 E+00	2.5 E-04	1.6 E-05	1.0 E-05	5.0 E-07	1.0 E-05	1.0 E-07
LUR1 (2)	1.0 E+00	7.0 E-03	1.3 E-04	4.2 E-03	4.6 E-04	9.2 E-03	9.2 E-05
EUR1 (3)	1.0 E+00	8.1 E-02	1.3 E-03	8.1 E-02	4.1 E-03	8.1 E-02	8.1 E-04
EUR2 (4)	1.0 E+00	1.8 E-02	3.0 E-04	1.8 E-02	9.0 E-04	1.8 E-02	1.8 E-04
EUR3 (5)	1.0 E+00	2.0 E-02	3.3 E-04	2.0 E-02	1.0 E-03	2.0 E-02	2.0 E-04
EUR4 (6)	1.0 E+00	9.0 E-02	1.4 E-03	9.0 E-02	4.5 E-03	9.0 E-02	9.0 E-04
LUR2 (8)	1.0 E+00	7.0 E-02	1.3 E-03	4.2 E-02	4.6 E-03	9.2 E-02	9.2 E-04
EUR5 (9)	1.0 E+00	8.1 E-01	1.3 E-02	8.1 E-01	4.1 E-02	8.1 E-01	8.1 E-03
EUR6 (10)	1.0 E+00	1.8 E-01	3.0 E-03	1.8 E-01	9.0 E-03	1.8 E-01	1.8 E-03
EUR7 (11)	1.0 E+00	2.0 E-01	3.2 E-03	2.0 E-01	1.0 E-02	2.0 E-01	2.0 E-03
EUR8 (12)	1.0 E+00	9.0 E-01	1.4 E-02	9.0 E-01	4.5 E-02	9.0 E-01	9.0 E-03
EIR1 (13)	1.0 E+00	1.8 E-02	2.9 E-04	1.8 E-02	9.0 E-04	1.8 E-02	1.8 E-04
EIR2 (14)	1.0 E+00	1.8 E-01	2.9 E-03	1.8 E-01	9.0 E-03	1.8 E-01	1.8 E-03

Note: CET End-States = Confinement Event Tree Sequence Number on CET diagrams (See Section 7.0, Figure 7.1).

A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release

Table applies to Sequences 1, 2, 3, 5, and 8 (see Table 7-1).

Table 8-5. Environmental Release Fractions for Transients (Fuel Damage State Z).

Release Mode (CET End-State)	Fission Product Escape Fractions						
	Kr-Xe	Iodine	CH_3I	Cesium	Ruthenium	Tellurium	Others
A1 (1,7)	1.0 E+00	1.1 E-05	7.0 E-07	2.5 E-07	5.0 E-07	1.0 E-05	1.0 E-07
LUR1 (2)	1.0 E+00	3.7 E-04	5.8 E-06	1.9 E-04	4.6 E-04	9.2 E-03	9.2 E-05
EUR1 (3)	1.0 E+00	3.6 E-03	5.8 E-05	2.0 E-03	4.1 E-03	8.1 E-02	8.1 E-04
EUR2 (4)	1.0 E+00	8.0 E-04	1.3 E-05	4.5 E-04	9.0 E-04	1.8 E-02	1.8 E-04
EUR3 (5)	1.0 E+00	8.8 E-04	1.4 E-05	5.0 E-04	1.0 E-03	2.0 E-02	2.0 E-04
EUR4 (6)	1.0 E+00	4.0 E-03	6.3 E-05	2.3 E-03	4.5 E-03	9.0 E-02	9.0 E-04
LUR2 (8)	1.0 E+00	3.7 E-03	5.8 E-05	1.9 E-03	4.6 E-03	9.2 E-02	9.2 E-04
EUR5 (9)	1.0 E+00	3.6 E-02	5.8 E-04	2.0 E-02	4.1 E-02	8.1 E-01	8.1 E-03
EUR6 (10)	1.0 E+00	8.0 E-03	1.3 E-04	4.5 E-03	9.0 E-03	1.8 E-01	1.8 E-03
EUR7 (11)	1.0 E+00	8.8 E-03	1.4 E-04	5.0 E-03	1.0 E-02	2.0 E-01	2.0 E-03
EUR8 (12)	1.0 E+00	4.0 E-02	6.3 E-04	2.3 E-02	4.5 E-02	9.0 E-01	9.0 E-03
EIR1 (13)	1.0 E+00	7.9 E-04	1.3 E-05	4.5 E-04	9.0 E-04	1.8 E-02	1.8 E-04
EIR2 (14)	1.0 E+00	7.9 E-03	1.3 E-04	4.5 E-03	9.0 E-03	1.8 E-01	1.8 E-03

Note: CET End-States = Confinement Event Tree Sequence Number on CET diagrams (See Section 7.0, Figure 7.1).

A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release

Table applies to Sequences 6, 7, 10, 11, and 12 (see Table 7-1).

Table 8-6. Environmental Release Fractions for Rapid Transients (Fuel Damage State Z).

Release Mode (CET End-State)	Fission Product Escape Fractions						
	Kr-Xe	Iodine	CH_3I	Cesium	Ruthenium	Tellurium	Others
A1 (1,7)	1.0 E+00	3.6 E-05	2.3 E-06	7.2 E-06	5.0 E-07	1.0 E-05	1.0 E-07
LUR1 (2)	1.0 E+00	9.0 E-04	1.9 E-05	3.0 E-03	4.6 E-04	9.2 E-03	9.2 E-05
EUR1 (3)	1.0 E+00	1.2 E-02	1.9 E-04	5.8 E-02	4.1 E-03	8.1 E-02	8.1 E-04
EUR2 (4)	1.0 E+00	2.6 E-03	4.4 E-05	1.3 E-02	9.0 E-04	1.8 E-02	1.8 E-04
EUR3 (5)	1.0 E+00	2.9 E-03	4.6 E-05	1.4 E-02	1.0 E-03	2.0 E-02	2.0 E-04
EUR4 (6)	1.0 E+00	1.3 E-02	2.1 E-04	6.4 E-02	4.5 E-03	9.0 E-02	9.0 E-04
LUR2 (8)	1.0 E+00	9.0 E-03	1.9 E-04	3.0 E-02	4.6 E-03	9.2 E-02	9.2 E-04
EUR5 (9)	1.0 E+00	1.2 E-01	1.9 E-03	5.8 E-01	4.1 E-02	8.1 E-01	8.1 E-03
EUR6 (10)	1.0 E+00	2.6 E-02	4.4 E-04	1.3 E-01	9.0 E-03	1.8 E-01	1.8 E-03
EUR7 (11)	1.0 E+00	2.9 E-02	4.6 E-04	1.4 E-01	1.0 E-02	2.0 E-01	2.0 E-03
EUR8 (12)	1.0 E+00	1.3 E-01	2.1 E-03	6.4 E-01	4.5 E-02	9.0 E-01	9.0 E-03
EIR1 (13)	1.0 E+00	2.6 E-03	4.2 E-05	1.3 E-02	9.0 E-04	1.8 E-02	1.8 E-04
EIR2 (14)	1.0 E+00	2.6 E-02	4.2 E-04	1.3 E-01	9.0 E-03	1.8 E-01	1.8 E-03

Note: CET End-States = Confinement Event Tree Sequence Number on CET diagrams (See Section 7.0, Figure 7.1).

A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release

Table applies to Sequence 4 (see Table 7-1).

Table 8-4, 8-5, or 8-6, as appropriate. These combinations and the source term for each combination are listed in Table 8-7. The source terms in Table 8-4 are applied to sequences involving an initial breach of the reactor coolant system (LOCA events) and to seismically-initiated transients (for which no credit is taken for fission product removal in the dump tank). Table 8-5 is applied to transient initiators, except for seismically initiated transients and transients without reactor trip. The principal difference between Tables 8-4 and 8-5 is credit is taken in Table 8-5 for fission product removal in the dump tank. This credit is taken because fuel damage and release of fission products are expected to occur during most transient sequences with the reactor coolant system intact. Table 8-6 is applied to non-seismically initiated transients without reactor trip. Table 8-6 takes limited credit for fission product removal in the dump tank, as is discussed earlier in this section.

Notations in the first column of Table 8-7 represent the plant accident sequence number from Section 7.0 (Table 7-1) and the confinement release mode designator from Table 8-2. Table 8-2 keys the confinement release modes to the confinement event tree end states.

8.3 Source Term Categories for Consequence Calculations

The source term bins were reduced further for the purpose of consequence calculations. Generally, source term categories expected to produce consequences (or having source term magnitude) within approximately a factor of 3 of each other (1/2 decade) were combined. The initial reduced set of source term categories (Table 8-7) was collapsed to eight bins for consequence calculations. These eight bins or source term categories are presented in Table 8-8. Also indicated in Table 8-8 are the plant sequence/release mode combinations (from Table 8-7), which were grouped within each of the final collapsed release categories. A summary discussion of the types of sequences included in each of the collapsed release categories follows.

o Release Category N-1. Release category N-1 contains seismically-initiated events that lead to full fuel damage (Z state). Sequences included are:

1. Those that result in structural failure of 181-N/182-N with consequential loss of ECCS, GSCS, and fog sprays (Sequence number 1 from Table 7-1)

2. Seismic events that lead to loss of pressure tubes, GSCS tubes, and the capability for effective ECCS and GSCS backup injection (Sequence number 2 from Table 7-1) and loss of fog sprays due to independent system failure.

For these seismically initiated sequences, it is assumed that the dump tank and the filters in the exhaust path from confinement are not available. Radionuclides are vented to confinement without the dump tank scrubbing. For the higher frequency seismically initiated events with fog spray unavailability (Case 1/EUR7), there is a reasonable likelihood of venting to confinement through the dump tank. However, a conservative

