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PROJECT ADMINISTRATION

Clinch River Breeder Reactor Plant

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AN EXECUTIVE SUMMARY OF THE PSAR FOR THE CLINCH RIVER BREEDER REACTOR PLANT

FEBRUARY 1975

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Prepared for the Project Management Corporation as part of the U.S. Atomic Energy Commission Liquid Metal Fast Breeder Reactor Demonstration Program

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1. INTRODUCTION

The purpose of this Executive Summary is to present, in a readily assimilable form, the salient features of the CRBRP PSAR. To achieve this, the Summary is subdivided into two Parts and is supplemented by four Appendices.

In Part I are presented, in condensed form, the important contents of each PSAR Chapter and Appendix. Each summary is intended to convey:

a. An overview of the material contained in the Chapter or Appendix.
b. An appreciation of the more significant commitments that the Project is making in that Chapter with emphasis on changes made since the Reference Design Report was issued.
c. Any other items worthy of note in relation to that Chapter. (Note that, although third level design margin requirements are treated in each Chapter of the PSAR, in this Summary they are given in Chapter 15 and are not mentioned in the other Chapter Summaries).

In Part II are given synopses of items of special interest from the Licensing viewpoint. The majority of these are treating matters which cut across a number of Chapters and it has been considered appropriate to focus on each of these items outside the framework of the Chapters of the PSAR.

Appendix A of this Summary comprises an audit of the CRBRP PSAR against the Regulatory Guide on Standard Format and Content of Safety Analysis Reports, LMFBR Edition. Throughout this Summary, that Guide is termed the SFAC, the acronym by which it is most generally known within the Project. Every attempt has been made to comply with the requirements of the SFAC, as modified by agreements reached with the Directorate of Licensing on March 28, 1974. The modifications agreed at that meeting are itemized in Reference 1, and this Reference is known among Project participants as 'the Errata to the SFAC'; in this Summary it will be referred to simply as 'the Errata'. In some cases, as will be seen from Appendix A, it has not been possible to comply fully with the SFAC or Errata requirements and reasons are given for each instance of non-compliance.

Appendix B comprises an assessment of the applicability of Regulatory Guides to CRBR. A previous assessment was issued in Reference 2. Appendix B has modified Reference 2 to incorporate comments received from the Directorate of Licensing in Reference 3 and has extended the assessment to cover certain of the more recently issued guides.

Appendix C has been extracted from Section 1.1 of the PSAR. It is this Section of the PSAR which establishes the overall basis for the Construction Project application. Since it is brief, and readily assimilable, and because of its importance to the role of the PSAR, it has been considered worthy of reproduction here in its entirety.

Appendix D gives an overview of the Reliability Program, including material extracted directly from Appendix C of the PSAR.

2. CONCLUSIONS

A considerable volume of material is given in the PSAR, which in many instances, significantly exceeds the level of detail provided in commercial reactor PSAR's. There are items required by the SFAC which are not provided in this PSAR, as shown in Appendix A of this Summary; however, many items not supplied are omitted by agreement with Regulatory. The remaining items are, in some cases, not appropriate to the CRBR design and in others are not such as to represent docketing issues.

As an Application for a Construction Permit for the first LMFBR to be licensed in this country, this PSAR represents an adequate body of information. However it should be recognized that there are areas in which supplementary material will be required as the design evolves; in this respect the CRBRP application is no different from that of any other plant. It is also worth noting that a more realistic comparison than that with an LWR PSAR would be that with the Summit PSAR (Delmarva's HTGR), on which a construction permit is about to be awarded. In that instance a number of items were permitted to be left unresolved at the CP stage because of the stage of evolution of a first-of-a-kind plant.

The QA measures necessary to comply with RDT Standard F2-2 have been taken; all affected design groups have been actively involved in preparation of the PSAR and are satisfied with the presentation of the design therein. However, it must be recognized that, with a rapidly evolving design, it is not possible for the PSAR to be completely up to date in all respects; some of the design material presented is no longer current and will need to be updated by future supplements. In no case are the design changes involved such as to necessitate a radical departure from the established Project position.

It is ARD's clear recommendation that the PSAR be approved, for release to final print and submittal to Regulatory. The final editing of Appendices C and F can be completed, with customer participation, during the period that the remainder is in print.

References

1. Letter P. Bradbury to A. E. Mravca, "PSAR Responsibilities, Format and Content", WR 40285, May 6, 1974.
2. Letter P. Bradbury to A. E. Mravca, "Preliminary Review of Existing AEC Regulatory Guides for Applicability to CRBR", WR 40268, April 30, 1974.
3. Letter R. P. Denise to P. S. Van Nort, "Project No. 471, Regulatory Guides Applicability to LMFBR", March 27, 1974.

PART I: CHAPTER AND APPENDIX SUMMARIES

CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.A Summary

In Chapter 1 of the PSAR the applicant establishes the overall basis for the Construction Permit application, gives certain general information regarding the plant and the organizations responsible for its design and construction, and identifies key items of research and development work necessary. Section 1.1 of the PSAR, which is only a few pages in length, is reproduced in its entirety as Appendix C of this Summary, and will not be mentioned further here.

Sections 1.2 and 1.3 comprise respectively a general overview of the plant design (including GA drawings, taken from the Reference Design Report, but updated to include the Overflow Heat Removal Service (OHRS) and the lower reactor cavity) and a comparison of the more important parameters with those of other LMFBR's in the world. Also included is a fairly detailed comparison, in tabular form, of the CRBR and FFTF designs.

Section 1.4 details the various organizations participating in the Project (PMC/RRD/TVA/ARD/GE/AI/Burns and Roe), and their interrelationships.

Section 1.5 itemizes the safety related research and development programs, identified on Table 1.5.1 of the PSAR which is reproduced below. For each of these programs a criterion of success is identified and potential fallback options, in the event the program produces an unexpected result, are given. Each item also contains a schedule, with milestones, indicating that the majority of the work will be completed before issuance of a Construction Permit and all of it before issuance of an Operating License.

1.B Key Commitments

The key commitments entered into in Chapter 1 are:

a. Design power level 975 MW, stretch power level is 1121 MW. It is not intended that the CP be obtained on the basis of stretch power capability.
b. Two loop operation is intended, but the power level is not a CP issue.
c. The organizational arrangements are:

PMC	Overall integration and technical direction of BOP activities
RRD	Technical direction of NSSS activities
TVA	Ultimate owner and part licensee
ARD	Lead Reactor Manufacturer for NSSS
GE	Subcontractor to ARD
AI	Subcontractor to ARD
Burns and Roe	Architect Engineer

The relationship between the customer organizations is indicated diagrammatically on Figure 1-1 below.

d. Reliability goal of 10^{-6}/yr for loss of core coolable geometry from all initiators, or the acceptance of core disruption as a design basis accident.

e. Intent to demonstrate incredibility of massive failures of the primary coolant boundary, or implementation of a fallback design.
f. Performance of all tests listed in Table 1.5-1 as a necessary part of the demonstration of the safety of the plant. This includes implementation of any fallbacks found to be necessary, as a result of the tests.

1.C Other Special Items

The major item of interest is the structure of the parallel design approach, covered in Section 1.1 and reproduced in Appendix C of this Summary.

PMC BOARD

PSC

(CECO) (TVA) (AEC)

PMC GENERAL MANAGER

TVA
(Operations)

ASSIGNED LEAD ROLE INCLUDES:

- Supervision of plant safety, operation & maintenance
- Site & transportation facilities
- Support for safety, environment & licensing matters
- Purchasing energy
- O&M input to design and construction
- O&M training program
- Operator licenses
- O&M budget, costs & schedules
- Operation services
- Reporting on O&M cost & schedule
- On-site fuel management
- Integrating overall test, oper., maintenance, & demo programs

PMC
(Overall Integration & BOP)

ASSIGNED LEAD ROLE INCLUDES:

- Integration of overall plant activities per project requirements
- BOP per approved project requirements
- Technical supervision & contract administration of BOP activities
- Contract services for the plant
- BOP budgets, costs & schedules
- Integration of overall budgets, costs & schedules
- Administration of project funds
- Site evaluation program
- Safety, licensing & environmental matters
- Coordinating overall test, operational & demo requirements
- Test, operating & maintenance procedures for BOP

(NSSS-LRP)

ASSIGNED LEAD ROLE INCLUDES:

- NSSS per approved project requirements
- Tech. supervision & contract admin. of NSSS activities
- Support from base program
- Reporting on NSSS cost and schedules
- Supporting safety, licensing & other technical matters
- Test, operating, maintenance, & demo procedures for NSSS
- Providing fuel
- Reflecting project results into base program
- NSSS budget, costs & schedules

ADDITIONALLY TVA, CECO, PMC AND AEC WILL:

- Provide timely reports and requested information
- Provide notice of problem areas and significant decisions
- Avoid duplicating capabilities of other participants
- Make maximum use of available expertise
- Provide opportunity for reviews, inspections and audits
- Provide resources to meet commitments
- Provide means of assuring that commitments are being met
- Provide documentation of specific detailed plans, standards, practices, designs and procedures
- Develop formal procedures to implement the foregoing

Figure 1-1. Project Lead Roles

TABLE 1.5-1

FURTHER TECHNICAL INFORMATION REQUIRED

PSAR SECTION	SECTION HEADING & TASKS
1.5	Introduction
1.5.1	Information Concerning the Adequacy of a New Design
1.5.1.1	Shutdown Systems Reliability
1.5.1.2	Decay Heat Removal & Structural Reliability
1.5.1.3	Secondary Control Rod System Test
	Guide Tube Verification
	Latch System Tests
1.5.1.4	Overflow Heat Removal System Test
1.5.1.5	Radial Blanket Failure Threshold
	Failed Radial Blanket Rod Evaluations
	Radial Blanket Assembly Local Flow Blockage Evaluation
1.5.1.6	Sodium-Water Reaction Pressure Relief Test
1.5.2	Information Concerning Margin of Conservatism of Proven Design
1.5.2.1	Pipe Integrity Assessment
	Fracture Mechanics Study
	Characteristics of Sodium-Induced Corrosion
	Pipe Reliability
	Sodium Leak Detection Feature Test
1.5.2.2	Reactor Thermal & Hydraulic Tests
	Large Bundle Partial Blockages Evaluations
	Inlet Module Blockage Prevention Test
	Inlet Plenum Bubble Dispersion Test
	Inlet Plenum Particle Mobility Test
1.5.2.3	Core Restraint System Tests
	Full-Core Restraint System Test
1.5.2.4	Critical Experiments for Reactivity Coefficients, Control Rod Worth & Fuel Assembly Movement
1.5.2.5	Source Range Flux Monitoring System Tests
1.5.2.6	Ex-Vessel Transfer Machine Heat Removal Tests

CHAPTER 2 SITE CHARACTERISTICS

This Chapter was submitted to Regulatory, with the Environmental Report, in October 1974 and is not addressed in this Summary. However, two items relating to Chapter 2, Site Meteorology and Site Seismology, are addressed in Part II of this Summary, since they are of special interest to the Project Managers.

CHAPTER 3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.A Summary

This chapter identifies all the plant features important to safety and describes their design criteria, design and analysis procedures, and applicable codes/standards/specifications. Information on the testing and surveillance requirements for these plant features is also provided in this chapter.

A set of CRBRP General Design Criteria (GDC) is presented and the conformance by the plant design is discussed in detail. These are based on the 10CFR50 criteria but are not identical to them, because of design differences between LMFBR's and LWR's. The modifications have, with few exceptions, already been agreed with REG.

A Safety-Classification system, specially developed for this plant and comparable to LWR practice, is presented in this chapter. A summary table of all the safety-related systems, equipment, and structures, their safety classes, applicable and actually-used code classes is provided.

Wind and tornado loadings are quoted. The design basis tornado is defined, in accordance with REG Guide 1.76, as having 360 mph velocity (290 rotational and 70 translational). The radius of maximum rotational wind is quoted as 150 ft. and a pressure drop of 3.0 psi at 2.0 psi/sec is specified. Missiles are specified, resulting from the tornado, identical to those used for Sequoyah. The design basis wind is quoted as 90 mph, consistent with ANSI A58.1-1972.

Flood protection, against a maximum flood level (MFL) at Elevation 815', is described. All seismic Category I items are protected, either by elevation or by waterproofing of structures. The structures themselves are demonstrated as being capable of withstanding the hydrostatic forces resulting from the flood.

A fairly detailed discussion of missile protection is included. This considers rotational missiles from the following sources:

- Winds and tornadoes (see above)
- Turbine failure (no details available)
- PHTS pump missiles (retain within pump tank)
- IHTS pump missiles (retain within pump tank)
- Steam generator recirculating pump missiles (retain within the system)
- SGAHRS missiles (none expected, rationale given in PSAR)

Also considered are pressure generated missiles from the steam generator and SGAHRS. Table 3.5-3 of the PSAR, reproduced below, indicates the type of information given. Methods of analysis of missile effects (including equations used), and means of protection are discussed.

Pipe whip, for the PHTS or IHTS, is not a major concern, because of the low pressures in those systems; this point is made. However, in the steam systems, where pressures are much higher, this is a significant consideration, and a detailed discussion of pipe whip analyses, and protective measures against the consequences of pipe whip, is given.

The remainder of Chapter 3, about 100 pages, comprises a treatment of the various elements of the seismic design of the plant, and is supported by the inclusion of the Seismic Design Criteria (WARD-D-0037) as an Appendix. The SSE is defined as 0.18g, the OBE as 0.09g. Included in these portions are:

- Seismic response spectra (vertical and horizontal, for a range of damping values).
- Damping values to be used in dynamic analysis
- Soil structure interaction
- Methods of seismic analysis for systems, structures and components
- Seismic instrumentation
- Methods of control of the seismic design
- Design of Category I structures

3.B Key Commitments

Apart from items already covered in 3.A, the following key commitments are made in Chapter 3.

a. Load combinations that will be used in design of internal structures. Some of the most significant are:

$$\left.\begin{array}{l} U = D + L + To + A \\ U = 0.9\ D + TO + A \end{array}\right\} \text{ for concrete structures}$$

$$1.65 = D + L + To + A \text{ - for steel structures}$$

Where U is concrete section strength requirement
D is dead load
L is live load
To is thermal load
A is HCDA load
S is steel section strength requirement

b. List of Seismic Category I structures:

Reactor Containment Building
Reactor Service Building
Steam Generator Building
Diesel Generator Building
Emergency Cooling Tower Basin
Diesel Fuel Storage Tank Foundation
Electrical manholes (for safety related cables)

3.C Other Special Items

1. Plans and approaches for certain mechanical and electrical systems and/or components qualifications are not yet fully developed or confined. More work will be required in this area, and is being initiated.

2. There are a number of areas in which the effects of accidents on the operation of safety related components have not been fully covered as yet. This is treated as a special item in Part II of this Summary.

3. Reg. Guides 1.70.9 and 1.70.10 were issued late November 1974. These require additional information relating to design of Seismic Category I structures and wind and tornado loadings. Some of this material is not supplied, this is considered reasonable in view of the recent date of these Reg. Guides.

4. Reg. Guide 17a16 (Missile Barrier Design Procedures) was issued in December, 1974. Some of the material required by this Guide is not supplied, this is considered reasonable in view of the recent date of this Reg. Guide.

CHAPTER 4 REACTOR

4.A Summary

This Chapter covers the reactor vessel internals. The design presented in this Chapter is summarized below.

A schematic elevation of the reactor is shown in Figure 4-1. In addition to the vessel internals described in this Chapter, this figure also identifies the reactor vessel, closure head and inlet and outlet nozzles discussed in Chapter 5. The reactor internals are comprised of removeable fuel, blanket, and control assemblies, removeable radial shielding and the upper and lower internals structures which provide support and positioning for the core and the core restraint system.

The lower internals consists of the core support plate and cone, the core barrel, horizontal baffle, fixed radial shielding, and inlet and modules. Most of these components are shown in Figure 4-2. The core barrel provides support for the upper and lower core restraint former rings and the bypass modules provide support for the removable radial shielding. Together these comprise the core restraint system. The lower internals structure is welded into the reactor vessel.

The upper internals structure consists primarily of the four lifting columns, two transverse interconnected plates and thirty-five outlet modules and flow chimneys. This structure, which is shown in Figure 4-3, provides lateral stabilization for the control rod shroud and outlet module flow tubes, supports the in-vessel instrumentation and provides mechanical backup holddown for the core assemblies. The shroud and flow conduits are designed to mitigate transient temperature effects on the structure from the reactor core effluent. The upper internals structure is supported from the intermediate rotating plug of the vessel closure and is radially keyed to the upper core restraint former ring attached to the core barrel.

The active fuelled region is 36 inches long and the equivalent diameter is 73.6 inches. The fuel region consists of two radial enrichment zones with a total initial fissile plutonium loading of ∿1150 Kg. The reactor has two independent, diverse, fast acting control systems. The primary system has 15 mechanically scram assisted control rods while the secondary system has 4 hydraulically scram assisted control rods. Each system is independently capable of shutting down the reactor from full power to hot standby conditions. Each of the core assemblies and the removeable radial shields have two load pad areas which match the elevation of the core restraint former rings to position the core and restrain core assembly motion during operation. The fuel, blanket and control assemblies each contain a tag gas to permit detection and identification of failed elements. Fuel transfer and storage positions are provided in the annulus between the core barrel and the reactor vessel. A plan view of the reactor details is shown in Figure 4-4.

In addition to providing a detailed description of the reactor design, Chapter 4 also provides the nuclear, thermal-hydraulic and structural analysis results to support the discussed design features. Where final analyses are not available, the plans for future efforts to complete the required analysis are presented.

4.B Changes Since Reference Design Report

1. A new upper internals configuration, with improved structural strength, from that presented in the Reference Design Report is presented in the PSAR. The reason for the change in design is that the structural analysis of the Reference Design Report configuration indicated that design would not perform its required functions for the lifetime (30 years).

2. A new core support structure design to provide anti-blockage features has been incorporated into the reactor design since the Reference Design Report. This design as presented in the PSAR prevents large debris from completely blocking flow to any of the inlet modules. The design presented does not include the post accident debris retention features which are under discussion within the Project.

3. An increase in the diagnostic instrumentation coverage to provide one thermocouple at the outlet of each fuel and radial blanket assembly has been provided. This expansion of the in-vessel instrumentation has occurred since the Reference Design Report.

4. A design to increase the B_{10} loading in the secondary control rods from the Reference Design Report value of 3.5 Kg to 4.0 Kg is being developed. The nuclear analysis presented in the PSAR is based on this 4 Kg loading. The increased loading is added to provide flexibility to assure additional shutdown margin in the event that more detailed analysis or increased fuel loadings indicate such margin is necessary.

5. The Reference Design Report indicated that LWR discharge plutonium would be used as the CRBRP fuel. Because the capability to fabricate this plutonium will not be available in time for the first cores, the first core feeds will utilize FFTF plutonium. The impact on nuclear performance due to the use of FFTF grade plutonium is discussed in Section 4.3, and shown to be insignificant with respect to safety related considerations.

6. In addition to the expanded instrumentation discussed in Item 3 above, the control assemblies will now contain tag gas to permit detection of failed poison elements.

4.C Other Special Items

Other than those design changes discussed in the preceding section, there have been no new commitments generated by Chapter 4 of the PSAR. However, since the last draft of the PSAR, REG Guide 1.70.12* has been issued. Implementation of this guide into the PSAR will require an estimated 3 to 6

*Regulatory Guide 1.70.12 Information for Safety Analysis Reports, Reactor Materials, Issued December 1974.

man-months of effort to compile and collect the required information on the materials to be utilized (for example, a description of the processes, inspections and tests on austenitic stainless steel components of the control rod system, to assure freedom from increases susceptibility to intergrannular stress-corrosion cracking), in the control and reactor internals systems. ARD proposes not to delay the PSAR to incorporate this material but to initiate efforts so that this information can be provided during the Q1 and Q2 periods.

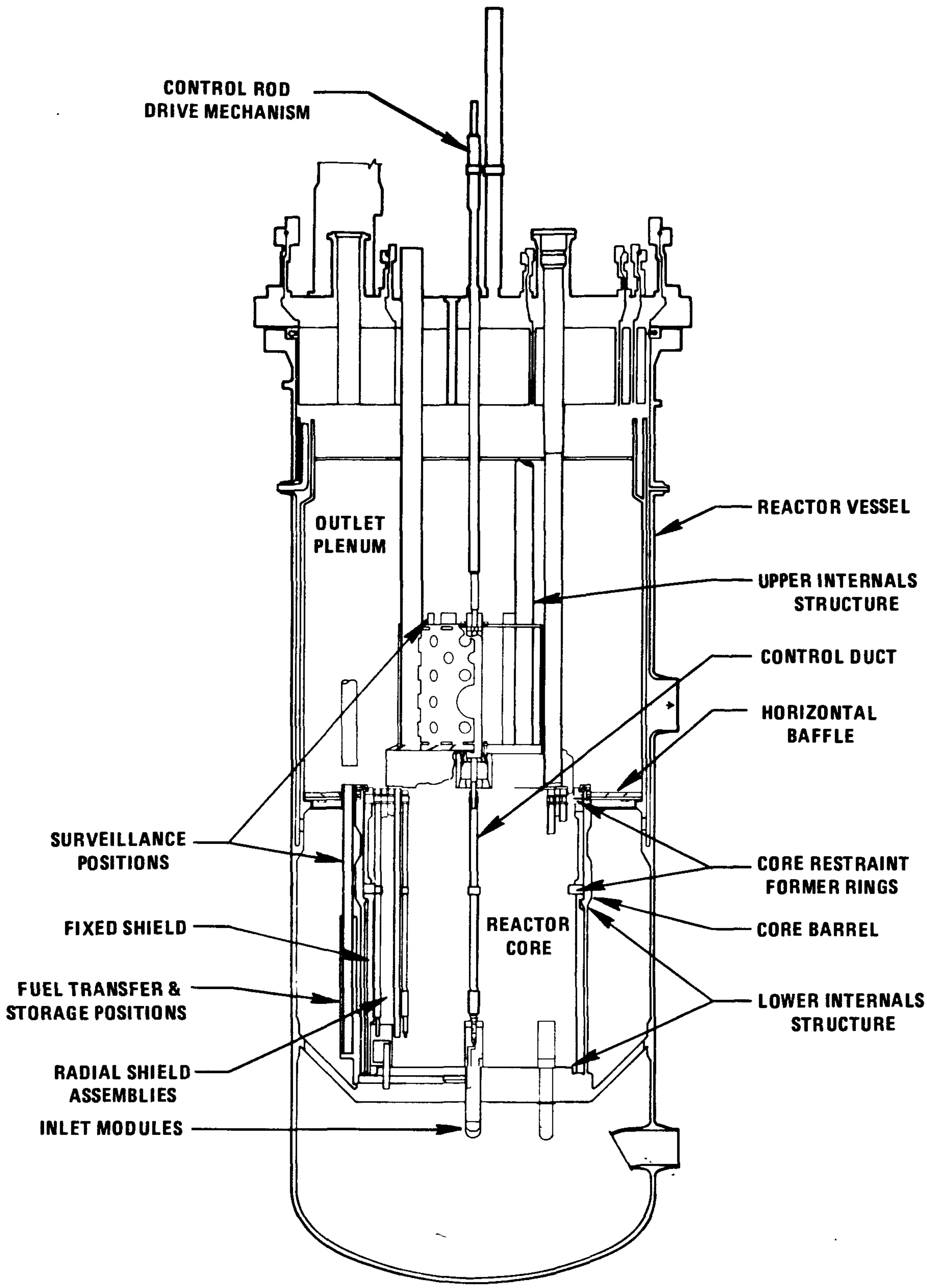


Figure 4-1. Reactor Elevation

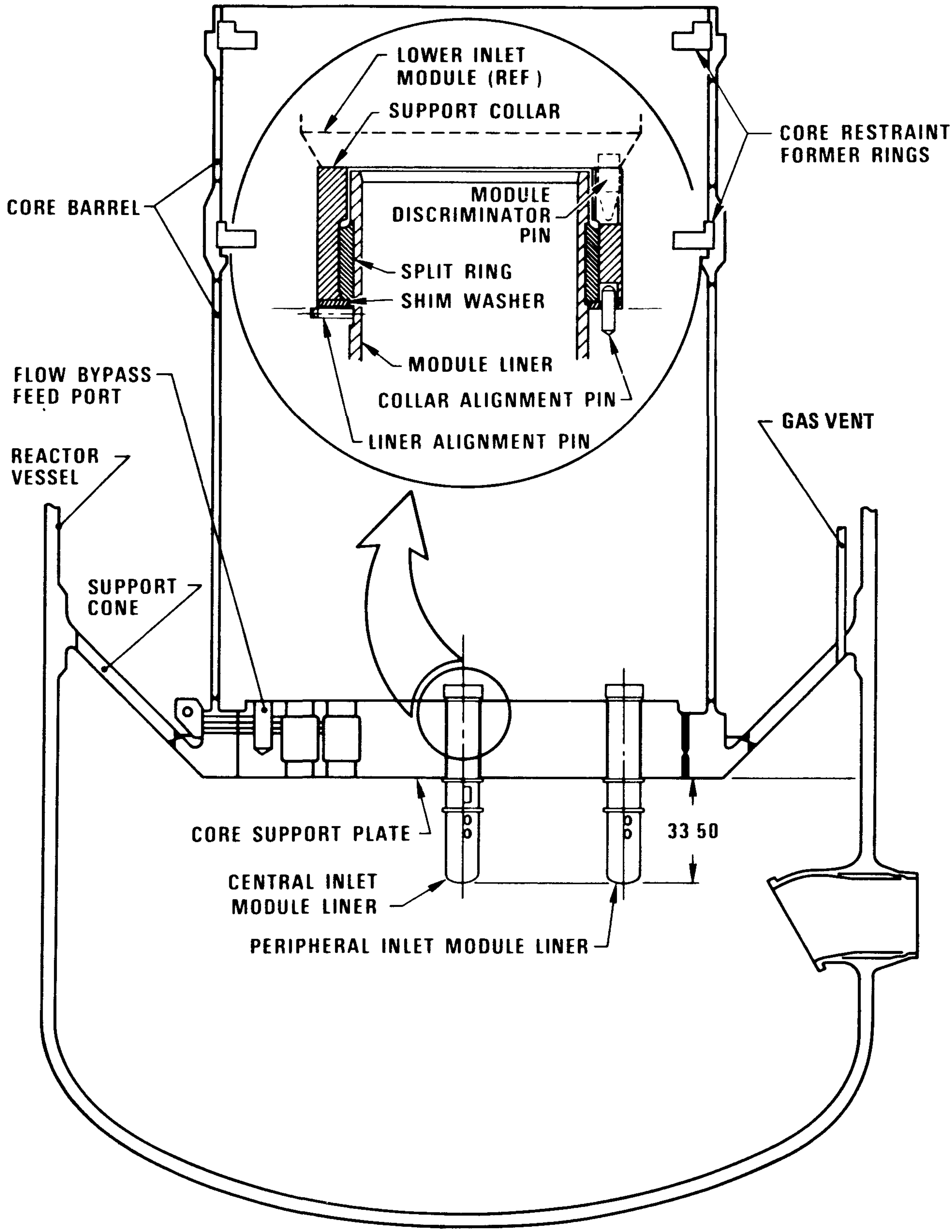


Figure 4-2. Lower Internals Structure

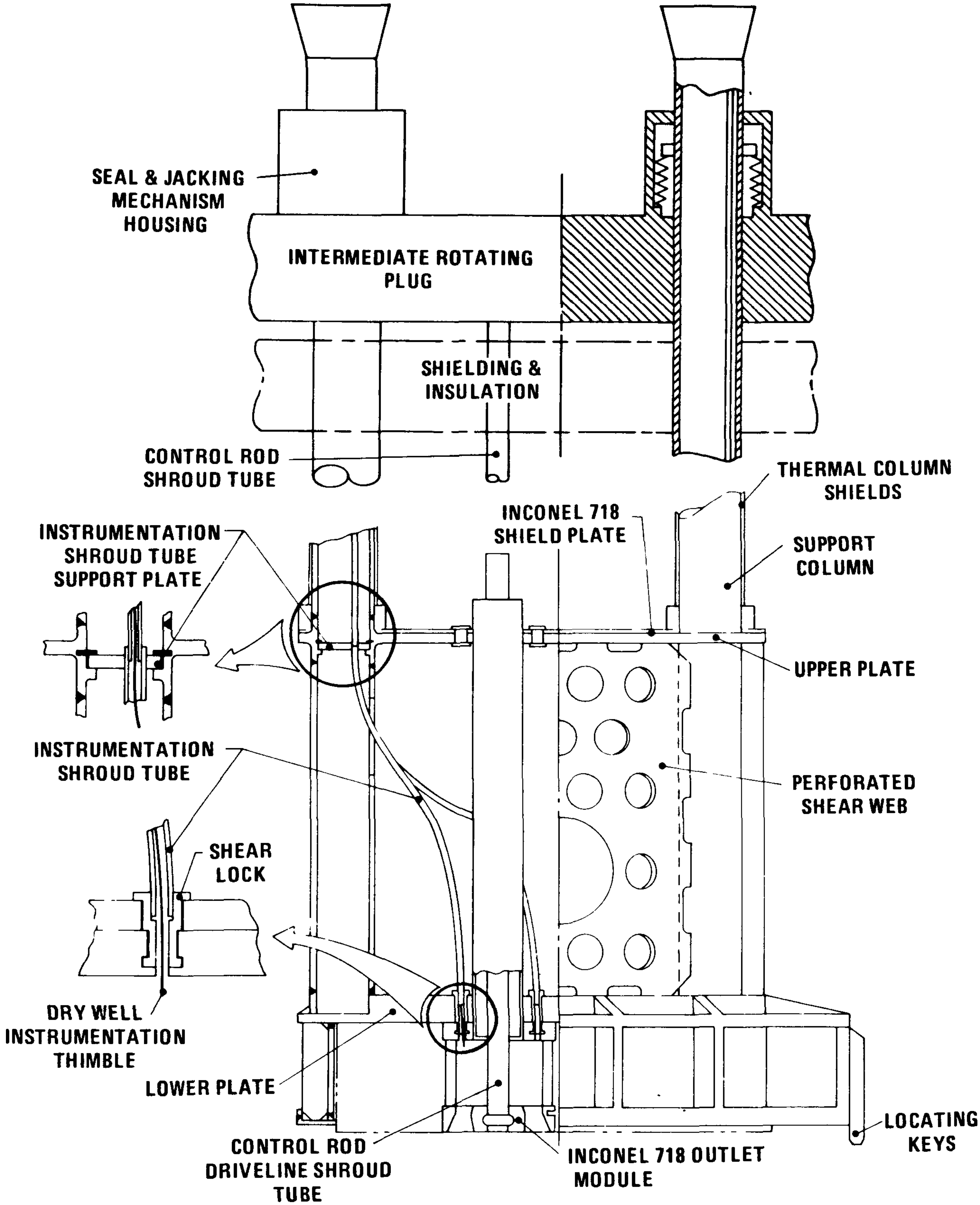
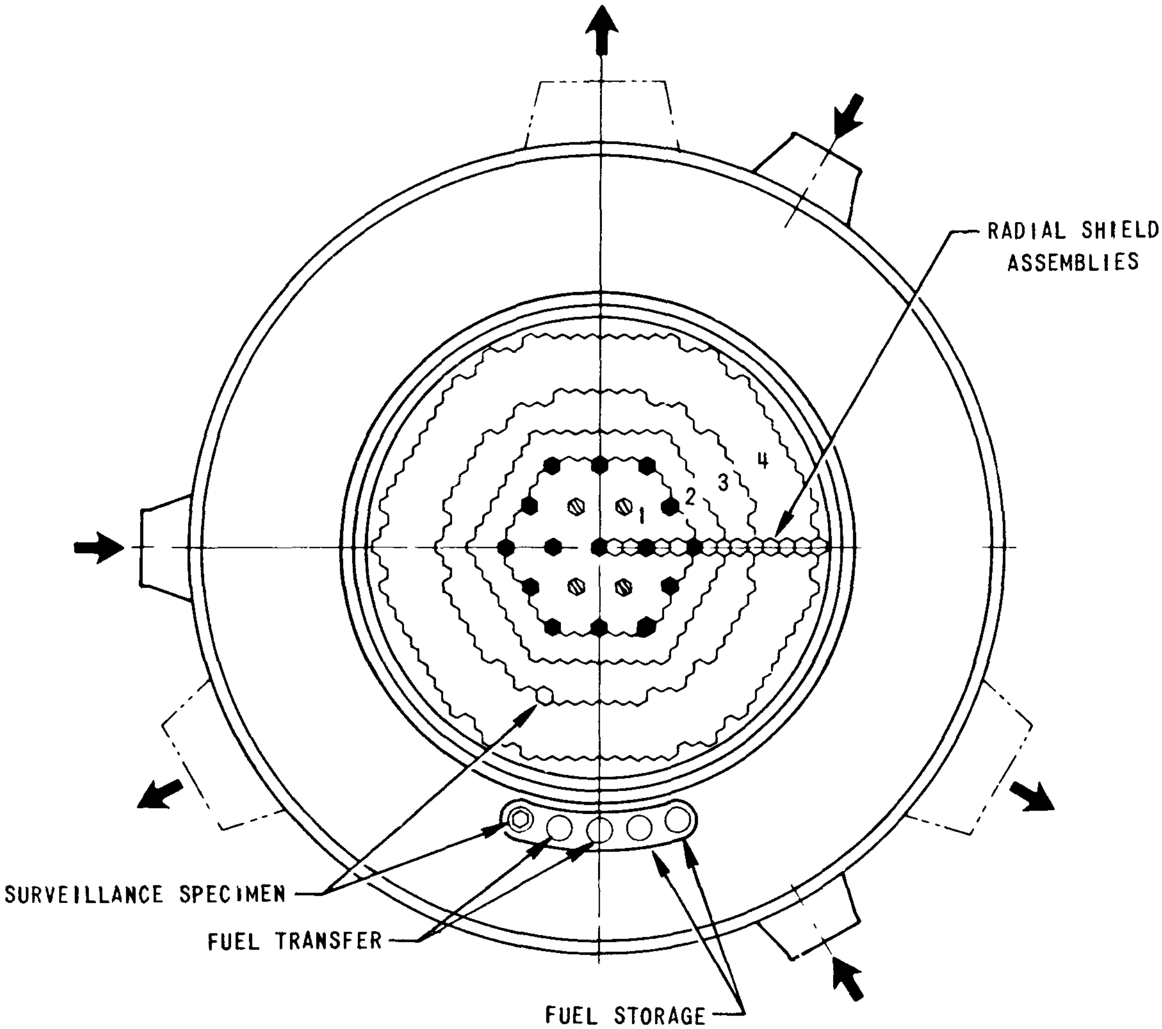


Figure 4-3. Upper Internals Structure



CORE MAP LEGEND

	TYPE OF ASSY	NO.
1	INNER CORE FUEL	108
2	OUTER CORE FUEL	90
3	RADIAL BLANKET	150
4	RADIAL SHIELDING	324
⬢	PRIMARY CONTROL	15
⊘	SECONDARY CONTROL	4

Figure 4-4 Plan View of Reactor

CHAPTER 5 HEAT TRANSPORT AND CONNECTED SYSTEMS

5.A Summary

Chapter 5 contains (1) the design bases, (2) system design description and (3) design evaluation of the following systems:

Section	
5.2	Reactor Vessel, Closure Head and Guard Vessel
5.3	Primary Heat transport System (PHTS)
5.4	Intermediate Heat Transport System (IHTS)
5.5	Steam Generation System (SGS)
	Including Sodium Water Reaction Pressure Relief Subsystem (SWRPRS)
	Sodium Dump Subsystem
	Water Dump Subsystem
5.6	Residual Heat Removal Systems
	Including Steam Generator Auxiliary Heat Removal System (SGAHRS)
	Overflow Heat Removal Service (OHRS)

The design bases sections typically include performance objectives (e.g., natural circulation, two loop operation), seismic criteria, applicable codes (e.g., ASME, Section III, Class 1 Code cases, Case 1592), surveillance requirements, materials considerations (e.g., welding filler metal requirements), and instrumentation requirements, (usually a brief statement and reference to Chapter 7).

The system design description sections typically include discussion of design methods, material properties, surveillance and in-service inspection programs, components and leak detection systems.

The major emphasis of the Chapter is on the design evaluation portions. The methods, data, assumptions and criteria to be used in system evaluation are given. The results of the analyses themselves will be available for inclusion in the FSAR. Consideration is generally given but not limited to stress evaluation plans, pump speed and integrity, operation of valves, component support, thermal and hydraulic characteristics of components, coolant boundary integrity, IHX and steam generator module tube leaks, materials compatibility and performance and pressure relief provisions.

A portion of Section 5.2, "Features for Improved Reliability", includes discussions of Reactor Vessel Thermal and Nozzle Liners, Internal Elbows in the Inlet Plenum, Closure Head Crush Tubes, Plug Seals, the Omega Seal and Surveillance and Inservice Inspection.

The descriptions of the components which make up the OHRS are provided in Section 9.3 of the PSAR as part of the Auxiliary Liquid Metal System. Section 5.6 gives the bases and describes the operation of those components for heat removal service.

In addition to addressing the reactor vessel closure head, and heat transport system themselves, the Chapter provides an overall system evaluation including startup and shutdown, load following characteristics, transient effects and a preliminary summary of the plant design duty cycle.

5.B Changes Since Reference Design Report

1. The Recirculating Feedwater Heater has been omitted. The effectiveness of the recirculating feedwater heater of SGAHRS in reducing thermal transients in the steam drum and evaporator has been questioned, and it may have a

detrimental effect on reliability. The heater is on HOLD in SDD 52 and has been omitted in the PSAR, by Project agreement.

2. The individual capacity of the two electric motor driven auxiliary feedwater pumps of SGAHRS is stated in SDD 52 and the PSAR as 50% of required flow rather than 100% as in the Reference Design Report. The current arrangement of two 50% capacity, motor driven pumps and one 100% capacity, turbine driven pump, meets the AEC criteria for "initiating event plus one failure". (There is the implication that a pump cannot be taken out of service during plant operation.)

3. The isolation valve in the water/steam piping at the outlet of each evaporator was deleted in the PSAR and SDD 53. It was found that this leads to less severe thermal transients in the duty cycle with no other impact on plant safety.

4. The piping wall thickness of the 24" PHTS cold leg was increased from 0.375 to 0.500 inches in SDD 51 and the PSAR.

5.C Other Special Items

Areas of particular interest in Chapter 5 include: surveillance and in-service inspection, mitigation of the consequences of reactor coolant boundary leaks, the PHTS "leak-before-break" position, natural circulation, mitigation of the consequences of sodium water reactors and provisions for decay heat removal. The last item is discussed as a separate item in Part II of this Summary. A brief description of the PSAR treatment of each of the other items follows.

a. Surveillance and In-Service Inspection (SISI)

The intention to expose surveillance samples of reactor vessel materials, either inside or outside the reactor vessel itself, to provide a means of monitoring potential material degradation is stated, though the surveillance program is not described in detail. The applicability of Appendix H of 10CFR50 is shown to be limited since that Appendix was written for ferritic materials and the CRBRP reactor vessel is austenitic. Visual inspection programs for inside and outside welds of the reactor vessel and the outside of the heat transport systems during plant shutdowns and viewing within the PHTS cells and pipeways to check evidence of sodium leakage and insulation and hanger integrity during plant operation are discussed. General and periodic NDT or other metallurgical inspections are not planned, though the intention to perform such inspections, where and when an opportunity arises, is stated. Any requirement for monitoring surveillance samples of PHTS piping material must be shown by ongoing development programs. If such a requirement is identified, a surveillance program will be developed in accord with the philosophy of Appendix H to 10CFR50.

b. Mitigation of the Consequences of Reactor Coolant Boundary Leaks

The role of guard vessels, check valves, low pony motor shutoff head and elevated piping is to limit the consequences of a leak if that unexpected event should occur.

If a leak occurs in a component or piping within a guard vessel, the vessel will fill with sodium until it reaches a level equal to that in the reactor vessel. The volume between each PHTS component and its guard vessel is sized to prevent the reactor sodium level from dropping below the reactor vessel outlet nozzles and to prevent sodium spillage as a result of pony motor flow. For breaks in certain locations, the check valve prevents the operating pony motors from forcing significant bypass flow out through the inlet nozzle of the breached loop. However, even if the check valve fails, the remaining two pony motors can provide sufficient core cooling.

The only possible location for a leak outside the guard vessels, is in the elevated piping. If such a leak occurs, the sodium level in the reactor remains just below the level of the leak which is higher than the reactor minimum safe sodium level inherent in the elevated piping design.

Coolant spilled outside the guard vessels will fall into either the lined reactor cavity or in the lined cell of one loop which is separated from the cells of other loops. Coolant spilled either in or outside of a guard vessel will spill into an inerted atmosphere which minimizes the degree of combustion that can occur.

c. PHTS "Leak-Before-Break" Position

The project position of "leak before break" in the PHTS is supported by discussion of the PHTS piping materials. Considerations include rigorous QA programs for all phases of design, fabrication, installation and testing, the chemical and radiation environment of the piping, the thermal duty cycle of the system, seismic loadings, dead weight and internal pressure. A pre-existing crack, much larger than that which would be detected and allowed by the standards applied, is shown to extend less than 10^{-6} inches over the life of the plant. In addition, it is shown that even if this prediction of crack propagation were grossly in error, a through wall crack with a length of 15.4 inches for the cold leg and 33.4 inches for the hot leg would be required before the crack would bulge open under operating stresses. Even then, the ends of the crack would not tear in a gross manner to cause a double ended guillotine, or equivalent, failure. The leak detection system development program is referenced, indicating that the system will be capable of detecting a leak before significant corrosion damage from sodium reaction products could occur.

d. Natural Circulation

Adequate transport and removal of decay heat by natural circulation is a performance objective of all heat transport systems as follows:

PHTS and IHTS	3 loops after rated power operation 2 loops after rated or 2/3 rated power operation 1 loop after 2/3 rated power operation
SGS and SGAHRS	1 loop after rated or 2/3 rated power operation (SGAHRS requires forced circulation on its air side to provide the required 15 MWt of cooling in each protected air cooled condenser (PACC). The three loops combined are capable of 15 MWt cooling with natural circulation on the air side.)

Hydraulic profiles of the various systems (See Figure 5-1) and a brief discussion are provided in support of the design, but performance analysis is left for the FSAR.

Mitigation of the Consequences of Sodium Water Reactions

Chapter 15 of the PSAR provides a discussion of the mechanisms of sodium water reaction (SWR) initiation and propagation, showing that the failure of a single steam generator module tube followed by failure of the six surrounding tubes can conservatively be considered the maximum which can occur in the CRBRP.

Chapter 5 describes the assumptions and analysis techniques used to determine the pressure transient in the IHTS components. A table of pressures expected at various points in the system is given for one tube, two tube and seven tube leaks. The IHTS is required to withstand those pressure transients. Analysis to demonstrate that system capability is left to the FSAR.

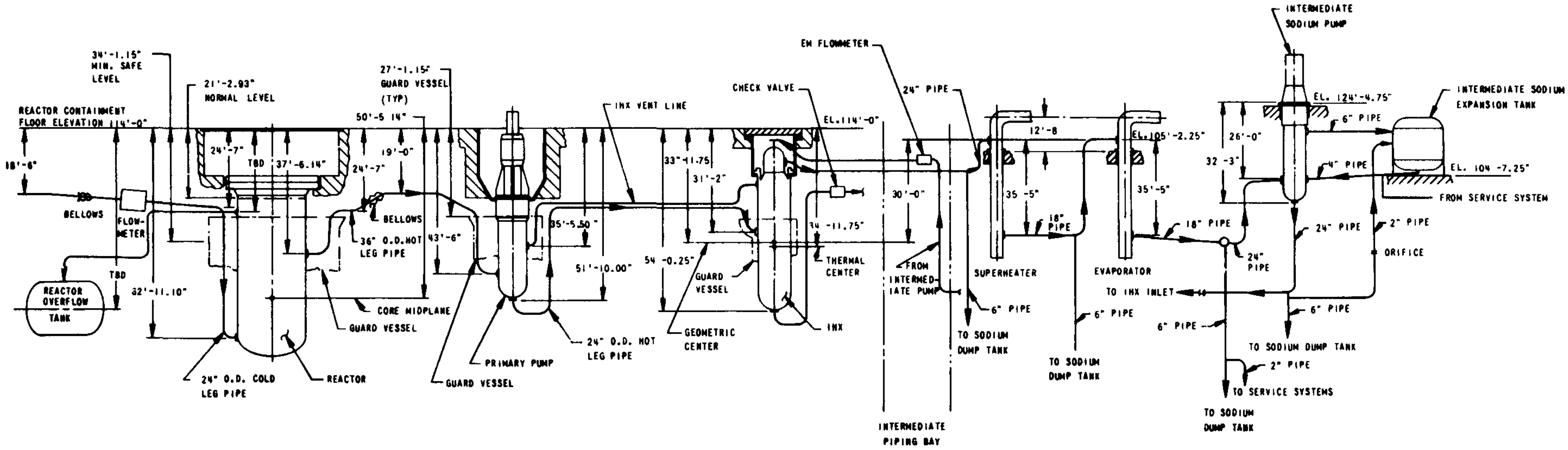


Figure 5.1 HTS Hydraulic Profile

6667-14

CHAPTER 6 ENGINEERED SAFETY FEATURES

6.A Summary

This chapter presents detailed information on three Engineered Safety Features. They are: the Containment System, the Containment Isolation System and the Control Room Habitability System.

The containment functional design is described and the containment design basis accident is identified as a Primary Sodium In-Containment Storage Tank Failure during maintenance. The accident condition in-containment pressure and temperature transients are provided and the calculated radioactivities in the containment atmosphere are presented in Table 6.2-1 (reproduced below). Details of calculated site boundary doses are provided and are shown well below the 10CFR100 guideline exposures. (See Table 6.2-1 below)

The Containment Isolation System design bases and design features are discussed in this chapter. A summary table of the types and numbers and status of isolation valves is provided. The design details of the instrumentation and control equipment of the system are provided in Chapter 7 of the PSAR.

The design of the Habitability System for the control room is described in this chapter including the concrete shielding and the Heating and Ventilating System. The design bases and design features of the system are provided. A detailed design evaluation of the system to demonstrate the capability to meet the General Design Criterion 19 (i.e., to assure access and safe occupancy of the control room under accident conditions) is presented.

6.B Changes Since Reference Design Report

None

6.C Other Items of Interest

The selected design concept of the HVAC system for the Containment uses continuous venting with the containment operating pressure at atmospheric. Most LWR plants with single containment and the FFTF have a nominal subatmospheric containment operating pressure. Such a design feature provides an obvious advantage in that one can easily defend that the control over potential release during normal operation is well assured. The selection of an atmospheric containment operating pressure may call for extra effort in defending the overall containment concept and capability.

The CRBRP Containment System has no provision for post-accident containment heat removal or containment atmosphere cleanup. This is based upon preliminary safety analysis. Needs for post-accident containment atmosphere cleanup system are subject to continuing evaluation. In any event, comprehensive and convincing evidence in terms of design and analysis data must be developed to assure that all potential abnormal conditions have been properly covered.

In addition, a project effort appears to be needed to address the need, if any, for combustible gas control inside the containment.

TABLE 6.2-1

REACTOR CONTAINMENT DESIGN BASIS ACCIDENTS

I. Primary Sodium In-Containment Storage Tank Failure During Maintenance

Na Spill: 32,000 gallons @ 400°F
Pool : 830 Sq. Ft.
Hatch Opening: 21 Sq. Ft.
Max. RCB Pressure: 1.8 psig
Max RCB Wall Temperature: 240°F

POTENTIAL OFF-SITE DOSES

	Dose (Rem)		
Organ	Guidelines of 10CFR100	Site Boundary (0.41 mi) 2-Hour	Low Population Zone (5.0 mi) 30-Days
Beta Skin		1.18E-8*	5.66E-8
Whole Body**	25	4.55E-6	2.17E-5
Thyroid	300	2.61E-5	1.25E-4
Bone	150+	1.21E-4	5.76E-4
Lung	75+	2.83E-5	1.36E-4

*1.18E-8 = 1.18×10^{-8}

**Includes both inhalation and external gamma exposure

+Not covered in 10CFR100; used as guideline values

CHAPTER 7 INSTRUMENTATION AND CONTROLS

7.A Summary

This chapter discusses the Instrumentation and Control Systems provided for the CRBRP. Particular emphasis is placed on discussions of safety related systems, which include the Plant Protection System (PPS) and the safety related display instrumentation required to maintain the plant in a safe shutdown condition. The Plant Protection System includes all equipment necessary to initiate and carry to completion reactor, heat transport and balance of plant shutdown; containment isolation; and decay heat removal. Safety related display instrumentation assures that the operator has sufficient information to perform required manual safety functions and monitor the safety status of the plant. Major control systems not required for safety are described and analyses are included to demonstrate that even gross failure of these systems does not prevent Plant Protection System action. Analyses are also included to demonstrate that the requirements of the AEC General Design Criteria, IEEE Standard 279-1971, applicable AEC Regulatory Guides and other appropriate criteria and standards are satisfied.

The Reactor Shutdown System (RSS) performs the functions of reactor, heat transport and balance of plant shutdown. The Reactor Shutdown System consists of two independent and diverse systems, the Primary and Secondary Shutdown Systems. All anticipated and unlikely events can be terminated by either system without exceeding the specified limits, even if the most reactive control rod in the system cannot be inserted. In addition, the Primary System acting alone can terminate all extremely unlikely events without exceeding specified limits even if the most reactive control rod in the system cannot be inserted. To assure independence of the shutdown systems, mechanical and electrical isolation of redundant components is provided. Functional or equipment diversity is included in the design of instrumentation and electronic equipment, and the Primary Shutdown System uses a different logic design from that of the Secondary Shutdown System. Sufficient redundancy is included in each system to prevent single random failure degradation of either the Primary or Secondary System. Both the Primary Shutdown System and the Secondary Shutdown System are designed to provide on-line testing capability.

A typical Primary Shutdown System Subsystem is shown in Figure 7-1. The Primary Shutdown System is composed of 24 subsystems. Heat transport system pump trip and BOP trip is accomplished by auxiliary circuits from the scram breakers. As shown in Figure 7-1, electrical isolation within the Primary RSS is accomplished by optical coupling, and buffered outputs are provided for non-PPS use of PPS signals.

A typical Secondary Shutdown System is shown in Figure 7-2. In the Secondary Shutdown System, the sensed variables are signal conditioned and compared to specified limits by equipment which is different from the Primary RRS equipment. The secondary logic is configured in general rather than local coincidence to provide additional protection against common mode failure. As shown in Figure 7-2, electrical isolation within the Secondary RSS is accomplished by transformer coupling and buffered outputs are provided for non-PPS use of PPS signals. As for the primary system, heat transport system pump trip and BOP trip is accomplished by auxiliary circuits from the final scram elements.

The Containment Isolation System (CIS) is comprised of redundant instrumentation which senses the need for closure of valves in lines which are directly connected to containment atmosphere (Figure 7-3 shows a block diagram of the system). The CIS is designed for automatic activation of the valves in lines directly connected to the containment atmosphere and valves which require closure in less than 10 minutes to remain within limits (10CFR100 radiological guidelines). When closure is not required in less than 10 minutes, manual actuation is provided. Radiation Sensor are provided in two areas: the exhaust duct of the containment ventilation and the head access area. Three independent, redundant sensors are provided at each location. If the signal is greater than the setpoint, a comparator trip is initiated. The logic for automatic containment isolation is functionally identical to that used in the Secondary Reactor Shutdown System.

All PPS equipment is of quality construction with RDT Standard C16-1T and IEEE Standard 279-1971, the primary controlling documents.

7.B Significant Changes From Reference Design Report

1. The power supply and PPS breaker arrangement for the Heat Transport System Pumps has changed. The reference design report indicated cycloconverters for speed control and the possibility of paralleled breakers for HTS breaker testing. Chapter 7 shows the current design which uses speed control generators (M-G Sets) and a switch for test purposes.

2. The CIS design is defined in greater detail than in the Reference Design Report, and has changed. The containment isolation system described in the reference design report did not identify the sensor locations (inputs) and described two containment isolation (output) functions: (1) principal containment and (2) cell atmosphere processing/nitrogen exhaust containment. The system described in Chapter 7 of the PSAR reflects current design with sensors in the exhaust duct of containment ventilation and the head access area and one output which is principal containment isolation. Many additional design details are also given in this chapter.

7.C Other Special Items

1. The concept of safe shutdown monitoring (and instrumentation) has been introduced. This is the equipment which assures that the operator has sufficient information to monitor the safety status of the plant and maintain it in a safe shutdown condition. Possible examples are primary hot leg temperature and low range nuclear flux. Further design work will be required to completely define requirements in this area.

2. Regulatory Guide 1.75 (Physical Independence of Electric Systems) is introduced and applicable portions can be met. However, Regulatory Guide 1.75 is a potential problem, and is discussed further in Chapter 8.

3. The potential for refueling interlocks being required for safety is introduced in Section 7.6.2. If such safety interlocks are required, the design could be costly.

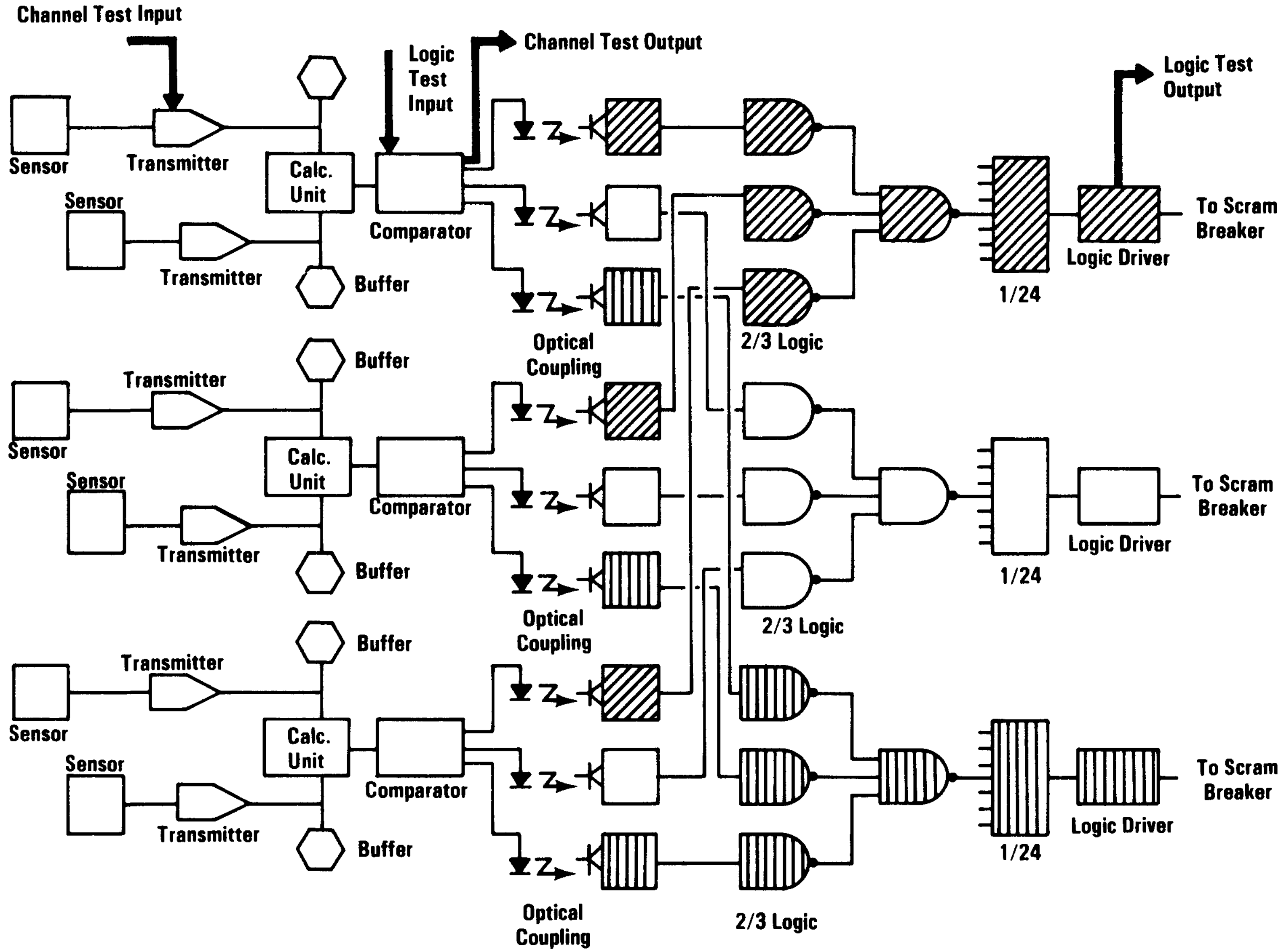


Figure 7-1. Typical Primary Reactor Shutdown System

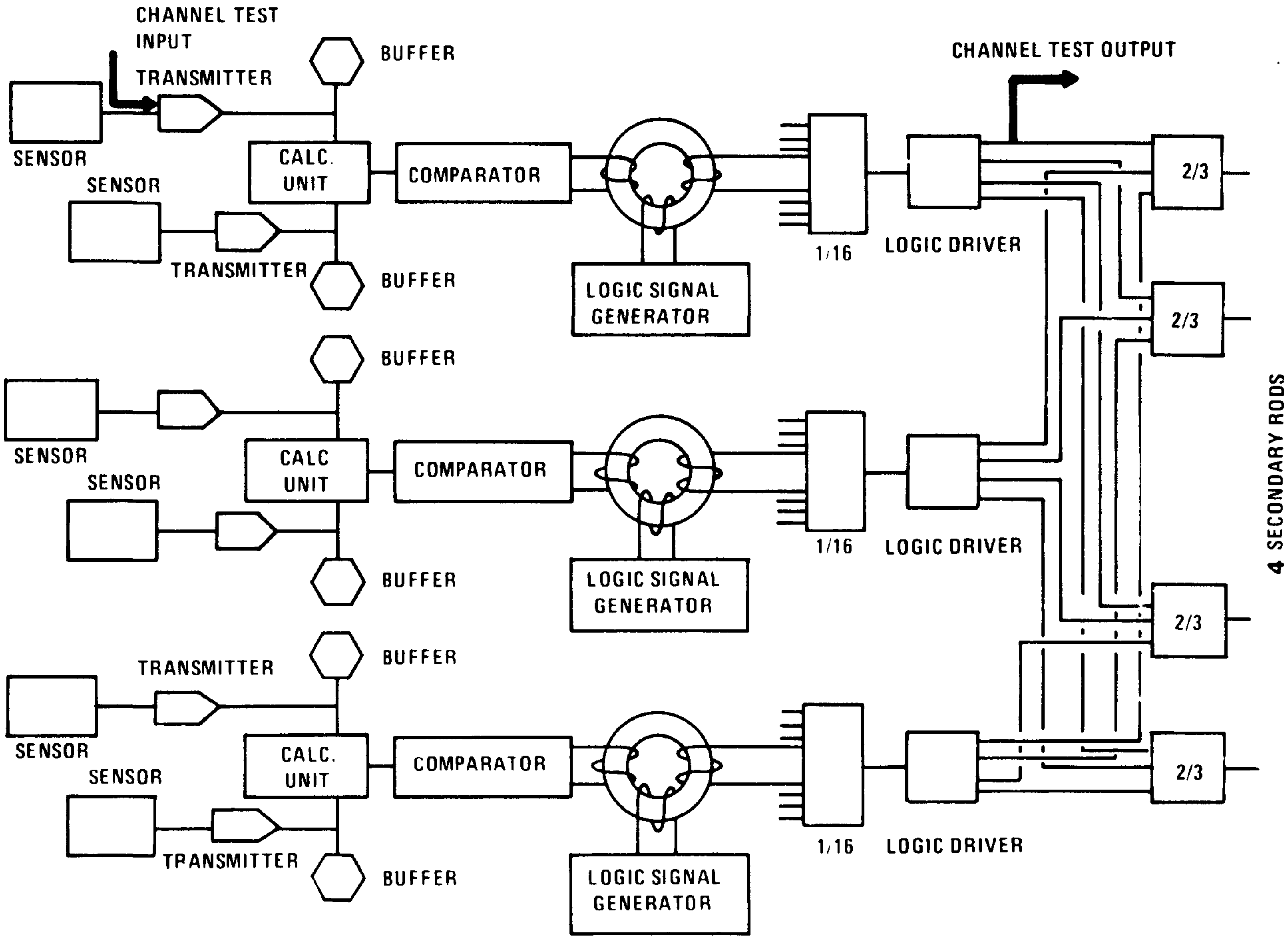


Figure 7-2. Typical Secondary Reactor Shutdown System

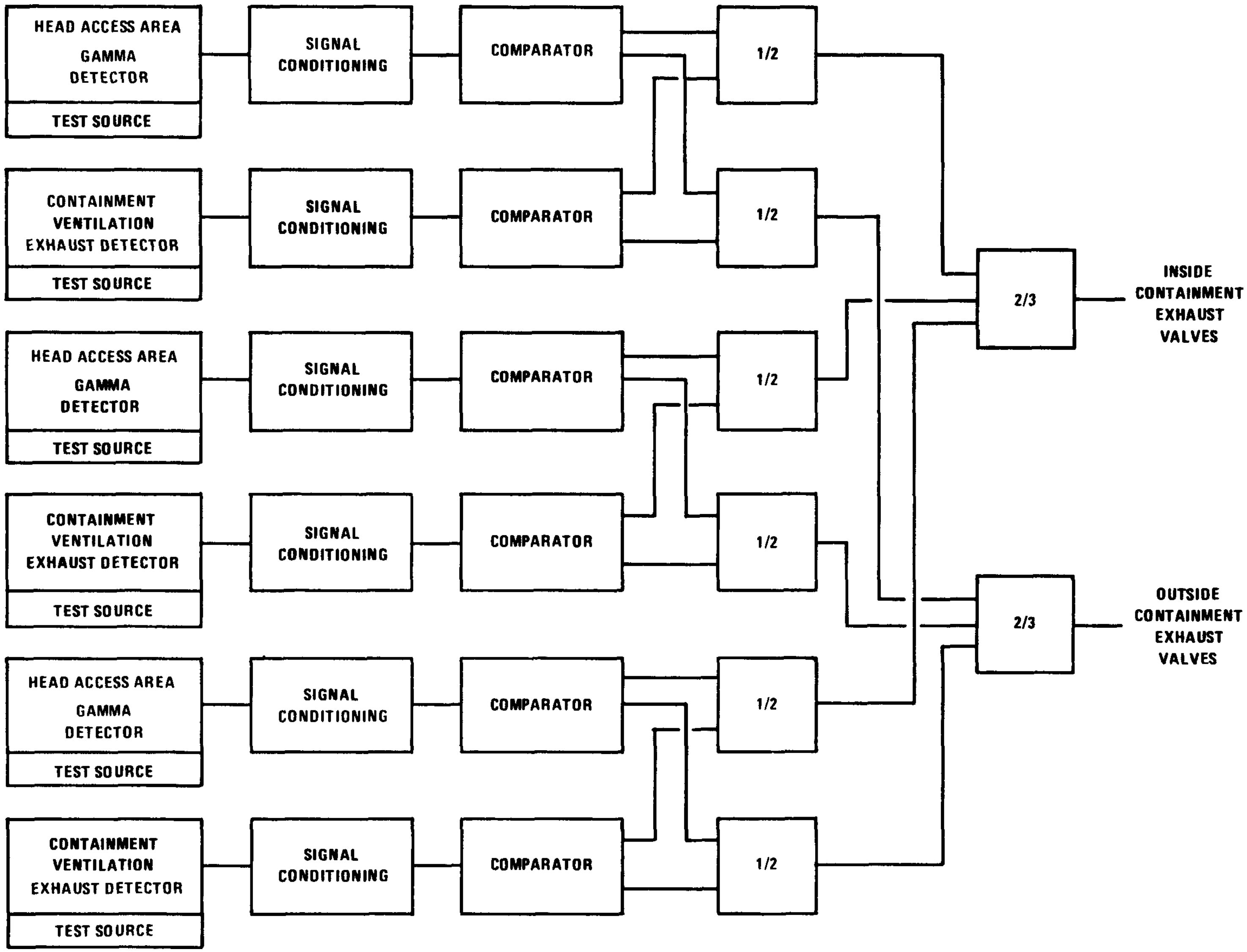


Figure 7-3. Containment Isolation System Block Diagram

CHAPTER 8 ELECTRICAL SYSTEMS

8.A Summary

This Chapter discusses the offsite and onsite electrical power sources and distribution systems. These include:

a. The transmission lines and switchyards connecting the CRBRP to the TVA grid
b. The normal AC distribution system
c. Emergency AC and DC power supplies
d. The safety related AC and DC distribution systems
e. The safety related and non-safety related loads supplied by the emergency power supplies (including a table of safety related loads and timing sequences).