Table 8-7. Source Terms for the Dominant Release Modes

Plant Sequence/ Release Mode	Time of Release*	Duration of Release*	Radionuclide Escape Fractions to Environment (Fraction of Inventory) Kr-Xe	Iodine	CH_3I	Cs	Ru	Te	Others
3/EUR1	5 min	1.0 E+00	1.0 E+00	8.1 E-02	1.3 E-03	8.1 E-02	4.1 E-03	8.1 E-02	8.1 E-04
3/EUR3	5 min	1.0 E+00	1.0 E+00	2.0 E-02	3.3 E-04	2.0 E-02	1.0 E-03	2.0 E-02	2.0 E-04
4/EUR1	5 min	1.0 E+00	1.0 E+00	1.2 E-02	1.9 E-04	5.8 E-02	4.1 E-03	8.1 E-02	8.1 E-04
4/EUR3	5 min	1.0 E+00	1.0 E+00	2.9 E-03	4.6 E-05	1.4 E-02	1.0 E-03	2.0 E-02	2.2 E-04
1/EUR7	2.0 E+00	2.4 E+01	1.0 E+00	2.0 E-01	3.2 E-03	2.0 E-01	1.0 E-02	2.0 E-01	2.0 E-03
2/EUR3	2.0 E+00	6.0 E+00	1.0 E+00	2.0 E-02	3.3 E-04	2.0 E-02	1.0 E-03	2.0 E-02	2.0 E-04
2/EUR7	2.0 E+00	2.4 E+01	1.0 E+00	2.0 E-01	3.2 E-03	2.0 E-01	1.0 E-02	2.0 E-01	2.0 E-03
9(V-SEQ)	5 min	4.0 E+00	5.0 E-01	6.0 E-02	9.5 E-04	3.8 E-02	3.4 E-03	6.8 E-02	6.8 E-04
6,7/EUR2	5.0 E-01	8.0 E+00	5.0 E-01	4.0 E-04	6.5 E-06	2.3 E-04	4.5 E-04	9.0 E-03	9.0 E-05
6,7/EUR3	5.0 E-01	8.0 E+00	5.0 E-01	4.4 E-04	7.0 E-06	2.5 E-04	5.0 E-04	1.0 E-02	1.0 E-04
6,7/EIR1	5.0 E-01	8.0 E+00	5.0 E-01	3.9 E-04	6.5 E-06	2.3 E-04	4.5 E-04	9.0 E-03	9.0 E-05
6,7/LUR2	5.0 E-01	2.4 E+01	5.0 E-01	2.0 E-03	2.9 E-05	9.5 E-04	2.3 E-03	4.6 E-02	4.6 E-04
8/EUR2	1.0 E+00	8.0 E+00	5.0 E-01	9.0 E-03	1.5 E-04	9.0 E-03	4.5 E-04	9.0 E-03	9.0 E-05
8/EUR3	1.0 E+00	8.0 E+00	5.0 E-01	1.0 E-02	1.6 E-04	1.0 E-02	5.0 E-04	1.0 E-02	1.0 E-04
8/EIR1	1.0 E+00	8.0 E+00	5.0 E-01	9.0 E-03	1.4 E-04	9.0 E-03	4.5 E-04	9.0 E-03	9.0 E-05
8/LUR2	2.0 E+00	2.4 E+01	5.0 E-01	3.5 E-02	6.0 E-04	2.1 E-02	2.3 E-03	4.6 E-02	4.6 E-04
5/EUR3	2.0 E+00	6.0 E+00	5.0 E-01	1.0 E-02	1.7 E-04	1.0 E-02	5.0 E-04	1.0 E-02	1.0 E-04
10,11/EUR2	5.0 E-01	8.0 E+00	2.0 E-02	1.6 E-05	2.6 E-07	9.0 E-06	1.8 E-05	3.6 E-04	3.6 E-06
10,11/EUR3	5.0 E-01	8.0 E+00	2.0 E-02	1.8 E-05	2.8 E-07	1.0 E-05	2.0 E-05	4.0 E-04	4.0 E-06
10,11/EIR1	5.0 E-01	8.0 E+00	2.0 E-02	1.6 E-05	2.6 E-07	9.0 E-06	1.8 E-05	3.6 E-04	3.6 E-06
10,11/LUR2	5.0 E-01	2.4 E+01	2.0 E-02	7.4 E-05	1.2 E-06	3.8 E-05	9.2 E-05	1.8 E-03	1.8 E-05
12/EIR2	5 min	2.4 E+01	1.0 E-03	7.9 E-06	1.3 E-07	4.5 E-06	9.0 E-06	1.8 E-04	1.8 E-06
6,7/A1	1.0 E+00	5.0 E-01	5.0 E-01	5.5 E-06	3.5 E-07	1.3 E-07	2.5 E-07	5.0 E-06	5.0 E-08
8/A1	1.0 E+00	5.0 E-01	5.0 E-01	1.3 E-04	8.0 E-06	5.0 E-06	2.5 E-07	5.0 E-06	5.0 E-08
10,11/A1	5.0 E-01	2.0 E-02	2.0 E-02	2.2 E-07	1.4 E-08	5.0 E-09	1.0 E-08	2.0 E-07	2.0 E-09

NOTES: Source terms identified take account of the dominant sequence progression insofar as it affects the potential release paths from the fuel to the confinement system to the environment. For example, if the event sequence is a LOCA, the dump tank is not considered a viable removal mechanism. For a very fast developing transient event scenario, the pressure-tube integrity is not considered to

* Times in hours unless otherwise indicated.

A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release

Table 8-8. Collapsed Source Term Categories for Environmental Release Used for Consequence Calculations.

Release Category	Plant Sequence/ Confinement Release Mode	Release Frequency (/yr)	Time of Release	Duration of Release (h)	Fission Product Escape Fractions (Fraction of Inventory)					
					Kr-Xe	Iodine	Cs	Ru	Te	Others
N-1	1/EUR7 2/EUR7	3.3 E-05	2.0 h	24	1.0 E+00	2.0 E-01	2.0 E-01	1.0 E-02	2.0 E-01	2.0 E-03
N-2	8/LUR2	2.5 E-10	2.0 h	24	5.0 E-01	3.5 E-02	2.1 E-02	2.3 E-03	4.6 E-02	4.6 E-01
N-3	9/V Sequence	1.0 E-09	5 min	4	5.0 E-01	6.1 E-02	3.8 E-02	3.4 E-03	6.8 E-02	6.8 E-04
N-4	6/LUR2 7/LUR2	1.8 E-9	30 min	24	5.0 E-01	2.0 E-03	9.5 E-03	2.3 E-03	4.6 E-02	4.6 E-04
N-5	3/EUR1 4/EUR1 3/EUR3 4/EUR3 2/EUR3	3.5 E-6	5 min	1	1.0 E+00	2.0 E-02	2.0 E-02	1.0 E-03	2.0 E-02	2.0 E-04
N-6	8/EIR1 8/EUR2 8/EUR3 6/EIR1 7/EIR2 6/EUR2 7/EUR2 6/EUR3 7/EUR3 5/EUR3	7.0 E-6	1.0 h	8	5.0 E-01	9.0 E-03	9.0 E-03	9.0 E-04	9.0 E-03	9.0 E-05

Table 8-8. Collapsed Source Term Categories for Environmental Release Used for Consequence Calculations. (Cont.)

Release Category	Plant Sequence/ Confinement Release Mode	Release Frequency (/yr)	Time of Release	Duration of Release (h)	Fission Product Escape Fractions (Fraction of Inventory) Kr-Xe	Iodine	Cs	Ru	Te	Others
N-7	10/EIR1 11/EUR1 10/EUR2 11/EUR2 10/EUR3 11/EUR3 10/LUR2 11/LUR2 6/A1 7/A1 8/A1	1.7 E-05	30 min	24	2.0 E-02	7.4 E-05	3.7 E-05	9.2 E-05	1.8 E-03	1.8 E-05
N-8	12/EIR2 10/A1 11/A1	7.2 E-05	5 min	24	1.0 E-03	8.0 E-06	4.5 E-06	9.0 E-06	1.8 E-04	1.8 E-06
Bounding Case	Bounding	4 E-05	1 h	10	1.0	1.0	1.0	5.0 E-02	1.0	1.0 E-02

A1 = Confinement Success
EUR = Early Unfiltered Release
LUR = Late Unfiltered Release
EIR = Early Isolation Release

approach has been adopted here, in which it is assumed that radionuclides are released directly to the confinement.

- Release Category N-2. This release category is initiated by a large LOCA and fuel damage results from loss of ECCS. Release from confinement results from delayed burning of the fission product removal filters for sequences where fog sprays did not function thereby loading the filters. Burning is a result of loss of filter sprays.

- Release Category N-3. This category is for sequences where the check valves, which are the boundary between the reactor coolant system and the ECCS system, fail open. When the ECCS inlet valves (V-3s) open, the supply line is overpressurized and a LOCA occurs with blowdown through the ECCS high lift diesel pump expansion joints in 182-N. Following fuel damage, there are competing flow paths through V-4s to the dump tank back to the confinement and through the break in the ECCS system. For the purposes of the analysis, 50% of the released fission products are assumed to bypass the confinement through the break in the ECCS system (the longer of the two flow paths). The sequence is parallel to the PWR "V" sequence although different in detail for N Reactor systems.

- Release Category N-4. This release category includes reactor trip events in which post-trip core cooling demands are not met as a result of failure of ECCS, and either high-pressure injection or primary-to-secondary cooling. Such failures can lead to damage of 30% of the core potentially releasing 50% of the fission product inventory. Coupled with the plant damage states are loss of fog sprays and failure of the filters by burning as result of loss of filter sprays.

- Release Category N-5. These sequences are initiated by seismic events which lead either to failure of the pressure tubes as a result of biological shield uplift or failure of the reactor trip system. Non-seismically initiated transients with failure of reactor trip are also included. Such events lead to release of radionuclides to confinement. For these sequences, fog spray is available. Two confinement release conditions are considered:

    1. Release due to confinement overpressure failure

    2. Release due to filter bypass.

- Release Category N-6. This release category represents a group of sequences characterized by loss of the core cooling function concurrent with failure of the confinement isolation function. Fog sprays are operational for these sequences, resulting in scrubbing of 90% of the fission products, except for noble gases, before release to the environment. Specific sequences included from Table 8-7 are:

    1. Reactor trip events with failure of either high pressure injection or primary to secondary cooling

and ECCS; in addition, filters are not available in the confinement vent path to the stack

2. Same plant failure states as (1), with failure of confinement steam vent valves to close after weatherheads initially open

3. Same plant failure states as (1), with early confinement isolation failure

4. Seismic events with failure of ECCS and early confinement isolation failure

5. Large LOCA with ECCS failure; in addition, filters are not available in the confinement exhaust path

6. Large LOCA with ECCS failure; in addition, confinement steam vent valves do not reclose after initially opening

7. Large LOCA with ECCS failure; in addition, there is an early confinement isolation failure.

o Release Category N-7. This release category represents a group of slowly developing fuel damage sequences. Confinement failure modes generally involve loss of the isolation function, although one included sequence has a late unfiltered release. Most of the events are slowly developing transient events so credit is taken for fission product removal by the dump tank. Fog sprays are functional for all sequences.

This release category also includes sequences with fuel damage state Y for which there are no confinement system failures. Specific sequences included in this release category from Table 8-7 are:

1. Reactor trip events with loss of cooling to a single riser and failure of the filters in the confinement exhaust path to the stack

2. Reactor trip events with loss of cooling to a single riser with failure of the confinement steam vent valves to close after weatherheads initially open

3. Reactor trip events with loss of cooling to a single riser and early loss of the confinement isolation function

4. Reactor trip events with loss of cooling to a single riser; releases to the environment result from late failure of the filter due to burning

5. Sequences with fuel damage state Y and no confinement system failures; noble gases can escape from confinement to the environment.

- Release Category N-8. Release category N-8 involves melting in a single tube with release of fission products from the fuel prior to pressure-tube failure. Failure of fog sprays results in lifting of the weatherheads and a release to the environment. This release category also includes sequences with the more severe fuel damage state W but where all confinement systems work and release to the environment is limited to noble gases.

- Bounding Case. For the purpose of bounding the risk from N Reactor operation, a bounding source term category is defined in which it is assumed that all of the volatile radionuclide inventory of the core (100% of noble gases, iodine, cesium, and tellurium) is released directly to the environment. Releases to the environment for less volatile fission products are assumed to be 1% of the core inventory. The assumption of no confinement deposition for all species is believed to adequately compensate for possibly greater releases of the less volatile species.

8.4 References

1. Gumprecht, R. O., 1968, Mathematical Basis of Computer Code RIBD, DUN-4136.

2. Marr, D. R., 1975, A User's Manual for Computer Code RIBD-II A Fission Product Inventory Code, HEDL-TME 75-26, Westinghouse Hanford Company, Richland, Washington.

3. Roblyer, S. P., 1978, Accident Confinement Response and Dose Calculations of Single Pressure Tube Events for NUSAR, UNI-667, UNC Nuclear Industries, Inc., Richland, Washington.

4. Roblyer, S. P., 1978, Confinement Response and Dose Calculations for the Hypothetical Accident for N Reactor, UNI-669, UNC Nuclear Industries, Inc., Richland, Washington.

5. Birney, K. R., 1987, Fission Product Release Considerations, (memorandum to W. J. Quapp, UNC Nuclear Industries, Inc., Richland, Washington, February 27, 1987).

6. Wilcox, A. D., 1987, Confinement Filter Burden From Cold Leg Rupture Accident, (memorandum to G. R. Smith, UNC Nuclear Industries, Inc., Richland, Washington, May 28, 1987).

7. Roblyer, S. P., 1986, Filter Iodine Loading, Hypothetical Accident, (memorandum to G. L. Smith, UNC Nuclear Industries, Inc., Richland, Washington, June 16, 1986).

8. Silberberg, M. et al., 1985, Reassessment of the Technical Bases for Estimating Source Terms, NUREG-0956, U.S. Nuclear Regulatory Commission, Washington, D.C.

9. Parker, G. W., et al., 1967, Out of Pile Studies of Fission Product Release from Overheated Fuel at ORNL, 1955-1965, ORNL 3981, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

10. NRC, 1981, Technical Basis for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, U.S. Nuclear Regulatory Commission, Washington, D.C.

11. NRC, 1987, Reactor Risk Reference Document, NUREG-1150, draft report for comment, U.S. Nuclear Regulatory Commission, Washington, D.C.

12. Bloom, G. R., R. K. Hilliard, J. D. McCormack, L. D. Muhlestein, to be published, Aerosol Behavior Tests Under LWR Containment Bypass Conditions, CSTF Tests CB-1, CB-2 and CB-3, Westinghouse Hanford Company, Richland, Washington.

13. Row, T. H., et al., 1969, Design Consideration of Reactor Containment Spray Systems, ORNL-TM-2412, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

14. NRC, 1978, Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System, Air Filtration, and Adsorption Units of Light Water Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C.

9.0 CONSEQUENCE ANALYSIS

For this study, consequence calculations were performed for releases to air pathways. Consequence calculations were not performed for release to liquid pathways because the expected frequencies of such releases are low, as are the expected consequences (see the discussion in Section 9.3).

9.1 Release Categories and Category Frequencies

A collapsed set of fission product release categories for use in consequence calculations is developed in Section 8.3. These release categories are summarized in Table 9-1, along with the supporting information needed for input to the consequence calculations (with the CRAC2 Code).

9.2 Methodology for Consequence Calculations

Potential health consequences from N Reactor fission product releases were estimated using the NRC computer program CRAC2. This program models radionuclide release, downwind transport, and downwind deposition in estimating radiological health effects. The version of CRAC2 used for the present analysis is that distributed by Sandia National Laboratory[1,2] as updated through August 1986. The main inputs to the CRAC2 program include source term, meteorological data, economic data, population distribution, land usage, and evacuation parameters. Other data are also required, including health effects response functions; however, these data sets are not site-dependent and default values supplied with the CRAC2 computer program package were used. The following paragraphs describe the inputs to CRAC2 not described elsewhere.

Radiation exposure calculations were performed for offsite populations; exposure to onsite personnel is not included in the exposure estimates.

9.2.1 Atmospheric Dispersion

The atmospheric dispersion model used in CRAC2 for the present analysis utilizes a meteorological data file containing hourly observations of meteorological conditions in sequential order for a period of 1 yr. The present analysis used data from the Hanford Meteorological Station for 1986, as shown in Table 9-2.

The atmospheric dispersion model used in CRAC2 is based on the straight-line Gaussian dispersion model with a three-sigma top-hat approximation. This approximation, in effect, averages the airborne material over a distance of three sigma y (horizontal cross-wind standard deviation of concentration) about the plume centerline. The plume is depleted as it travels downwind, with a deposition velocity of 0.01 m/sec applied to all radionuclides except noble gases (which do not deposit). The CRAC2 computer program applies a weather sequence sampling scheme to the

Table 9-1. Summary of Source Term Release Categories for Consequence Calculations.

Source Term Release Category	Release Frequency (yr)	Release Time	Release Duration (h)	Release Elevation G=Ground S=Stack	Plant State/ Confinement Release Mode*	Fission Product Escape Fractions					
						Noble Gases	Iodine	Cs	Ru	Te	Others
N-1	3.3 E- 5	2 h	24	G	1,2/EUR7	1.0	2 E-01	2 E-01	1 E-02	2 E-01	2.0 E-03
N-2	2.5 E-10	2 h	24	S	8/LUR2	5 E-01	3.5 E-02	2.1 E-02	2.3 E-03	4.6 E-02	4.6 E-04
N-3	1.0 E-09	5 min	4	G	9/V Sequence	5 E-01	6.1 E-02	3.8 E-02	3.4 E-03	6.8 E-02	6.8 E-04
N-4	1.8 E-09	30 min	24	G/S	6,7/LUR2	5 E-01	2 E-03	9.5 E-03	2.3 E-03	4.6 E-02	4.6 E-04
N-5	3.5 E-06	5 min	8	G	3,4/EUR1 3,4/EUR3 2/EUR3	1.0	2 E-02	2 E-02	1 E-03	2 E-02	2 E-04
N-6	7.0 E-06	1 h	8	G/S	6,7/EUR2 6,7/EUR3 6,7/EIR1 5/EUR3 8/EIR1 8/EUR2 8/EUR3	5 E-01	9 E-03	9 E-03	9 E-04	9 E-03	9 E-05
N-7	1.7 E-05	30 min	24	G	10,11/EIR1 10,11/EUR2 10,11/EUR3 10,11/LUR2 6,7,8/A1	2 E-02	7.4 E-05	3.7 E-05	9.2 E-05	1.8 E-03	1.8 E-05
N-8	7.2 E-05	5 min	24	G	12/EIR2 10,11/A1	1 E-03	8.0 E-06	4.5 E-06	9.0 E-06	1.8 E-04	1.8 E-06
BOUNDING CASE	4 E-05(1)	1 h	10	G	Bounding	1.0	1.0	1.0	5.0 E-02	1.0	1.0 E-02

*From Table 8-7

(1) Bounding case frequency, equivalent to the calculated frequency of 100% fuel damage.

Table 9-2. Summary of Types of Hanford Meteorological Data Provided for CRAC2 Calculations.

Sector	Sector Wind Frequency (%)	Direction
1	2.9	≥348.75° to <11.25° (where 0° is true N)
2	3.6	≥11.25° to <33.75°
3	7.4	≥33.75° to <56.25°
4	8.45	≥56.25° to <78.75°
5	14.9	≥78.75° to <101.25°
6	15.6	≥101.25° to <123.75°
7	6.8	≥123.75° to <146.25°
8	4.1	≥146.25° to <168.75°
9	3.8	≥168.75° to <191.25°
10	6.4	≥191.25° to <213.75°
11	4.2	≥213.75° to <236.25°
12	3.9	≥236.25° to <258.75°
13	4.7	≥258.75° to <281.25°
14	4.9	≥281.25° to <303.75°
15	4.9	≥303.75° to <326.25°
16	3.2	≥326.25° to <348.75°

Other information provided for calculations:

1. Date and hour for data.

2. Average wind direction and wind speed for the hour measured at the 50-ft level of the Hanford Meteorology Station (HMS) tower. Wind speeds are reported in units of meters/second.

3. Atmospheric stability category for the hour. (Using the Environmental Protection Agency STAR classification scheme. Wind distribution of Pasguill stability classes STAR Program National Climatic Center, Ashville, NC, based on article "Estimation of the Dispersion of Windborne Material," Meterology Magazine 90:33-49.)

4. Hourly precipitation (in units of 100ths of in./hr).

meteorological database to ensure that the worst case, low frequency events are included in the calculations.

9.2.2 Population Distribution

The CRAC2 computer program requires the population data be provided on a radial grid defined by 16 compass directions with up to 34 concentric radii. The analysis was performed with spacing between radii varying from 0.5 mi to 150 mi, with the larger spacings at greater distances from the site. Population data are shown in Table 9-3 for distance intervals to 500 mi. Population data actually used in the CRAC2 program are defined for a finer grid than in Table 9-3. Data for 0 to 50 miles were derived from PNL-3777.(3) Data for 50 to 500 mi were derived by Sandia National Laboratories from 1980 census data.

9.2.3 Evacuation

Reduction in dose, resulting from evacuation of people from the area surrounding the reactor, is included in the calculations. The emergency response model provided with the CRAC2 computer program was used. This model includes a time delay before movement, followed by evacuation radially away from the reactor at a constant speed. All persons within the designated evacuation area are assumed to move as a group after the delay time. The relative position of the evacuation area and the plume is considered in evaluating the duration of exposure to the plume and contaminated ground (from deposition of particulates from the plume).

Because there are few people within 10 mi from N Reactor, evacuation has little effect on either the population exposure or total latent fatalities. Also, for the same reason, variations in the evacuation model parameters have little effect on the predicted fatalities from early exposures.

9.2.4 Health Effects Data

There are two categories of health effects considered by the CRAC2 program: (1) early somatic effects; and (2) late somatic effects. Early somatic effects include mortalities and morbidities occuring within a matter of days and up to 1 yr after exposure. The calculation of late somatic effects includes latent cancer fatalities plus, benign and cancerous thyroid nodules.

Early health effects are based on dose received by bone marrow, lungs, and gastrointestinal tract, with specific dose response functions defined for each organ. Contributions to dose are included for external exposures to the passing cloud and from ground contamination, and for inhalation of material in the cloud. The period for which the dose is received varies from 7 d for the gastrointestinal tract to 1 yr for the lungs.

Late somatic effects are based on early (acute) uptake plus uptake in years following the accident (chronic). Late effects are estimated for exposure of bone, bone marrow, lungs, breast, gastrointestinal tract, thyroid, and a general category for other cancers. Doses are converted to

Table 9-3. Population Distribution for N Reactor.*

Sector	Compass Dir	0-10	10-20	20-30	30-40	40-50	50-60	60-70	70-100	100-200	200-350	350-500	Totals
		Number of People Within Distance Interval (Mi.)											
1	NORTH	36	953	420	1,492	7,583	0	230	4,824	22,089	160,000	250,000	447,627
2	NNE	5	285	561	18,531	1,350	305	739	7,830	21,776	160,000	775,000	986,382
3	NE	0	624	1,031	2,691	259	464	1,226	3,757	234,280	160,000	210,000	614,332
4	ENE	0	620	5,884	1,129	429	0	2,770	2,405	248,658	92,966	45,605	400,466
5	EAST	0	294	625	2,742	605	476	198	4,501	124,376	167,709	216,633	518,159
6	ESE	0	306	1,493	596	247	480	341	7,302	15,875	18,713	149,799	195,152
7	SE	0	54	2,113	28,922	5,001	951	2,029	51,054	37,284	329,736	247,616	704,760
8	SSE	0	0	35,127	50,292	3,354	14,474	5,556	22,325	14,771	41,225	29,006	216,130
9	SOUTH	0	127	4,592	2,041	176	6,466	920	3,445	7,551	15,225	246,609	287,152
10	SSW	0	258	1,676	12,603	625	0	392	1,090	64,003	254,618	441,179	776,444
11	SW	0	547	4,946	16,747	469	43	0	9,304	139,816	789,371	33,250	994,493
12	WSW	0	680	1,699	8,297	15,274	2,804	279	1,119	1,343,410	89,799	0	1,463,361
13	WEST	0	395	936	5,149	75,686	11,971	332	1,602	245,467	66,459	0	407,997
14	WNW	54	573	377	490	1,598	15,807	0	5,247	2,092,771	41,292	0	2,158,209
15	NW	74	277	425	515	683	6,116	19,020	11,958	363,540	1,600,000	200,000	2,202,608
16	NNW	64	277	438	1,030	4,696	3,964	13,411	9,933	3,336	160,000	250,000	447,149
	Totals	** 233	6,270	62,343	153,267	118,035	64,321	47,443	147,696	4,979,003	4,147,113	3,094,697	12,820,421

*CRAC Population Distribution based on 1980 Census.
**A 1987 Grant County Survey shows a population of 178.

probability of cancer, using the central estimate method developed for the Reactor Safety Study.[4] The central estimate method applies dose effectiveness factors defined as a function of total dose and dose rate. The basic model for calculating cancer fatalities assumes a latent period in which no effects are found, followed by a plateau period in which the incidence rate is assumed to be constant. No effects are assumed to occur after the plateau period.

9.2.5 Economic Data and Land Usage

Economic data were not input to the present calculations because the desired consequences were limited to certain health effects and total population doses.

9.3 Radionuclide Release to Liquid Pathways

Consequence estimates for radionuclide release to liquid pathways are not included in this study for two reasons. First, as explained in the following, there is a low probability that such releases would occur in significant quantities. Second, the consequences of releases to the liquid pathways are lower than the consequences of comparable air pathway releases and are largely controllable.

In the event of a severe accident, water from several sources and with a large radionuclide content can accumulate on the floor of confinement. Sources of water include spilled reactor coolant, ECCS water, and fog-spray water. Systems are provided to pump water from the confinement floor and from the emergency dump tank to an outside basin of 11-Mgal capacity. Once this basin is filled, water is diverted to a separate 5-Mgal basin. The larger basin is being added as part of the current N Reactor upgrade program and will be lined to minimize percolation to the groundwater system. Any release to the river via the groundwater system would require substantial transport time (with radioactive decay) and would be accompanied by significant retention of radionuclides in the soil.[5]

Another possible route for release to the river is direct overflow from confinement. If the drainage system fails to pump water from confinement, collected water would underflow the "banana" wall in the refueling canal and enter the fuel handling area. The capacity of the fuel handling area and connected spaces is approximately 5-Mgal. It is judged very unlikely that all sump pumping capacity would be lost and the available collection volume would be exceeded.