The Reserve AC Power Supply (two transmission lines connected to the CRBRP through the Reserve Switchyard) meets the AEC and IEEE requirements for separation and redundancy. The Preferred AC Power Supply (two transmission lines and the CRBRP main generator connected to the CRBRP through the Generating Switchyard) is not required to and does not meet those AEC and IEEE requirements since plant loads are automatically connected to the Reserve AC Power Supply in the event of a failure in the Preferred AC Power Supply.

8.B Changes Since Reference Design Report

The electrical system design presented in the PSAR deviates from the Reference Design Report in switchyard configuration. The Reference Design Report showed a single switchyard for all grid connections. The PSAR shows two, the Generating Switchyard which transmits power during plant operation and is the normal source of supply to the plant auxiliaries during operation and shut-down periods; and the Reserve Switchyard which is used only in the case of Generating Switchyard failure (or planned outage). This represents the currently approved design.

8.C Other Special Items

Regulatory Guide 1.75, (Physical Independence of Electric Systems) based on IEEE Standard 384-1974 sets requirements for separation of safety related and non-safety related equipment which are considered excessive by the nuclear industry. The current design of the CRBRP does not comply with the Regulatory Guide, though it does comply with IEEE 384.

The following statement is made in the PSAR:

"AEC Regulatory Guide 1.75

This guide is recent (February, 1974) and experience in its application was not available during the preparation of the reference design. A preliminary review of the guide indicates that Class IE systems conform to the criteria established in the guide and in Appendix 1 of the guide.

Non-Class IE systems which come within the definition of associated circuits in Section 3.2 of Appendix 1 may not conform to the intent of the criteria of Appendix 1. Also, the method of application of Section 3.8 is not clear. Studies are currently in progress to determine the extent of applicability and the potential safety implications on the existing reference design."

This is consistent with the position adopted by LWR suppliers.

CHAPTER 9 AUXILIARY SYSTEMS

9.A Summary

This chapter discusses Auxiliary Systems which provide a wide variety of normal and emergency services to the plant. A condensed listing and brief description of the systems follows:

1. Fuel Storage and Handling - A description of the handling (including the operational sequence of New and Spent Fuel Assemblies is provided. Equipment, cells and vessels are described in detail. Decay heat removal from the Ex-Vessel Storage Tank is discussed.

2. Maintenance - Tools, fixtures and procedures for the transport, storage, inspection, repair and removal of sodium wetted and radioactive components are described. Special attention is paid to the cleaning of large sodium wetted components (see also Part II, Section 9).

3. Auxiliary Liquid Metal Systems - Liquid metal systems for receiving Na and NaK, Processing primary and intermediate Na, Storage and cooling are described.

4. Piping and Equipment Electrical Heating - Discusses the design of electrical heaters, mountings and power controllers to heat several sodium containing systems.

5. Inert Gas Receiving and Processing - Details for the supply of argon and nitrogen to cells and systems throughout the plant are provided.

6. Heating, Ventilating and Air Conditioning - The requirements for air quality throughout the plant are stated. The system permits personnel access to various plant areas for maintenance under normal operation.

7. Auxiliary Coolant Fluid System - This system provides redundant cooling to various safety related coolers. Cooling to non-safety related coolers is also provided. Dowtherm J is the cooling medium.

8. Water Systems - These systems provide normal and emergency chilled water for air-conditioning and unit coolers, general plant service and auxiliary equipment in the Turbine Generator Building. Also included is a discussion of the River Water System.

9. Compressed Air System - Various subsystems furnish instrument, service and breathing air for the plant.

10. Communications: Lighting - Normal and Emergency systems are provided to support operation or shutdown of the Plant.

11. Plant Fire Protection System - Means are supplied to fight conventional as well as sodium fires.

12. Diesel Generator Auxiliary System - This system supplies on-site power generation for use by plant systems in the event of loss of off-site power. The system is internally redundant. Special consideration is given to component starting, cooling , lubrication, and testing.

9.B Significant Changes from the Reference Design

1. An additional requirement for the Auxiliary Liquid Metal Systems is to provide for reactor decay heat removal in the event of loss of the steam generators. This requirement instituted the creation of the overflow heat exchanger (OHX). The OHX is positioned between two liquid metal systems. The tube side is part of the Primary Sodium Processing System. The shell side is part of one NaK cooling loop of the Ex-vessel Fuel Storage Processing System.

   In the event of loss of the Steam Generators, the OHX must be manually valved into the Primary Overflow system. Heat is transferred to the shell side NaK and dissipated in the EVST Nak airblast. This method of decay heat removal is termed the Overflow Heat Removal Service (OHRS), and is capable of removing decay heat after 24 hours from shutdown.

9.C Other Special Items

1. Use of Water in Containment for Cleaning of Heavy Components

   This is discussed in Part II, Section 9 of this Summary.

2. Reactor refueling with Open Containment and Ex-containment fuel store.

   This is discussed in Part II, Section 8 of this summary.

CHAPTER 10 STEAM AND POWER CONVERSION SYSTEMS

10.A Summary

Chapter 10 contains discussion of the following, generally non-safety related, systems:

- a. Turbine-Generator
- b. Main Steam Supply and Turbine Bypass Systems
- c. Condensate and Feedwater System
- d. Demineralizing System
- e. Steam Drum Blowdown System
- f. Turbine Gland Sealing System
- g. Circulating Water System
- h. Condenser Air Removal System

The major concerns in evaluating these systems are (1) the possibility of causing or propagating a failure to a safety related system (e.g., via pipe breaks or turbine missiles), and (2) the release of tritium from the water systems.

10.B Changes Since Reference Design Report

None

10.C Other Special Items

The CRBRP turbine-generator will be a first-of-a-kind turbine (375 MWt at 3600 RPM) so there is a lack of available data on generation of turbine-generator missiles. The PSAR cites the configuration of the turbine generator (axis perpendicular to RCB) and the similarity of the CRBRP unit to that of other nuclear plants as indications that analyses of the future will identify no significant hazard from turbine generator missiles.

11.A Summary

This chapter addresses the waste processing systems provided (liquid, gaseous and solid).

The principal modes of radioactivity production and/or release to the primary coolant and reactor cover gas are presented. These source terms form sources of radioactivity which the radioactive waste management system is designed to control. Sources of radioactivity considered include tritium production, fission product and potential plutonium release from failed fuel, sodium activation and corrosion product activation. Deposition of non-gaseous sources into primary sodium, cold traps and onto plant surfaces is analyzed. Fuel defect limit assumptions are derived in Part II, Section 7. The system designs presented are summarized below.

A. Liquid Waste System

A design objective of this system is to purify and reuse waste liquids where possible and to minimize the total activity in liquid effluents with virtually all of the liquid radwaste being solidified. The source of the liquid radwaste is considered as (a) small sodium spillages, plant drains, laboratory drains, etc. and (b) the washing of large components, for the low level activity system and intermediate activity system respectively. Each system has an evaporator-demineralizer set that will provide an overall decontamination factor of 10^5. Under normal conditions, liquid radwaste will be released into the cooling tower blowdown stream and eventually the Clinch River. Such release under normal conditions is associated only with the low activity level system, and will be accomplished only after monitoring of the radwaste storage tanks to assure that activity levels are in compliance with appropriate Federal and State regulations.

Also considered are the off-normal events of discharge of some intermediate level activity for eventual release into the Clinch River. The section assumes both systems release into the Clinch River after dilution and compares concentrations to MPC's of 10CFR20. Non-tritium releases are shown to be decades below the concentration limits, tritium releases are well below the 10CFR20 limits.

Estimates are made and presented of the dose effects associated with this design condition of the superposition of normal low activity and off-normal intermediate activity system releases. The calculations show that doses associated with "normal" operations are decades below both natural radioactivity levels and dose limits described in 10CFR20. These estimates include the contribution of BOP tritium in the cooling tower blowdown.

B. Gaseous Waste System

The design objective of the system is that the levels of radioactive material in the plant effluents to the environment shall be kept as low as practicable. Plant design objectives include conformance with the requirements of 10CFR20. The design of RAPS (Recycle Argon Processing Subsystem) and CAPS (Cell Atmosphere Processing Subsystem) are described in detail, including activity inventories in the components.

Based on a set of estimates and conservative assumptions of reactor cover gas leakage, buffered head seal leakage, primary piping leakages, RAPS-CAPS component leakage and intermediate bay cell leakage and tritium release from the Turbine Generator Building, estimates are made of the activity concentrations in the ventilation streams for plant buildings and head access area. In addition, dose rates at the site boundary are calculated.

The dose rates are based on normal operation with design value of 1% failed fuel. Ventilation stream concentrations are calculated for the design 1% failed fuel condition and expected condition of 0.1% failed fuel. The equations utilized in calculating the inventory terms are discussed. Ventilation streams are calculated to be less than 0.1% MPC as in 10CFR20 for the design base condition. Annual Site boundary doses for the design operating condition are shown to be a factor of 2500 below the requirements of 10CFR20 for unrestricted areas. Estimates include the release of BOP tritium.

C. Process and Effluent Radiological Monitoring

Monitors discussed are stated to be in accordance with AEC General Design Criterion No. 64 and general design criteria for the CRBRP. Radiation monitoring of process systems provides early warning of equipment malfunction, potential radiological hazards and prevents releases of activity to the environment in excess of 10CFR20 limits. Monitoring of liquid and gaseous effluent under normal operating conditions will be in accordance with AEC Regulatory Guide 1.21 and any activity release will be within limits established in 10CFR20.

Locations and sensitivities of the process and effluent monitors are provided.

D. Solid Waste

The design objective of the solid radwaste system is to release no radioactivity to the environment. The section presents the basic approach of the system is solidify the liquid radwaste with cement or concrete and to load all solid radwaste into canisters that satisfy DOT and CFR regulations. Expected amounts of the constituents of the solid radwaste system, their associated activities and associated number of shipments per year are included.

E. Off-Site Radiological Monitoring Program

Pre-operational and operational off-site radiological monitoring programs are discussed. The capability of the environmental monitoring program to detect design-level releases from plant effluents is uncertain because of the insignificant quantities which will be released. The program will have the capability of detecting any significant buildup of radioactive materials in the environment above and beyond that which is already present. A background of 110-130 mr/yr for the site is expected.

Dose models utilized in the program will be continually re-evaluated in light of the data resulting from the offsite monitoring program to ensure that all significant pathways are included in the calculations. The sampling techniques, locations and frequency of sampling for the program are provided.

11.B Changes Since Reference Design Report

None

11.C Other Special Items

1. Failed Fuel-Fission Product Release Models

   The project has agreed to a model that is felt to be an improvement upon Kayser's model. The CRBRP model has experimental input (GE capsule tests) unavailable to Kayser.

2. Meteorology

   Dose assessments presently utilize off-site weather data in the analysis. Analysis of one-year of on-site data will probably be available from WESD by mid-Feb. The results of the data are expected to reveal an added element of conservatism in present PSAR calculations. NRC should be made aware of this. More detailed discussion of this item is provided in Part II Section 2 of this Summary.

3. Sodium Disposal Methods

   In the description of the solid radwaste system, sodium disposal methods is a TBD item. NRC has already asked a question on this matter in a list of agenda items discussed at their CRBRP site visit. The project responded by stating that the project is reviewing developing technologies and alternative methods and will utilize, as appropriate, the HEDL R&D program in support of the FFTF project. NRC stated that they were satisfied with the response. It is planned that a more definitive response will be prepared after PSAR submittal.

CHAPTER 12 RADIATION PROTECTION

12.A Summary

This chapter discusses the means provided to assure the radiation protection of operating personnel. The shielding, ventilation and operational radiation monitoring design as well as the health physics program are included below.

Shielding objectives for the CRBRP are discussed, including the specific shield design parameters. Bases for zoning criteria are discussed. Source terms for the shielding design are discussed as being based on maximum operating conditions including biases for uncertainty. A special case of shielding design is control room shielding design, where the shield and heating and ventilation system are designed to limit the dose to operating personnel to 5 REM following a major radioactivity release. The intent of the bases is to conform to criterion 19, Appendix A of 10CFR50.

Source terms of items such as liquid radwaste tanks, RAPS and CAPS components, solid radwaste drums, the EVST, control room, cold traps, are listed and discussed.

Dose Rates and annual doses at restricted locations of the plant and the resulting expected manrem value associated for the plant are provided. The estimated value of 280 man-rem per plant year is well within the range of values associated with LWR's.

Zone maps are presented reflecting the criteria established and source terms provided within the section. Analytical techniques, basic nuclear data, and shielding design, verification and testing are either discussed or referenced.

Design objectives of the heating and ventilation and air conditioning are in compliance with 10CFR20 Appendix B, Table 1. Concentrations in the ventilation stream of the normally accessible Head Access Area and Intermediate Sodium Piping Cells are discussed. Both are shown to be less than 0.1 MPC. Inhalation doses are derived from the concentrations.

Airborne radioactivity monitoring and sampling is discussed, including basis for location and purpose.

Health Physics Program objectives are stated and facilities and equipment are discussed. The program objective is to maintain plant personnel exposures as low as practicable.

12.B Changes Since Reference Design Report

None

12.C Other Special Items

Chapter 12 was written on the basis of the SFAC. However, in Nov. 1974, Reg. Guide 1.70.3 ("Additional Information on Radiation Protection") was issued restructuring the format of Chapter 12, including requests for additional information. In particular, guide 1.70.3 calls for assurance that occupational radiation exposure be as low as practicable and Chapter 12 should include a discussion of detailed plans and procedures that will show how the design is directed toward reducing both the need for maintenance and the time spent where maintenance is required. Dose assessments associated with operation, maintenance, radwaste handling, refueling and inservice inspection should be provided.
The project has appropriately conformed to the guidelines in existence as the PSAR effort was being undertaken. The project may well be asked to supply information called for in Guide 1.70.3 not presently in Chapter 12. Because of the recent date of issuance of the Guide it is not considered necessary to include its requirements in this PSAR.

CHAPTER 13 CONDUCT OF OPERATIONS

13.A Summary

This chapter describes the framework within which the operation of the plant will be conducted.

Section 13.1 describes the organizational structure of the applicant and identifies PMC and TVA as co-applicants, with TVA having responsibility for the safe operation of the plant. Included in this section is a description of the various positions in the plant with the required qualifications.

Section 13.2 is a description of the proposed training programs important to safety with the appropriate responsibilities for implementation.

Section 13.3 covers emergency planning. In addition to the organizational structure and responsibility for emergency response is the commitment to submit the actual TVA Radiological Emergency Plan (REP) for the CRBRP as a separate document with the FSAR.

Section 13.4 covers Review and Audit. Reference is made to Chapter 17.

Section 13.5 defines, with appropriate diagrams, the structure for implementing plant procedures and instructions, as well as defining the various procedures and instructions.

Section 13.6 covers plant records. With the exception of a treatment of Plant History, the remainder of this section is deferred to the FSAR.

Section 13.7 covers Industrial Security with specific reference to site security, personnel control, and plant access. Section 13.7.3.7 on Tests and Inspections is deferred to the FSAR.

13.B Commitments

The following commitments are indicated in Chapter 13:

a. Submit the "TVA Radiological Emergency Plan (REP)" as a separate document with the FSAR.
b. Submit the CRBRP Physical Security Plan as a separate document with the FSAR. Note that this does not comply with Reg. Guide 1.70.15, issued December, 1974, which requires a separate submittal, covering Preliminary Planning, at PSAR stage. This non-compliance is considered reasonable in view of the recent date of Reg. Guide 1.70.15.

13.C Other Special Items

Reg. Guide 1.70.14 was issued in December, 1974. This requires material, related to Emergency Planning beyond that currently supplied in Chapter 13. This non-compliance is considered reasonable in view of the recent date of Reg. Guide 1.70.14.

CHAPTER 14 INITIAL TESTS AND OPERATION

14.A Summary

This chapter of the PSAR is intended to provide, to the extent possible, information relating to the period of initial operation and testing. As described in the SFAC, the bulk of this material is not required until it is issued in the FSAR. However, certain administrative subjects are requested for the PSAR and these have been addressed.

Specifically, TVA has been assigned the responsibility for detailed planning, scheduling, coordination and conducting of plant testing with the assistance of the ARD technical staff. TVA also has been assigned the responsibility for recording and reporting the test results.

Upon satisfactory completion of the construction tests on a particular system or clearly defined portion thereof, the system shall be turned over by the constructor to PMC, ready for acceptance testing.

The Acceptance Test Program has been divided into four distinct categories which are described as follows:

Category	Effect on Plant	Examples of Systems Covered	*Responsibility for Test Specification	Responsibility for Test Procecure
A	Direct	Reactor, PHTS	ARD	CRBRP Test Eng. (TV/
B	Direct	IHTS, HTS Instrumentation, Steam/Water System, SG System, Fuel Handling	B&R, AI, GE	CRBRP Test Eng. (TV.
C	Limited	Radwaste, vacuum, receiver gas cooling	ARD, B&R, AI, GE	CRBRP Test Eng. (TV.
D	None	Lighting, telephones, waste water treatment, multiplexer	ARD, B&R, AI, GE	CRBRP Test Eng. (TV/

*Normally the design contractor will be assigned

14.B Changes Since Reference Design Report

None

14.C Other Special Items

Project approval of the Joint Test Group composition and responsibilities has not been received. Section 14.A indicates how this has been covered in the PSAR. Since this is not a basic issue for the CP review, it is recommended that this remain as is. Any subsequent change can be submitted either as a supplement to the PSAR or at the FSAR stage.

CHAPTER 15 ACCIDENT ANALYSIS

15.A Summary

Chapter 15 reports the accident analyses conducted for the plant, on the basis of which it is demonstrated that no identified accidents will result in a site boundary dose in excess of 10CFR100 guidelines. Section 15.1.1 contains a collation of third level design margin requirements and a preliminary assessment of the feasibility of compliance with these. Because of the significance of this Section, it is discussed separately at the end of this Section, in 15.B.

Section 15.1.2 discusses the fuel cladding failure criteria used for evaluation. It is shown in this Section that, provided a cladding hot spot temperature of 1600°F is not exceeded in any transient, then that transient should not result in cladding failure. However, if a temperature in excess of 1600°F is reached, then that particular transient must be evaluated on an individual basis.

Sections 15.2 and 15.3 cover, respectively, the identified events which could result in reactivity insertion or in reduction in core cooling. In each case, the events are categorized as Anticipated, Unlikely or Extremely Unlikely. A summary of the results of these studies appears in Tables 15.2-1 and 15.3-1 of the PSAR, which are reproduced below. The results of these analyses indicate there are no deleterious consequences associated with any of the undercooling events presented in this Section.

Section 15.4 discusses the potential local failure events that could occur to the fuel, radial blanket and control assemblies. The major items addressed are:

- Stochastic Failures
- Overenriched Assemblies
- Flow Blockages

The results show that none of the potential events presented leads to either propagation of fuel pin failures or of assembly-to-assembly failures.

Section 15.5 discusses Fuel Handling and Storage Events. Some typical fuel handling problems discussed in this section are:

- Dropped Fuel Assembly
- Insertion into an Occupied Position
- Fuel Cladding Failure and Subsequent Fission Gas Release During Refueling
- Reactor Cover Gas Release

The results of the analyses show that none of the events presented generate site boundary or low population zone doses in excess of the suggested 10CFR guidelines.

Section 15.6 covers Sodium Spills. Sodium spills were postulated to occur in the:

- Reactor Containment Building (overflow tank cell, reactor cavity and the PHTS cell)
- Reactor Service Building (cooling equipment cell)
- Steam Generator Building (Intermediate Bay)

The potential spills and resulting fires that were discussed in this section did not produce site boundary or low population zone doses in excess of the suggested 10CFR100 guidelines. All temperatures and pressures generated by the fires were within the Design Basis limits for the cells concerned. Note that Primary and Intermediate HTS integrity considerations, which relate to this Section, are covered in Part II, Sections 5 and 6.

Section 15.7 comprises those events that did not fall within any of the previously discussed categories. Some of the more prominent events addressed are:

- Loss of One DC System
- Loss of Instrument Air
- Rupture of the RAPS Surge Vessel
- Liquid Radwaste System Failure
- Shipping Cask Drop from Maximum Possible Height
- Maximum Possible Conventional Fire, Flood and Storm

The results of the analyses show no adverse consequences associated with any of the events described in this section.

15.B Third Level Design Margin Requirements

The third level design margin requirements were derived from generic HCDA analysis, FFTF experience and engineering judgement. These are quoted in the design Chapters (3 through 12) of the PSAR, and collated together in Section 15.1.1. This Section begins with a reminder and amplification of the safety philosophy contained in Section 1.1 and gives specific examples of first and second level design features. Sub-Section 15.1.1.3 gives a 20 page discussion of the third level design requirements and their implications and concludes with a series of Tables and Figures specifying the numerical requirements. The derivation of these requirements is treated in some detail. Some examples are quoted below:

- The core support structure and reactor vessel shall be able to accommodate, without failure, the dynamic loading shown in Figure 15.1.1-5 (not reproduced here) on the upper surface of the core support structure and attenuate these loads to values which are acceptable to the supporting concrete as quantified below.

- The vessel support ledge shall be able to accommodate a load of 50×10^6 lbs in either the upward of downward direction.

- The IHX upper shell shall be able to accommodate the dynamic loading shown in Figure 15.1.1-19 (not reproduced here).

- The vertical clearance between the reactor vessel and guard vessel shall be at least 6 inches to allow postulated vessel downward motion.

- Clearance above head mounted components shall permit a 6 inch head lift at the outer bolt circle and a 10 inch maximum vertical lift at the center of the head.

- The design shall be capable of sustaining temperatures up to 1250°F for as long as 300 hours in the vessel, nozzles and core support structure without exceeding creep rupture strength, where the only imposed loading is weight.

- The Reactor Containment Building shall be provided with isolation valves that can be closed without release of radioactivity, following detection of high radioactivity levels in the building heating and ventilating system. The closure time requirement for the inlet and exhaust isolation valves is 4 seconds from the time of detection of high radiation levels in the heating and ventilating system assuming a 10 second transport time from the serving point to the valve.

The above are only a small sample of the total listing of requirements, but are indicative of the level of detail provided. In order to calibrate these requirements against actual CRBRP HCDA analyses, a comparison is given, in tabular form, in Table 15.1.1-1 (reproduced below). As explained in the text of Chapter 15, in this Table the first column gives the third level design margin requirements, and the second gives results of a conservative CRBRP HCDA analysis chosen to test the design capability taken from Appendix D.

TABLE 15.1.1-1

COMPARISON OF HCDA PRIMARY SYSTEM LOADINGS

	Reference Case	Structural Evaluation Fuel Vapor Expansion Case
Work Energy to One Atmos. (MW-sec)	300	1324
Initial Core Pressure (PSIG)	2972	2175
Residual Bubble Pressure (PSIG)	290	347
Max. Vessel Strain in Core Region (%)	1.8	3.2
Max. Upper Vessel Wall Radial Strain (%)	3.9	10.0
Core Barrel Strain (%)	8.8	9.6
Peak Outlet Nozzle Pressure (PSIG)		
Before Slug Impact	420	464-493
After Slug Impact	761	652-725
Peak Inlet Nozzle Pressure (PSIG) (REXCO Averaged)	435	493
Peak Force on CSS (10^6 LBF)	52.5	57.9
Impulse on CSS to Slug Impact (10^6 LBF-Sec)	2.1	2.0
Impulse of CSS to System Equil (10^6 LBF-Sec)	2.3	3.8
Peak Force on Head (10^6 LBF)	135	108
Avg. Force for Second Peak (10^6 LBF)	29	49
Peak Inlet Piping Pressure (PSIA)	717	607
Peak Primary Piping Pressure (PSIA)	720	∿770
Peak Pump Inlet Pressure (PSIA)	590	∿580
Peak IHX Shell Pressure (PSIA)	522	∿772
Peak Check Valve Pressure (PSIA)	703	∿763

Table 15.2-1 REACTIVITY INSERTION DESIGN EVENTS

Section No.	Event	Max. Clad. Temp.* Primary Scram	Max. Clad. Temp.* Secondary Scram	Comments
15.2	Reactivity insert. design events			
15.2.1	Anticipated events			
15.2.1.1	Control assembly withdrawal @ startup	NA (See 15.2.1.1)	1383°F	Temp. shown for 1¢/sec. withdrawal. Resultant Temp. less than operating condition. (full power)
15.2.1.2	Control assembly withdrawal @ power	1510°F	1610°F	Based on extremely small withdrawal rate - Results are within the guidelines of Table 15.1.2-3
15.2.1.3	Seismic reactivity insertion (core, radial blanket and control rod) - OBE	1440°F	∿1440°F	Based on postulated 30¢ step reactivity insertion - Results are within guidelines of Table 15.1.2-3
15.2.1.4	Small reactivity insertions	1500°F	1560°F	For 2¢/sec insertion case - Results are within guidelines of Table 15.1.2-3
15.2.1.5	Inadvertent drop of single control rod at full power	Less than init.cond.	Less than init. cond.	Results fall within guidelines of Table 15.1.2-3.
15.2.2	Unlikely Events			
15.2.2.1	Loss of hydraulic holddown	1415°F	1420°F	Results are within guidelines of Table 15.1.2-3
15.2.2.2	Core radial movement	1470°F	1510°F	For non-seismic conditions - Results fall within guidelines of Table 15.1.2-3
15.2.2.3	Mal-operation of reactor plant controllers	<1510°F	<1610°F	Less than limiting condition shown in Table 15.2-1.2-1
15.2.3	Extremely unlikely events			

*Fuel pin inside diameter cladding temperature (under wire wrap)

Table 15.2-1 REACTIVITY INSERTION DESIGN EVENTS (Continued)

Section No.	Event	Max. Clad. Temp.* Primary Scram	Max. Clad. Temp.* Secondary Scram	Comments
15.2.3.1	Cold sodium insertion	Less than init. cond.	Less than init. cond.	Results fall within the guidelines of Table 15.1.2-3
15.2.3.2	Large gas bubble through core	<1480°F	<1480°F	Results fall within the guidelines of Table 15.1.2-3
15.2.3.3	Seismic reactivity insertion (core, radial blanket and control rod) - SSE	<1505°F	NA	Based on postulated 60¢ step reactivity insertion - Results fall within the guidelines of Table 15.1.2-3
15.2.3.4	Control assembly withdrawal at startup - max. mech. speed	NA (See 15.2.3.4)	800°F	For 20¢/sec reactivity insertion - Results fall within the guidelines of Table 15.1.2-3
15.2.3.5	Control assembly withdrawal at power - max. mech. speed	1420°F	1460°F	For 20¢/sec reactivity insertion - Results fall within the guidelines of Table 15.1.2-3

*Fuel pin inside diameter cladding temperature (under wire wrap)

Table 15.3-1 UNDERCOOLING EVENTS

Section No.	Event	Max. Clad. Temp.* Primary Scram	Secondary Scram	Comments
15.3	Undercooling Design Events			
15.3.1	Anticipated Events			
15.3.1.1	Loss of off-site electrical power	1410°F	1630°F	Primary shutdown within upset umbrella, temp. Spike associated with sec. shutdown is considerable less severe than the umbrella transient (See Section 15.3.1.1)
15.3.1.2	Spurious primary pump trip	1390°F	1445°F	Within the umbrella
15.3.1.3	Spurious intermediate pump trip	<1365°F	<1365°F	Core sees only normal trip
15.3.1.4	Inadvertent closure of one evaporator or superheater	<1365°F	<1365°F	Core sees only normal trip
15.3.1.5	Turbine Trip	<1365°F	<1365°F	Temperature decreasing continuously
15.3.1.6	Loss of normal feedwater	<1365°F	<1365°F	Core sees only normal trip
15.3.1.7	Inadvertent actuation of the sodium/water reaction system	<1365°F	<1365°F	Core sees only normal trip
15.3.2	Unlikely events			
15.3.2.1	Single primary pump seizure	1400°F	1470°F	Within the umbrella
15.3.2.2	Single intermediate loop pump seizure	<1365°F	<1365°F	Core sees only normal trip
15.3.2.3	Small water-to-sodium leaks in steam generator tubes	<1365°F	<1365°F	Core sees only normal trip
15.3.2.4	Failure of the steam dump system	<1365°F	<1365°F	Core sees only normal trip

Table 15.3-1 UNDERCOOLING EVENTS (Continued)

Section No.	Event	Max. Clad. Temp.* Primary Scram	Secondary Scram	Comments
15.3.3	Extremely unlikely events			
15.3.3.1	Steam or feedwater line pipe break	<1365°F	<1365°F	Core sees only normal trip
15.3.3.2	Loss of normal shutdown cooling system	<1365°F	<1365°F	Core sees only normal trip
15.3.3.3	Large sodium/water reaction	<1365°F	<1365°F	Core sees only normal trip
15.3.3.4	Design bases leak in primary loop	no effect	no effect	No effect on reactor core or primary system temperatures or pressures
15.3.3.5	Intermediate heat transport system pipe rupture	no effect	no effect	Core temperatures would not increase

*Fuel pin cladding midwall temperature (under wire wrap)

CHAPTER 16 TECHNICAL SPECIFICATIONS

16.A Summary

The technical specifications, which regulate the operation and maintenance of a nuclear power plant become an integral part of the plant license, and as such form the basis of a continuing relationship between the licensee and the regulatory agency. They are proposed by the applicant and ultimately imposed upon the plant operation in the interest of the health and safety of the public.

Because of the special nature of the material in this chapter and the present state of the design, it is neither possible nor prudent to produce final technical specifications for the essential plant parameters. Rather, for the PSAR, Chapter 16 has been written to identify the essential systems and parameters which require technical specifictions in an LMFBR without attempting to provide the final values for the essential parameters. For those systems where the design is sufficiently detailed, technical specifications have been written.

As required by the Standard Format and Content, the chapter is divided into six major sections.

16.1 Definitions

16.2 Safety Limits and Limiting Safety System Settings

16.3 Limiting Conditions for Operation

16.4 Surveillance Requirements

16.5 Design Features

16.6 Administrative Controls

Section 16.1 is essentially complete and defines those special conditions and terms as they apply to CRBRP.

Section 16.2 covers the Safety Limits and Limiting Safety System Setting. The only safety limit which has been identified is the combination of thermal power and primary coolant flow which will prevent clad melting and thereby maintain a coolable core geometry. No specific values are given for these parameters.

For the Limiting Safety System Settings, the Plant Protection System protective functions have been identified without specifying the actual trip settings.

Section 16.3 provides the technical specifications for the Limiting Conditions for Operation of each of the major systems. The intent of this section is to identify the lowest functional capability or performance level of equipment required for safe operation of the plant.

Section 16.4 is concerned with the surveillance requirements for the various systems and components. Technical specifications are written to identify the tests, calibrations and inspections which are necessary to assure that the quality of the systems and components is maintained.

Section 16.5 is used to describe the major design features of the plant. By including these descriptions as a part of the technical specifications, a change in any of these features requires the same procedure as a change in any of the other technical specifications. In this way, the regulatory agency is able to control major changes in safety related systems. The subjects covered in this section are:

1. Site
2. Containment
3. Reactor
4. Heat Transport System and Residual Heat Removal
5. Fuel Storage and Handling

The final section in this chapter, 16.6, is a description of the administrative controls which are necessary to assure safe operation of the plant.

16.B Changes Since Reference Design

None

16.C Other Special Items

None

CHAPTER 17 QUALITY ASSURANCE

17.A Summary

This chapter describes the program of plans and actions related to quality assurance for the CRBRP. The chapter defines the Project Quality Assurance philosophy, provides a description of the organization and discusses the implementation of programs to assure quality performance throughout the design and construction phases of the CRBRP. The chapter has been written in concert with the format of REG Guide 1.70.6 (July 1974) which significantly expanded the amount of material required by the SFAC.