The relative effectiveness of soluble fission products (such as iodine, cesium, and strontium) in generating population dose is discussed in References 5 and 6. In these references it is concluded that on a per-curie basis, significantly greater population exposure results from releases to airborne pathways than to liquid pathways.

9.4 Results of Consequence Calculations

Consequences were calculated for three consequence indices: (1) early fatalities; (2) population exposures in person-Rem; and (3) latent fatalities. Selected results from consequence calculations (for the fission product release data summarized in Table 9-1) are listed in Table 9-4. Peak values are listed for each of the consequence indices. The calculations show the results are insensitive to whether the release is from the stack or at ground level. Calculations also predict no early fatalities for any of the credible release categories. This is due in part to the low offsite population near the reactor and the large distance between the plant and the offsite population.

The mean health effects shown in Table 9-4 are the weighted average of 1,600 individual consequence calculations for each source term. These 1,600 calculations are based on a sampling of meteorological conditions from hourly readings over a 1-yr period. The sampling procedure is semi-random because a number of meteorological conditions are chosen from a limited number of sets or bins that represent the complete spectrum, ranging from benign to very adverse meteorology. In this way a number of severe cases is guaranteed to be selected for calculation. In each consequence index the peak value is the worst result for the 1,600 individual calculations. Peak health effects are generally two to three orders of magnitude less likely to occur than the mean health effects.

Detailed output from the CRAC calculations was used for the overall risk calculations presented in Section 10.0.

9.5 References

1. Ritchie, L. T., J. D. Johnson, and R. M. Blond, 1983, Calculations of Reactor Accident Consequences Version 2 CRAC2: Computer Code, NUREG/CR-2326 (SAND81-1994), U.S. Nuclear Regulatory Commission, Washington, DC.

2. Ritchie, L. T., D. K. Alpert, R. P. Burke, J. D. Johnson, R. M. Ostmeyer, D. C. Aldrich, and R. M. Blond, 1984, CRAC2 Model Description, NUREG/CR-2552, (SAND82-0342), U.S. Nuclear Regulatory Commission, Washington, DC.

3. McCormack, W. D., 1982, Hanford Dose Overview Program: Standardized Methods and Data for Hanford Environmental Dose Calculations, PNL-3777, Pacific Northwest Laboratory, Richland, WA.

4. NRC, 1975, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Power Plants, Apendix VI, Calculation of Reactor Accident Consequences," WASH-1400 (NUREG 75/104), U.S. Nuclear Regulatory Commission, Washington, D.C.

5. NRC, 1978, Liquid Pathway Generic Study, NUREG-0440, U.S. Nuclear Regulatory Commission, Washington, D.C.

6. Westinghouse, 1977, Liquid Pathway Generic Study, Topical Report 22A60, Westinghouse (Offshore Power Systems).

Table 9-4. Summary of Results of N Reactor Consequence Calculations for Risk Dominant Fission Product Release Categories.

Release Category	General Characteristics	Release Elevation	Early Fatalities Mean	Early Fatalities Peak	Whole Body Population Dose Person-Rem Mean	Whole Body Population Dose Person-Rem Peak	Latent Fatalities* Mean	Latent Fatalities* Peak	Release Category Frequency/yr
N-1	Seismic Initiator, LOCA and/or ECCS/Spray Failure	Ground	0	0	6 E+05	4 E+06	40	300	3.3 E-05
N-1	Seismic Initiator, LOCA and/or ECCS/Spray Failure	Stack	0	0	7 E+05	6 E+06	50	400	3.3 E-05
N-4	Reactor Trip with Loss of Post-Trip Core Cooling	Ground	0	0	7 E+04	6 E+05	5	40	1.8 E-09
N-4	Reactor Trip with Loss of Post-trip Core Cooling	Stack	0	0	8 E+04	6 E+05	5	40	1.8 E-09
N-5	Seismically initiated LOCA or Trip Failure with Loss of Spray	Ground	0	0	2 E+05	2 E+06	10	140	3.5 E-06
N-6	Loss of Core Cooling with Loss of Confinement Isolation	Ground/ Stack	0	0	5 E+04	4 E+05	3	30	7.0 E-06
---	Bounding	Ground	<1	140	2 E+06	2 E+07	160	1,400*	4 E-05**

* The expected number of cancer fatalities from other causes for the same population and time period (50 years) is approximately 1.4 million.

** Bounding case frequency, equivalent to the calculated frequency of 100% fuel damage.

10.0 RISK ESTIMATES

Consequences calculated in Section 9.0 are combined in this section with source term category frequencies to develop overall risk estimates for severe accidents that could occur during operation of N Reactor. In addition, risk estimates are presented for a bounding source term case. Generally, the source term estimates in this study are conservatively high so the risk values are also expected to be conservatively high.

10.1 Radiological Risk Estimates

Consequence calculations presented in Section 9.0 indicate no early fatalities for credible accident sequences. Risk estimates for the population dose and latent cancer fatality indices are shown in Figures 10-1 and 10-2. The curves have similar shapes because the latent fatalities calculations in the CRAC health effects model are based on total radiation exposure. Also shown in Figures 10-1 and 10-2 are the contributions from those release categories that are significant contributors to the overall risk curve. It is apparent that release category N-1 is the principal contributor to total risk with some contribution from release categories N-5 and N-6. Table 10-1 lists the release categories, the principal sequences contributing to them, and the frequencies associated with these principal sequences.

The principal sequences contributing to release category N-1 (from the standpoint of frequency) are the 1/EUR7 and 2/EUR7 sequences. Both are seismically initiated sequences, with the higher frequency 1/EUR7 sequence resulting from failure of 181-N/182-N as a result of the seismic event. Loss of these buildings leads to loss of the fog spray, ECCS, and GSCS systems. Released fission products resulting from fuel damage will at least partially flow through the dump tank if the tank is functional following a seismic event. For the purpose of the scoping analysis in this study, no fission product scrubbing credit is taken for the dump tank because the tank response to an earthquake was not analyzed.

The other sequence contributing to category N-1 (2/EUR7) results from failure of the pressure tubes as a result of the seismic event. This sequence has significantly lower frequency than 1/EUR7 discussed previously. For Sequence 2/EUR7, radionuclides released as a result of fuel damage vent to the confinement. Fog sprays are unavailable in both sequences.

In summary, the principal sequences leading to the dominant N-1 fission product release category are initiated by seismic events. It is assumed conservatively for these sequences, the dump tank will not be available for scrubbing released fission products. With the dump tank available for the frequency-dominant sequence, consequences and thus risk values could be lower by one to two orders of magnitude.

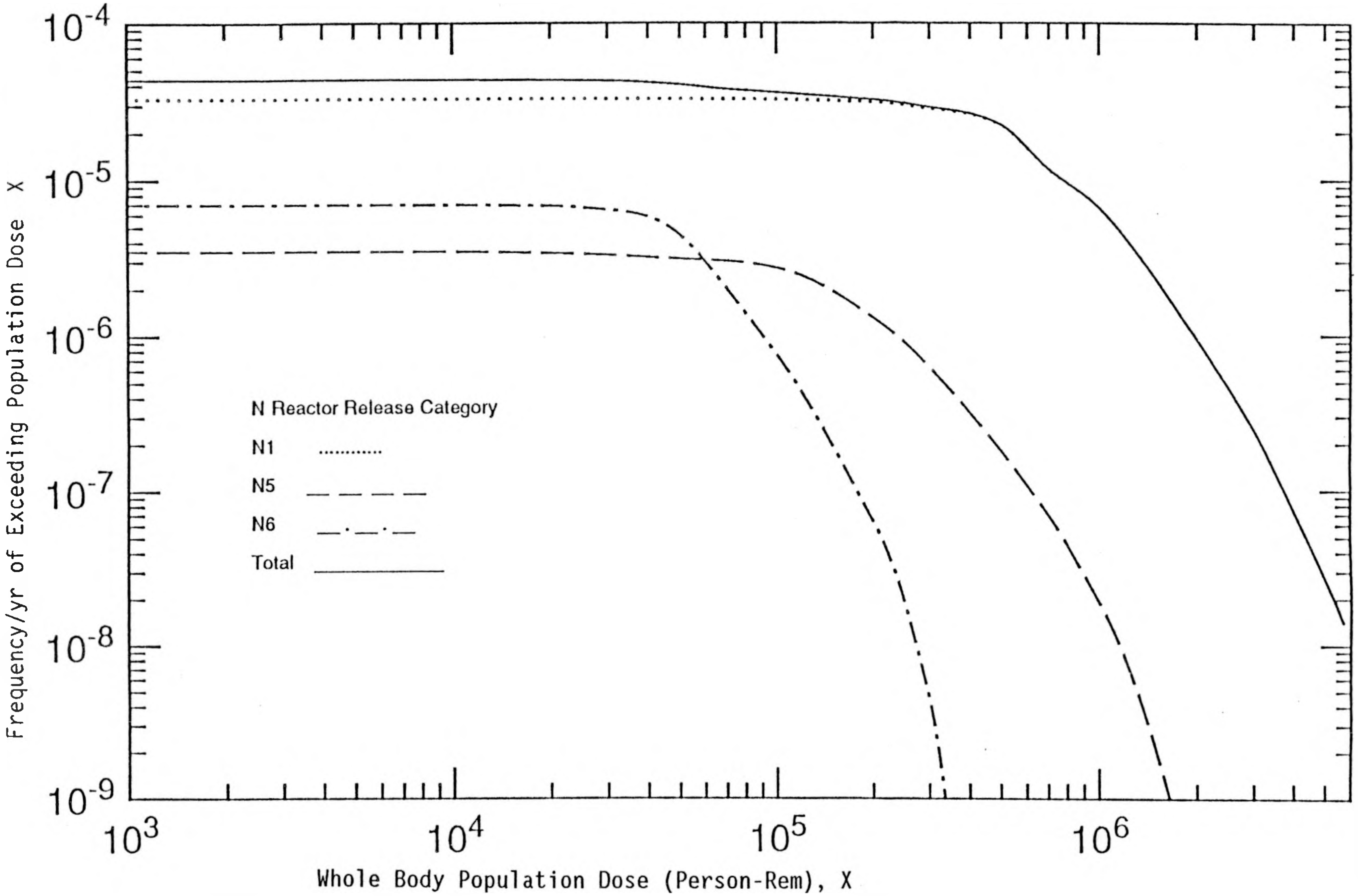


Figure 10-1. N Reactor Risk Curve for Population Dose.

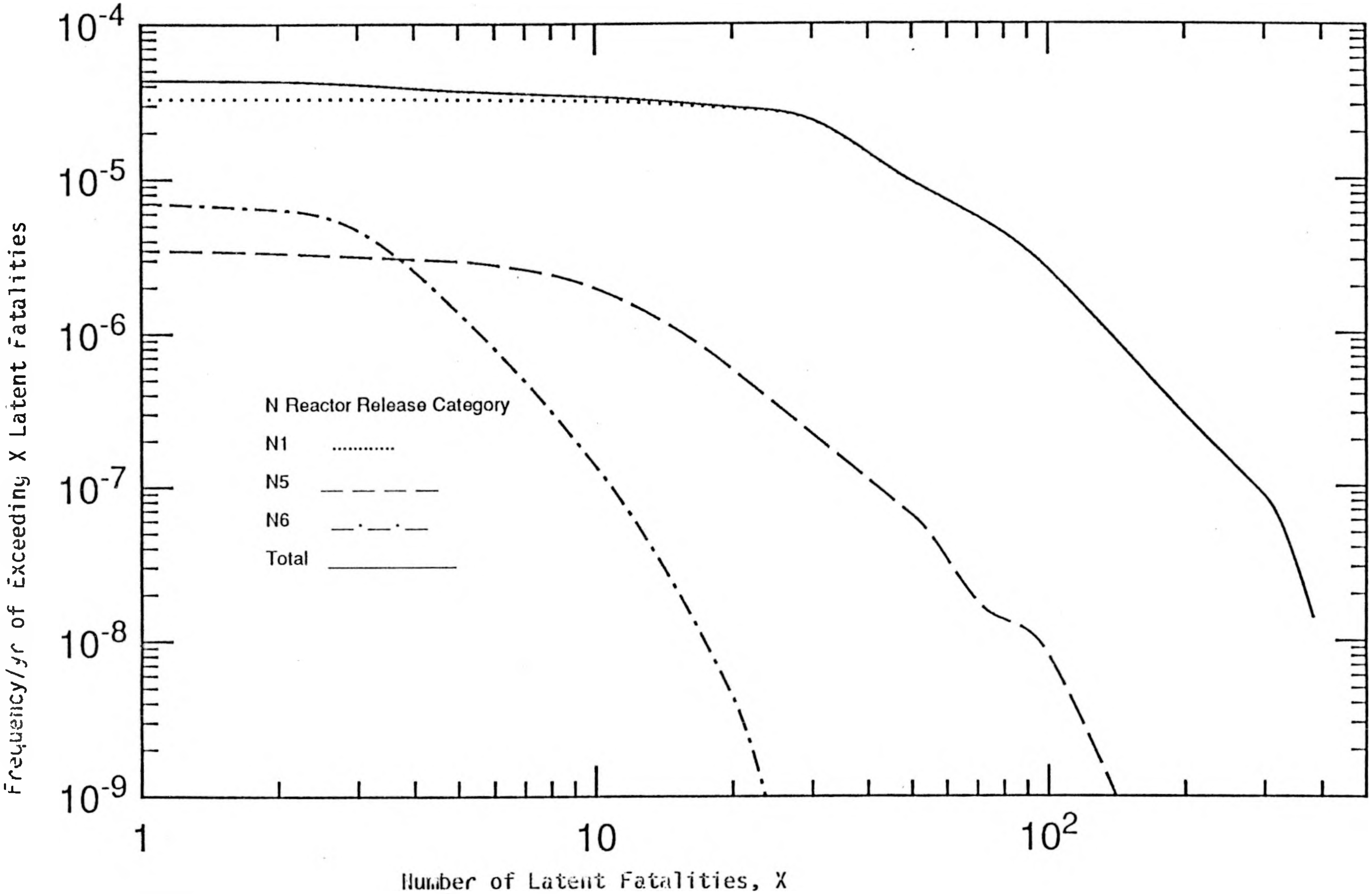


Figure 10-2. N Reactor Risk Curve for Latent Fatalities.

Table 10-1. Release Categories and Dominant Contributors

Release Category	Frequency (Per Year)	Dominant Contributing Sequences (1)	Frequency (Per Yr)
N-1	3.3 E-05	1/EUR7	3.3 E-05
		2/EUR7	5.8 E-09
N-2	2.5 E-10	8/LUR2	2.5 E-10
N-3	1.0 E-09	9(2)	1.0 E-09
N-4	1.8 E-09	6/LUR2	9.6 E-10
		7/LUR2	8.4 E-10
N-5	3.5 E-06	3/EUR1	8.6 E-08
		4/EUR1	2.0 E-07
		3/EUR3	8.6 E-07
		4/EUR3	2.6 E-08
		2/EUR3	2.3 E-06
N-6	7.0 E-06	8/EIR1	8.0 E-10
		8/EUR2	2.9 E-10
		8/EUR3	2.7 E-08
		6/EIR1	3.0 E-09
		7/EIR1	2.7 E-09
		6/EUR2	1.1 E-09
		7/EUR2	9.8 E-10
		6/EUR3	1.0 E-07
		7/EUR3	9.1 E-08
		5/EUR3	6.8 E-06
N-7	1.7 E-05	10/EIR1	1.9 E-09
		11/EIR1	1.7 E-09
		10/EUR2	7.1 E-10
		11/EUR2	6.3 E-10
		10/EUR3	6.6 E-08
		11/EUR3	5.9 E-08
		10/LUR2	6.1 E-10
		11/LUR2	5.4 E-10
		6/A1	7.8 E-06
		7/A1	6.9 E-06
		8/A1	2.1 E-06
N-8	7.2 E-05	12/EIR2	6.3 E-05
		10/A1	5.0 E-06
		11/A1	4.4 E-06

Notes

(1) Format is plant sequence number/confinement failure mode.
(2) Sequence 9 results in partial direct bypass of confinement via ECCS supply line.

10.2 Comparison to Risk Calculated in other PRA Studies

There were two studies selected for comparison with the risk estimates calculated in this study. These are the WASH-1400 study,[1] in which risk estimates were calculated for the Surry PWR plant and the Peach Bottom Boiling-Water Reactor (BWR) plant, and the Indian Point Probabilistic Safety Study (IPPSS) in which risk estimates were calculated for the Indian Point 2 and Indian Point 3 plants.[2] In the IPPSS, risk estimates were calculated for the two plants at the same site, Indian Point 2 and Indian Point 3; calculated risks were significantly lower for Indian Point 3. WASH-1400 was selected for comparison because it represents the initial, definitive study of risk from commercial nuclear power plants. Indian Point was selected because the results reported in the IPPSS formed the evidentiary basis for an extensive public hearing in which the Nuclear Regulatory Commission found the risks to the public from plant operation to be acceptably low.[3] This is the only public hearing in which a finding of this nature was based largely on a PRA.

In the WASH-1400 and IPPSS studies, early fatalities were predicted for the low probability/large radionuclide release accident sequences. The results of the early fatality calculations for the WASH-1400 plants and for Indian Point 3 are summarized in Figure 10-3. The calculations in this study indicate no early fatalities at a probability equal to or greater than 1 E-09/yr.

Results of calculations for latent cancer fatalities for the WASH-1400 plants, Indian Point 3, and N Reactor are plotted in Figure 10-4. The figure shows latent fatality risks for N Reactor are significantly lower than those calculated in these earlier studies for these commercial nuclear power stations (which are regarded as the bench marks from the standpoint of plant siting acceptability). The lower radiological risks calculated for N Reactor result from the combination of plant design features and the remote location of the plant.

The risk calculations in WASH-1400 included only events initiated by internal plant upsets or initiators; the Indian Point 3 and N Reactor studies include accidents initiated by both internal and external events (seismic event only for N Reactor). For both Indian Point 3 and N Reactor, accident sequences initiated by external events are found to be the principal contributors to risk. The total risks calculated in several other power reactor PRAs have also been dominated by externally initiated events.

10.3 Comparison of Results to NRC Safety Goals

The NRC has issued a set of safety goals in the form of nuclear power plant design objectives. The objectives set forth in the NRC Safety Goals for Operation of Nuclear Power Plants[4] are as follows.

- o The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents members of the U.S. population are generally exposed. Based on recent statistics[6] the accidental death risk for the U.S.

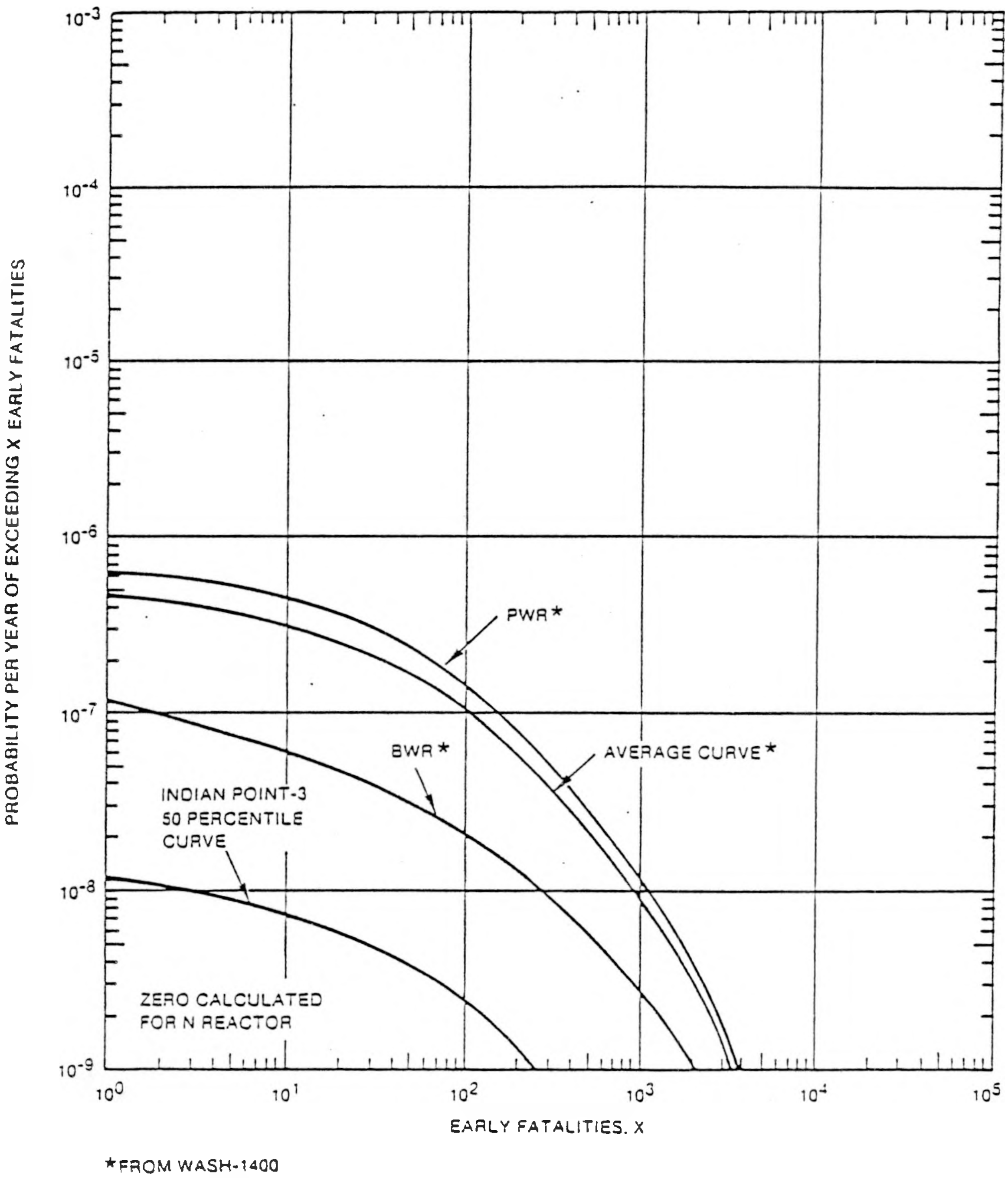


Figure 10-3. Comparative Risk for Early Fatalities.

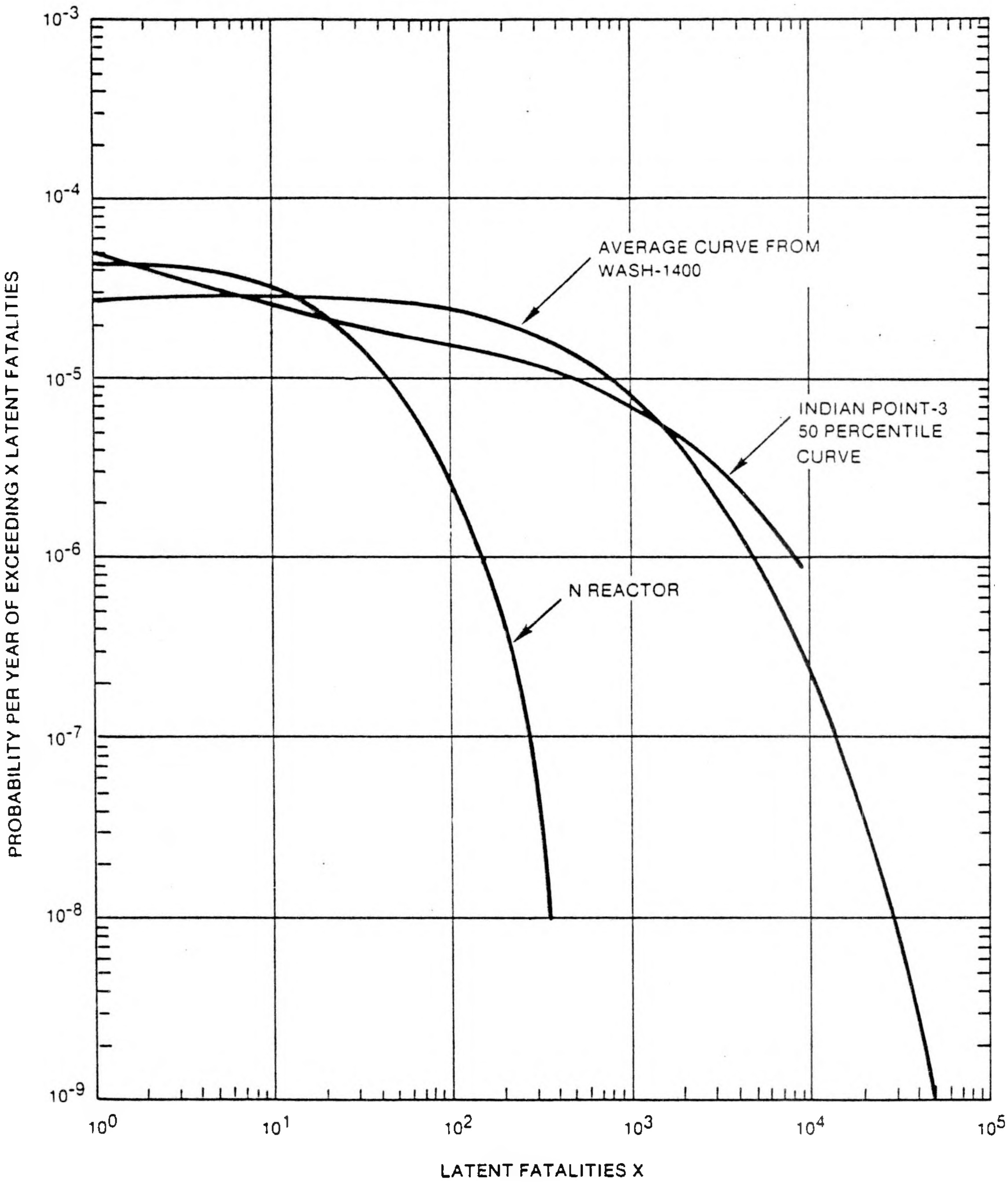


Figure 10-4. Comparative Risk for Latent Fatalities.

population is approximately 4 E-04/yr, thus the safety goal for prompt fatalities would be 4 E-07/yr.