The basic chapter and its appendices provide a detailed discussion of how implementation of quality requirements is delegated down through the project organization and defines the means utilized to assure compliance with these requirements. The disciplines discussed in detail in each of the appendices are as follows:

1. Organization
2. Quality Assurance Program
3. Design Control
4. Procurement Document Control
5. Instructions, Procedures and Drawings
6. Document Control
7. Control of Purchased Material, Equipment, and Services
8. Identification and Control of Materials, Parts and Components
9. Control of Special Processes
10. Inspection
11. Test Control
12. Control of Measuring and Test Equipment
13. Handling, Storage and Shipping
14. Inspection, Test and Operating Status
15. Nonconforming Materials, Parts or Components
16. Corrective Action
17. Quality Assurance Records
18. Audits

17.B Other Special Items

1. This Chapter has been submitted to the Nuclear Regulatory Commission (2-5-75).

2. The organizational relationships are given as shown on Figure 17-1.

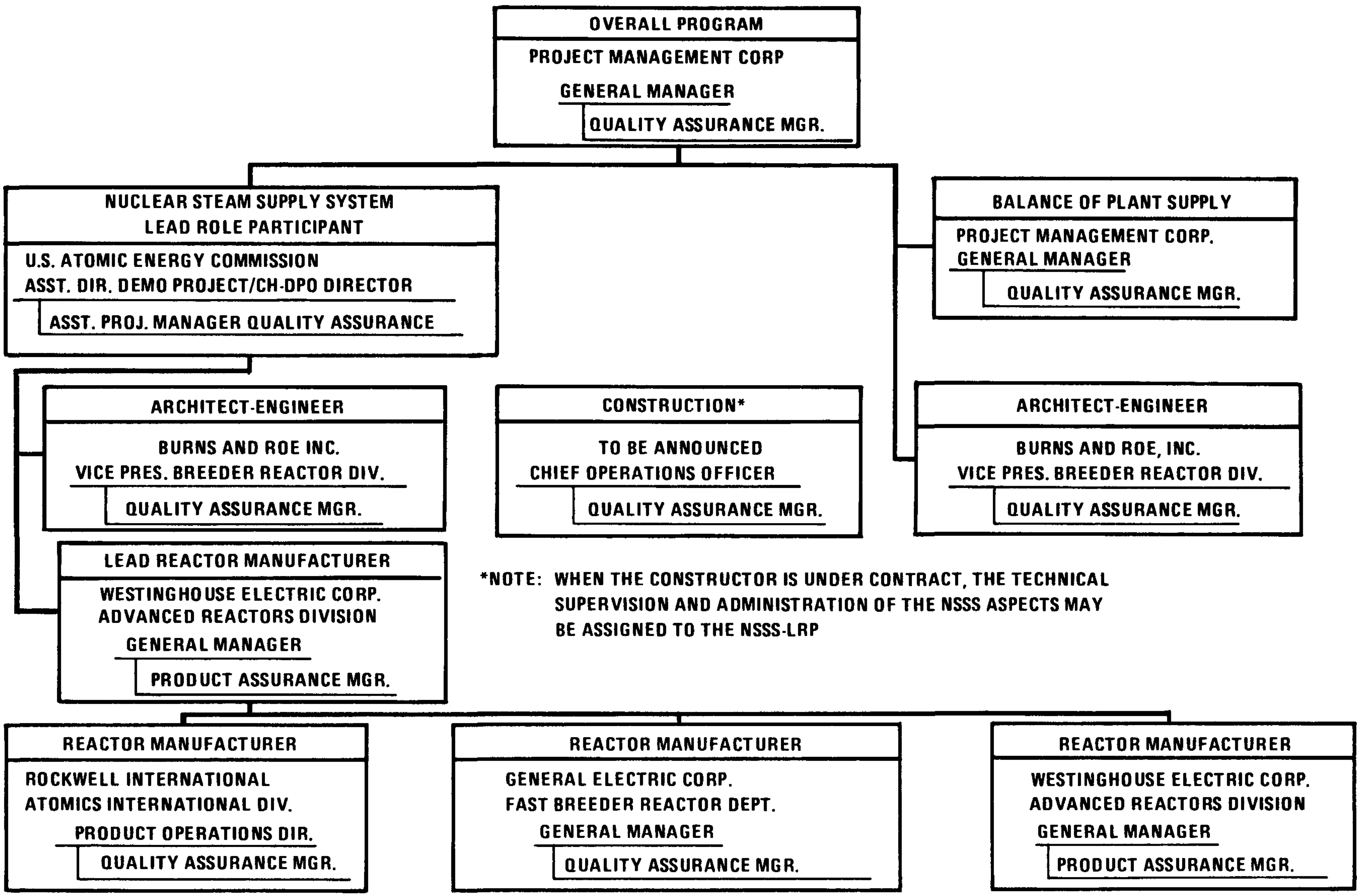


Figure 17-1. CRBRP Organizational Relationships

APPENDIX A COMPUTER CODES

A.1 Summary

Appendix A provides brief abstracts of the 103 computer codes used or identified to be used in the analysis of the CRBRP. For those codes which have been determined to be non-proprietary (approximately 2/3) references, available in the open literature or at the user's location, have been cited to provide a source for supplementary information.

A.2 Other Special Items

For those codes which have been deemed proprietary (37 thus far) a system or systems may need to be established to make these codes available to REG at the code originator's location. ARD plans to implement a system similar to that which is already in force at WNES-PWR which provides access to REG for use of those codes required in the analysis of the Emergency Core Cooling System as described in Appendix K of 10CFR50.

The WNES-PWR system is to place such proprietary codes into a locked safe at the WNES-PWR site to which only REG has a key. REG is then permitted to come on site at any reasonable time to utilize said material. A commitment has been made to REG to provide a reasonable amount of computer time to run sample cases of the codes.

APPENDIX B GENERAL PLANT TRANSIENT DATA

B.1 Summary

This Appendix comprises a listing of the preliminary duty cycle events (normal, upset, emergency and faulted) for the plant with a discussion of how the selection of "umbrella" transients allows simplification of the design duty cycle in a conservative manner.

The listing of events is prefaced with a disclaimer; "...inclusion of an item in this list signifies that it has been utilized for design purposes, but not that the event itself is necessarily regarded either as being credible, or to be expected as frequently as indicated."

Example descriptions are provided below for events likely to be of interest.

Upset Event 17 - Three Loop Natural Circulation

From initial conditions of full power operation, complete loss of forced sodium circulation in all loops is assumed. A reactor/turbine trip is initiated by primary pump under-voltage relays. Steam pressures increase causing some relief of steam through the power operated relief and safety valves. Sodium pumps coast down and stop and natural circulation flow is established in all sodium loops. Auxiliary feedwater flow is established from the auxiliary feedwater portion of the steam generator auxiliary heat removal system based on low drum level signals. The turbine driven auxiliary feed pumps take suction from the protected storage tank to maintain drum levels. Terminal conditions include decay heat removal through SGAHRS.

Emergency Event 7 - One Loop Natural Circulation from Initial Two Loop Operation

Initial operation at a reduced power of TBD% is assumed, using Loops A & B. Loop C is assumed to be shut down. Upon loss of forced sodium circulation, the reactor and turbine are tripped and a transient similar to U-17 occurs in Loops A and B. Primary flow in Loop B is then assumed to be lost for natural circulation due to an unspecified event. Sodium temperatures in Loop B are then reduced, providing a small down transient at the superheater inlet and outlet and at the evaporator inlet. Sodium temperatures in the hot leg of Loop A increase slightly as the decay heat load to this loop is doubled and flow increases slightly. Superheater temperatures and the evaporator inlet temperature increase correspondingly in Loop A. This event encompasses the three to two loop natural circulation transient.

Emergency Event 6 - Design Basis Steam Generator Sodium-Water Reaction

This event consists of an instantaneous rupture of evaporator or superheater tubes, which results in rupture disk actuation, automatic isolation and blowdown of all evaporator modules and the superheater in the affected loop, and manual activation of the sodium rapid dump system. In addition, a trip of the reactor, turbine, and sodium pumps occurs. The intermediate sodium system experiences a pressure transient resulting from the reaction. This event is classified as a fault for the affected steam generator module. For the rest of the loop, the occurrence is classified as an emergency event.

The plant is tripped on the same signal as that which activated the emergency blowdown system. For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the two remaining loops.

B.2 Variance From Reference Design Report

The only deviations from the Reference Design Report are: (1) inclusion of OHRS events, (2) omission or modification of events affecting the evaporator module outlet isolation valves (the valves no longer exist - see Chapter 5), and the inclusion of Test Event - 1 which is blockage of dummy core assemblies during pre-operational testing.

APPENDIX C RELIABILITY PROGRAM

In this Appendix are given full details of the reliability programs for the shutdown systems and decay heat removal system. A summary of the material in the Appendix is given below. Further details are given in Part II, Section 3 of this Summary, in Section 1.1 of the PSAR (reproduced in Appendix C of this Summary), and in Appendix D of this Summary (reproduced from Appendix C of the PSAR).

C.1 Rationale

This section of the Appendix sets out the criteria for success of the programs in terms of reliability goals and their allocations. The following goals are stated as allocations to the three main systems:

System	Unreliability Goal Per Reactor Year
Primary Shutdown	10^{-4}
Secondary Shutdown	$5x10^{-4}$
Combined Shutdown (Primary and Secondary)	$5x10^{-8}$
Decay Heat Removal	$8x10^{-7}$

These are related to the need to demonstrate the probability of less than 10^{-6}/ year of an event resulting in loss of core coolable geometry, and hence to eliminate core disruptive accidents as a basis for design. A detailed treatment of the rationale for selection of this goal, and for concentration on the three systems indicated above, is given in Section C.1 of Appendix C to the PSAR.

C.2 Descriptions of Programs

The overall programs are given in summary form early in the Appendix, and given a much more detailed treatment (about 100 pages) in the body of the Appendix. The items covered are:

- Reliability manual, as a guide to reliability assessment, and its part in the overall project evolution. The manual covers all the methods of assessment (FMEA, FTA, Monte Carlo Simulation, Bayerian techniques, etc.) and the management procedures required for their implementaiton.

- Analyses already initiated (conditions necessary to assurance of core coolable geometry, rod worth requirements and uncertainties, speed of response requirements, etc.).

- Development of a data base. The first phase is the collation of existing relevant data, and of data from FFTF and program test data prior to CRBR operation; the second phase will be CRBR operating experience.

- Specific features of the Shutdown System test program. This includes the use of test rigs for SCRAM testing, environmental tests, etc. Specifics are given below.

Existing test facilities to be used:

a. ARD's GPL-2 high temperature sodium loop;
b. GE's DTL (Drive Test Loop), also high temperature sodium;
c. ARD's Multipurpose Hydraulic Loop (MPHL) for control rod dashpot and damper tests, and other tests not requiring sodium:
d. GE's Hydraulic Test Rig (HTR) for static flow testing of the control assembly

Proposed future modifications to existing facilities and additional facilities:

a. CRDM Spring Test Rig
b. Bellows Test Facility
c. Friction and Wear Seismic Test Rig
d. Partial Control Room Mockup

This section includes conceptual arrangement drawings of the proposed modifications or additional facilities.

- Future programmatic work including the decay heat removal and reliability program and their test programs.

- Specific features of the Shutdown Heat Removal Systems Program test plan.

- Overall Reliability Program schedule in block diagram form, showing that data being obtained before award of an Operating License.

C.3 Preliminary Assessment of Reliability

A full treatment of the methodology and assumptions is given. This includes a brief summary of the design, which corresponds with the design discussed elsewhere in the PSAR. Where items are currently not fully established, demonstrated conservatism has been used in the assessment. The results quoted are:

System	Current Unavailability Estimate
Primary Shutdown	$5x10^{-5}$
Secondary Shutdown	$6x10^{-5}$
Decay Heat Removal	$7X10^{-7}$

Comparison of these results with the goals quoted above shows the feasibility of meeting these goals in the final assessment.

APPENDIX D EVALUATION OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS FOR THE CLINCH RIVER BREEDER REACTOR PLANT

This Appendix is a compendium of core disruptive accident analyses conducted for CRBR, including selection of initiators, analyses performed and mechanical and radiological consequences.

The early part of the Appendix considers the range of potential initiators, namely:

- Reactivity insertions as either a ramp or a step
- Core voiding by entrained gas bubbles
- Control rod ejection
- Local assembly faults
- Loss of control material
- Loss of primary pumping power

and concludes with the selection of a reactivity ramp of 10 ¢/sec, and a flow coastdown event as the two candidates to be examined further. These are termed, respectively, the TOP and LOF events (transient overpower and loss of flow). The results of analyses of these events are summarized in Table D2-1, reproduced below.

An extensive treatment of the methods of analysis is given, including input assumptions and areas of uncertainty. The Codes used are identified:

SAS 2B	Calculation of energy release and shutdown
VENUS II	Core disassembly phase
REXCO-HEP	Mechanical loads on vessel and internals
PLAP	Modification of REXCO-HEP output into vessel nozzle pressure time histories
TRANSWRAP	Uses PLAP output to give mechanical loads on primary system
HAA 3	Release of radioactive material into the containment space, with due allowance for plate-out, settling, leakage, agglomeration, etc.
COMRADEX	Uses HAA 3 output to calculate site boundary doses

Details of the design configuration and design parameters used as input to the analyses are given, including design drawings. For reasons of timing, some of these data do not correspond precisely with data quoted elsewhere in the PSAR, and a comment on the sensitivity of the conclusions to these changes (concluding negligible sensitivity) is given.

Some 200 pages of the Appendix (about half the total volume) is concerned with a detailed treatment of the analyses conducted, and results obtained, in terms of energetics, and mechanical loads. Included, for example, are tables and figures showing:

- Energy partition among the various components
- Reactor configuration at various times during the excursion
- Pressure time histories at a number of locations

Also included in this section is a discussion on the experimental verification of the theoretical models used, including the tests conducted at the Stanford Research Institute on a scale model of the FFTF. These showed that there is good reason for confidence in the results of the REXCO-HEP Code as a realistic but conservative model.

Some treatment of post accident heat removal capability is included, with a statement of estimated capacities for containment of core debris within the primary system (Table D 8-1, reproduced below). The Appendix recognizes that the loss of flow accidents may not be coolable within the vessel with the reference design and notes that modifications to improve the post accident debris retention capability of the core support structures are being investigated.

Next, the radiological consequences are examined, and results quoted in Tables D 9-1, D 9-2 and D 9-3 reproduced below. Except for the most extreme assumptions of head leakage, these are shown to be within the guidelines of 10CFR100 at the site boundary. The radiological analyses are based on retaining the debris within the vessel,except for head leakage.

The final section includes a discussion of uncertainties and requirements for further technical information.

The analyses in Appendix D are based on the reference design. Therefore, the effects of a sealed head access area or an ex-vessel core catcher are not included. These design features are included in Appendix F.

TABLE D2-1

HCDA ENERGY SUMMARY

ENERGY CHARACTERISTICS	UNITS	REACTIVITY INSERTION		LOSS OF FLOW REPRESENTATIVE CASE	BASIS FOR STRUCTURAL EVALUATION
		UPPER BOUND	EXPECTED		
Thermal Energy Above 298°K	MJ	10,800	5,520	13,500	17,900
Thermal Energy Above Steady State Full Power	MJ	8,480	2,807	11,050	15,450
Molten Fuel Energy Above Solidus	MJ	3,060	287	5,620	10,000
Molten Fuel Mass	KG	5,800	1,060	∿7,000	7,400
Available Fuel Work Energy	MJ				
Expansion to One Bar[1]		155	∿ 0	521	1,320
Expansion to 20 Bar		37	∿ 0	151	470

NOTE: 1) For reference only; system dynamic equilibrium occurs at ∿20 bar.

TABLE D8-1

ESTIMATED CAPACITIES FOR CONTAINMENT OF CORE DEBRIS WITHIN THE PRIMARY SYSTEM OF THE CRBR

(Total inventory assumed to be 7575 kg fuel + 2900 kg blanket, 1/2 of the total axial blanket)

Total Volume as particulate debris bed 73.1 ft^3

Decay Power Fraction	Time After Shutdown	Limiting Debris Depth In.	Limiting Debris Volume $ft.^3$	Fraction of Debris Inventory
		Particulate Debris on Upper Thermal Baffle		
0.08	~0 sec.	1.06	21.9	0.30
*0.04	0.5 min.	2.12	43.8	0.60
0.02	12 min.	3.70	76.5	1.05
		Particulate Debris in Primary System Piping		
*0.04	0.5 min.	2.12	51.8	0.71
0.02	12 min.	3.70	117.8	1.61
0.01	3 hr.	4.59	161.6	2.21
		Particulate Debris in Lower Reactor Vessel Head		
0.02	12 min.	3.70	5.85	0.08
*0.01	3 hr.	4.59	8.96	0.123
0.0075	7 hr.	4.94	10.36	0.142
0.005	1 day	5.43	12.48	0.171

*Most likely values of decay power.

TABLE D9-1

HCDA RADIOLOGICAL CASES ANALYZED

Case	HCDA Type	Head Leak Rate (%/Day @ 20 Atm)	Sodium Leaked (lb)	RCB Pressure (psig)
1	TOP*	10	0.037	1
2	TOP*	10^2	0.37	1
3	TOP*	10^3	3.7	1
4	TOP*	10^4	37	1
5	TOP*	∞	>5600***	10
6	LOF**	10	0.037	1
7	LOF**	10^2	0.37	1
8	LOF**	10^3	3.7	1
9	LOF**	10^4	37	1
10	LOF**	∞	>5600***	10

* TOP-HCDA - Sodium vapor bubble formed
Material released to cover gas
100% Noble Gases
4% Halogens
0.24% Solid Fission Products
2.85 kg PuO_2 (11.4 kg fuel)

** LOF-HCDA - Fuel vapor bubble formed
Material released to cover gas
100% Noble Gases
6.3% Halogens
1.5% Solid Fission Products
28.6 kg PuO_2 (114 kg fuel)

*** Sodium release was assumed to raise the RCB pressure to 10 psig. Burning 5600 lbs. of sodium as an ideal spray with all the energy released used to heat the RCB atmosphere results in a 10 psig pressure increase.

TABLE D9-2

RADIOLOGICAL CONSEQUENCES OF
HYPOTHETICAL TOP-HCDA'S

	10 CFR 100	Expected	Off-Site Doses (REM) Upper Bound Base Parametric Cases 1	2	3	4	5
2 Hr S.B. (0.41 mi)							
Bone	150	0	0.120	1.19	10.78	66.02	184.6
Thyroid	300	0	0.052	0.519	4.72	28.92	81.50
Lung	75	0	0.0077	0.0759	0.690	4.22	11.82
Whole Body	25	0.038	0.029	0.27	1.64	3.28	3.75
30 Day LPZ (5 mi)							
Bone	150	0	0.025	0.246	2.22	13.4	10.35
Thyroid	300	0	0.009	0.092	0.831	0.502	4.26
Lung	75	0	0.0016	0.015	0.139	0.842	0.65
Whole Body**	25	0.0015	0.0035	0.014	0.034	0.063	0.13
Head Leak Rate (%/day at 20 atm.)			10	10^2	10^3	10^4	∞
RCB Pressure (psig)		1	1	1	1	1	10

**Whole body dose includes direct dose and cloud gamma.

TABLE D9-3

RADIOLOGICAL CONSEQUENCES OF
HYPOTHETICAL LOF-HCDA

	10 CFR 100	Off-Site Doses (REM) Representative Parametric Cases				
		6	7	8	9	10
2 Hr S.B. (0.41 mi)						
Bone	150	0.093	0.926	9.11	131.7	1893
Thyroid	300	0.0063	0.063	0.616	8.90	128.
Lung	75	0.005	0.050	0.488	7.05	101.
Whole Body	25	0.028	0.258	1.54	3.04	5.
30 Day LPZ (5 mi)						
Bone	150	0.019	0.185	1.82	26.4	106.
Thyroid	300	0.001	0.011	0.105	1.52	6.
Lung	75	0.001	0.01	0.097	1.40	5.
Whole Body**	25	0.0035	0.014	0.031	0.062	0
Head Leak Rate (%/day at 20 atm.)		10	10^2	10^3	10^4	∞
RCB Pressure (psig)		1	1	1	1	1

**Whole body dose includes direct dose and cloud gamma.

APPENDIX E - PRIMARY PIPE RUPTURE FALLBACK POSITION

E.1 Introduction

The basic project position is that large pipe ruptures in the primary heat transport loops have a low enough probability so that such an occurrence should not be used as a basis for the design of the CRBRP. (Note that this is not the position adopted for IHTS rupture, as discussed in Part II Section 6 of this Summary). However, recognizing that this position cannot be fully supported at this point in time, a parallel design is being vigorously pursued which can be incorporated into the CRBRP if necessary without adversely affecting the startup schedule.

The objective of this section is to establish a pipe rupture accommodation program (as a fallback position) with the goal of developing a design which will mitigate the consequences of a large primary pipe rupture and assure acceptable core temperature and cell transient conditions. The approach, as described in the various subsections of this Appendix, is to:

1. Define the general requirements and key objectives for the parallel design.
2. Describe the current status of the program along with a brief description of the studies being conducted to establish the low probability of such pipe ruptures.
3. Provide an overall description of the program and key decision points.

The major requirement, predicated on the assumption that a double-ended rupture in the primary system must be accepted as a design basis, is that the modifications to the heat transport system and containment structures shall be designed as necessary to accommodate the consequences of postulated ruptures, including a double-ended break in the primary piping, for all anticipated 3-loop and 2-loop operating conditions.

Two of the principal key objectives of the program are that: modifications to the reactor vessel and/or the heat transport piping shall maximize capability for in-service inspection of the coolant boundary, consistent with the requirements of the ASME Boiler and Pressure Vessel Code and modifications to the existing design shall be capable of being built and installed in the plant without adversely affecting the start-up schedule.

E.2 Discussion of Program

Current design studies indicate that the pipe sleeve concept is the most promising design option for accommodation of core transients due to double-ended ruptures in the primary piping. The principal design features of the sleeve concept are shown in Figure E-1 below. Core transient analyses performed to date indicate the need for the pipe sleeve protection only between the reactor vessel and the top of the inlet downcomer. However, the design provides for a sleeve extending up to the flowmeter inlet to provide an additional safety margin.

The structural analysis of the pipe sleeve arrangements will be performed in conjunction with the analysis of the primary heat transport system during the preliminary design phase of the pipe rupture accommodation program. Preliminary calculations for double-ended ruptures at various sections in the inlet downcomer piping show that these loadings will result in stresses below allowable ASME Code limits for the reactor vessel and the sleeve.

In-service inspection will be conducted by visual inspection. The inspection of the primary piping protected with the pipe sleeve will consist of remote visual viewing in the HTS cell and the HTS pipeway between the PHTS cell and the reactor cavity.

Leak detectors will be provided at selected locations on the pipe and at the bottom of the vertical pipe sleeve runs. A detailed discussion of the various types of leak detectors (spark plug, aerosol detectors, radiation monitors, level detectors) proposed to be used in the primary heat transport system is provided in the PSAR Section 7.5.5.

E.3 Analysis

Analyses of core transients resulting from double-ended pipe ruptures in the primary heat transport loops have been performed for three-loop plant operation based on thermal/hydraulic design parameters. The preliminary results of the analysis for three-loop operation at full power indicate that a double-ended pipe rupture in the primary heat transport system can produce unacceptable core temperature transients only if the break occurs in the cold leg piping between the reactor inlet nozzle and the top of the downcomer. In this region of the PHTS piping, a double-ended break results in hot channel coolant temperatures exceeding saturation limits within a period of less than one second. Incorporation of the pipe sleeve concept mitigates this accident event. At the reactor inlet, representing the worst case location for pipe rupture, the coolant temperature was calculated to be 90°F below the calculated saturation temperature.

To conservatively predict the pressure transient resulting from the postulated ruptures, heat transfer from the discharged sodium to the inert atmosphere is assumed to be ideal. For this limiting case, the temperature of the PHTS cell or RC inert atmosphere is assumed to instantaneously increase to the temperature of the discharged sodium. The resultant cell/cavity pressure was also determined ideally assuming that the perfect gas law applies.

Preliminary analysis of the primary HTS and reactor cavity transients resulting from a double-ended pipe rupture has also been performed. The results indicate peak PHTS cell and RC pressures on the order of 25 psig, corresponding to an increase in the temperature of the inert atmosphere from 90°F to 1015°F, the peak hot-leg sodium temperature. In addition to the pressure and temperature transients imposed on the PHTS cells or RC following pipe rupture, the potential exists for bubbling gas into the primary system. Based on the geometric configuration of the primary heat transport system and the lower plenum pressure history following double-ended ruptures, preliminary analyses have shown that if the cell or cavity gas pressure following a pipe rupture is maintained below ∿10 psig, the potential for gas introduction to the lower plenum does not exist, If a postulated double-ended rupture is required to be treated as a basis for design with a peak cell temperature and pressure of 1015°F and 25 psig respectively, a different type of seal between the PHTS cells and the RC will be designed, which would yield under a pressure differential of 1 psi. Incorporation of these seals would enable the PHTS cells to be designed to 10 psig. The PHTS cell liners will be designed to accommodate the maximum temperature of 1015°F.

From a radiological standpoint, preliminary analysis indicates that a large margin (greater than a factor of 10^4) exists between the potential doses at the site boundary and low population zone and the applicable guideline limits.

E.4 Overall Program Description

A program has been established to develop and design plant modifications that will accommodate the consequences of postulated double-ended pipe ruptures in the primary heat transport piping. The details (schedule and key decision points) of the project program are shown in Figure E-2, reproduced below.

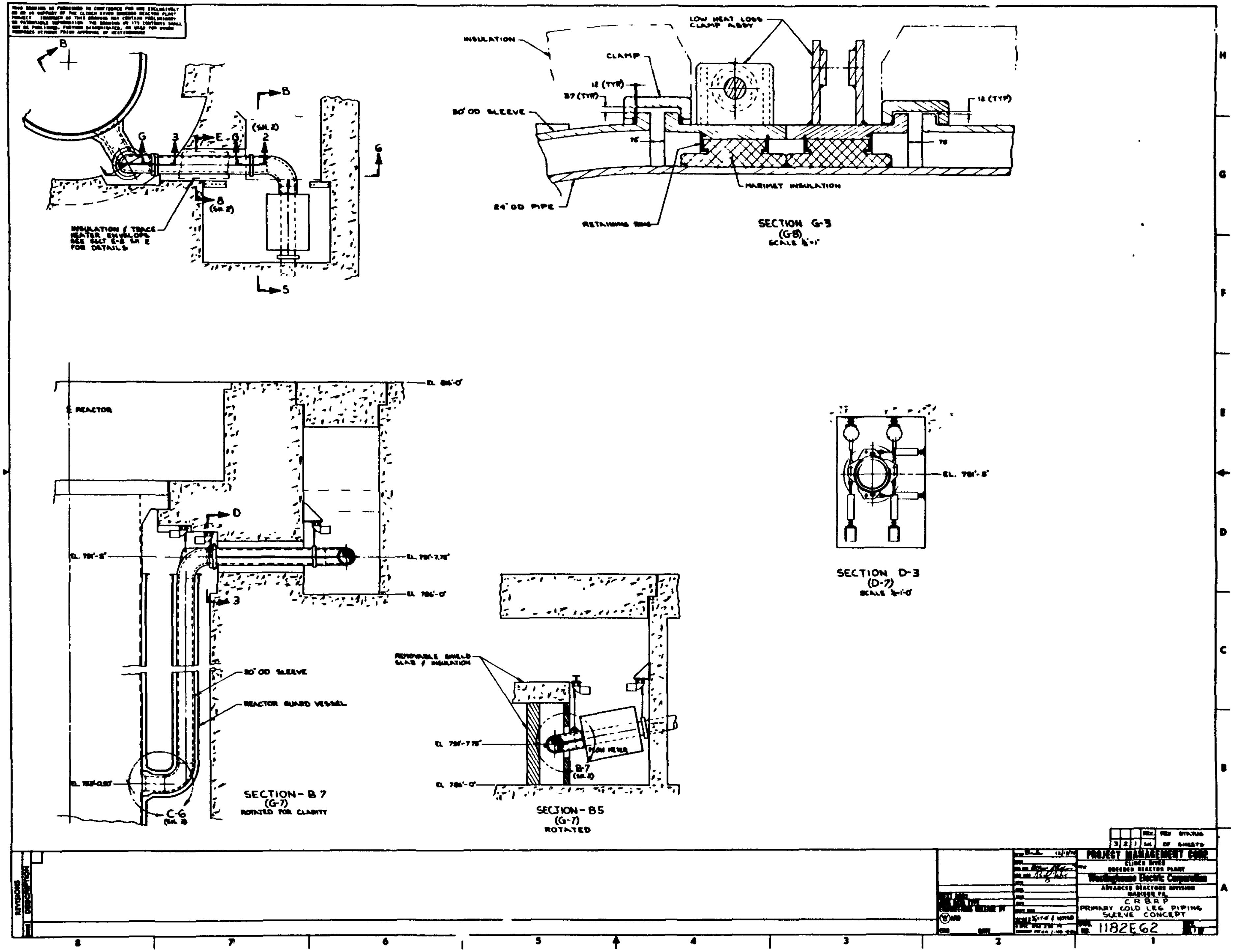

LOW HEAT LOSS CLAMP ASSY
INSULATION
CLAMP
30" OD SLEEVE
24" OD PIPE
RETAINING RING
MARIMET INSULATION
SECTION G-3
(G-8)
SECTION D-3
(D-7)
EL. 791'-8"
80" OD SLEEVE
REACTOR GUARD VESSEL
SECTION - B 7
(G-7)
ROTATED FOR CLARITY
REMOVABLE SHIELD SLAB & INSULATION
FLOW METER
SECTION - B5
(G-7)
ROTATED
EL. 786'-0"
PROJECT MANAGEMENT CORP.
CLINCH RIVER BREEDER REACTOR PLANT
Westinghouse Electric Corporation
ADVANCED REACTORS DIVISION
MADISON PA.
C R B R P
PRIMARY COLD LEG PIPING
SLEEVE CONCEPT
1182E62

Figure E-1

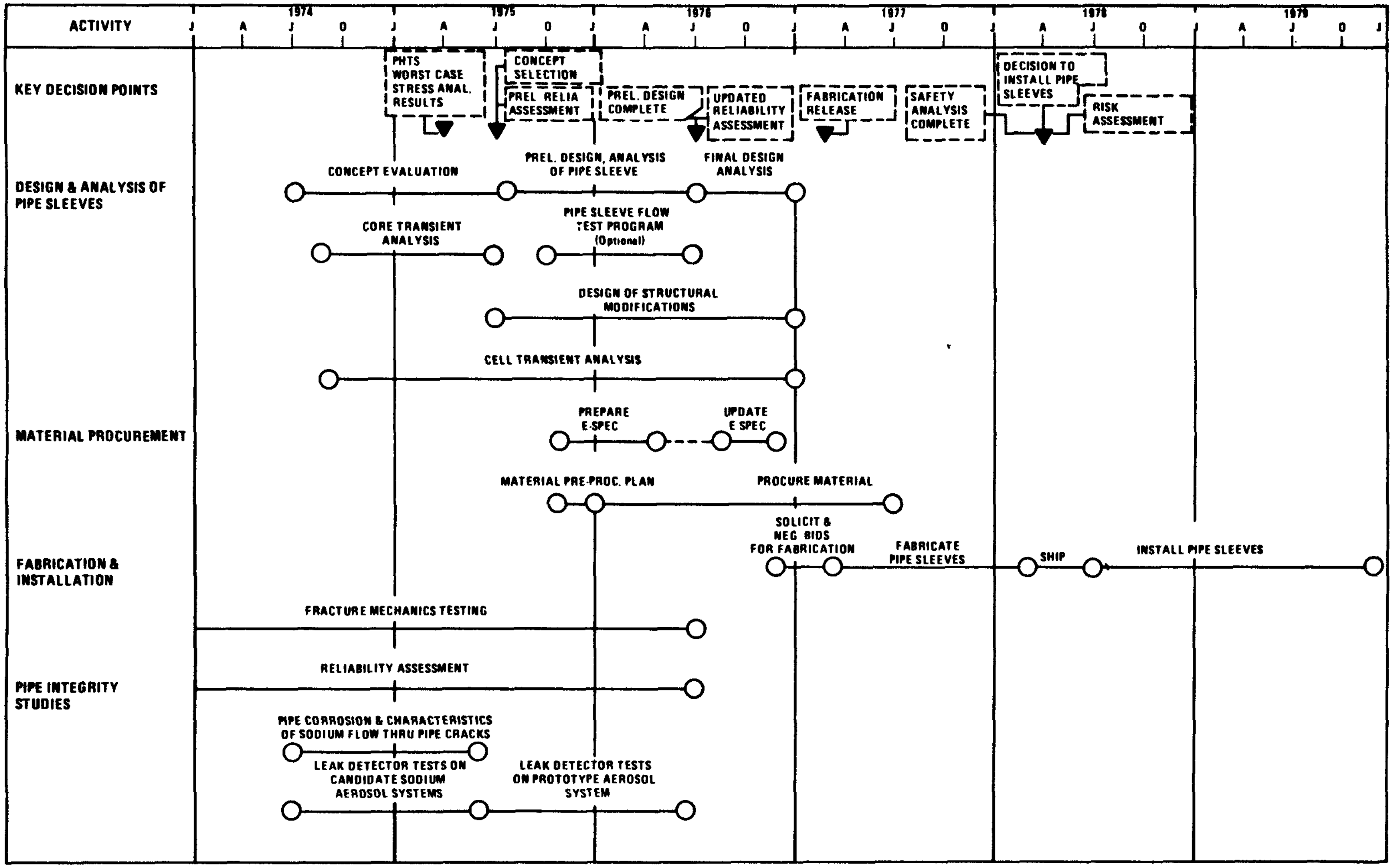


Figure E-2 Pipe Rupture Accomodation Program

APPENDIX F CORE DISRUPTIVE ACCIDENT ACCOMMODATION

F.1 Introduction

The project position remains firm, that any event capable of leading to a loss of core coolable geometry is so improbable that it should not be accepted as a design basis for the plant. This position, and the part played by Appendix F, is laid out in Section 1.1 of the PSAR (reproduced in Appendix C of this Summary), and repeated in the introduction to Appendix F. It is made clear that the treatment of a core disruptive accident as a design basis, in Appendix F, is being done to follow the agreements of Reference 1, and should not be interpreted as any change in the Project position. Following this statement, a CDA is discussed as a design basis event throughout the Appendix.