- o The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risk resulting from all other causes. Based on the Statistical Abstract of the U.S.,[6] the individual cancer risk is 2 E-03/yr, thus the safety goal for nuclear power risk is 2 E-06/yr.

In an earlier draft policy statement[5] the Commission also included the following design objective:

- o The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000/yr (1 E-04/yr) reactor operation.

The NRC safety goals are compared with the results of this study in Table 10-2. An estimate of latent fatality risk for N Reactor is 7 E-09 fatalities/yr for the offsite population within 10 miles from the reactor. The total offsite population within 10 miles is approximately 250 people(See Table 9-3). Thus the average latent fatality risk to an individual is of the magnitude of 3 E-11/yr. The average latent fatality risk to an individual within 50 mi from the plant is somewhat higher (7 E-10/yr) because winds blow more often in the directions of denser population beyond 10 mi. The latent fatality risk results are based on results from the CRAC2 consequence calculations for a specific population group. The latent fatality (or societal risk) values calculated for N Reactor are approximately four orders of magnitude less than the NRC goal (Table 10-2).

No early fatalities are calculated outside the plant boundary at a frequency of 1 E-09/yr or greater for credible accident sequences defined in this study; thus, this safety goal is met.

Estimated fuel damage frequencies are lower than the core melt frequency appearing in the earlier version of the NRC's proposed safety goal, even when end state Y ($\leq$30% fuel damage) is included in the total. Seismically initiated events are the principal contributors to the overall estimated fuel damage frequency for N Reactor.

10.4 Risk Estimates for the Bounding Release Case

A bounding calculation was performed to estimate the maximum possible risk from N Reactor operation. The boundary source term includes release of the entire core inventory of volatile fission products directly to the environment (i.e., with no credit for radionuclide retention by the confinement system). The frequency of this bounding release is assumed to be 4 E-05/year based on the frequency of sequences identified in Section 5.0 that lead to 100% fuel damage. This frequency was selected to ensure conservatism rather than representing a realistic estimate for a release of this magnitude.

Bounding case risk results are shown in Figures 10-5 and 10-6 for early and latent fatalities, respectively. Bounding case early fatalities for N

Table 10-2. Comparison of N Reactor Limited-Scope Risk Assessment Results with Nuclear Regulatory Commission Safety Goals.

Goals	Quantified design objective	N Reactor result
Primary Safety Goal[1]		
Prompt fatality risk in reactor vicinity	Limit increase in individual annual risk of accidental death to an increment of 4 E-07/yr. Apply to circular area within 1 mi from plant boundary.	No early fatalities outside plant boundary.
Latent fatality risks in reactor vicinity	Limit increase in individual annual risk of cancer death to an increment of no more than 2 E-06/yr. Apply to offsite population within 10 mi from plant.	3 E-11/yr
Secondary Draft Safety Goal[2]		
Likelihood of large-scale core melt	Likelihood of large-scale core melt should be less than 1 E-04/yr.	7 E-05/yr[3]

[1]From Reference 5
[2]From Reference 4
[3]Total calculated frequency of $\geq$30% fuel damage

PST87-1159-6

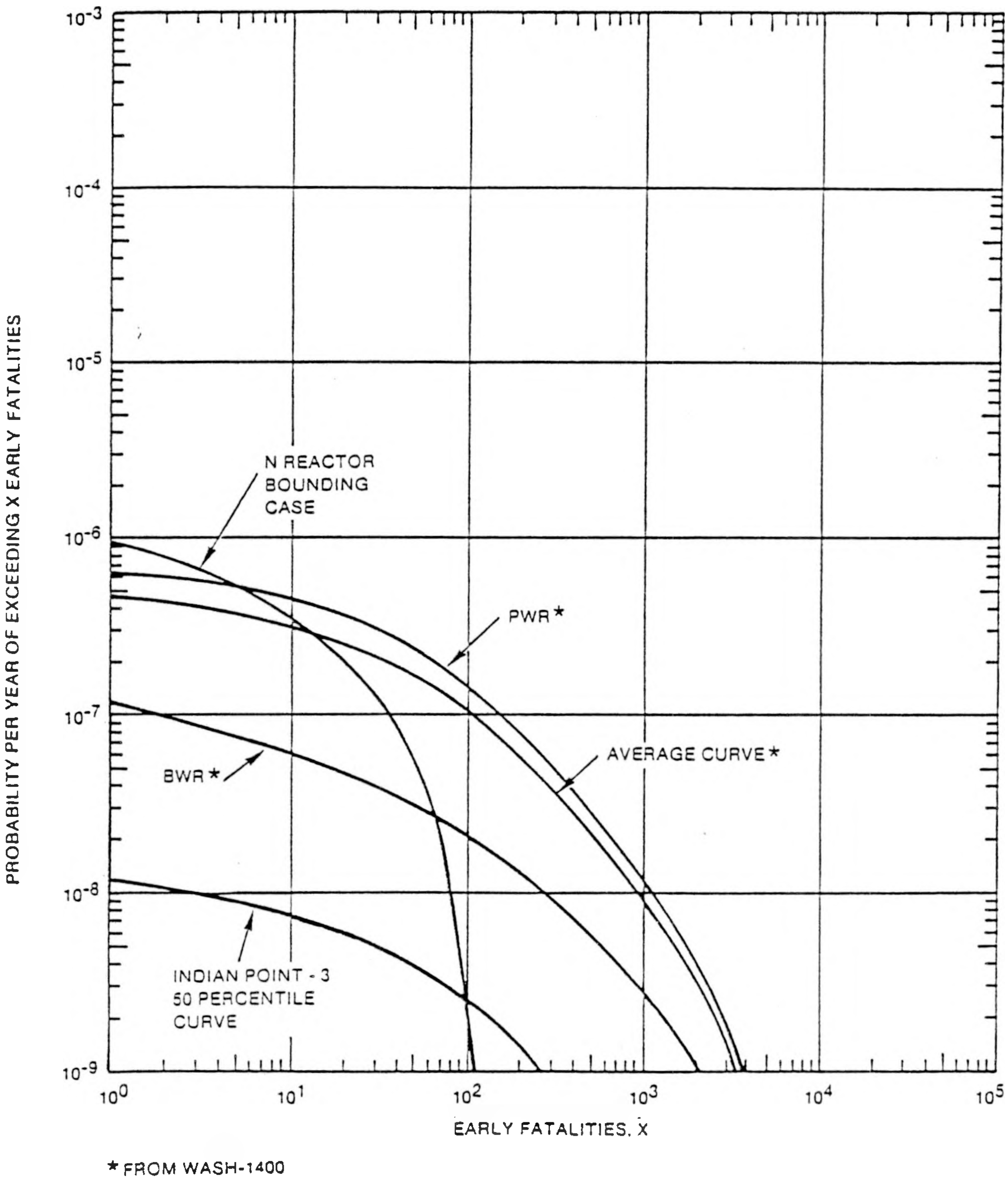


Figure 10-5. Comparative Risk for Early Fatalities - Maximum Bounding Release.

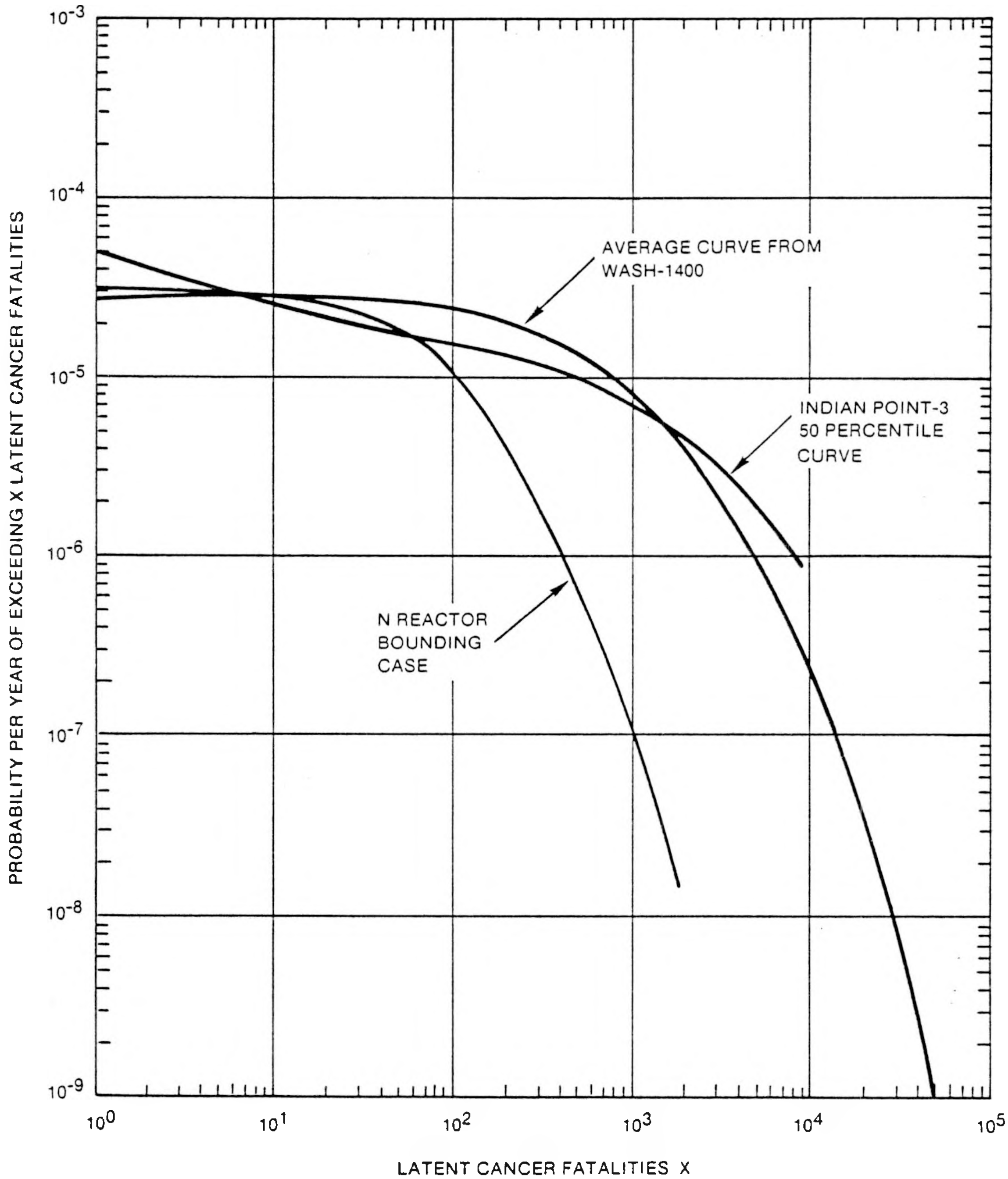


Figure 10-6. Comparative Risk for Latent Fatalities - Maximum Bounding Release

Reactor are higher than those calculated for Indian Point-3 and are lower than WASH-1400 results. Latent fatalities are lower than both WASH-1400 and Indian Point-3 results. Neither the WASH-1400 results nor the Indian Point-3 results are based on a bounding source term similar to that assumed for N Reactor.