It is also made clear that this Appendix represents a status report only, and that the final version of Appendix F will be submitted in September 1975. It has already been determined (meeting with NRC 1/24/75) that this will not be regarded as a docketing issue by NRC.

The Appendix is divided into two Parts, of which the first deals with the sealed HAA, and the second with the EVCC.

Part I - Sealed HAA

After a brief introduction, there follows a fairly detailed listing of criteria and design requirements for the sealed HAA. Some examples of the 18 requirements listed are:

- The HAA shall be sealed to limit the radiological effects of the design basis CDA below the guidelines of 10CFR100.
- The sealed HAA shall be designed as a seismic Category I structure.
- All gas lines and connections shall be capable of withstanding the pressure resulting from the sodium egress consequences of the design basis CDA.

There follows a five page discussion of the design program, supported by a schedule. This shows, for example, input from a mechanistic CDA assessment by May 1975, and a preliminary risk assessment and updated CDA analysis by mid 1977.

The remaining 12 pages of the text cover the current status of the program. Both air filled and inerted concepts are discussed, and several sealing concepts described. (Large dome, small dome, fabricated panel type structure and some variants of these) In each case design drawings are given, and a date of 8-1-76 is quoted for completion of preliminary design.

PART II: ITEMS OF SPECIAL INTEREST

Part II - Ex-Vessel Core Catcher (EVCC)

This Part is significantly more extensive than Part I (approximately 100 pages in volume). The Introduction includes a review of current EVCC studies at ANL, Interatom and elsewhere.

Design requirements are presented, some examples of which are given below:

- The EVCC shall be designed to prevent recriticality of the debris from the design basis CDA.

- The design shall assure that debris which penetrates the Guard Vessel reaches the Lower Cavity EVCC.

- Class 1E power supplies and controls shall be provided as required for the system to function.

- The EVCC system shall be designed to Seismic Category I requirements.

- The bed material shall have a high volumetric heat absorbing capability.

The initial conditions are discussed, in terms of melt through of the reactor vessel and guard vessel of 100% of the core and axial blankets as well as 50% of the radial blanket. The decay heat load from this mass is stated, and the effects of the reactor vessel sodium are discussed.

The program of activities is described and supported with schedules which show the compatibility of this effort with the overall plant construction schedule and with the time scales for availability of results from the reliability program.

Three candidate concepts are presented (sacrificial bed, crucible and suspended catch trays). These are briefly described, with some conceptual drawings and analyses relating to secondary criticality and heat loads.

A number of Addenda to this Part are presented, covering:

- Analysis of bed material melting

- Discussion of EVCC transient response

- Molten pool heat transfer

- Bed heat load

- 20 page description of an actively cooled sacrificial bed system, supported by design drawings

In each of these, numerical details are presented.

Reference 1. Letter from L. Manning Muntzing (Director of Regulation) to John A. Erlewine (USAEC General Manager) "CRBRP Licensing Review", January 2, 1975.

1 INTRODUCTION

In this Part are discussed a number of items which are of special interest from the Licensing standpoint. Their inclusion in this Part denotes, principally, that these are items which are either novel to this particular plant (e.g., use of water in containment), or have required significant discussion in arriving at a Project position (e.g., fuel defect limits). The intent is set on record the conclusions reached by the Project, in a readily assimilable form. The conclusions themselves are not in question, and are considered to represent a position which is acceptable to the designers and should not adversely affect licensability.

2 SITE CHARACTERISTICS

2.1 Seismology

2.1.1 Description of Item

The site is located in Zone 1, minor damage, on the USGS Seismic Probability Map of the United States. The seismic history of the southeastern United States indicates that there has been no seismic activity originating in the site area. The Safe Shutdown Earthquake (SSE) for the plant has been established as having a maximum horizontal ground acceleration of 0.18g and a simultaneous vertical acceleration of the same magnitude.

Based on historical data at the site vicinity, the maximum horizontal ground acceleration is less than one half of the SSE established. In accordance with 10CFR100 Appendix A, an OBE of 1/2 of the SSE or 0.09g is selected.

This Item is included in this Section because the derivation of the SSE is of considerable significance to the design of the plant.

2.1.2 Synopsis of Treatment

The Seismology information for the site is provided in Section 2.5 of Chpater 2 of the PSAR. This includes basic geologic and seismic information, earthquake history, and the basis and evaluations relating to establishing the SSE for the plant.

Specification of an SSE is dictated by a combination of requirements of 10CFR50 Appendix A GDC 2, Regulatory Guide 1.29 and 10CFR100 Appendix A. This information impacts all the PSAR "Design" Chapters and Chapter 15, "Accident Analyses". It affects the design requirements for all the safety related systems, equipment, and structures of the plant.

A common difficulty encountered in nuclear plant site seismic evaluation is lack of historical data. This plant site is no exception. Before the year 1800, much of the region was sparsely populated such that the epicenters were identified with the scattered towns, which could be tens of miles from the actual epicentral locations. After 1800 and up to about 1960, epicentral locations were largely based upon intensity estimates. Because of this lack of historical data, the most severe earthquake associated with tectonic province in which the site is located is determined in a conservative manner. Were this not the case, a smaller SSE may possibly be specified for the plant.

2.1.3 Consideration of Alternatives

Two alternatives, a lower SSE and a higher SSE for the plant, are discussed below.

1. *Attempt to justify a lower SSE (0.14g is used as an example since this was at one time proposed for Sequoyah).*

   The tectonic structures in the site general region are considered by experts as ancient and inactive, which is also supported by the recent site investigation carried out by Law Engineering. No known correlation between tectonic structures and epicentral locations can be established (See Section IV.(a).(6) of Appendix A to 10CFR100). Furthermore, no evidence for any capable faulting within 200 miles of the site has been observed (See Section IV.(a).(8) of Appendix A to 10CFR100). Earthquakes are thus identified with the tectonic province, a procedure set forth in Section V.(a),(1).(ii) of Appendix A to 10CFR100 and the same as used in the determination of SSE for the Sequoyah Nuclear Plant which is about 60 miles from the Clinch River site. The Clinch River site is located in the same tectonic province as Sequoyah. This province has been designated as the Southern Appalachian Tectonic Province by the AEC in their evaluation of the Sequoyah Nuclear Plant. The maximum historic quake reported in this province was assigned an intensity of Modified Mercalli (MM) VIII and occurred in 1897 in Giles County, Virginia, about 220 miles NE of the Clinch River site. The same earthquake was used in the determination of SSEs for both this plant and Sequoyah, pursuant to Section V.(a).(1).(ii) of Appendix A to 10CFR100.

The Sequoyah SSE was originally estimated to be 0.14g. However, as the results of a position taken by the Regulatory geological-seismological consultants, it was later raised to 0.18g. Largely because of this precedent, it is rather doubtful that the alternative (i.e., a 0.14g SSE) would be acceptable to Regulatory for this plant.

2. *Adopt a higher value for SSE.*

This alternative appears to be unnecessarily over-conservative.

Firstly, the maximum historic quake reported (and used in the SSE determination as discussed above) was assigned an intensity of MM VIII, although it was believed by some that it should have been rated as MM VII.

Secondly, it is assumed that this highest intensity earthquake occurred at the site, although it actually occurred at some 220 miles away from the site. This is pursuant to the requirement of Section V.(a).(1).(ii) of Appendix A to 10CFR100; it is nevertheless additional conservatism which was necessary largely due to lack of historical data. The SSE specified for Sequoyah serves as a good precedent. In addition, the Sequovah SSE specifies a maximum vertical ground acceleration of 0.12g or 2/3 of the horizontal component, while the SSE for the Clinch River has both specified as 0.18g.

In view of the above, the alternative of a higher SSE appears to be over-conservative.

2.1.4 Conclusion

The site seismic evaluation for the plant as reported in the CRBRP PSAR is considered appropriate and consistent with those undertaken by essentially all the LWRs. The SSE of 0.18g as determined for the plant is considered reasonable and conservative.

2.2 Meteorology

2.2.1 Description

Reg. Guide 4.2 ("Preparation of Environmental Reports for Nuclear Power Plants") calls for meteorology based on at least one year on-site data. The guide also states that if at the time of filing for a CP the detailed information may not be available, "the applicant may present information applicable to the general site area when available from the U.S. Weather Bureau or other authoratative sources". Present PSAR meteorological description is based on off-site data on which Regulatory has expressed doubt concerning applicability.

2.2.2 Synopsis of Treatment

PSAR calculations in chapters such as 11 (Normal Operations) and 15 (Accidents) utilize meteorological data taken from the Oak Ridge data station located 4.5 miles northeast of the CRBRP. An on-site monitoring program was initiated on April 11, 1973 with the intent of accumulating and analyzing data acceptable for use in analyses in the ER. "Acceptable" in terms of Reg. guide 1.23 implies 90% data recovery.

Analysis of the first 6 months of the on-site program indicated data recovery below the acceptable recovery level. This data is presently in the ER and PSAR in appendix form for "comparison" only. The problem in the on-site data collection process was identified, and in January 1975, TVA provided WESD with 1 year (Aug. '73 to Aug. '74) of data with an 88% recovery rate. WESD should have its analysis of the new data completed by mid-February ready to respond to a critical item for ER-PSAR Chapter 2, Docketing.

2.2.3 Consideration of Alternatives

1) Utilizing CRBRP on-site data with sub-par data recovery rate.

Not a sound alternative. Data recovery (70%) was not in conformance with Reg. guide 1.23 (90% recovery).

2) Utilizing off-site data only.

This alternative would fail to inform NRC of the up-to-date on-site data accumulation effort of the project and as such is not appropriate. In addition, NRC was not convinced of the applicability of the off-site data to the CRBR site.

3) *Delay the PSAR until a set of on-site data with acceptable recovery rate was assured.*

It was not appropriate for the Project to delay submittal for this reason. If necessary, the PSAR could be modified to include acceptable on-site meteorological data when it became available.

4) *Utilize off-site data in calculations, present the six months of sub-par on-site data in appendix form.*

This approach does not result in delay of PSAR submittal while letting NRC know of the project efforts to accumulate acceptable on-site data.

2.2.4 Conclusion

Alternative (4) was selected as the course of action. The new on-site data being analyzed by WESD is expected to show greater dispersion characteristics than does the present data presented in chapter two. Thus, an increased conservatism will be contained in the present PSAR analysis. NRC should be made aware of this fact.

3. RELIABILITY PROGRAM TREATMENT

3.1 Description of Item

The item discussed here is the overall reliability program, including that for both shutdown systems and the decay heat removal system. It is included in this Part because of its central significance to the CP application for this Project. Only a very brief treatment is given at this point, since the majority of the Introduction to Appendix C of this PSAR is included as Appendix D of this Executive Summary.

3.2 Synopsis of Treatment

The PSAR coverage of this item is addressed in Section 1.1 of the PSAR, reproduced as Appendix C of this Summary. In particular, Table 1.1-1 and 1.1-2 indicate the various Sections of the PSAR which contain relevant material. Because this subject is also treated in some detail in Part I of this Summary, this synopsis has been abbreviated.

The position adopted by the Project is that Shutdown System and Decay Heat Removal System reliabilities must be high in order to prevent an accident resulting in loss of core coolable geometry. A specific goal of 10^{-6}/yr for the overall probability in loss of core coolable geometry is quoted, and preliminary assessments of the reliabilities of the systems are given (0.3x10^{-8}/yr for failure to scram and 0.7x10^{-6} for loss of decay heat removal).

At the time of writing there is not complete Project agreement on the manner of implementation of RDT Standard F9-2 (Reliability Assurance). This Standard demands that reliability requirements be placed on component vendors in the E-Specs. The alternative approach, favored by the designers, is to place requirements on the System design, and to assess the reliability of the system based on vendor data. It is expected that this will be resolved within the first half of February 1975, following the LRM recommendation.

3.3 Consideration of Alternatives

1. *Submit the reliability program as a Topical Report, either with or separately from the PSAR.*

   This was considered the wrong course of action because of the central role played by the reliability program in this application. Regulatory is aware that this material will be in the PSAR.

2. *Omit the preliminary assessments of reliability of the shutdown and decay heat removal systems.*

   These were included because final data will not be available until after the date of the Construction Permit. It was considered necessary to demonstrate the feasibility of meeting the goals. The results of the assessments will undoubtedly change as the program evolves and in no sense should the numbers quoted be regarded as final. However, this inclusion enhances confidence in the reliability approach.

3. *Accept a core disruptive accident as a design basis, and omit the reliability program entirely.*

   The phenomenology of core disruption is not sufficiently well understood that assured protection can be provided from it at this time. Further, the cost of providing such protection is a burden that the LMFBR industry should not be expected to bear, because of inherent confidence in the exceedingly low probability of such events. It is judged to be a far better use of funds to direct effort towards improvements in reliability and assurance of prevention of such accidents, than to entertain expensive development programs aimed at providing protection against such a low probability event.

3.4 Conclusion

The approach adopted in the PSAR is considered appropriate because:

a. Technically, prevention of core disruption is certainly preferable to provision of protection.
b. The central position of the reliability program in this application makes it necessary to include it in the PSAR.
c. The feasibility of meeting the reliability goals needs to be established at this stage, so that preliminary reliability assessments need to be included.

4 CORE DISRUPTIVE ACCIDENT TREATMENT

4.1 Description of Item

The item under discussion is the total treatment of core disruptive accidents in this PSAR. The place of the parallel design is covered in Section 1.1 of the PSAR, reproduced as Appendix C of this Summary, and will not be addressed here.

The item is included in this Part because it is clearly a key item in the licensing of LMFBR's in this country. Although mention has been made of it in the Chapter and Appendix Summaries, it is considered worth while to collate this material into one place and reiterate the major points.

4.2 Synopsis of Treatment

In Section 1.1 (reproduced as Appendix C of this Summary) are contained introductory remarks on the method of treatment, including an overview of the significance of the third level margin requirements, and the parallel design approach. Each of Chapters 3 through 12 contains a statement (without justification or demonstration of feasibility of compliance) of the third level margin requirements appropriate to the System under discussion. In Section 15.1.1 (Chapter 15) these requirements are collated to give visibility of the totality of the third level margin requirements for the plant as a whole. Section 15.1.1 also discusses the derivation of these requirements, from generic studies and FFTF experience and feasibility of compliance with these requirements.

Appendix D of the PSAR reports analyses of core disruptive accidents performed for CRBR, including mechanical consequences and radiological considerations. Some of the results of these analyses are compared, in Section 15.1.1, with the third level design margin requirements and the latter requirements are shown to be conservative in most cases.

Appendix F of the PSAR will be submitted in Fall of 1975, and will demonstrate the capability of the plant to accommodate a core disruptive accident as a design basis. The version of Appendix F being submitted at this time is preliminary only, and to be regarded as a status report of activities in this area.

4.3 Consideration of Alternatives

1. *Submit Appendix D as a Topical Report, either with or separately from the PSAR.*

   Regulatory have repeatedly expressed interest in seeing core disruptive accident analyses for CRBR. The inclusion of this material as an Appendix is a part of the parallel design approach negotiated with Regulatory. Regulatory has already reviewed an early draft of Appendix D and is expecting this material in the PSAR.

2. *Accept a core disruptive accident as a design basis, omit the reliability program, and treat core disruption throughout the PSAR in the same way as the LWR's treat a LOCA.*

   See discussion under 3.3, Item 3.

3. *Await completion of Appendix F before submittal of the PSAR.*

   Regulatory have been approached and have stated that non-completion of Appendix F at the time of initial submittal will not, in itself, constitute a reason for refusal to docket.

4.4 Conclusion

The present method of treatment is one which has evolved during production of the PSAR and is a principal reason for the delay in submittal. It has evolved largely by negotiation with Regulatory and should not be modified without Regulatory involvement.

5 PRIMARY BOUNDARY INTEGRITY CONSIDERATIONS

5.1 Description of Item

The item under discussion is the treatment in the PSAR of the integrity of the PHTS boundary. It is included as a special item because the approach adopted (incredibility of the event) differs from that adopted for LWR's.

5.2 Synopsis of Treatment

A synopsis of the treatment of PHTS integrity is given in Section 1.1 of the PSAR, which appears as Appendix C of this Summary. No further discussion is given here.

5.3 Consideration of Alternatives

The only alternative is to accept a PHTS rupture as a design basis accident (either for the entire PHTS or a portion of it).

This is the approach adopted in the 'parallel design' covered in Appendix E. (See Summary of Appendix E in Part I of this Summary for a more detailed discussion). To do this would be to accept an undesirable precedent. There is considerable confidence that the case for incredibility can be made (for the PHTS only, the IHTS case differs, see next section), and the associated R and D programs are within a few months of completion. Thus it behoves the Project to hold to its position for the Reference Design at PSAR submittal.

5.4 Conclusion

The present method of treatment is appropriate, since it is defensible and the only possible alternative is one which would set an unnecessary and undesirable precedent for the LMFBR industry.

6 INTERMEDIATE BOUNDARY INTEGRITY CONSIDERATIONS

6.1 Description of Item

The item under discussion is the treatment in the PSAR of the integrity of that part of the IHTS which is exposed to an air atmosphere. It is included as a special item for two reasons. Firstly, because the treatment differs from that of the PHTS, secondly because the position has only recently developed, and its implications may not yet have been appreciated by senior management.

6.2 Synopsis of Treatment

In Chapter 5 of the PSAR appears a discussion of materials properties. This is in a similar level of detail to that of the PHTS, except in the areas of OA and of crack propagation characteristics. The QA treatment is deliberately in less detail, so as not to overcommit the Project beyond what is feasible - to apply the exhaustive QA needed for the PHTS to every part of the IHTS would be extremely costly, and probably not justified. Crack propagation analyses are less detailed because corrosion, in the air atmosphere surrounding this system, is of greater significance.

Leak detection arrangements are discussed in Section 7.5.5, in a similar level of detail to that for the PHTS, and include the use of smoke detectors and of visual inspection.

In Section 15.6 a guillotine break of an IHTS is treated as an Extremely Unlikely, (but not incredible), Event. This significant difference from the treatment of the PHTS is explained in terms of the existing corrosion data, which indicates that, at operating temperatures in an air atmosphere, it may be possible to corrode through a pipe in a matter of hours.

The core temperature transient, and off-site radiological doses, are shown to be negligible. However, the spillage of sodium results in significant pressure buildup in the air atmosphere of the intermediate bay; a method of venting is identified. Consideration is given to protection of safety related equipment from the direct effects of spill and it is shown that protection is provided. Also considered is the potential for sodium concrete reactions and for contact of sodium with the groundwater. Protection, in the form of liners, is demonstrated.

6.3 Consideration of Alternatives

1. *Claim incredibility of IHTS ruptures.*

   With the scarcity of applicable corrosion data, this is not a viable option. It is, however, recommended that efforts be made to obtain further corrosion data. This is under active discussion between ARD and RRD.

2. *Enhance the status of the leak detectors to Engineered Safety Features.*

The multiple redundancy, testing and reliability requirements, together with the current state-of-the art in leak detection methods make this an undesirable alternative.

6.4 Conclusion

The current method of treatment is appropriate in that it represents a defensible case without major perturbation to the design of the plant. However, it does not preclude the possibility of supplementing the PSAR at some future time, should it be possible to show that the present corrosion data are unnecessarily conservative.

7 FAILED FUEL LIMITS

7.1 Description of Item

This section deals with the "current" project position associated with the CRBRP design basis for continued plant operability with failed fuel. During the past few months, there has been considerable discussion between project participants regarding CRBRP failed fuel limits and the objective of consistency with;

1. Commercial practice for fuel design of large nuclear power plants.
2. Meeting as low-as-practical release guidelines
3. Controlling costs of the CRBRP system and building designs.
4. "Hands-on-maintenance" philosophy
5. Plant availability requirements
6. Licensability
7. FFTF design basis.

7.2 Synopsis of Treatment

Tentative agreement between RRD, PMC and ARD was reached on Jan. 28, 1975. The agreement reached was to utilize as a design basis for fission product release the occurrence of defects in the cladding of fuel rods generating 1% of the core power. The design basis for plutonium circulating in the reactor coolant was established early in November 1974 as 100 ppb.

Details of the fission product release models are discussed in Chapter 11 of the PSAR. The results of the agreed on position are consistent with items 1 through 6 above, however, the model differs from FFTF in that FFTF considers gaseous fission products only, and limits the plutonium circulating in the coolant to 10 ppb.

The failed fuel model and calculated radioactivity concentrations are used in PSAR Chapters 4, 6, 9, 11, 12, 15 and 16, and appendices D, E, and F, as well as in the CRBRP Environmental Report (ER). With regard to the ER, the current position will require some modifications to the data in the ER, however, the overall conclusion (site related dose data) will not be affected.

7.3 Consideration of Alternatives

1. *Utilize a 1% failed fuel model based on "Kayser" release fraction for fission products and peak plutonium buildup of 1 ppm in the reactor coolant.*

   This approach was rejected by RRD since they were concerned that this approach would have a serious impact on system design requirements and eliminate the advertised "hands-on" maintenance approach.

2. *Utilize a 1% failed fuel model with fission product gas release only and peak plutonium buildup of 10 ppb in the reactor coolant.*

   PMC and ARD rejected this approach, since it was felt that such an approach would result in requiring numerous plant shutdowns and outages.

3. *Utilize a 1% failed fuel model with 90% of the failures in the gas plena zone and the remaining 10% in the fueled zone and peak plutonium buildup of 100 ppb.*

   PMC rejected this approach in that they felt that said distribution of fuel rod failures was not a defensible position and was not born out by experience.

7.4 Conclusion

The position adopted is a compromise between differing viewpoints. The points at issue, however, affect economics and ease of operation, rather than licensability. The course of action taken is one which should be readily licensable.

8. OPEN CONTAINMENT FUEL HANDLING AND EX-CONTAINMENT FUEL STORAGE

8.1 Description

The current fuel handling system requries that the 44.5 foot diameter equipment hatch be removed in order to transfer fuel from the reactor to the Ex-Vessel Storage Tank (EVST). In addition, the fuel handling system requires that during fuel transfer operations the tornadic integrity of the Reactor Service Building (RSB) be maintained by securing the access opening in the RSB with a tornado missile hardened door. Following the transfer operation, the irradiated fuel is stored in the EVST located in the RSB. Cooling provisions for the EVST are accomplished with redundant cooling loops.

The CRBRP is the first domestic sodium-cooled reactor proposing refueling and spent fuel storage in a building with direct communication to the environment and as such will require an in-depth defense against all possible fuel handling accidents. The LRM is currently performing such an in-depth review.

8.2 Synopsis of Treatment

Accidental release of radioactivity to the environment during fuel handling operations from radioactive systems are precluded as follows:

1. Primary Heat Transport System Fires During Maintenance - During those periods when the containment building hatch is open and the primary loop contains sodium, the HTS cells must remain inerted.

2. Release of Cover Gas Radioactivity - Before violating containment integrity, the concentration of radioactivity in the cover gas will be measured, and reduced if required, to ensure that in the event of a release of the total cover gas, site boundary dose limits would not be exceeded.

3. Accidental Dropping of an Irradiated Fuel Assembly and Subsequent Fuel Melting - Accidental dropping of an irradiated fuel assembly is prevented by the geometry of transfer machines, by the use of four (4) sequential interlocks and a design provision to prevent unlatching a grapple under load conditions.

4. Irradiated Fuel Assembly Melting Due to Loss of EVST Cooling - Loss of cooling to the EVST is precluded by the use of redundant cooling loops.

5. EVST Cooling Loop Sodium Fire - Each cooling loop is located in a cell with an inerted atmosphere to limit any potential for combustion. In addition, the maximum plutonium concentration in the EVST sodium will be limited to 100 ppb.

8.3 Consideration of Alternatives

1. *Refueling and storage within containment and use of large equipment air lock.*

   This alternative was originally proposed for the CRBRP, however, as a result of cost and design optimization analysis performed late in 1973, this alternative was rejected.

2. *Design the RSB as a leaktight building.*

   In order to minimize the impact of ex-containment refueling on both the fuel handling design and licensing requirement, consideration was given to making the RSB a leaktight structure. As a result of analysis performed by B&R indicating that it was impractical to design the RSB as a leaktight building, this alternative was rejected.

3. *Place leaktight requirements on both the RCB and RSB cells and designing the RSB as a confinement building.*

   This alternative was rejected at this time since it was felt (RRD) that the requirement for leaktight cells would compromise plant availability in that periodic tests of cell leakage would be required.

8.4 Conclusion

The current system was selected since it represents a defensible licensing portion and is the most commercially attractive system. However, it should be pointed out that the selected system will probably generate many detailed licensing questions that an in-containment system would not.

9 USE OF WATER IN CONTAINMENT FOR CLEANING OF COMPONENTS

9.1 Description

The Sodium Removal and Decontamination System provides for the cleaning and rinsing of large sodium wetted components. Sodium is removed in a controlled manner by passing moist N_2 (<2% H_2O vapor) through the large component cleaning vessel (LCCV) containing the component to be cleaned. Rinsing is accomplished by filling the vessel with water (40,000 gal. maximum) to a level that completely immerses the component and draining the vessel.

Use of large quantities (40,000 gallons) of water in containment is clearly a subject of interest, since this has not previously been done in any LMFBR design.

9.2 Synopsis of Treatment

The treatment of this subject in the PSAR is brief, and no detailed safety evaluation is presented. The moist gas process is identified, and the system is given descriptive treatment only.

9.3 Consideration of Alternatives

1. *Use of ethanol rather than moist gas as the cleaning agent.*

   This alternative is still under active consideration by the Project. There are concerns with the introduction of a volatile and flammable material into containment, which need to be examined. It is possible that when this evaluation is complete (about mid 1975) it will be determined that ethanol should not be used within containment.

2. *Locate the LCCV outside containment.*

   From the Licensing standpoint this is attractive, since considerations of protection against containment pressurization no longer arise. However, for reasons of economics it is highly desirable to demonstrate the licensability of an in-containment cleaning cell, to obviate the need for transport of heavy components outside containment.

3. *Delay PSAR submittal until the safety evaluation is complete.*

   This is not felt to be justified. Some design modifications will likely be required as a result of the safety evaluation, but these are not expected to result in a major change to the concept.

9.4 Conclusions

There is much further work still to be done in order to demonstrate the acceptability of an in-containment, moist gas cleaning process. AI has been requested to produce a complete evaluation, covering both moist gas and ethanol, by mid 1975, and final decisions cannot be made until after that time.

At present there is no reason to change the Reference Design, and if that design should change subsequently, a PSAR supplement could be prepared during this calendar year.

For these reasons, the present treatment is regarded as acceptable.

10 EFFECTS OF ACCIDENTS ON SAFETY RELATED EQUIPMENT

10.1 Description of Item

The item discussed is the effects of events, such as sodium spills or steam/feedwater line breaks on safety related equipment required to operate in the event of such an accident. The effects on surrounding structures are also included. This topic merits discussion because it is important to the licensing process and because more information on this will be necessary as the design evolves.

10.2 Synopsis of Treatment

The design of structures etc. is covered in Chapters 3 and 6. Included in these are loading requirements, a description of protective features such as cell liners, sodium trays in the steam generator building, and design basis events. Ventilation provisions are covered in Chapter 9. Chapter 15 treats the consequences of accidents. The methods of protection are sometimes treated in the form of requirements, with little in the way of specific design detail. Examples are:

a. Venting of the steam generator building will be provided, as necessary, to protect against pressure buildup as a consequence of IHTS line failures. The precise method of venting the building has not been established at this time, because the analysis of IHTS pipe ruptures has only recently been completed. Further work is needed to define precisely what degree of venting is needed, and how it should be provided; however, the feasibility of venting is not in question.

b. The operability of primary pump pony motors under sodium spill conditions has not yet been fully demonstrated and is not discussed in detail in the PSAR. This will be given in a future PSAR supplement. This information would have been included if available, but the design has not progressed to this stage. It is not expected that this will result in a problem, because the conditions under which pony motor operation is required for safe shutdown do not encompass conditions in which a sodium spill could have an adverse effect on their environment. The pony motors are situated above the operating floor; they are needed for safe shutdown following full power operation, at which time the primary HTS cells are inerted, and not open to the outer containment.

c. Evaluation of cell liner capability to accommodate sodium spills including the efficacy of liner venting provisions is not complete at this time and is not addressed in the PSAR. This will be given in a future PSAR supplement.

This information would have been included if available, but the evaluations are not yet completed. Based on FFTF experience, it is not expected that significant design modifications will be required in this area.

10.3 Consideration of Alternatives

The only alternative to the present course of action would be to delay submittal of the PSAR until all this information is available. This is not considered to be justifiable, since the feasibility of meeting requirements is not in question. Thus the requirements of 10CFR50 for PSAR production have been met.

11 DECAY HEAT REMOVAL PROVISIONS

11.1 Description of Item

The adequacy, redundancy and overall reliability of decay heat removal systems are primary concerns in the Licensing process. The PSAR shows that the CRBRP decay heat removal scheme is adequate and meets the Project's overall reliability criterion.

11.2 Synopsis of Treatment

The PSAR presents the CRBRP decay heat removal scheme in five sections:

1. Chapter 5, "Heat Transport and Connected Systems", provides a description of the Primary and Intermediate Heat Transport Systems (PHTS and IHTS), the Steam Generation System (SGS), the Steam Generator Auxiliary Heat Removal System (SGAHRS) and the Overflow Heat Removal Service (OHRS).

2. Chapter 7, "Instrumentation and Controls", provides a description of the safety related systems which control and monitor operation of SGAHRS and OHRS.

3. Chapter 9, "Auxiliary Systems", provides a description of the OHRS components as components of the Auxiliary Liquid Metal System, but refers to Chapter 5 for discussion of the OHR service.

4. Chapter 10, "Steam and Power Conversion Systems", provides a discussion of the non-safety related turbine bypass main condenser, Condensate and Feedwater Systems.

5. Chapter 15, "Accident Analysis", shows that for all postulated events, the CRBRP design will provide adequate cooling to the core.

6. Appendix C, "Reliability Program", provides a discussion of and gives the preliminary results of the program to quantitatively evaluate the reliability of the integrated decay heat removal scheme.

A synopsis of the relevant points in each section follows.

Chapter 5

The functioning of the PHTS and IHTS with pony motor flow to remove decay heat and sensible heat after all plant events is a major performance objective. This includes the qualification of the primary and intermediate coolant pumps to operate at pony motor speed after a safe shutdown earthquake. There is also a performance requirement that the PHTS and IHTS provide adequate cooling by natural circulation on three or two loops following rated power operation and with two or one loops following operation on two loops. With pony motor flow, two operating loops will provide adequate cooling even in the event that the third loop has a pump seizure compounded with a check valve failure to close.

The performance objective of the SGS is to remove adequate decay and sensible heat from the IHTS by natural circulation under all postulated PHTS and IHTS operating modes. This includes performance during use of main condenser cooling, venting of steam through relief valves and PACC cooling.

The SGAHRS objectives are to 1) provide auxiliary feedwater supply in case of failure of the (non-safety related) Condensate and Feedwater Systems and 2) provide cooling by venting steam and/or condensation in the protected air cooled condensers (PACCs) in case of unavailability main condenser cooling. The SGAHRS protected water storage tank and PACCs are sized to provide adequate short and long term decay heat removal capacity using one SGS and SGAHRS loop with natural circulation on the steam/water side and forced circulation on the air side of the PACC for any postulated operating mode of the PHTS and IHTS.

The OHRS is to provide a backup to the SGS for decay heat removal and substantially improves the reliability of the decay heat removal scheme. The performance of OHRS is divided into two categories. If the OHRS assumes the decay heat load 24 or more hours after reactor shutdown, the event is classified as an emergency plant event. If the load is assumed between one and 24 hours after reactor shutdown, the event is a faulted plant event. In either case, at least one primary pump must be operating at pony motor flow to provide circulation through the core. The rated capacity of OHRS is based on operation of all pumps blowers and heat exchangers in the OHRS train; i.e., there is no redundancy for OHRS operation at rated capacity.