Bounding case calculations provide an upper limit on the possible magnitude of severe accident consequences. When combined with a conservatively high frequency of occurrence, the bounding calculations provide an estimate of the maximum possible risk from N Reactor operation.

10.5 References

1. NRC, 1975, Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,WASH-1400, U.S. Nuclear Regulatory Commission, Washington, D.C.

2. 1982, Indian Point Probabilistic Safety Study, copywrited by Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., New York, New York.

3. Paladino, N. J., et al., 1985, Nuclear Regulatory Commission Decision CLI-85- 06, Docket No. 50-247-SP and 50-286-SP, Washington, D.C.

4. NRC, 1986, Safety Goals for Operation of Nuclear Power Plants, Policy Statement, U.S. Nuclear Regulatory Commission, Washington, D.C.

5. NRC, 1983, Safety Goals for Nuclear Power Plant Operation, NUREG-0880, Revision 1, for comment, U.S. Nuclear Regulatory Commission, Washington, D.C.

6. 1986, Statistical Abstract of the U.S., U.S. Department of Commerce, Bureau of the Census, 106th Edition, Washington, D.C.

APPENDIX A

FRAGILIITY ESTIMATES FOR N REACTOR COMPONENTS AND STRUCTURES

National
Technical
Systems

NTS Engineering
6695 East Pacific Coast Highway
Long Beach, CA 90803
(213) 493-6651

April 1, 1987

Dr. Alan M. Kolaczkowski
Supervisor, Reliability and Safety Analysis
Science Applications International Corporation
2109 Air Park Road S.E.
Albuquerque, New Mexico 87106

References

1. Seismic and Pressure Tube Rupture Evaluation of DOE N Reactor, Phase I--Analyses and Evaluations, UNC Nuclear Industries Report UNI-3153, Feb., 1985.

2. N Reactor Core Seismic and Pressure Tube Rupture Event Safety Evaluation Criteria, UNC Nuclear Industries Report UNI-3171, July 1985.

3. Seismic and Pressure Tube Rupture Evaluation of DOE N Reactor, UNC Nuclear Industries Report UNI-3178, Vols. I and II, Aug., 1985.

4. Seismic Response of N Reactor Core-Independent Review of Phase II Work, UNC Nuclear Industries Report UNI-3179, Aug., 1985.

Dear Alan:

In accordance with your request, we have evaluated the seismic capacities of several additional components and structures for the N-Reactor. Among these components and structures are the graphite core, the silo, the steam generators, and Building 182. These evaluations have been based on very limited information for the most part, and consequently still retain significant amounts of uncertainty. In some cases, there may also be some conservatism although we have tried to minimize this.

1. FRAGILITIES FOR THE N-REACTOR CORE ASSEMBLY

Several reports (1)(2)(3)(4) on the seismic and pressure tube rupture analysis of the N Reactor core and surrounding shield walls were reviewed to determine the relative seismic resistance of the various elements that control the conditional probability of failure of core elements and to develop approximate fragility curves in support of the mini PRA activity.

1) Pressure tube structure integrity
2) Graphite cooling tube structural integrity
3) Ability to scram the reactor

These concerns are, in turn, influenced by the integrity of the graphite block assembly, the primary shield walls, the inlet/outlet shield walls and the reactor pedestal. Structural failure and relative deflections are both of concern in assessing failure modes. Results of the Phase I and Phase II seismic studies (1)(3) conducted by Impell Corporation were used to develop approximate fragility descriptions for the above concerns.

The Impell report summarizes the results of complex analysis of the core assembly to a seismic input defined at the soil/foundation interface. Since the reports are summaries, much of the intermediate detail is not contained and assumptions must be made on these intermediate details. The Impell analysis is deterministic and is stated to be conservative. An independent study sponsored by LLNL (4) examined a number of parameter variations in the soil modeling and structural modeling and used a different computer code for soil structure interaction analysis. Their conclusion was the Impell analysis was reasonable but they also showed that large variations in response can be expected depending upon the structural modeling and soil properties used in the models. The fragility descriptions in this memo are based upon the Impell results, and incorporate the uncertainties in those results.

The fragility descriptions are portrayed as a best estimate (median) capacity relative to the free field peak ground acceleration and logarithmic standard deviations β_R and β_U representing the randomness and uncertainty in the estimated capacity. Not all of the details of the analyses and actual physical hardware are provided in the Impell reports. The consequences of exceeding some of the displacement limits imposed upon the core assembly and the basis for the limits are not developed in the reports. Corsequently the fragility descriptions provided are approximate. If the lower bound of the fragility descriptions is expected to exceed 1.0g peak ground acceleration, this fact is stated and the median value and uncertainties are not calculated These estimates can be improved with further study of the analytical models and selected drawing detials. They are being provided as a first cut for purposes of assessing the significance and frequency of each failure mode.

The fragility descriptions for seismic induced loading are uncoupled from the pressure tube rupture event. As will be shown, the pressure tubes have a large margin against seismic induced failure and such events are not considered to be concurrent.

1.1 Pressure Tubes

The most conservative Impell model assumes that graphite block keys are failed in the active core zone. This case produces the highest tube stresses and strains but the calculated values are low relative to their allowables. Thus, for the conservative conditions modeled, the lower bound on pressure tube capacity is expected to exceed 1.0g peak ground acceleration and seismic induced pressure tube failure is not judged to be a controlling failure mode unless other sources of failure cause the pressure tube rupture.

1.2 Graphite Cooling Tubes

The graphite core is not keyed to the core pedestal in the NS direction. Thus, the tubes carry the inertial load of the grpahite core as it slides as a rigid body. The resulting SSE stresses and strains are low however for the specified SSE and the lower bound seismic capacity is expected to exceed 1.0g. Seismic induced pressure tube failure is not judged to be a controlling failure mode unless other failure modes cause the tube failures.

1.3 Ball Channels

The ball channels are limited by deflection and the most critical area is between the top of the core and the primary shield. The stated deflection limit is 1.25 inches and the computed vector of the NS and EW core displacements relative to the primary shield structure is 96% of this value. This value is taken from the Phase 2 report that considers the keys in the active core region to be inactive. The Phase I report contained results of an as-built core model, as well as an inactive key model. Under the best conditions with the keys active, the displacement was 80% of the worst case with inactive keys. There is some degree of conservatism in the calculated displacements and likely some conservatism in the allowable displacement. The relative displacement limit is dominated by displacement of the graphite reflector assembly. In developing the fragility description it was assured that the stated allowable deflection was a lower bound on failure of balls to drop and that the median value was 25% higher. Accounting for the conservatisms and uncertainty in the soil structure interaction modeling and core assembly modeling, the median ground acceleration capacity and its uncertainty were estimated to be:

$$\check{A} = 0.4g$$
$$\beta_R = 0.2$$
$$\beta_U = 0.25$$

1.4 Horizontal Control Rods

The Phase II model results are based on having no active keys in the active region of the core. This affects the displacement between the active core and reflector zones but this displacement does not appear to be as controlling as the displacement between the primary shield and the core. The Phase I report shows that there is little difference in the displacement between the primary shield and the core for all conditions analyzed so the estimated capacity is based on the displacement limit of 1.0 inches between the shield and core and the calculated Phase II displacements. The calculated displacement for the SSE is 0.63 inches. There is some conservatism in the calculated response and probably in the allowable displacement. The median failure displacement is assumed to be 25% greater than the allowable and the allowable is assumed to be lower bound displacement limit. Accounting for the conservatisms and uncertainty in soil structure interaction modeling and core assembly modeling, the median peak ground acceleration was estimated to be:

$$\check{A} = 0.6g$$
$$\beta_R = 0.2$$
$$\beta_U = 0.25$$

The failure mode is assumed to be a failure to scram. However, since the relative displacement is cyclic in nature, it is not clear that the HCR's would not enter the core between peak displacement excursions. Note that the displacement limit applies to the first 1½ seconds of the earthquake. The HCR's can supposedly be driven into the core in this time period. The displacements after 1½ seconds are larger and if there is any doubt about the scram timing, some additional uncertainty in the fragility description should be considered.

1.5 Graphite Keys and T Blocks

The calculated stresses are low and the capacity of these elements are not considered controlling.

1.6 Primary Shield/Core Impact

There is a 1.5 inch clearance between the graphite core and the primary shield. Based upon the Phase II calculations, the median input acceleration level is about 0.4. The variability associated with this value is estimated to be:

$$\beta_R = 0.2$$
$$\beta_U = 0.25$$

Consequences of impact are likely not serious until g levels considerably in excess of the impact threshold are experienced. Based upon the the reported stresses for the pressure tubes and graphite cooling tubes and the calculated SSE displacements, the tubes are not expected to fail at the impact displacement of 1.5 inches.

The affect of impact upon the graphite reflector structural integrity cannot be determined from the current analysis. For purposes of making an estimate of consequences, assume severe damage to the reflector blocks at about 1.5 times the impact threshold. This could impede scram but the damage would occur later in time than the previously described deflection limits for the HCR's. It is difficult to imagine damage to the reflector blocks causing a tube rupture since the graphite cooling tubes manage to carry the whole graphite core assembly in the NS direction.

The shield wall uplift discussed later may be a more critical condition so for purposes of the mini PRA, consequences of impact should not be modeled.

1.7 Primary and Inlet/Outlet Shield Walls

The Phase II Impell report indicates small margins exist for shear and bending of the shield walls. The calculated shears and moments were compared to allowable limits in ACI 349-1976 and the margins were stated to be 1.16 to 1.19. The ϕ factors were not used in the allowables on the basis that the increase in concrete strength with age would compensate for not using the ϕ factors. No credit was taken for reinforcing steel in the concrete. It was not clear what load factors were use in the ACI 349 evaluation so without examining the detailed calculations or obtaining drawings of the shields, conducting independent strength calculations for the specified moment and shears, the basis for the allowable and the actual margin to failure not known. There is ductility available in the shield walls but, if the walls yield, the displacement will increase and all analyses of relative deflections between the core and shield walls are no longer representative. These increases in relative deflections would affect the stresses in the pressure tubes, graphite cooling tubes and the displacements which are functional limits for horizontal control rods and ball channels. We think that the primary shield wall is not a problem but would like to review the drawings and strength calculations to verify this assumption.

1.8 Reactor Core Pedestal

The calculated margin for the pedestal is about the same as the shield walls. The pedestal supports both the core and the primary shield walls so that inelastic deformation in the pedestal does not directly result in differential motion between the core and the shield wall. We feel that the reactor pedestal is not a problem area but would like to review the drawings and strength calculations to verify this assumption.

1.9 Uplift Of Primary Shield

The analysis for the SSE indicates that uplift of the primary shield wall would occur at about 0.3g. The consequences of uplift are not certain. Uplift will increase the relative displacements between the core and the shield walls which in turn will affect the ball channel operability and control rod operability. Large amounts of uplift could cause tubes to break. Once uplift occurs another major nonlinearity is introduced and the current analytical results cannot be extrapolated.

For the frequency range of the earthquake that drives the uplift, there is considerable margin in the analysis. The median uplift point is exactly to be 0.6g which is about the same value as estimated for the HCR displacement fragility and greater than estimated for the ball channels. For purposes of scoping, assume that the fragility descriptions for ball channel function and HCR function previously stated is not altered by uplift. Further assume that at 1.5 times the uplift threshold that multiple failure of pressure tubes and graphite cooling tubes will occur. Thus, the fragility description for graphite cooling tubes and pressure tubes, resulting from uplift are:

$$\check{A} = 0.9$$
$$\beta_R = 0.25$$
$$\beta_U = 0.5$$

1.10 Summary

The following table summarizes the fragility descriptions for the core for the assumed failure modes and consequences.