Chapter 7

All SGAHRS Auxiliary Feedwater Pumps and Protected Air Cooled Condenser blowers are started and appropriate valves operated in sequence, automatically upon indication of low steam drum level or high steam-to-feedwater flow ratio. The activating circuit is comprised of 2/3 logic for each initiating signal (low steam drum level or high steam-to-feedwater flow ratio), 1/2 logic combining the outputs of the 2/3 logic, and 1/3 logic carrying the signal to the actuated device. The operator cannot manually override the actuating signals.

The Outlet Steam Isolation Subsystem provides isolation of steam system pipe breaks to insure loop-to-loop independence.

Initiation of OHRS is manual. The earliest OHRS can carry the decay heat load is one hour after reactor shutdown which provides ample time for operator action.

Chapter 9

The components which provide OHRS (except for the OHRS Heat Exchanger) normally provide reactor coolant makeup or cooling for the Ex-Vessel Storage Tank (EVST). The components are sized to provide removal of the required reactor decay heat in addition to heat generated by spent fuel in the EVST.

Chapter 10

The (non-safety related) turbine bypass system and main condenser are sized to provide condensation of 80% of the full rated power of the plant. The Plant Protection System automatically turns on SGAHRS if the bypass, condenser or feedwater systems become unavailable, see above under Chapter 7.

Chapter 15

The accident analyses show for all plant events considered, that the PHTS, IHTS, SGS and SGAHRS provide adequate redundancy to prevent loss of core cooling. (Only three loop plant operation is considered.)

In none of the accidents is there dependence on natural circulation since no analysis has yet been completed to show that natural circulation will provide adequate cooling for recently modified decay heat generation rates. The calculations for the PSAR are based on the "old" decay heat generation rates, but it is conservatively assumed (though not stated) that there will be no dependence on natural circulation. (However, the 'natural circulation events' listed in the Duty Cycle, are included in Appendix B of the PSAR).

Appendix C

A program is underway to quantitatively show that the combined decay heat removal systems have an unreliability of less than 8×10^{-7} failures per year. (See Part II, Section 3 of this Summary). The program uses methods similar to WASH-1400, "the Rasmussen Report"; assignment of failure rates to each component, generation of failure mode and effects analysis, consideration of common mode failures and statistical calculation of the overall system unreliability.

The program to date shows a decay heat removal unreliability of approximately 7×10^{-7} per year. Further refinement of the system model is expected to lower the calculated unreliability.

11.3 Consideration of Alternatives

1. *Use updated decay heat generation rates.*

   The updated decay heat generation rates (which show a greater production of decay heat) were not available until well into the PSAR review and were not substantiated until recently. It is intended that when the analyses for the plant with the new curves are completed, the results will be transmitted to REG as an amendment to the PSAR.

2. *Provide analyses of natural circulation performance.*

The adequacy of natural circulation for decay heat removal may be impacted by the new decay heat generation rates. Natural circulation decay heat removal is not required for safety reasons as long as the primary and intermediate coolant pumps will operate following all plant events. Rather than providing unnecessary analysis whose conclusions might later be challenged, no analysis is provided, since the safety of the plant does not depend on natural circulation.

11.4 Conclusions

The analyses now provided in the PSAR contain adequate information which, when supplemented to reflect the new decay heat generation rates, will demonstrate to the NRC that abundant and redundant decay heat removal capability is provided by the design.

12 CORE INSTRUMENTATION

12.1 Description of Item

The in-core instrumentation includes temperature, sodium level and vibration sensors for instrumenting the reactor parameters required for the Plant Protection System, the Plant Control System, and reactor temperature surveillance and design verification. The in-vessel instruments provided in the design consist of:

a. Sodium level sensors - used to provide the level signals to the Reactor Shutdown System (3 active sensors plus one spare) and a minimum safe level sensor which supplies a signal to the Primary Control System. The sensors are drywell induction probes sensitive over their entire length.

b. Temperature sensors - used to provide signals for use by the Plant Control System, and for surveillance and design verification purposes. A total of 392 instruments are provided: 198 at the exits of each fuel assembly, 150 at the exits of each radial blanket assembly, 3 on the periphery of the core, 26 in the sodium in the core outlet plenum and 15 in the upper internals. Twenty-one of the 198 fuel assembly sensors provide signals to the Plant Control System, the remaining 371 sensors are only used for surveillance and design verification. The signals from the 21 sensors are also available for surveillance and design verification purposes. The sensors are 1/8 inch chromel alumel ungrounded, stainless steel sheathed thermocouples.

c. Vibration sensors - used to provide design verification of the upper internals by measuring the flow induced vibration of the structure. Two accelerometers mounted on the upper internals structure furnish these measurements.

Of these instruments, only the 348 thermocouples above the fuel and radial blanket assembly outlets are discussed here. This number of instruments is provided to comply with an RRD/ARD agreement to fully instrument the core. The purpose for this expanded surveillance instrumentation coverage is to enhance the ability to detect flow blockages or enrichment or loading errors in the fuel and radial blanket assemblies.

12.2 Synopsis of Treatment

The core instrumentation is discussed in three places in the PSAR. The requirements on the instruments are given in Sections 4.4.5 and 7.5.3. Their description is presented in Section 7.5.3 and the discussion of their performance is given in Section 15.4.1.3. The treatments are necessarily brief and rather general in nature because the final instrument locations and their capabilities to perform in these locations and environment are not well defined.

The information presented indicates that the instruments (thermocouples) will be able to detect a 10 to 15°F increase in the mixed mean outlet temperature which corresponds to an approximate 5% reduction in the flow to a fuel assembly rod bundle. The maximum clad temperature is estimated to be approximately 3150°F which is more than 250°F below the 1600°F limit quoted in Section 15.1.2 (See under Part I of this Summary). Therefore, to produce a condition of incipient clad failure would require a blockage in excess of 80% of the assembly flow area which should be detectable with the existing full coverage instrumentation. It is intended that further studies will be conducted in this area.

12.3 Considerations of Alternatives

1. *The only viable alternative to the present course of action would be to delay the PSAR pending completion of the further studies mentioned above.*

   The studies will require development work and design analysis, which will not be completed for several months. This is a delay in PSAR submittal which is not considered justifiable. The position now presented is acceptable.

12.4 Conclusion

Based on the analysis which indicates that slow local blockages should be detectable before fuel clad damage occurs with the existing instrumentation it seems that the existing presentation provides adequate and appropriate coverage for submittal in the PSAR.

13 SURVEILLANCE AND IN-SERVICE INSPECTION (SISI)

13.1 Description of Item

This item is concerned with provisions for surveillance and in-service inspection of the primary coolant boundary. It is included here because LMFBR requirements for SISI have not yet been fully developed, so that CRBRP will be setting precedents in this area.

13.2 Synopsis of Treatment

The commitment to a SISI program in the PSAR is a general one for most sodium containing components and piping. The commitment is to a largely visual inspection program, plus provisions for surveillance samples to be placed in the reactor vessel (locations are provided both in the Upper Internals and the Lower Internals). The visual inspection programs will be similar, in terms of overall schedule requirements, with Section XI of the ASME Code (specified for LWRs) where practical.

Any metallurgical inspection will be performed when components are removed for maintenance.

Appendix H of 10CFR50 is shown to be inapplicable since it specifies a surveillance program for ferritic reactor vessel materials; the CRBR vessel is austenitic.

That visual inspection is adequate, is supported by the low crack growth rates demonstrated in the PSAR presentation on primary coolant boundary integrity (See Page 17 of this Summary).

13.3 Consideration of Alternatives

1. *Commit to complete compliance with ASME Code Section XI.*

   This cannot be done because the inspection required by the section includes use of ultrasonic or radiographic techniques which cannot be used for the CRBRP; ultrasonic testing does not provide accurate results for the large grain structure of Type 304 and 316 SS welds and radiographic films would become fogged if used near the highly activated materials of the primary system.

2. *Adopt Regulatory Guide 1.87, "Construction Criteria for Class 1 Components in Elevated Temperature Reactors", as interim guidance.*

This regulatory guide is one which is not supported by the ACRS and for which comments were made by the industry but not incorporated by REG. The guide calls for accommodation of "any required inservice inspection and surveillance programs to monitor and alert for material or component degradation such as creep rupture, creep deformation...Representative environmental factors of concern...are the effects of the cooling fluid...and/or impurities; irradiation effects...; and aging from prolonged exposure to elevated temperature."

That requirement is vague, yet suggests that surveillance programs are required in areas such as the CRBRP Primary Heat Transport System where Project engineers have determined that data available from materials development programs show such surveillance is unnecessary.

Adoption of Regulatory Guide 1.87 would be an ill-defined commitment which the Project does not support, and which the ACRS refused to endorse.

3. *Await the results of ASME Code subcommittee on considering a standard for LMFBR inservice inspection.*

The subcommittee is preparing a draft standard which is not likely to be published, even for tentative use, until after the CRBRP construction permit is issued. The subcommittee in considering the unique characteristics of LMFBRs, has rejected the approach of merely amending the LWR standards and may agree to the visual inspection approach proposed by the Project.

The subcommittee schedule for drafting the standard is inconsistent with the PSAR submittal date, and the subcommittee's work is not advanced far enough to make commitment to their future standard a meaningful position.

13.4 Conclusion

The approach used in the PSAR reflects concern for providing adequate SISI and is consistent with what is hoped for from the ASME Code subcommittee in the future. No standards which provide appropriate guidance for SISI of LMFBR's exist for reference.

APPENDICES

APPENDIX A AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

A.1 Introduction

Section 50.34 of the USAEC's regulations 10CFR Part 50 specifies in general terms the required content of Safety Analysis Reports (SARs) submitted in applications for construction permits (CP) and operating licenses (OL) for nuclear power plants. In February 1974, the USAEC issued the LMFBR Edition of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (SFAC). This edition of SFAC identifies the principal detailed information required by the Regulatory in its evaluation of application for CPs and OLs for LMFBR nuclear power plants.

While conformance with the SFAC is not required, it represents a format and content acceptable to the Regulatory. The CRBRP PSAR has been prepared by using this edition of SFAC as a guide. This audit was undertaken to make a complete and thorough check of the PSAR content as provided versus the SFAC requirements.

A.2 Description of Audit

Each PSAR chapter was covered in this audit, except Chapters 1, 2 and 17. Chapter 2 has already been submitted and is under review by Regulatory. Chapters 1 and 17 have complied with all the information requirements of the SFAC. For each chapter audited, cognizance has been taken of LWR practice in the preparation of that same chapter for their PSAR submittal. Consideration has also been taken of the lack of applicability of a number of existing guides, codes and standards identified in the SFAC and of the fact that the SFAC recognizes the needs for, and allows, certain non-provisions at the time of submission, providing the bases and criteria for their development and completion are included.

The audit has also taken cognizance of the differences between the SFAC and the Errata. The Errata was based on a meeting with Regulatory in March 1974. Thus the deletion of items per the Errata is acceptable to Regulatory.

A.3 Findings and Summary

The findings of this audit are itemized and tabulated on the Audit Sheets for each Chapter audited. These are attached as Enclosure I. Chapter 16, "Technical Specifications", is of a preliminary version, as agreed with Regulatory. Any detailed audit findings for this chapter at this time can hardly reflect the desired uniformity at which this audit has aimed to attain. Therefore, no audit sheets are included for this chapter.

To present an overview of the findings of this audit, a summary table is also prepared. This is included as Table 1, "Summary of Non-Provision versus Reason Category".

In conclusion, the sufficiency of provision in this PSAR is considered comparable, in general, to most LWR PSARs and as a matter of fact, greater than the latter in several instances. Considering the thousands of items of information required by the SFAC, the non-provisions identified appear to be minimal. Most importantly, the principal purpose of the PSAR is to provide sufficient information needed to understand the basis upon which the conclusion that the plant can be built and operated safely has been reached. In that context, any significance the non-provisions identified may have appears to be minor.

TABLE 1 - SUMMARY OF NON-PROVISION VERSUS REASON CATEGORY

Information Category	Total	REASON CATEGORY					
		By Errata (1)	Design Or Purchase Progress	Not Applicable (N.A.)	N.A., But May Change	Customer Direction	Other (2) Reasons
Design Bases	2		1	1			
Design/Analysis Details	6		5		1		
Design Description and/or Evaluation	10	1	8		1		
P & IDs, Other Drawings	5		5				
Test Programs	4	1	2			1	
Surveillance and In-Service Inspection Programs	5		4	1			
Accident Analyses	9	1	2	6			
Other Information (3)	3		1		1		1
TOTAL	44	3	28	8	3	1	1

Notes:

(1) See Section 2, "Description of Audit".

(2) "Other Reasons" include non-existing of applicable codes/standards, material selections and first-of-a-kind plant design characteristics, etc.

(3) "Other Information" includes those items not of the general categories such as, identification of vendors, comparative evaluation against other nuclear plants, etc.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 3 : DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

DATED 1-20-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
3.6.4	Complete information on a Typical Dynamic Analysis	Part of the information already provided. Additional information is being developed and will be supplemented.
3.8.2.4	Details of the design and analysis procedures used in designing the containment.	Purchase Progress.
3.11.1	"Worst Case" environmental conditions for each location of safety related mechanical and electrical equipment.	Design Progress.
3.11.2	Qualification tests and analysis to be performed on safety related equipment.	Design Progress.
3.11.4	"Operational" environmental conditions for each location of safety related equipment.	Design Progress.

CHAPTER 4 : REACTOR

DATED 1-27-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
4.3.1	Describe considerations of ATWS effects in backup systems design.	Not applicable to the CRBR due to the degree of conservatism in the shutdown systems design.
4.4.3.8	Evaluate energy release and potential should physical burnout of fuel elements occur.	By Errata.
4.4.3.9	Evaluate energy release and resulting pressure pulse due to MFCI.	By Errata. (Also, covered in Section 15.4).
4.4.3.10	Discuss fuel rods behavior in the event of coolant flow blockage.	By Errata. (Also, covered in Section 15.4).

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 5 : HEAT TRANSPORT SYSTEM AND CONNECTED SYSTEMS

DATED 1-21-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
5.2.1.9	Identify analytical methods for	There are parts of the Primary
	stress evaluation for components and	System boundary designed
	branch connections involving mixing	specifically for mixing streams
	liquid metals of different	of sodium of different
	temperatures.	temperatures. Should such needs
		arise as design progresses, the
		relevant information will
		be supplemented.
		Information on mixing componnets
		in IHTS and SGS is not yet
		available and will be
		supplemented.
5.2.1.11	Details of Stress Analysis Plan.	Design Progress. Preliminary
		information provided.
5.2.1.18	Provide sketches of system showing	Design Progress.
	points of large changes in flexibility	
	and stress levels relating to Seismic	
	Category I design.	
5.2.4.4	Describe material surveillance program	No material surveillance
	and indicate conformance to 10CFR50	program as defined in 10CFR50
	Appendix H.	Appendix H is required due to
		material selections and CRBR design

CHAPTER 5 : HEAT TRANSPORT SYSTEM AND CONNECTED SYSTEMS

DATED 1-21-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
5.2.9	Provide information on In-Service	Certain ASME-XI inspection
	Inspection Program and discuss	techniques are not feasible
	compliance with ASME-XI.	to LMFBR's such as the
		CRBR. Code requirements are
		in the early development.
		CRBR program details will
		be supplemented.
5.4.5	Identify reactor vessel fabricator.	Purchase Progress.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 6 : ENGINEERED SAFETY FEATURES

DATED 1-20-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
6.2.3	Detailed discussion of post-	Preliminary safety analyses
	accident containment cleanup system.	indicate no such needs. This
		is, however, subject to
		continuing evaluation.
		Should such needs arise,
		relevent information
		will be supplemented.
6.2.1.4	Certain information on containment	Design and Purchase Progress.
	in-service test program, including:	
	• Selected test frequency	
	• Test methods	
	• Requirements for acceptability	
	• Structural design provisions for	
	integrated leakage rate tests	
	anytime in plant life.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 7: INSTRUMENTATION AND CONTROLS

DATED 1-21-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
7.1.1	Identify I&C systems design	There is no such previous
	similarities to and differences	nuclear power plant.
	from those of a nuclear power	
	plant of similar design recently	
	issued CP or OL.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 8 : ELECTRIC POWER

DATED 1-22-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
8.3.11 and	Certain detailed information	Design Progress.
8.3.1.2	required for Onsite AC and DC Power	
	Systems description, including:	
	basis for the power required for each	
	safety load, motor insulation,	
	starting torque, grounding requirements	
	etc.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 9 : AUXILIARY SYSTEMS

DATED 1-23-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
9.1.3.4	Inspection and testing requirements	Design Progress.
	for fuel storage cooling and	
	cleanup systems.	
9.4.2	Inspection and testing requirements	Design Progress.
	for the RSB HVAC system.	
9.4.3	Information on HVAC system for the	Design Progress.
	radwaste area.	
9.5.2.3	Inspection and testing requirements	By Errata.
	for the Communications Systems.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 10 : STEAM AND POWER CONVERSION SYSTEM

DATED 1-22-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
10.2.3	Detailed information on Turbine Missiles.	Design and Purchase Progress.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 11 : RADIOACTIVE WASTE MANAGEMENT

DATED 1-23-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
11.2.7	Identify liquid waste release points on a site plot plan.	Design Progress.
11.4.2	Provide setpoints and bases associated for effluent release monitoring systems.	Design Progress.
11.5.3	Description of all equipment in the solid radwaste system.	Conceptual description provided. Further information will be supplemented.
11.5.4	Expected curie content in the solid wastes at the time of shipment.	Design Progress.
11.5.6	Identify exact locations of solid radwastes storage facilities on a plot plan. State expected decay by such storage.	Design Progress.
11.6.6	Statistical Sensitivity of the Off-Site Radiological Monitoring Program	Design Progress.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 12 : RADIATION PROTECTION

DATED 1-23-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
12.2.6	Expected annual inhalation doses	Design Progress.
	to plant personnel and peak air	
	concentrations in each building.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 13 : CONDUCT OF OPERATIONS

DATED 1-28-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
13.7.3.7	Provide a list of tests and	By Errata.
	inspections to demonstrate compliance	
	for the plant industrial security	
	plan.	

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 14 : INITIAL TESTS AND OPERATIONS

DATED 1-27-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
14.1	Preoperational testing program.	Customer Direction.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 15: ACCIDENT ANALYSES

DATED 1-28-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
Item 7*	Loss of all AC power to station auxiliaries and loss of emergency diesels.	Consistent with LWR practice.
Item 9	Excessive heat removal due to feed-water system malfunctions.	Design Progress.
Item 33	Core flow maldistribution due to fuel loading error.	Not applicable due to CRBR design.
Item 36	Inadvertent closure of either reactor coolant valves or inter-mediate valves.	There are no primary or IHTS coolant loop isolation valves.
Item 40	Plugging of reactor overflow line.	Not credible, but subject to continuing evaluation.
Item 42	Accidental opening of valves to a drained isolated loop.	Not applicable; based upon preliminary CRBR design.
Item 45	Misloaded fuel assembly.	Not applicable due to CRBR design.

*Item numbers in Table 15-1 of SFAC.

AUDIT OF CRBRP PSAR VERSUS SFAC REQUIREMENTS

CHAPTER 15 : ACCIDENT ANALYSES

DATED 1-28-75

SIGNED

SFAC Section No.	Item Required By SFAC But Not Provided	Reason For Non-Provision
Item 50	Inadvertent closure of floor valve	Design Progress.
	on canister during fuel handling.	
Item 59	Inadvertent opening of fuel	Not applicable; based
	handling machine valve during	upon preliminary design.
	transfer.	
Item 64	Loss of all power to fuel handling	Not applicable, based
	machine.	upon preliminary design.

APPENDIX B - APPLICABILITY OF REGULATORY GUIDES

This Appendix comprises, verbatim, a report prepared by ARD Licensing on "Review of Existing AEC Regulatory Guides (Division 1, Power Reactor) for applicability to the Clinch River Breeder Reactor Plant".

B.1 Introduction, Scope and Purpose

This report describes a preliminary review of the existing AEC Regulatory Guides for Applicability to the Clinch River Breeder Reactor Plant (CRBRP). The review covers the Division 1 (Power Reactor) Regulatory Guides only. These include 83 Regulatory Guides, 1.1 through 1.83.

The AEC Regulatory Guides are intended to describe the Regulatory position as to how the requirements of a given AEC regulation have been satisfied. These requirements are set forth in Appendix A to 10CFR Part 50 for design of nuclear power plants and in various parts of Chapter I of 10CFR for construction, operation, and quality assurance, in addition to design. Some of the detailed requirements, however, address directly the light-water-cooled nuclear power plants. Consequently, a number of the existing Regulatory Guides may or may not apply to the CRBRP, mainly due to the differences in designs between the LMFBR plants and the LWR plants.

In order to assure that the design of the CRBRP will appropriately meet the requirements of the AEC regulations and to make maximum use of the Regulatory Guides, this preliminary review is undertaken:

(1) to assess the applicability, if any, of the existing Regulatory Guides to the CRBRP; and

(2) to identify the needs for changes such that an existing Guide will properly cover the CRBRP or for issuance of new Guides that directly apply to the CRBRP.

B.2 Evaluation of Applicability and Identification of Changes Needed

A percentage-rating scale is used to evaluate the applicability of the Regulatory Guides 1.1 through 1.83. The assessment is made both in the content of "Intent" and of "Detailed Provisions" of the Regulatory Guides. The definitions of the percentage rating used are as follows:

0%	=	Not Applicable
25%	=	Major Portion Not Applicable
50%	=	Partially Applicable
75%	=	Major Portion Applicable
95%	=	Essentially Fully Applicable
100%	=	Fully or Directly Applicable

It is important to note that both the applicability evaluation and the needed-changes identification are made based upon the selected design of the CRBRP at the time of this review. However, wherever practical, and/or the emphasis on the CRBRP is not compromised, the assessment is then made in the context of an LMFBR plant in general.

B.3 Results of the Review

The evaluated applicability and the identified changes required as concluded from the review are presented in Table I.

TABLE I

EVALUATION OF APPLICABILITIES OF EXISTING AEC REGULATORY GUIDES TO THE CLINCH RIVER BREEDER REACTOR PLANT

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR BEING NOT APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (formerly Safety Guide 1)	0.0	0.0	(No equivalent system pumps in the CRBRP)
1.2	Thermal Shock to Reactor Pressure Vessels (formerly Safety Guide 2)	0.0	0.0	(No comparable emergency core cooling system, nor any large quantities of cold coolant injection involved on the CRBR)
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Revision 1, 6/73, of Safety Guide 3)	0.0	0.0	A separate new guide for LMFBRs needs to be developed. Major changes required include: 1. Emphasis on loss of coolant accident is not applicable to the CRBRP. 2. Acceptable assumptions related to the accident release, taking into consideration the LMFBR characteristics as appropriate, and 3. Addition of provisions to allow credit for reduction in the amount of release available for leakage(s) due to plate-out and settling.
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Revision 1, 6/73, of former Safety Guide 4)	0.0	0.0	Same as 1.3 above
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (formerly Safety Guide 5)	0.0	0.0	(No comparable radiological consequences involved for a steam line break in the CRBRP)
1.6	Independence Between Redundant Standby (Onsite) Power Sources & Between Their Distribution Systems (formerly Safety Guide 6)	100%	100%	Consistent with the (Proposed) CRBRP, GDC 17.
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident (formerly Safety Guide 7)	0.0	0.0	There is no zirconium-water reaction, nor containment spray reaction with metals in the CRBRP. Also, emphasis on loss of coolant accident is not applicable to the CRBRP. However, need for monitoring of combustible gases is to be assessed.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.8	Personnel Selection and Training (formerly Safety Guide 8)	100%	100%	ANSI N18.1 equally applies to the CRBRP
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies (formerly Safety Guide 9)	100%	100%	Intent consistent with the Proposed GDC 17 The detailed provisions are equally applicable to the CRBRP
1.10	Mechanical (Caldweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1/2/73, of former Safety Guide 10)	100%	100%	This Guide is directly applicable. The procedures set forth in this Guide for testing & sampling of mechanical splices in reinforcing bars are considered equally applicable to the Category I concrete structures of any nuclear power plant.
1.11	Instrument Lines Penetrating Primary Reactor Containment (formerly Safety Guide 11)	100%	100%	The intent of this Guide is consistent with GDC 55 and GDC 56 of the Proposed CRBRP GDC. The provisions stated in this Guide for demonstrating the acceptability of instrument line containment penetrations are considered equally applicable to the CRBRP.
1.12	Instrumentation for Earthquakes (formerly Safety Guide 12)	100%	100%	The intent of this Guide is consistent with 10 CFR 50.36(c), which applies equally to any nuclear power plant. The provisions set forth in this Guide relating to a suitable program for the seismic instrumentation required are considered equally applicable to the CRBRP as appropriate.
1.13	Fuel Storage Facility Design Basis (formerly Safety Guide 13)	100%	50%	The intent of this Guide is consistent with GDC 61 of the Proposed CRBRP GDC. The detailed provisions of this Guide would be 90% applicable to an LMFBR plant using ex-containment water pool spent fuel storage. The only modification required would be related to Provision C.4 in that the inventory of radioactive materials available from leakage should be based on assumptions consistent with the characteristics of an LMFBR, rather than Regulatory Guide 1.25 (also see evaluation of Regulatory Guide 1.25 below). The CRBRP is presently using an ex-containment sodium-cooled EVST design. Consequently the detailed provisions of this Guide is estimated to be about 50% applicable. To make the Guide fully applicable to the CRBRP, appropriate changes are required to supplement and/or modify Provisions C.3, C.4 and C.8.
1.14	Reactor Coolant Pump Flywheel Integrity (formerly Safety Guide 14)	0.0	0.0	(This Guide is related to flywheels of reactor coolant pump motors in LWRs and is not applicable to the CRBRP.)
1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1) 12/28/72, of former Safety Guide 15)	100%	100%	This Guide is wholly applicable to the CRBRP.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY: INTENT	% RATING OF APPLICABILITY: DETAILED PROVISIONS	REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
1.16	Reporting of Operating Information (Revision 1, 10/73, of former Safety Guide 16)	100%	50%	This Guide is partially applicable to the CRBRP. The changes required include the following: 1. The parameter list in Provision C.1.a.(3).(f) needs minor modification. 2. In Table 1, the report items related to "Fracture Toughness" and "Reactor Vessel Material Surveillance" need modification for full applicability to the CRBRP. This is due to the reason that both Appendices G and H to 10 CFR 50 may be not applicable or only partially applicable. This in turn depends on the materials selection for the vessel system which is not yet firm in certain areas.
1.17	Protection of Nuclear Plants Against Industrial Sabotage, (Revision 1, 6/73, of former Safety Guide 17)	100%	100%	This Guide is considered fully applicable to the CRBRP.
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12/28/72 of former Safety Guide 18)	0.0	0.0	The containment design selection is steel so that this is not applicable to the CRBRP.
1.19	Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8/11/72, of former Safety Guide 19)	0.0	0.0	(Same as 1.18 above.)
1.20	Vibration Measurements on Reactor Internals (formerly Safety Guide 20)	100%	50%	The intent of this Guide is applicable, however the testing details given are not appropriate to LMFBR's.
1.21	Measuring & Reporting of Effluents from Nuclear Power Plants (formerly Safety Guide 21)	100%	75%	The intent of this Guide is equally applicable to the CRBRP. The provisions in this Guide are only applicable to the CRBRP, where appropriate.
1.22	Periodic Testing of Protection System Actuation Function (formerly Safety Guide 22)	100%	100%	The intent of this Guide is consistent with the Proposed CRBRP GDC.
1.23	Onsite Meteorological Programs (formerly Safety Guide 23)	100%	95%	The intent and provisions of this Guide are considered generally applicable. Although in the "Discussion" section of this Guide references are made to Safety Guides 3 and 4 which were prepared for LWRs, the detailed provisions as set forth in the "Regulatory Position" section of the Guide have no requirements strictly and exclusively based upon these two LWR guides. (Also see Regulatory Position C.6.d of this Guide.)
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure (formerly Safety Guide 24)	100%	0.0	This Guide was specifically prepared for PWR plants, although the basic intent is considered generally applicable. The detailed provisions are considered not applicable to the CRBRP.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling & Storage Facility for Boiling & Pressurized Water Reactors (formerly Safety Guide 25)	50%	0.0	For applicability to LMFBRs, major changes in Provisions C.1 and C.3 of this Guide are needed. Due to basic differences in fuel handling and storage designs between the CRBRP and the LWRs, this guide is not rated as the detailed provisions of the Guide are largely not applicable.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.26	Quality Control Classifications & Standards (formerly Safety Guide 26)	100%	25%	The intent of this Guide is equally applicable to the LMFBR plants. The detailed provisions of this Guide are basically not applicable to the CRBRP. This will be addressed in the PSAR per Section 3.2.2 of the SFAC.
1.27	Ultimate Heat Sink (formerly Safety Guide 27)	100%	100%	The intent of this Guide is considered generally applicable. Due to design differences, however, the detailed provisions of this Guide are applicable only where appropriate.
1.28	Quality Assurance Program Requirements (Design & Construction) (formerly Safety Guide 28)	100%	100%	This Guide, which is mainly to concur on the requirements as set forth in ANSI N45.2.11 (Draft No. 3, Rev., 1 July, 1973), is equally applicable to the CRBRP.
1.29	Seismic Design Classification (Revision 1, 8/73, of former Safety Guide 29)	100%	50%	The basic intent of this Guide is equally applicable to the CRBRP. In their present version, the detailed provisions described in this Guide are not directly applicable to the CRBRP. This will be addressed in the PSAR per Section 3.2.1 of the SFAC.
1.30	Quality Assurance Requirements for the Installation, Inspection, & Testing of Instrumentation & Electric Equipment (formerly Safety Guide 30)	100%	100%	Both the intent and provisions (basically IEEE Std-336) are directly applicable to the CRBRP.
1.31	Control of Stainless Steel Welding (Revision 1, 6/73, of former Safety Guide 31)	100%	100%	Although this Guide was prepared for application to LWRs, it is equally applicable to the CRBRP.
1.32	Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generation Stations" (formerly Safety Guide 32)	100%	100%	The intent and provisions of this Guide are equally applicable to the CRBRP, as appropriate.
1.33	Quality Assurance Program Requirements (Operation) (formerly Safety Guide 33)	100%	50%	The intent of this Guide is equally applicable to the CRBRP. The detailed provisions, in their present form, do not adequately cover the LMFBR plants. For direct applicability to the CRBRP, the following changes are required: 1. A new section needs to be added to Appendix A of this Guide. This new section should address the "Procedures for Startup, Operation, & Shutdown of Safety Related LMFBR Systems". 2. Also in Appendix A, the present Sections F,G,H, and I need to be revised in order to cover these conditions and activities characteristic of LMFBR plants.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.34	Control of Electroslag Weld Properties (12/28/72)	100%	100%	This Guide, describing an acceptable method for assuring materials control & control of special process related to fabricating electroslag welds for nuclear components, is equally applicable to the CRBRP. Actual use of this Guide, however, is expected to be very limited, if any. One possible use is for the core support. It is anticipated that "Up-John" or "Subvert" will be the special process to be used on the CRBRP.
1.35	Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures (2/5/73)	0.0	0.0	(This Guide, relating to Prestressed Concrete Containment, is not applicable to the CRBRP.)
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (2/23/73)	100%	50%	This Guide addresses the selection and use of nonmetallic Thermal insulation to minimize promotion of stress-corrosion cracking in the stainless steel portions of the reactor coolant boundary and other systems important to safety. Parts of the detailed provisions of the Guide are applicable where appropriate to the CRBRP.
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73)	0.0	0.0	In the context of "on-site cleaning" as intended by this Guide, the provisions set forth in ANSI N45.2.1-1973 which forms the basis of this Guide are not expected to be applicable to most of the liquid-metal systems of this plant. At this point in time, it is anticipated that these fluid systems components will be cleaned, prior to installation, in the fabricator's shop. This shop cleaning may be water cleaning, and the requirements and control will be comparable to ANSI N45.2.1-1973. On site pre-operation cleaning, to which this Guide refers, if any, will be minimal and will be done by hand. Because of the above reasons, this Guide is not rated.
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, & Handling of Items for Water-Cooled Nuclear Power Plants (3/16/73)	100%	100%	The intent of this Guide is consistent with Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants". The provisions are mainly based upon ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Phase". This Guide is considered equally applicable to the CRBRP.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3/16/73)	100%	100%	The intent of this Guide is consistent with Appendix B to 10 CFR 50. The provisions are mainly based upon ANSI N45.2.3-1973. This Guide is considered directly applicable to the CRBRP.
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (3/16/73)	100%	25%	This Guide is intended mainly to concur on the requirements set forth in IEEE Std-334-1971, subject to additional provisions. The basic intent of the guide is generally applicable. However, changes and supplements to IEEE Std-334-1971 appropriate to LMFBRs are needed in order to be applicable to the CRBRP.
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments (3/16/73)	100%	100%	This Guide describes an acceptable method of verifying the proper assignments of redundant load groups to the related on-site power sources. It is considered equally applicable to the CRBRP.
1.42	Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors (6/73)	50%	0.0	The detailed provisions, developed primarily for LWR plants, do not apply to the CRBRP.
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (5/73)	100%	100%	This Guide is related to selection and control of welding processes used for cladding ferritic steel components with austenitic stainless steel. It is equally applicable to the CRBRP, as appropriate.
1.44	Control of the Use of Sensitized Stainless Steel (5/73)	0.0	0.0	The intent of this Guide relates to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion. It was developed primarily for LWRs. For the S.S. materials to be used for the primary system components in the CRBRP, sensitization will occur. On the other hand, the high operating temperatures limit the use of materials of low carbon content. The solution is therefore mainly to rely upon control for cleanliness and protection against contaminants.

No.	TITLE	% RATING OF APPLICABILITY: INTENT	% RATING OF APPLICABILITY: DETAILED PROVISIONS	REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
1.45	Reactor Coolant Pressure Boundary Leakage Detection System (5/73)	50%	0.0	The basic intent of this Guide is considered generally applicable, but the Guide was prepared to address the LWR coolant systems. The detailed provisions of this Guide are largely not applicable to an LMFBR plant.
1.46	Protection Against Pipe Whip Inside Containment (5/73)	100%	0.0	The basic intent of this Guide is considered generally applicable. The detailed provisions of this Guide, however, was developed primarily for LWR plants.
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (5/73)	100%	100%	This Guide is considered equally applicable to CRBRP.
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components (5/73)	100%	50%	The basic intent of delineating acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents and specified seismic events for the design of Seismic Category I fluid system components is considered generally applicable to all nuclear power plants. The detailed provisions of this Guide were developed primarily for LWR plants. They need to be supplemented and/or modified for direct application to the CRBRP.
1.49	Power Levels of Nuclear Power Plants (Revision 1, 12/73)	100%	100%	This Guide is generally applicable. (It should be noted that, due to the projected power levels of this plant, this Guide has no impact on the CRBRP.)
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (5/73)	100%	100%	This Guide describes an acceptable method with regard to the control of welding for low-alloy steel components during initial fabrication. It is considered applicable to CRBRP, as appropriate.
1.51	Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components (5/73)	100%	0.0	The intent of this Guide is equally applicable to CRBRP. The detailed provisions in this Guide may not be directly applicable. Where feasible with regard to the state-of-the-art of the specified examination method, the intent of requirements set forth in the ASME-XI as well as this Guide will be met. However, certain significant differences exist between LWRs and the CRBRP (e.g., low pressure system) and some specified examination methods (e.g., volumetric) have been found not feasible due to certain component material (e.g., UT on stainless steel) and/or the special environment (e.g., high radiation level, high-temperature sodium coolant, etc.) characteristic of the CRBRP. In these cases, alternative requirements wherever practicable & justifiable will be considered & proposed.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.52	Design, Testing, & Maintenance Criteria for Atmosphere Clean-up System Air Filtration and absorption Units of Light-Water-Cooled Nuclear Power Plants (6/73)	100%	100%	
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (6/73)	100%	100%	This Guide is considered applicable to CRBRP.
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (6/73)	100%	100%	The intent of this Guide is considered applicable to the CRBRP. It is applicable as appropriate.
1.55	Concrete Placement in Category I Structures (6/73)	100%	100%	This Guide is considered equally applicable to any nuclear power plant.
1.56	Maintenance of Water Purity in Boiling Water Reactors (6/73)	0.0	0.0	(This Guide was developed for BWRs and is not applicable to the CRBRP.)
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (6/73)	0.0	0.0	This Guide was specifically prepared for and limited to those LWR plants of which the containment system comprises a metal containment that is completely enclosed within a Seismic Category I structure (e.g.) a concrete shield building). It is, therefore, generally applicable to those plants which use this particular type of containment system. Due to containment selection, this Guide is not rated as it is not applicable.
1.58	Qualification of Nuclear Power Plant Inspection, Examination, & Testing Personnel (8/73)	100%	100%	This Guide is considered generally applicable to CRBRP, as appropriate.
1.59	Design Basis Floods for Nuclear Power Plants (8/73)	100%	100%	This Guide is equally applicable to CRBRP, as appropriate.
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Revision 1, 12/73)	100%	100%	This Guide is considered equally applicable to CRBRP, as appropriate.
1.61	Damping Values for Seismic Design of Nuclear Power Plants (10/73)	100%	100%	This Guide is equally applicable to CRBRP, as appropriate.
1.52	Manual Initiation of Protective Actions (10/73)	100%	100%	This Guide describes an acceptable method for complying with the requirements of IEEE Std 279-1971 (Section 4.17). It is considered equally applicable to the CRBRP.
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants (10/73)	ˆ00%	100%	This Guide concurs with IEEE Std 317-1972 and supplements it with four additional provisions. It is considered equally applicable to CRBRP as appropriate.
1.64	Quality Assurance Program Requirements for the Design of Nuclear Power Plants (10/73)	100%	100%	This Guide is mainly to concur on the QA requirements during the design phase as set forth in ANSI N45.2.11 (Draft No. 3, Rev. 1 - July 1973). It is considered equally applicable to the CRBRP.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE
		INTENT	DETAILED PROVISIONS	
1.65	Materials & Inspection for Reactor Vessel Closure Studs (10/73)	0.0	0.0	This Guide was prepared primarily for LWRs. Due to differences in loading characteristics, it is considered essentially not directly applicable to the CRBRP.
1.66	Nondestructive Examination of Tubular Products (10/73)	50%	50%	This Guide was developed and intended primarily for application to tubular products used for ASME-III Code Class 1 components on LWRs. The corresponding CRBRP components are expected to be of austenitic steel. The state-of-the-art of the UT examination, as specified by the Guide, has not been capable of producing meaningful results. The CRBRP, however, is anticipated to meet the requirements as set forth in NB-2550 of ASME-III for the examination addressed by the Guide.
1.67	Installation of Over-Pressure Protection Devices (10/73)	100%	50%	Code Case 1569, which forms the basis of this Guide, has covered four categories. Only the open systems, however, are treated in detail. Closed discharge systems are essentially left undefined. According to the selected design of the CRBRP at this time, the Guide is expected to be applicable only in the design of steam line safety valves. The Guide is therefore considered as partially applicable to the CRBRP in terms of the detailed provisions.
1.68	Preoperational & Initial Start-up Test Programs for Water-Cooled Power Reactors (11/73)	50%	25%	This Guide was developed primarily for LWR plants. In order to properly cover the LMFBR plants, the detailed provisions of this Guide need to be supplemented and modified by taking into consideration characteristics of LMFBR plants. Specifically, this includes modifications of and supplements to appropriate items included in Appendices A and C to this Guide.
1.69	Concrete Radiation Shields for Nuclear Power Plants (1/74)	100%	100%	This Guide is considered applicable to CRBRP.
1.70.1	Additional Information-Hydrological Considerations for Nuclear Power Plants (12/73)-To: Standard Format & Content of Safety Analysis Reports of Nuclear Power Plants (Revision 1, Regulatory Guide 1.70, 10/72)	100%	100%	The provisions of this Guide have already been incorporated in the "Standard Format & Content of Safety Analysis Reports for Nuclear Power Plants - LMFBR Edition", issued February 1974.
1.70.2	Additional Information-Air Filtration Systems & Containment Sumps for Nuclear Power Plants (11/73)	50%	25%	Provision B.1 set forth in this Guide is considered applicable, as appropriate. In particular, in order to make Provision B.1 applicable to LMFBRs, major and appropriate changes are required with regard to the Positions in Regulatory Guide 1.52 which is referenced. Provision B.2 is considered not applicable.
1.70.3	Additional Information - Radioactive Materials Safety for Nuclear Power Plants	100%	100%	This Guide is considered generally applicable to all nuclear power plants.
1.70.4	Additional Information - Fire Protection Considerations for Nuclear Power Plants	100%	100%	This Guide is considered generally applicable to all nuclear power plants.
1.71	Welder Qualification for Limited Accessibility Areas (1/74)	100%	100%	This Guide relates to control of welding for nuclear components and is considered generally applicable.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
1.72	Spray Pond Plastic Piping (1/74)	0.0	0.0	It is anticipated that there will be no spray pond in the CRBRP.
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (1/74)	100%	75%	This Guide is mainly based upon IEEE Std. 382-1972 and is considered equally applicable to any nuclear power plant, where appropriate. In order to be properly applicable to LMFBRs, modifications and supplements to IEEE Std. 382-1972 appropriate to LMFBRs are required.
1.74	Quality Assurance Terms and Definitions	100%	100%	This Guide is applicable to CRBRP.
1.75	Physical Independence of Electric Systems	---	---	This Guide is not rated since the LWR vendors are still discussing its implications with REG.
1.76	Design Basis Tornado for Nuclear Power Plants	100%	100%	This Guide describes design basis tornadoes, for nuclear power plants, acceptable to the Regulatory for three regions within the contiguous United States. It is generally applicable and is applicable to the CRBRP as appropriate.
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	0.0	0.0	This Guide was specifically prepared for PWR plants in regard to acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide fueled cores. It is not applicable to the CRBRP.
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	100%	50%	This Guide describes acceptable assumptions and criteria to be used in the evaluation of control room habitability during and after a postulated hazardous chemical release. Requirements of the Guide are dependent upon actual or projected presence of certain specified chemicals within five miles of the plant or in frequent transit within the same distance. Preliminary design of the CRBRP control room habitability system has been assessed for a hypothetical and most limiting radiological consequence. Chemical toxicity will be assessed.
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	0.0	0.0	This Guide was specifically prepared for PWR plants in regard to acceptable preoperational testing programs for ECCs. It is not applicable to the CRBRP.
1.80	Preoperational Testing of Instrument Air Systems	0.0	0.0	This Guide describes an acceptable preoperational testing program for verifying the opera bility of safety-related instrument air system. On the CRBRP, except those portions penetrating the containment and being considered as parts and appurtenance thereof, safety-related instrument air system parts are yet to be identified.

TABLE I (Cont'd)

No.	TITLE	% RATING OF APPLICABILITY		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE.
		INTENT	DETAILED PROVISIONS	
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	0.0	0.0	This Guide addresses the USAEC's requirements with regard to the sharing of onsite emergency and shutdown electric systems for multi-unit nuclear power plants. It is not applicable to the CRBRP.
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	0.0	0.0	This Guide applies to PWRs only. It is not applicable to the CRBRP.
1.83	Inservice Inspection of Presurized Water Reactor Steam Generator Tubes	0.0	0.0	This Guide applies only to PWRs. It is not applicable to the CRBRP.

APPENDIX C - INTRODUCTION TO PSAR

This appendix comprises material from Section 1.1 of the PSAR.

1.1 INTRODUCTION

The unique characteristics of the Clinch River Breeder Reactor Plant (CRBRP) are such as to require more introductory information than that specified in the LMFBR Edition of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. For this reason, this Introduction has been subdivided into three main sub-sections. The first (1.1.1) of these gives the information required by the Standard Format and Content, while the second (1.1.2) discusses those aspects of the application unique to the role of the Clinch River Breeder Reactor Plant in the U.S. LMFBR Program. Both of these sub-sections must be read to gain a full understanding of the nature of this application. In particular, the second of these describes the 'parallel' approach adopted in this application and shows how two applications are covered in this PSAR. The third section (1.1.3) descrives the Project Safety Philosophy in terms of the three levels of safety, which form the basis for design evaluations within the Project.

1.1.1 General Information

This Preliminary Safety Analysis Report (PSAR) is submitted in support of a joint application, by Project Management Corporation (PMC) and the Tennessee Valley Authority (TVA), for a permit to construct an LMFBR Demonstration Plant, known as the Clinch River Breeder Reactor Plant. Since this is a Demonstration Plant, the application is made for a Class 104 permit, as specified in 10CFR50.21.

The plant will consist of a single generating unit, employing a liquid metal cooled fast breeder reactor nuclear steam supply system (NSSS). Westinghouse Electric Corporation (Advanced Reactors Division) is responsible for the design of the NSSS and of the steel containment, under the technical direction of the USAEC's Division of Reactor Research and Development. The General Electric Company and the Atomics International Division of Rockwell International Corporation have major sub-contracts, related to the NSSS, from Westinghouse. Burns and Roe is responsible for the design of the balance-of-plant (BOP) and other functions normally associated with the architect-engineer (e.g., characterization of the site seismology, etc.), under the direction of PMC. The plant will ultimately be operated by TVA. Further amplification of the relationships between these participants in the Project is found in Section 1.4 of this PSAR.

The Clinch River Site is in east central Tennessee in the eastern part of Roane County and within the town limits of Oak Ridge, 25 miles west of Knoxville. The site is on a peninsula bounded on the south by the Clinch River and on the north by AEC's Oak Ridge Reservation. Complete details of the site location, layout and characteristics are given in Chapter 2 of this PSAR.

The design power level for the plant is 975 MW(Th), corresponding to a generation of 380 MW(E). This power level is discussed under the terms 'thermal hydraulic' or 'T and H' conditions in various sections of the PSAR. It is this power level which forms the basis for the present application, and in particular, 'T and H' conditions were assumed for the safety analyses presented in Chapter 15. However, the permanent components of the plant (heat transport system, core support structure, BOP, etc.) have been designed for additional capability, namely for a power level of 1121MW(Th) corresponding to 439 MW(E). These latter conditions are referred to as 'stretch' conditions in this PSAR, and in various sections, components are shown to be capable of accommodating 'stretch' conditions. Although 'stretch' conditions do not form the basis for the present application, it is intended that at some future date a supplementary application will be made to increase the power level to these 'stretch' conditions.

The plant is designed with three main coolant loops and the intended mode of operation is that all three loops should be continuously in service. However, it is recognized that maintenance work may require operation with only two loops in service. The power level appropriate to two loop operation will be established at the time of application for the Operating License for the Plant, although it is an objective to attain two-thirds of full power output in this condition. However, at the construction permit stage, the basis of the application is that sufficient redundancy has been provided in the heat removal system to permit two loop operation, in principle, at a power level yet to be defined.

The scheduled completion date for the Plant is September 1981. The corresponding date anticipated for commercial operation is early 1983.

1.1.2 Unique Features of this Application

1.1.2.1 Emphasis on Quantitative Reliability

The role of the CRBRP is unique in that it is a one-of-a-kind plant, with the objective of demonstrating the commercial feasibility of utilizing large LMFBR's for power production in the U.S.A. Facilities, such as the Enrico Fermi Plant, the Experimental Breeder Reactor 2(EBR 2) and the Fast Flux Test Facility (FFTF) can be regarded as predecessors in that they have contributed to the technology needs of this Project. However they cannot be regarded as providing precedents in general on which to judge the design of CRBRP, because of significant design differences (In a sense the FFTF can be regarded as a precedent, because there are a number of similarities in the designs of the two plants, as shown in detail in Section 1.3.2). Moreover, the philosophy upon which this application is based differs in important respects from that appropriate to standard water reactor plants (and earlier LMFBR's), although a level of safety at least equal to that of such plants is required.

Fundamental to this application is an emphasis on the use of quantitative reliability requirements and evaluations. The philosophy embodied in this emphasis is that, by assuring a high degree of reliability, particularly in safety related components, the probability of occurrence of an event resulting in melting of a significant fraction of the reactor core is so low (less than 10^{-6} per reactor year) that it should not be regarded as a basis for design of the plant. A detailed treatment of the application of the reliability program, and preliminary quantitative reliability estimates, are given in Appendix C of this PSAR. This Appendix must, therefore, be regarded as central to the application, and a fundamental part of the foundation for design of the plant. The subject is treated in an Appendix, rather than the main body of the text, for two reasons:

- Wherever possible, the requirements of the Standard Format and Content of Safety Analysis Reports have been followed. Apart from Section 1.5 of this Standard Format, where development needs related to the reliability program are given, there is no place in the Standard Format appropriate to this material, which is unique to this application.

- It is necessary to proper comprehension of this program that all relevant material be collated and presented in a coherent fashion.

It is the firm resolve of the Management of the CRBRP that the reliability program will be pursued and used (with design modifications if necessary) to demonstrate that all possible initiators of core melting accidents (termed 'loss of coolable geometry' in this PSAR) have such an integrated acceptably low probability of occurrence, (less than one chance in a million per year) that they s-ould not be used as a basis for design. However, it has been considered prudent to assure that the plant has some capability for accommodation of such accidents, even though they are not regarded as credible. To this end, specific design requirements have been placed on particula components, systems and structures which are above and beyond those required by consideration of the plant duty cycle. These requirements are termed 'third level design margin requirements' (see Section 1.1.3 for a description of the three levels of safety, from which the term 'third level' is derived) in this PSAR. Each of the design Chapters (i.e., Chapters 3 through 12) contains a statement of the third level margin requirements appropriate to the area of design covered by that Chapter. In Section 15.1.1 (Chapter 15) these requirements are collated to give visibility of the totality of the third level margin requirements for the plant as a whole. Section 15.1.1 also discusses the derivation of these requirements, and the compliance of the design in meeting these requirements.

These requirements were derived, as shown in Section 5.1.1, from generic considerations of core disruptions, based largely on experience gained from the design of the FFTF. However, since their adoption as requirements, analyses of core disruptive accidents for CRBRP have been conducted, and are reported in Appendix D to this PSAR. Section 15.1.1 contains a comparison of the more significant third level design margin requirements with data derived from Appendix D and shows that the margin requirements do not differ substantially from, and are in most cases more conservative than, the Appendix D data.

Chapters 1 through 17 and Appendices A through D (Appendix A is a compilation of computer code abstracts, and Appendix B comprises the plant duty cycle) constitute the application for a construction permit on the basis that loss of coolable geometry should not form a basis for design. All pages in these Chapters and Appendices are colored white and the design to which they refer is termed the 'Reference Design'.

1.1.2.2 <u>Parallel Design Approach</u>

A detailed discussion of these potential initiators is given in the introduction to Appendix D of this PSAR, and so only a brief treatment will be given here. The first of these, fuel failure propagation, is addressed extensively in Section 15.4 of the PSAR, where it is shown that, irrespective of the probability of such propagation, the phenomenology is such that an autocatalytic mechanism involving more than a single assembly does not exist.

Section 15.2 of the PSAR gives an analysis of the consequences of bubble entry into the reactor core. It is shown in that section that a gas bubble approximately inches high and embracing rows of fuel assemblies can be tolerated without core damage resulting. This size of bubble is much larger than the very small bubbles that might actually enter the core. In this connection it should be noted that there are many design features to prevent the entry of bubbles of any significant size into the core; these features are discussed in Section 15.2.

The next two items are those toward which the reliability programs covered in Appendix C are addressed.

Generically, it is possible to hypothesize the following initiating mechanisms for a loss of coolable geometry:

- Fuel failure propagation not detected by Shutdown System sensing equipment.

- Large scale voiding of the reactor core.

- Failure of both of the two scram systems when required to shut down the plant during a transient.

- Failure of the decay heat removal function following a reactor shutdown.

- Massive failure of the primary coolant boundary.

The Project believes that these can be demonstrated to be of such low probability that they should not be regarded as a basis for design and in addition will survey specific initiators of these general adverse conditions. However, certain development programs necessary to confirm this position have yet to be completed. These are described in Section 1.5 of the PSAR, as a part of the application for the Reference Design. Apart from Section 1.5, material relevant to a full understanding of each of these areas is contained in the PSAR as shown in Table 1.1-1 through 1.1-3.

Notwithstanding this position, the Project has agreed (Reference 1) that pending completion of these development programs a Backup Design will be evolving in parallel with the Reference Design. In evolution of the Backup Design it is arbitrarily assumed that each of these two initiating mechanisms is credible, and design features necessary for protection of the public from the consequences of such events are presented in the manner described below. Evolution of the Backup Design will continue unless and until the results of the development programs justify cessation of this effort. Backup features will not be included in Reference Design unless they add significantly to public safety.

Massive failures of the primary coolant boundary are not considered appropriate as bases for design of the plant; because of the properties of the stainless steel piping used in the CRBRP. A full understanding of the rationale behind this statement can be gained from study of the PSAR sections listed in Table 1.1-3. It will be noted that there are certain development programs identified, in Section 1.5 of the PSAR, but that completion of these is expected during 1975. Pending such completion, a Backup Design effort is beign maintained, as indicated below.

Appendix E treats a massive failure of the primary coolant boundary and shows the preliminary design of features that could be incorporated into the CRBRP if the development programs should unexpectedly fail to confirm an exceedingly low probability for such an event. With these features incorporated, Appendix E demonstrates that a massive failure of the primary coolant boundary will not result in loss of coolable geometry. A supplement to Appendix E comprises pages which, when inserted into the main body of the PSAR in place of the currently existing pages, will convert the PSAR into a PSAR appropriate to the Backup Design, in which massive failures of the primary coolant boundary form a basis for design. For clarity of presentation, Appendix E, including the supplement, is printed on green paper.

Appendix F will be completed and submitted in September 1975. This Appendix will present the Backup Design, with loss of coolable geometry of the core treated as a design basis. A specific accident, or a number of accidents, will be presented as design bases. Appropriate design modifications (such as a sealed head access area, and/or an ex-vessel core retention device) will be shown such that, with their incorporation, protection of the public from the consequences of such an event, within the guidelines of 10CFR100 can be demonstrated. A supplement to Appendix F will comprise pages which, when inserted into the main body of the PSAR in place of the currently existing pages, will convert the PSAR into a PSAR appropriate to the Backup Design, in which loss of coolable geometry forms a basis for design. For clarity of presentation, Appendix F, including the supplement, will be printed on yellow paper. The version of Appendix F submitted at this time is to be regarded as a report on status of activities in relation to the Backup Design, and will be replaced with the completed version of Appendix F in September 1975.

1.1.2.3 Preliminary Conclusions from the Reliability Assessments and Parallel Design Studies

At this time, it is not possible to give final reliability data. However, preliminary assessments of both shutdown system and decay heat removal system reliabilities have been conducted, and are reported in Appendix C. These assessments, which were developed using conservative assumptions where data is not yet available, concluded the following preliminary estimates of system unavailability:

Primary Shutdown	$5x10^{-5}$ failures on demand per reactor year
Secondary Shutdown	$6x10^{-5}$ failures on demand per reactor year
Decay Heat Removal	$7x10^{-7}$ failures on demand per reactor year

Since the two shutdown systems are redundant, these data, when combined, indicate an integrated failure probability, on demand, of less than 10^{-6} per year. Thus there is considerable confidence that the original goal, 10^{-6} failures per year quoted in Section 1.1.2.1, is attainable and will be achieved in the design and demonstrated by the reliability program.

However, careful planning of the design and fabrication sequences associated with the Reference and Backup Designs, show that, should the 10^{-6} goal not be realized, in spite of the current high level of confidence, then it will be possible to implement the Backup Design features, if shown to be necessary, with minimal impact on overall schedule. This is illustrated by the schedule given on Figure 1.1-1. From this Figure it can be seen that, ignoring the various interim milestones, the last date for a decision to implement the Backup Design features, without impacting Project schedules occurs in 1979. The same Figure shows that the reliability program will, by that date, have produced sufficient data for a meaningful, more final, evaluation of the failure probabilities of the systems involved.

In summary, then, the Project has tangible evidence that the reliability goals set are attainable. If, nevertheless, against expectation, the goals should not be met, then the Backup Design features can be implemented with minimal schedule impact.

TABLE 1.1-1 SCRAM SYSTEM DESIGN AND RELIABILITY IN THE PSAR

Section	Item(s) Discussed
1.2.6	General Discussion of the Design
1.3	Comparison With Other Designs, in Particular, the FFTF
1.5.1.1	Shutdown System Reliability Program (Overview Only)
1.5.1.3	Secondary Control Rod System Test
1.5.2.5	Critical Experiments for Reactivity Coefficients and Control Rod Worth
3.10	Seismic Design of Category I Instrumentation and Electrical Equipment
4.2.3	Mechanical Design of Reactivity Control Systems
4.3	Nuclear Design
7.1/7.2	Electrical Design of Reactor Shutdown Systems, Including a Failure Mode and Effects Analysis
15.1/15.2/15.3	Accident Analysis, From Which the Performance of the Systems Can Be Judged
Appendix B	Plant Duty Cycle, Indicating the Number of Demands Which Can Be Accommodated by the Design
Appendix C	Detailed Discussion of Reliability Programs and Preliminary Estimate of Scram System Reliability

TABLE 1.1-2 DECAY HEAT REMOVAL SYSTEM DESIGN AND RELIABILITY IN THE PSAR

Section	Item(s) Discussed
1.2.3/1.2.4/ 1.2.5/1.2.7	General Discussion of the Design
1.3	Comparison With Other Designs, in Particular, the FFTF
1.5.1.2	Shutdown Heat Removal Systems Reliability Program (Overview Only)
1.5.1.4	Overflow Heat Removal Development Test
Chapter 3	Design Criteria, Classification of Components, Methods of Analysis, Etc.
4.4.3.8	Thermal Description of the Overflow Heat Removal Service
Chapter 5	Detailed Description of Design
7.4.1	Steam Generator Auxiliary Heat Removal Instrumentation and Control System
7.6.3	Overflow Heat Removal Service Instrumentation and Control
Chapter 8	Electrical Power Supplies
9.1.3.1	Ex-Vessel Storage Tank Cooling System
9.3.2	Overflow and Makeup Circuit
15.3	Undercooling Design Events, Accident Analysis
Appendix B	Plant Duty Cycle
Appendix C	Detailed Discussion of Reliability Programs and Preliminary Estimate of Decay Heat Removal System Reliability

TABLE 1.1-3 PRIMARY COOLANT BOUNDARY INTEGRITY TREATMENT IN THE PSAR

Section	Item(s) Discussed
1.2	General Discussion of the Design
1.5.2.1	Development Programs Associated With Pipe Integrity Assessment
3.2	Classification of Components
5.1.2	Summary Description of the PHTS
5.3	Detailed Discussion and Evaluation of Design of the PHTS
5.3.2.2	Material Properties
5.3.3.6*	Coolant Boundary Integrity
5.3.3.10	Material Considerations, Including Chemistry
7.5.5.1	Sodium to Gas Leak Detection System
15.6	Consequences of Primary Boundary Leaks
Appendix B	Plant Duty Cycle
Appendix E	Consequences of Hypothesized Massive Failure of Primary Piping and Description of Design Features to Mitigate These Consequences

* This Section contains the principal collation of material relative to piping integrity.

1.1.2.4 Summary of Design Safety Approach for the CRBRP

This Section represents an updated version of the Design Safety Approach given in the Reference Design Report (Reference 2), to incorporate the agreements and understanding developed in Reference 1:

1. The following CRBRP Design Safety Approach is generally consistent with the three levels of safety concept used by AEC's Regulatory Branch to evaluate the adequacy, for licensing purposes, of nuclear power reactors.

    a. The first level focuses on the reliability of operation and prevention of accidents through the intrinsic features of the design, construction, and operation of the plant, including quality assurance,redundancy, testability, inspectability, maintainability, and failsafe features of the components and systems of the entire plant.

    b. The second level focuses on the protection against "Anticipated Faults" and "Unlikely Faults" (as defined in Table 1.1.3-1) which might occur despite the care taken in design, construction, and operation of the plant set forth in Level One above. This protection will ensure that the plant is placed in a safe condition following one of these faults.

    c. The third level focuses primarily on the determination of events to be classified as "Extremely Unlikely Faults" (as defined in Table 1.1.3-1) and their inclusion in the design basis. Table 1.1.3-2 contains a list of such "Extremely Unlikely Faults". These faults are of low probability and no such events are expected to occur during the plant lifetime. Even though they represent extreme and unlikely cases of failures, they have been analyzed using the same conservative assumptions as those employed in consideration of second level events. Additionally, as described in Item 2, Level Three includes consideration of severe accidents which are even less probable than extremely unlikely faults.

2. With respect to Level Three, in keeping with past practice for first-of-a-kind plants, the project plans to incorporate features designed on the basis of accommodating a range of events including those having an exceedingly low probability of occurrence. Extensive R&D programs are being undertaken with the objective of confirming that failure to scram and other potential sources for initiating severe accidents have a sufficiently low probability of occurrence that they need not be considered as bases for design. Nonetheless the project plans to incorporate features and margins in the design to mitigate accident consequences from loss of inplace coolable core geometry and these features and margins include:

a. Impulse energy absorption features in the head;

b. Primary system features (including supports) designed to accommodate above normal dynamic loadings (See Table 1.1.3-3);

c. Reactor core internals designed to enhance post accident cooling capability and reduce the potential for secondary criticality; and

d. A low leakage containment housing the entire reactor coolant system.

3. As a parallel effort, the project wll conduct detailed analysis and R&D work relative to low probability accidents involving a loss of in-place coolable geometry in order to gain a more complete understanding of their consequences. These studies will include design studies on features which have a potential for mitigating such consequences (for example, a sealed head access area,which may or may not be inerted, an ex-vessel core catcher or other consequence limiting feature). In the event that the reliability programs, discussed in 2 above, should be unable to demonstrate an acceptably low probability for an event leading to core disruption, a core disruptive accident will be selected and used as a design basis for the plant. The selection of such a design basis event will incorporate all existing understanding of the phenomenology of such events, to assure as much realism as possible in the selection.

The Reference Design will be evaluated for its capability to accommodate the design basis event. Should this evaluation show it to be necessary, the additional features, addressed above, will be incorporated into the design.

1.1.2.5 References

1. Letter from L. Manning Muntzing (Director of Regulation) to John A. Erlewine (USAEC General Manager) 'CRBRP Licensing Review', January 2, 1975.

2. Clinch River Breeder Reactor Plant, Reference Design Reports, June 1974.

TABLE 1.1.3-1

DEFINITION OF TRANSIENTS

1. Anticipated Fault

   An off-normal condition which individually may be expected to occur once or more during the plant lifetime.

2. Unlikely Fault

   An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all components and systems, events in this category may be expected to occur a number of times.

3. Extremely Unlikely Fault

   An off-normal condition of such low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures.

TABLE 1.1.3-2

TENTATIVE LIST OF EXTREMELY UNLIKELY FAULTS

Design Basis Earthquake, Flood, or Tornado

Large Intermediate or Steam System Pipe Rupture

Sodium Fire Above the Operating Floor

Large Sodium Spills Inside and Outside Containment

Loss of Cooling to a Control Assembly

Large Na-H_2O Reactions in the Steam Generator

TABLE 1.1.3-3

PRIMARY SYSTEM COMPONENTS DESIGNED TO ACCOMMODATE ABOVE NORMAL DYNAMIC LOADINGS

Reactor Vessel Walls and Nozzles

Core Support Structure

Reactor Vessel Support Ledge

Reactor Vessel Head

Intermediate Heat Exchanger and Supports

Primary Sodium Pumps and Supports

Check Valve

Primary Piping and Supports

Vessel Support Structure

RELIABILITY PROGRAM AND PARALLEL SCHEDULE

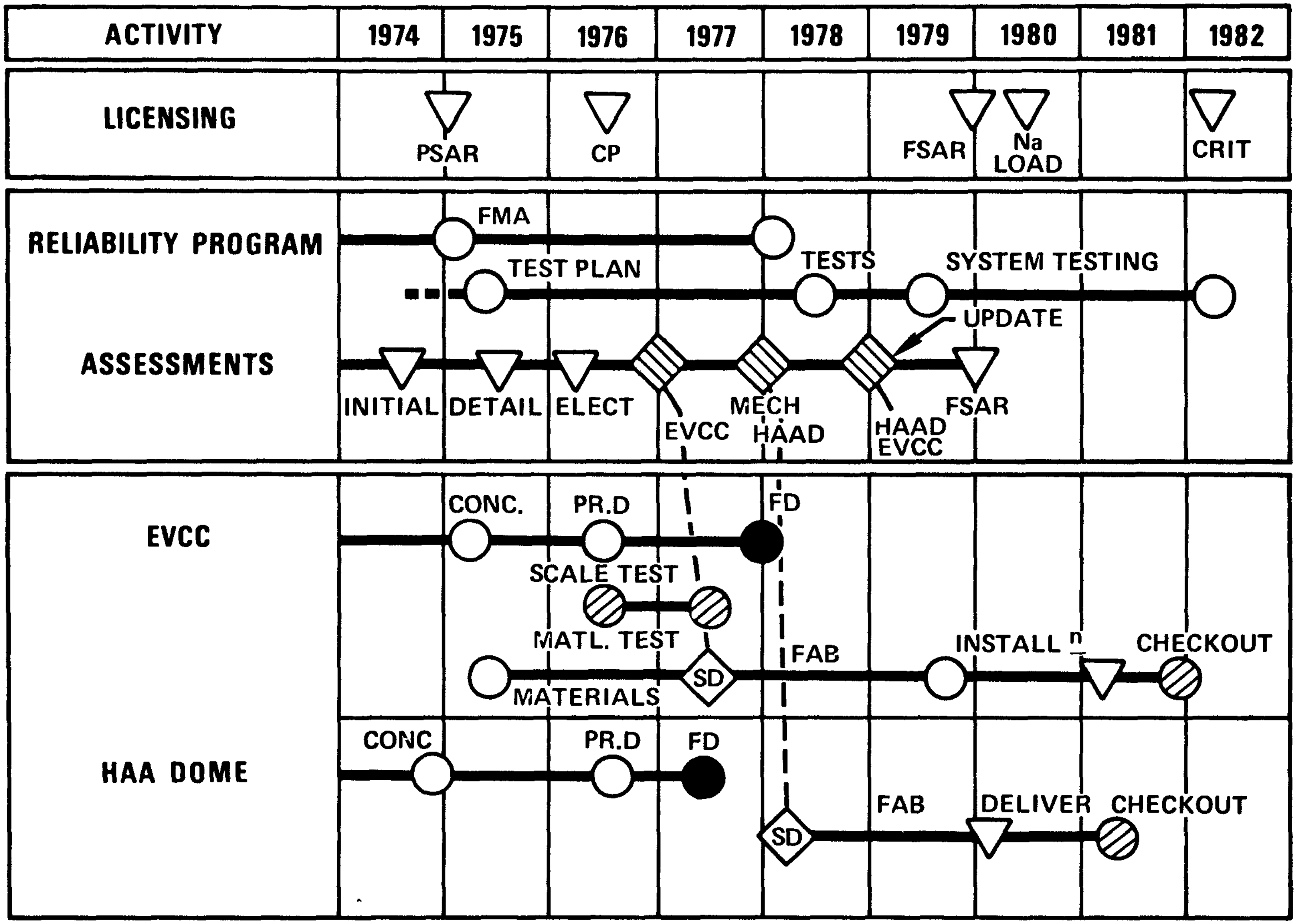


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Figure 1.1-1

D.1.0 Summary

The purpose of this section is to explain the objectives for the reliability programs in the context of plant safety, and to show how reliability goals are to be met.