Failure Mode	Consequences	$\check{A}$	β_R	β_U
Relative displacement between primary shield and reflector	Ball channels inoperable	0.4	0.2	0.25
Relative displacement between primary shield and reflector	Failure to scram with HCRS	0.6	0.2	0.25
Uplift of shield wall	Multiple tube rupture	0.9	0.25	0.5

2. CONCRETE WATER STORAGE TANK (SILO)

Details of the concrete water storage tank were limited to one drawing showing the reinforcing steel. The embedment of the tank and soil foundation properties, the piping connections and flexibility, and the results of the recent deterministic seismic analysis for the SSE were not available.

The controlling mode of failure of the silo is expected to result from pipe failure as a result of base-slab uplift due to structure rocking. Once uplift occurs, the resulting high toe pressure in the soil foundation can be expected to lead to soil degradation and further rotation. Some uplift can be tolerated depending on the flexibility of the attached piping. However, at higher deformations, fracture of the piping connections must be considered likely, followed by loss of the tank contents. In order to provide an adequate fragility description, additional information and possibly some nonlinear analysis is required. However, for the mini PRA, the seismic capacity for this mode of failure is estimated to be:

$$\check{A} = 0.55g$$
$$\beta_R = 0.2$$
$$\beta_U = 0.35$$

Structural failure of the concrete based on combined shear and flexure of the tank wall is estimated to have a median seismic capcity of about 0.7g. For this mode of failure, some concrete cracking with minor leakage of the unlined tank could be expected, but the cracking and leakage are expected to be small prior to about 0.7g. In view of the expected consequences and lower capacity, it is recommended the failure due to uplift be used for the mini PRA.

3. BUILDING 182-N HIGH LIFT PUMP HOUSE

No additional information on the masonry walls of building 182 was received beyond the brief visual examination during the site visit.

Thus, for the existing wall configuration with no strengthening of any connection details, the seismic fragility previously estimated is still considered representative.

$$\check{A} = 0.5g$$
$$\beta_R = 0.2$$
$$\beta_U = 0.4$$

If the masonry wall connections are strengthened to increase their seismic capacity, the next lower seismic failure mode of the structure is expected to be the failure of the diagonal steel bracing in the roof truss and vertical bracing in the walls. The steel structure was originally designed to withstand a 0.2g static lateral earthquake load or 15 psf wind loads. For some braces wind was found to control. A preliminary investigation of the seismic capacity of the steel structure indicates failure can be expected to initiate in a roof truss diagonal brace in the corner bay between columns D and E and 1 and 2. Failure of this diagonal brace can be expected to lead to progressive failure of the remaining bracing system followed by out of plane loading of the center portions of the walls. This will result in failure of the structure. The capacity for this mode of failure (assuming the masonry walls remain attached) is estimated to be:

$$\check{A} = 0.6g$$
$$\beta_R = 0.25$$
$$\beta_U = 0.3$$

It is our understanding that strengthening of the masonry attachments will be undertaken. No details of the proposed strengthening scheme were available. It is assumed that any strengthening can be expected to result in at least a 20 percent increase in the wall capacity such that the failure of the steel frame can be expected to control. For purposes of the mini PRA, it is recommended that the failure of the steel frame be assumed to control, provided the strengthening program can be verified.

4. STEAM GENERATOR

An estimate of the steam generator seismic fragility was developed since it is considered a likely controlling element in the steam system. No analytical results were available for the seismic analysis of the steam generator. The governing failure mode is assumed to be failure of the anchor bolts in the floating end support. Four 1-1/4 inch diameter bolts are used to anchor the roller support system to the concrete. The expected mode of failure is expected to result from shear of these bolts due to lateral inertia loads from the steam generator together with the nozzle loads from the attached piping. The estimated capacity for this mode of failure is:

$$\check{A} = 0.7g$$

$$\beta_R = 0.2$$

$$\beta_U = 0.35$$

Due to the limited time and resources, other items such as the pressurizer, Building 181-N and other components or structures which may be significant contributors to seismic risk were not evaluated. It is expected the pressurizer capacity will be somewhat greater than the steam generator, and since the consequences of loss of the pressurizer will be no worse than the steam generator, the steam generator capacity is expected to control this system. Building 181-N is reinforced concrete construction for the lower elevations. It apparently was designed to 0.1g static earthquake loads and is consequently fairly lightly reinforced. For purposes of the mini-PRA, its capacity can be assumed similar to Building 182-N.

I hope this provides sufficient information for your purposes for the mini-PRA. If you have any questions, please do not hesitate to call.

Very truly yours,

NTS ENGINEERING

Don

Donald A. Wesley
Senior Consultant

DAW/jlc
cc: Emil Lutz
Dave Aabye

APPENDIX B

ACRONYMS

ACRONYMS

A BUS; B BUS	See Section 3.9.1, Electrical Distribution System
A-HAT	Seismic Fragility Parameter, See Section 4.4.2.3
Al	Confinement Success CET Sequence Designator (See Table 8-2)
AC	Alternating Current
BETA-R	Seismic Fragility Parameter, See Section 4.4.2.3
BETA-U	Seismic Fragility Parameter, See Section 4.4.2.3
CET	Confinement Event Tree
CET 1,2...	Confinement Event Tree Sequence Number(s) or End States
CRW	Circulating Raw Water
CsI	Cesium Iodide
CV	Check Valve
DA	Discontinuous Action (Actions following reactor trip, Section 3.8.1)
DC	Direct Current
DOE	U. S. Department of Energy
ECCS	Emergency Core Cooling System
EIR	Early Isolation Release (for subsets e.g., EIR 1,2,3...see Table 8-2)
ESFAS	Engineered Safety Features Actuation System
EUR	Early Unfiltered Release (for subsets e.g., EUR 1,2,3...see Table 8-2)
EVENT "__"	Event Tree Headings Defined in Sections 5.0 and 7.0, and on Event Tree Diagrams
FFTF	Fast Flux Test Facility
g	Acceleration of Gravity, Appx. 32.2 F/sec^2
GSCS	Graphite and Shield Cooling System
H&V	Heating and Ventilation
HCR	Horizontal Control Rod
HEDL	Hanford Engineering Development Laboratory
HEPA	High Efficiency Particulate Air (Filters)
HLDP	High Lift Diesel Pumps
HMS	Hanford Meteorology Station
HPI	High Pressure Injection System
HPRW	High Pressure Raw Water System
I_2	Elemental Iodine
ID	Inside Diameter
kW	Kilowatt
LACE	Light Water Aerosol Containment Experiments
LANL	Los Alamos National Laboratory
LLDP	Low Lift Diesel Pumps
LLNL	Laurence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LPRW	Low Pressure Raw Water (Supply)
LUR	Late Unfiltered Release (for subsets e.g., LUR 1,2,3...see Table 8-2)
MM	Modified Mercalli (Seismic Intensity Scale)
MTU	Metric Ton Uranium
MVA	Megavolt - amperes
MW	Megawatt

MWD	Megawatt Days
MWt	Megawatt Thermal
N "__"	Source Term Release Categories, Section 8.3
NUSAR	N Reactor Updated Safety Analysis Report
NRC	U. S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OD	Outside Diameter
PAS	Plant Accident Sequence
PNL	DOE's Pacific Northwest Laboratory operated by Battelle Memorial Institute
PRA	Probabilistic Risk Assessment
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTS	Reactor Trip System
RWS	Raw Water System
RWS-1	Low Pressure Raw Water System
RWS-2	High Pressure Raw Water System
SAIC	Science Applications International Corporation
SSE	Safe Shutdown Earthquake
USSR	United Soviet Socialist Republic
V	Volt
V-3 VALVE	ECCS Inlet Valve to Reactor
V-4 VALVE	ECCS Discharge Valve from Reactor
W	Fuel Damage End State, $\leq$2% Fuel Damage (See Table 5-2)
W.G.	Water Gate
WHC	Westinghouse Hanford Company
WNP	Washington Nuclear Plant
WPPSS	Washington Public Power Supply System
X	Fuel Core Damage End State, $\leq$6% Fuel Damage (See Table 5-2)
Y	Fuel Core Damage End State, $\leq$30% Fuel Damage (See Table 5-2)
Z	Fuel Core Damage End State, 100% Fuel Damage (See Table 5-2)

DISTRIBUTION

DOE-RL
Defense Production Operations
Division

DP Simonson, Director FED/603

Defense Production Operations
Division
Reactor Operations Branch

KR Absher, Chief FED/210-B

Westinghouse NTSD
P.O. Box 355
Pittsburgh, PA 15230

DF Paddleford

Science Applications International Corp. (8)
2109 Air Park Road S.E.
Albuquerque, NM 87106

AM Kolaczkowski
BJ Shapiro

800 Oak Ridge Turnpike
Oak Ridge, TN 37830

CL Mason

5150 El Camino Real, Suite C-31
Los Altos, CA 94022

JE Kelly
ZT Mendoza

10210 Campus Point Drive
San Diego, CA 92121

JR Penland

802 Martintown Road, Suite 208
N. Augusta, SC 29841

DC Aabye
DH Walker

DISTRIBUTION (Cont)

Los Alamos National Laboratory
P. O. Box 1663, MS-K555
Los Alamos, NM 87545

A. Neuls

Sandia National Laboratories (2)
1515 Eubank, S.E.
Albuquerque, NM 87185

DM Kunsman
AL Camp

University of Washington
Department of Nuclear Engineering
BF-20
Seattle, WA 98195

N_ McCormick

PRD Consulting
1935 Sabin Drive
Idaho Falls, ID 83401

PR Davis

Pacific Northwest Laboratory (5)
P.O. Box 999
Richland, WA 99352

SW Heaberlin	SIGMA3/317
WW Little	SIGMA3/418
JB Martin	RTL/61/3000
RE Nightingale	RTL/520/25
DL Strenge	2400 STV/1214

MAC Technical Services Company
1933 Jadwin Avenue
Richland, WA 99352

LP Leach

DISTRIBUTION (Cont.)

Westinghouse Hanford Company (74)

MJ Anderson	337/2S5A
GR Armstrong	VITRO/L18
WH Arnold	WB-89
KD Birney	VITRO/L18
RM Black	2750-E/C-239
FG Buck	1101N/N17
PA Carlson	VITRO/L4
GA Coles	VITRO/L4
HL Debban	1101N/N5
JJ Dorian	1114N/S3
JS Dukelow	CDC1/221
DB Engelman	1101N/N18
GR Franz	1118N/B5
JM Goff	1125N/A3
LL Grumme	1117N/C1
JW Hagan	4706/107
FC Han	W/B-16
HW Heacock	3200GWW/L5
DI Herborn	CDC2/223
JA Hunter	3200GWW/L17
EN Hurd	3200GWW/L17
WM Jacobi	WX-1
GL Jones	3730/C11
HM Jones	1115N/E2
JP Kester	1101N/N3
MK Korenko	3763/C90
JJ Laidler	3200GWW/L11
EE Leitz	VITRO/L4
MK Mahaffey	309/C27
FJ Martin	VITRO/L18
WJ McShane	337/B12
KR Mikkelsen	1146N/E14
NR Miller	1156N/E20
RL Moffitt	1100N/R7
RD Morrison	1146N/E14
LW Muhlestein	4706/M19
JE Nolan	POB/3000
DM Ogden	VITRO/L18
EV Padgett	3200GWW/L17
CL Peckinpaugh	1117N/C3
RE Petersen	337/B66
PD Powell	326/A61
TB Powers	VITRO/L4
PR Praetorius	1104N/E9
DR Pratt (6)	1117N/N2
RJ Pyzel	1101N/N6
WJ Quapp	VITRO/L18
SP Roblyer	1117N/C12
RC Ross	3200GWW/L5
WG Ruff	1117N/C4
JP Schmidt	1117N/C4
LK Severud	337/B20
WF Sheely	3763/15
DE Simpson	337/3N
GL Smith	1118N/B2
DA Strege	1115N/26
JA Theis	CDC1/216
H. Toffer	FED/F26
VP Treadway	VITRO/200
AM Umek	1103N/T4
AE Waltar	326/A54
RD Warner	1100N/R8
SA Weber	1117N/C3
AD Wilcox	FED/F26
MD Zentner	VITRO/L4
Doc. Control (3)	W/C-123