D.1.1 Overall Objective

The purpose of the CRBR Reliability Program is to assure a negligible risk to the public from consequences associated with the nuclear core aspects of the plant and to confirm this negligible risk in a quantitative manner. Such risk can only be the result of public exposure to the fission product and fissile material inventory in the core. Apart from normal clean up of the primary system following any random fuel pin failures, the event which is common to any public exposure to core material is 'loss of coolable geometry' (LCG). Loss of coolable geometry in this sense is defined as the lower limit of core conditions beyond which there is no assurance that a core disruptive accident could not occur. It should be noted that the probability of occurrence of a LCG does not represent a statement of risk to the public. The risk element is dependent upon the amount of radioactive materials, if any, released to the environment should an LCG occur, and the population distribution and environmental conditions surrounding the plant. The amount of radioactive materials released, if any, would vary according to the severity of the LCG itself, the time during the fuel cycle in which the event occurred, and the effectiveness of the containment. The public risk associated with a LCG is seen to be some set of numbers, significantly smaller than the probability of LCG itself, relating to specific consequences wherein the effects of LCG are mitigated by the elements described above. However, since LCG must be a part of any significant consequence scenario, it has been used as a lower limit conservative basis for setting reliability requirements for the CRBR.

The criterion for setting CRBR reliability requirements was to specify requirements which are achievable, confirmable and publically acceptable. The overall numerical goal determined to satisfy this objective was: 'The probability of Loss of Coolable Geometry shall be less than one chance in a million per reactor year'. Confirmation of this requirement provides a firm basis for designating LCG as an insignificant event which does not need be considered as a design basis accident. This numerical requirement imposed upon the CRBR is much more stringent than is inferred from design objectives of current LWR's. However, comparisons made did not attempt to equate potential consequences related to the two reactor types. Consequences of LCG in CRBR are being addressed in a parallel task (Appendix D of the PSAR).

This one chance in a million risk is a valid goal since studies have shown that society has accepted risks of this order of probability, whatever the benefits or consequences, (1), (2) and Regulatory studies have also come to the same conclusion that such an order of failure rate is an acceptable one. (3), (4). Additionally, the project has risk analysis studies in place to quantify the consequences of such failures to the public and preliminary information indicates that these consequences will be minor.

D.1.1.1 Public Acceptance of Risk

Chauncey Starr in a number of papers has evaluated the social risk and social benefit from a large number of different activities, both imposed on society and voluntarily engaged in by society [References 1, 2]. He concluded that the public was less willing to accept high risks imposed on individuals by the state, than it was in accepting a risk voluntarily entered into, even though the apparently higher benefits in the latter case were often illusory. Despite this, he also concluded that the present acceptable risk from the production of electricity was of the order of 2×10^{-5} fatalities per person per year (Figure D-1). In later testimony before the JCAE [Reference 2], Starr concluded that the target risk level for nuclear stations should be substantially lower than the already accepted risk levels associated with fossil fuel for providing the same technical performance, and he concludes therefore that his estimates of the risk from a nuclear power plant, of the order of 0.1×10^{-5} compared to 10×10^{-5} for fossil plants, was well within the socially acceptable range of risk (Figure D-2).

One ought to note that these figures refer to fatalities, whereas the target goal for loss of coolable geometry is a long way from expressing damage to the public of even a minor consequence, let alone a fatality. Indeed if the integrated probability of loss of coolable geometry was 10^{-6}, then there would be some other probability that significant radioactivity could be released from the containment (say 10^{-x}) and a further probability that that released radioactivity might cause fatal hazard to any member of the public (say 10^{-y}). The overall probability of risk to the public would then be $10^{-(6+x+y)}$, significantly less than 10^{-6}, and probably of the order of 10^{-10} or 10^{-12}. It is this number that should be compared to Starr's 0.1×10^{-5}.

D.1.1.2 The Regulatory Position

Further, Regulatory documents have also substantiated a 10^{-6} figure as a conservative goal [References 3, 4]. The Regulatory report on the treatment of anticipated transients without scram [Reference 4] says:

> *The envelope of design basis accidents does not include all events that are conceivable for a nuclear power plant. It is not necessary, nor even possible, to design nuclear power plants, or any other man-made device or system, for all conceivable eventualities that are physically possible.. Thus, at the very low probability end of the spectrum of all possible events there is a residuum of conceivable accident sequences that could, if they occurred, lead to radiological consequences outside the plant boundary in excess of the Part 100 guidelines. These are the Class 9 accidents mentioned in AEC environmental impact statements. It is the judgement of the AEC that these events are of such low likelihood as to present in sum an acceptable risk in view of the benefits derived from the electricity produced by a nuclear power plant.*

and further that:

> *For an anticipated population of about one thousand nuclear plants in the United States by the end of the century, the safety objective will require that there be no greater than one chance in one million per year for an individual plant of an accident with potential consequences greater than Part 100 guidelines. Since plants now being designed and constructed are expected to have service lives approaching 40 years, and may thus be part of the century-end population, the staff believes it appropriate to consider their designs in the light of this future requirement. In view of the difficulty of determining such a low probability, the staff regards this number as an "aiming point", or design objective rather than as a fixed number that must be demonstrated for a given plant design.*

Based on this discussion, it could be argued that for a single plant of a new technology rather than 1000 in operation then it would be acceptable to use a target of one chance in a thousand to meet the same overall risk to the population if the consequences of failure are assumed to be similar. Indeed since in any technology, performance in reliability always improves as more becomes known, then as later plants come on line, the target goal could be raised from 10^{-3} to meet the regulatory figure of 10^{-6} for a large number of plants. This improvement of reliability as the LMFBR industry grows would have two components: the increase of experience proportional to the integrated plant years of operation, and the increased testing and development proportional to the period of growth. Increases in reliability of one or two orders of magnitude per decade could be expected in the early years. Against this discussion the conservatism of the present CRBRP goal of 10^{-6} against loss of coolable geometry is clear. The Regulatory position is based on operating data from a number of light water plants and so this conservatism for the LMFBR goal is appropriate to compensate the lack of operating data. It certainly conforms with the definitive statement made in Reference 3:

> *The safety objective is that the likelihood of all accidents with significant consequences not included in the design basis envelope should not be greater than one chance in one million per year, i.e., should not occur with a failure rate greater than 10^{-6} per year. For the particular potential failure path of ATWS, the staff believes that a failure rate of the order of one tenth of the overall safety objective is an appropriate objective.*

In fact, the CRBRP objective for scram reliability is indeed 1/10 of the overall goal or 10^{-7}.

Moreover, the assumption is made in the foregoing discussion that the consequences of failure in an LMFBR are no greater than the consequences of failure in a Light Water Reactor. However, the comparison made for the FFTF reactor [Reference 6] based on the risks calculated in the Rasmussen Report [Reference 7] show the risks to be up to two orders of magnitude less. Even for the CRBRP site and changed population distribution around that site we would expect a similar study also to show favorable results. Thus, a comparison of probability goals based on assumed equal consequences is probably still further conservative. Such a study of the consequences has been initiated by the CRBRP project and should provide initial results by mid-1975.

D.1.1.3 Summary of Acceptable Risk Level Considerations

In summary, the project agrees with the Regulatory published position that 10^{-6} failures per year is an acceptable goal for the loss of coolable geometry which has the potential for attainining a condition leading to a release of radioactivity from the primary system. Such a goal should be viewed against a much lower probability (by some 4 to 6 orders of magnitude) of endangering the public, and an expected public acceptance figure of some 4 to 6 orders of magnitude greater than the goal, an acceptance figure which agrees with a logical choice of a goal in a new technology. In this light the CRBRP reliability goal is very conservative and its attainment gives excellent surety of the safety of the plant from hypothetical core disruptive accidents.

To assure that the overall goal of less than one chance in a million per reactor year for LCG is met, all events which could contribute to this probability must be addressed (See Figure D.1-4). The two functional elements of the plant which have the greatest impact upon attainment of this goal are the shutdown system (SDS) and the shutdown heat removal system (SHRS) since they are generic to a large number of possible LCG initiators. Included in the reliability assessment of the SDS and SHRS is the structural adequacy of the systems themselves as well as those of interfacing components which could affect their functioning to prevent LCG when required. The reliability programs described within this appendix are directed toward confirmation that these two systems meet reliability objectives consistent with the overall goal.

These two systems have been selected for priority attention by the project since, not only are they the most important safety systems and are generic to a large number of accident situations, but they also require the priority establishment of extensive and time consuming test programs.

It is also recognized that potential for LCG not within the prevention capability of the SDS and SHRS must also be addressed. One such consideration within this category is undetected fuel failure propagation not initiated by a mechanism sensed by the SDS. Fuel failure propagation is a low probability event and is the subject of a separate study at this time (Section 15.4). Another example of this type is sabotage. In all cases the first defense to prevent LCG from mechanisms in this category is in the design itself which either eliminates their potential for occurrence or makes it acceptably small. Where required, strict administrative and security measures will also be employed. The overall goal, therefore, has been set recognizing that all initiators of LCG must be addressed and shown to be consistent with the overall goal; all potential initiators must have a combined probability of occurrence of less than one in a million per reactor year.

D.1.2 Goals (Numerical)

Initial numerical reliability allocations have been made which are consistent with the overall goal and which provide the designer with a realistic and challenging reliability objective. These goals were set recognizing the relative difficulty of achievement among the differing systems and are subject to change within the constraints of the overall goal for LCG. Because the most probable initiators which could cause LCG are prevented by the SDS and SHRS, these two systems have been allocated the largest portion of the goal.

Event/System	Goals (Failures to Prevent LCG/yr.)
Shutdown System Failure with Anticipated Transient	$<10^{-7}$
Shutdown Heat Removal System Failure following Shutdown	$<8 \times 10^{-7}$
Other Faults Leading Directly to LCG	$<10^{-7}$

To achieve these goals, a three pronged approach to success has been set into place for the reliability program.

- Numerical confirmation of an acceptably low random independent failure rate for the shutdown systems and shutdown heat removal systems.
- Confirmation that the reliability enhancement obtained from protective system redundancy is not lost due to potential common mode failures.
- Identification of all possible initiators of LCG, with numerical analyses where appropriate and management control where numerical analysis is deemed inappropriate.

An amplification of how Common Mode Failures (CMF) will be treated within the program is appropriate. The first thrust is a systematic search to identify and eliminate potential for CMF. This will utilize both test and analysis. Where it is impossible to completely eliminate a potential CMF (e.g., control rods are suspended from the same head and must pass through interfaces with finite clearances) a combination of test and analysis will be employed to ensure that the probability of occurrence is very very low indeed. The project does not believe that the probability of CMF can be adequately quantified at this time, but the project does believe that by suitable qualitative analyses and by adequate management control the probability of CMF can be made so low that it does not impact attainment of the system reliability goal. Thus, while CMF will not be allocated a portion of the overall numerical goal, sufficient test and analysis will be performed to assure that it does not impact attainment of that goal.

D.1.3 Current Reliability Assessment

The current reliability assessment is that the plant meets the overall objectives presented earlier. It is important to recognize that this conclusion is not based solely on the quantitative analysis. The basis for the conclusion of plant safety adequacy comprises four major elements, namely:

- The quantitative assessment based on available reliability methodology and hardware reliability information.
- The qualitative reliability activities within the Project which impose a systematic and disciplined method of plant design, improving on the more nearly intuitive method of conventional design. This approach serves to minimize the likelihood of design oversights, and in particular highlight the potential for common mode failures.
- The presence of redundancy and diversity in essential design features in the systems of interest.
- The ability to recommend design and procedural changes to enhance the reliability over and above that required to meet normal design practices.

D.1.3.1 Numerical Assessment

Seven or more numerical assessment are schedularly identified prior to FSAR submittal, three of these being prior to CP. The present analysis is a preliminary assessment to judge the balance of goal allocation and to guide the relative priorities. It also indicates that goals can be attained.

D.1.3.1.1 Shutdown System

An initial reliability assessment of the Shutdown Systems has been completed and is described in detail in Section
Results of this analysis provide high confidence that the SDS meets the numerical reliability goals associated with the prevention of loss of coolable geometry, which in the analysis is conservatively represented by prevention of sodium boiling.

Table D.1.1 summarizes the results of the analysis and compares the allocations to the various subsystems with the current numerical assessments for those subsystems. It should be noted that the results are presented in terms of unavailability which is an indication of the probability of failure per challenge to the system. It is expected that there will be fewer than one challenge per year that would result in LCG if the reactor did not scram, with an upper bound of two or three such challenges per year. Even for the worst case, the system more than satisfies the goal of less than 10^{-7} failures per year that could lead to LCG. This analysis uses conservative data for the electrical system obtained from analytical predictions for the FFTF hardware and commercial reactor experience. The mechanical system assessment is based upon response to a worst case reactivity transient. An individual rod unavailability of 0.01 was used for each rod in the primary and secondary system, and system availability determined by using a mathematical model (number of rods required versus the number available). Based on preliminary analysis and data from FFTF and LWR's, an unavailability of 0.01 is conservative and thus provides a conservative estimate for the mechanical systems.

D.1.3.1.2 Shutdown Heat Removal System

Based on the analysis presented in detail in Section D.3.2, it is concluded that the post-shutdown heat removal system is reliable enough that its dependability in preventing core disruptive accidents after a scram supports eliminating these accidents as design bases. The result of the initial reliability assessment of the shutdown heat removal system is that the probability of loss of coolable geometry due to failure of the heat removal equipment is 0.66×10^{-6} per reactor year. This result is consistent with the preliminary allocation discussed earlier of 0.8×10^{-6} for shutdown heat removal. For purposes of the heat removal reliability, a criterion of failure more conservative than loss of coolable geometry was used as a limit. The criterion which was applied was that sodium bulk temperature within the reactor vessel should not exceed about 1250°F. This temperature would not produce a loss of coolable geometry, but the integrity of the primary system might be in doubt - for long term operation at temperatures higher than this value. Analysis is continuing to confirm the acceptability of this limit. However, work to date gives good confidence that the 1250°F limit is not only acceptable but conservative and that a somewhat higher limit may later be able to be utilized. Operation of portions of the heat removal equipment at this temperature is, of course, treated as a faulted condition, implying that some of the equipment may not be reusable after being exposed to the 1250°F environment during a given shutdown. However, such faulted operation is appropriate for this type of failure analysis.

Table D.1.1

RESULTS OF CURRENT UNAVAILABILITY ASSESSMENT
FOR SHUT-DOWN SYSTEMS

		Electrical	Mechanical	System
Primary System	Allocated Unavailability	2.5×10^{-5}	7.5×10^{-5}	1.0×10^{-4}
	Current Assessment Unavailability	5×10^{-5}	1.5×10^{-5}*	6.5×10^{-5}
Secondary System	Allocated Unavailability	1.2×10^{-4}	3.8×10^{-4}	5×10^{-4}
	Current Assessment Unavailability	6×10^{-5}	2×10^{-5}*	8×10^{-5}

* Based on rod unavailability of 0.01 which will be confirmed by the test programs.

D.1.3.2 Qualitative Reliability Assurance Actions

A second source of confidence in the reliability of the systems under discussion are those activities underway in the design process that are planned to minimize the risk of design oversights. These activities guide the designer, who has traditionally followed a largely intuitive approach, through a systematic and disciplined review of the operating features of his design. These actions include failure mode and effects, fault tree, common mode failure, and single point failure analyses. Some of these analyses are quantified to indicate areas of special concern (and candidates for potential modification) and to serve as a data source for the numerical reliability analysis. An especially important objective of these analyses is to locate potential common mode failure sources, which are then eliminated or consciously, with adequate management attention, accommodated in the design. Absolute assurance that design errors are identified and corrected is not a realistic objective. However, conscientious use is being made of the best available tools provided by modern reliability technology to reduce design errors to the lowest practicable level. It is intended that these qualitative analytical programs will be demonstrably valuable in identifying and eliminating potential common mode failure initiators or contributors.

D.1.3.3 Design Redundancy and Diversity

A further consideration in support of the conclusion that these systems meet the stated objective of extremely high reliability is the inherent redundancy and diversity in the system designs.

D.1.3.3.1 Shutdown System

A significant factor in support of the high reliability assessment of the Shutdown Systems is the redundancy and diversity in the systems design. The systems consist of two independent control rod systems (Primary and Secondary) which have diversity to avoid common mode failures between them. Reactor shutdown can be achieved by either system with the other system completely inoperable even with a stuck rod in the operable system. To assure that the two shutdown systems are independent the two systems are mechanically and electrically isolated from one another. Each shutdown system has been designed to include sufficient redundancy to ensure that single internal random failures will not cause degradation of protection. The redundant components within each individual shutdown system are also mechanically and electrically isolated. The Primary Shutdown System uses a different plant parameter (except for flux monitoring - in that case different type sensors are used) than the Secondary Shutdown System to provide protection against any particular fault condition being sensed.

As noted above, the secondary control rod system concept has been selected with the intention of providing a shutdown system which is diverse relative to the primary shutdown system. Indeed this diversity between systems is provided even though it may not be possible to confirm the same random independent failure reliability for the secondary system as for the primary system. Table D.1.2 compares those principal features of the secondary control rod system and primary control rod system which are different between the two systems. The diversity between the two systems enhances the plant shutdown reliability by minimizing the potential for common mode failures, such as failures of parts or unlikely malfunctions such as life induced distortions common to the two systems.

D.1.3.3.2 Shutdown Heat Removal System

The key elements of shutdown heat removal system redundancy and diversity are:

- Post shutdown heat removal can follow any one of three parallel paths (the three heat transport loops) immediately after scram, and any one of four paths (the normal heat transport loops plus the overflow heat removal system [OHRS]) beginning about one hour after scram.

- When heat is removed through the normal heat transport loops, multiple ultimate heat sinks are available:

  a) Beyond the sodium/water heat exchangers, three heat sinks which are in most respects redundant and quite diverse in their functioning are available. The sinks are the main condenser, the power relief valves for steam venting to the atmosphere, and, after about an hour after scram, the protected air-cooled condensers (PACC) for steam-to-air heat transfer.

  b) Within the steam/water system, two sources of stored feedwater are available, as well as there being a main and an auxiliary feedwater pumping system.

- As a redundant path with the heat transport loops, decay heat can be dumped through the OHRS system. The OHRS system is an all liquid metal system, and therefore diverse in this important regard from the sodium/water interface in the normal heat transport paths. The OHRS system utilizes sodium/sodium, sodium/sodium-potassium, and sodium-potassium/air heat exchangers.

Table D.1.2 Shutdown System Diversity of Design

Control Assembly (CA)	Primary	Secondary
Control Rod	37 pin bundle	19 pin bundle
Guide Geometry	Hexagonal	Cylindrical
No. of Control Rods	19	4
Control Rod Driveline (CRD)		
Coupling to CA	Rigid Coupling	Flexible Collet Latch
Connection to CRDM	CRD Leadscrew to CRDM Roller Nuts	CRD Attached to CRDM Carriage with Pneumatic Activation of CRD Latch through Slender Rod
Disconnect from CA for Refueling	Manual	Automatic
Control Rod Drive Mechanism (CRDM)		
Type of Mechanisms	Collapsible Rotor-Roller Nut	Twin Ball Screw with Translating Carriage
Overall Mechanism Stroke	37 Inches	69 Inches
Scram Function		
Scram Release	Magnetic, Release CRDM Roller Nuts	Pneumatic, Release CRD Latch in CA
Scram Assist	Spring in CRDM	Hydraulic in CA
Scram Speed Versus Flow Rate	Increases with Decreasing Flow Rate	Decreasing with Decreasing Flow Rate
Scram Assist Length	14 Inches	Full Stroke
Scram Deceleration	Hydraulic Dashpot	Hydraulic Spring
Scram Motion through Upper Internals	Full Stroke	0.25 Inch

a) Within OHRS, all pumping is by electromagnetic pump, quite diverse from the mechanical pumps in the heat transport system sodium loops.

b) At a certain point after scram, the OHRS system becomes internally redundant, that is, half of its heat removal capacity is adequate to dissipate the decay heat production load. Equipment arrangements (principally pumps and heat exchangers) are such that true redundancy exists with the exception of elements like some common piping runs.

- Diesel generators are provided as redundant and diverse sources of power for heat transport and OHRS equipment requirements.

D.1.4 Programs for Verification and Improvement

While the normal design procedures will produce a reliable design and the preliminary assessment indicates that the various parts of the design do meet the reliability goals allocated to them, nevertheless this activity is not adequate to confirm the reliabilities with confidence or to have an ordered program whereby reliability can be improved and whereby weak design links can be identified and corrected. Hence, comprehensive programs have been established for the shutdown and heat removal systems to do just this.

D.1.4.1 Shutdown System

The Shutdown System Reliability Program is described in the PSAR, both in Section 1.5 of the main text and in Section C.4.1 of Appendix C of the PSAR. The purpose of the program is to confirm the reliability of the CRBRP shutdown systems; in particular that a failure to scram concurrently with an anticipated transient can be shown to be of sufficiently low probability in meeting the allocated goals that such a combination of events should not be treated as a basis for design.

The program provides a balanced effort of qualitative analytical assessment with component, subsystem and system testing to provide adequate data for system reliability quantitative evaluation. Four major tasks can be identified within this effort:

- A comprehensive set of reliability methods is being collected and developed into a manual for project-wide use. Included in this effort are: procedures for management of the reliability programs and guidelines for model and success-failure criteria development; methods for qualitative and quantitative reliability analysis and computer program development under the appropriate duty cycle conditions; and procedures for the collection and the use of data from both the CRBRP test and other relevant programs.

- The reliability analysis task uses two approaches:

  a) Qualitative analysis - to establish the fault paths leading to potential failure; to identify the potential for common mode failures; and to integrate the component and subsystem failure mode analyses into system level analysis to identify single failure points within each system.

  b) Quantitative analysis - to perform sensitivity analyses; to define reliability goals for subsystems and components; to iteratively perform updated reliability evaluations of components, subsystems and systems; to provide bases for test programs and interpretation of test results; to define a priority listing of component subsystem and system improvement areas.

  A part of this task is the definition of goals for each system as well as the definition of the transient events and envelope of conditions which directly affect the system reliability. The analytical effort includes the evolution of reliability block diagrams and mathematical models of the subsystems and components that affect shutdown reliability.

- The data base development task consists of the collection of reliability data, including applicable abnormal operating experience and maintenance problems from all types of reactors, as a source of dependable input for reliability assessment. Computer codes will be adapted or developed for the storage and selective retrieval of data from both the CRBRP and other applicable programs.

- The test phase of the program provides the relevant data necessary to define the overall CRBRP shutdown systems reliability when this data is integrated by the reliability analysis with data from other sources such as existing component and part data, and FFTF and CRBRP design verification test data. The test plan provides a balanced combination of tests involving varying levels of component, sub-system and systems tests of primary and secondary shutdown systems electrical and mechanical hardware.

D.1.4.2 Shutdown Heat Removal System

The Shutdown Heat Removal System Reliability Program is described in Section 1.5 of the body of the PSAR. The program is intended to confirm the reliability of the shutdown heat removal system, with emphasis on those items most in need of verification as indicated by the first assessment. That assessment essentially confirms the adequacy of the shutdown heat removal reliability program as originally planned with very few minor modifications to the proposed tests.

The program consists of four major tasks:

- Reliability analysis methods are being developed to supplement the main methods development and reliability manual preparation effort within the Shutdown System Reliability Program. The methods being developed within the heat removal program are those of unique application within that program, such as reliability analysis approaches for pressure vessels and heat exchangers and for heat transport boundary and support structures.

- Quantitative and qualitative analyses are performed under a reliability analysis task. Failure mode, fault tree, common mode failure, and single point failure analyses are included. The end items of greatest general interest to the LMFBR program are the overall system reliability assessments, which are scheduled for refinement and periodically issued updates.

- The data collection task supplements the reliability data bank task within the Shutdown Systems Reliability Program with failure and repair data specifically related to heat transport components. This task also has the objective of defining the most important data needs as a source of recommendations for testing within and outside this program.

- The test program task includes reliability testing of key components in the heat removal systems. The following components are included: the steam generator tubes, sodium leak detectors, intermediate loop pressure relief rupture discs, the power pressure relief valves, the steam generator auxiliary heat removal system (SGAHRS) instrumentation and controls, the protected air-cooled condenser (PACC) louver actuators, the turbine by-pass valve, isolation and control valves in the steam generator auxiliary heat removal system, the sodium pump bearings and pony motors, a segment of welded main loop sodium piping, and the most critical sodium component nozzle. The first two items, related to steam generator tube leakage and leak detection dependability, were added to the test program when the importance of these items was highlighted by the reliability analysis.

D.1.5 Achievement of Reliability Goals

The confirmation process associated with the previously defined reliability goals is structured to provide necessary confidence in final goal achievement at significant interim project decision points. By the time a construction permit is requested a detailed random independent and common mode failure analysis of the shutdown system will have been completed. The electrical portion of the analysis will be based upon the final design parameters and will utilize data from almost identical components in the FFTF. The mechanical design at this point in time will have been modified, if required, by earlier qualitative reliability analysis (FMEA, FTA) and numerical analysis will include the effects of a detailed evaluation of the design and transient response requirements for the SDS. Much of the conservatism inherent in the analysis presented in this appendix will have been reduced, with the result of very high reliability predictions and little doubt should exist that the in-place test program underway will confirm the final goals. Potential common mode failures will have been identified (through a systematic application of Failure Modes and Effects and Fault Tree Analyses) and they will either have been eliminated from the design or their probability of occurrence shown to not impact achievement of the reliability goal. By the time of FSAR submittal, the portion of the test program required to confirm random independent failure goals will be essentially complete. Certain aspects of the electrical SDS tests may still be in process, but until their completion, options such as a reduction in the maintenance test interval could be retained, which would provide the necessary reliability. Because it is recognized that CMF potential could be impacted by construction and operational practices, this portion of the program will be continually updated through criticality.

The Shutdown Heat Removal System Reliability Program will also provide interim results appropriate to plant construction milestones, namely, the Construction Permit, FSAR, and Initial Criticality. The program will produce the following major end items in support of the Construction Permit: an initial and updated numerical assessment of heat removal reliability, completed FMEAs on the systems and components of interest, significant progress with Fault Tree and Common Mode Failure Analyses, an initial assessment of the reliability implications of designing components to ASME Code Section III, and detailed plans for testing to be carried out under this program. All program tasks except the analysis of operational data will be completed before submittal of the Final Safety Analysis Report. These tasks include the final reliability assessment; completed Fault Tree, Common Mode, and Single Point Failure Analyses; and completion and interpretation of all planned testing. Before plant criticality, data gathered during operational shakedown of the plant equipment will be collected and studied for verification of prior reliability conclusions.

D.1.6 Conclusions

Plant safety reliability objectives were presented which are judged to conform to responsible regard for the health and safety of the public. These objectives are considered adequate to provide a basis for the licensing of the plant without treatment of a core disruptive accident, characterized here as loss of coolable geometry as a basis for design. Moreover, considerations were summarized, in anticipation of more detailed exposition in later portions of the PSAR Appendix, to provide reasonable assurance that the plant safety systems do indeed meet the objectives.

The arguments for the adequate provision of plant safety include:

- A quantitative assessment (based on available methods and data) of less than 0.9×10^{-6} likelihood of loss of coolable core geometry due to failure of the scram systems or heat removal equipment, the equipment of widest impact on plant safety.

- Analyses which serve to minimize the likelihood of system or component design oversights, with emphasis on reducing to the extent practically possible the number of potential common mode failure sources.

- A system design with the characteristics of equipment redundancy and true functional diversity, following sound engineering principles for the design of equipment with such stringent reliability requirements.

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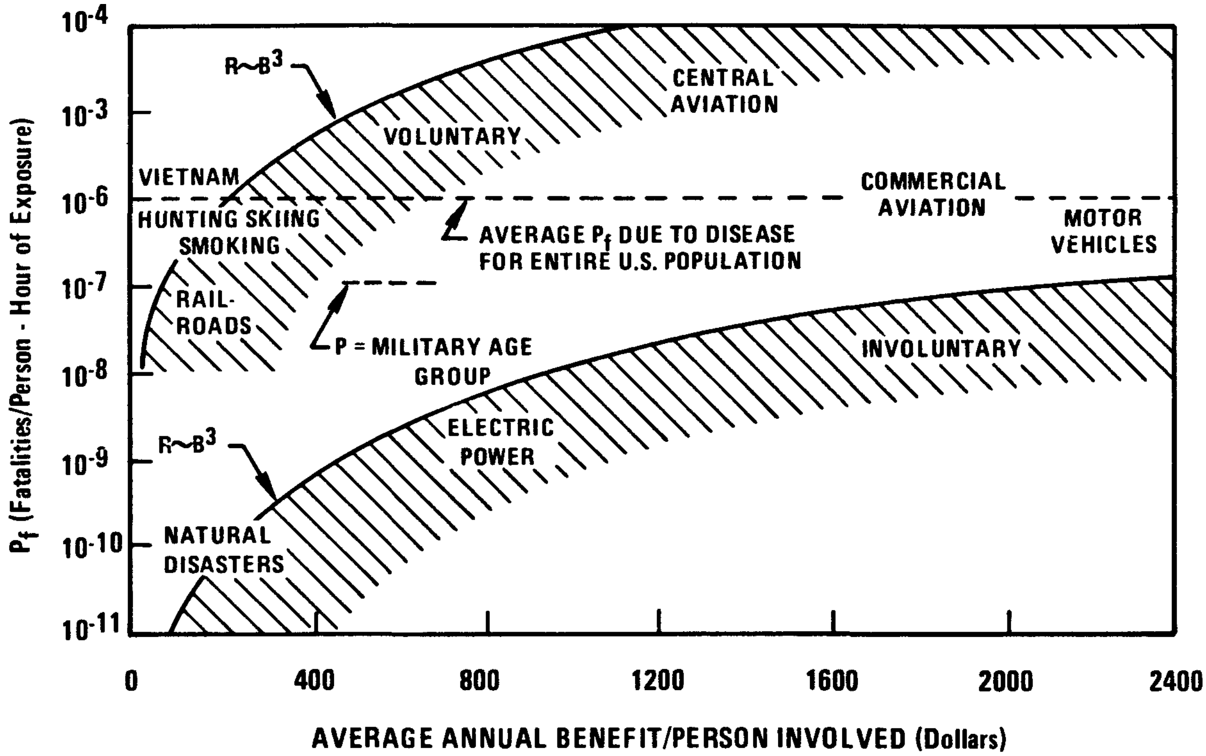

Figure D-1. Risk (R) Plotted Relative to Benefit (B) for Various Kinds of Voluntary And Involuntary Exposure

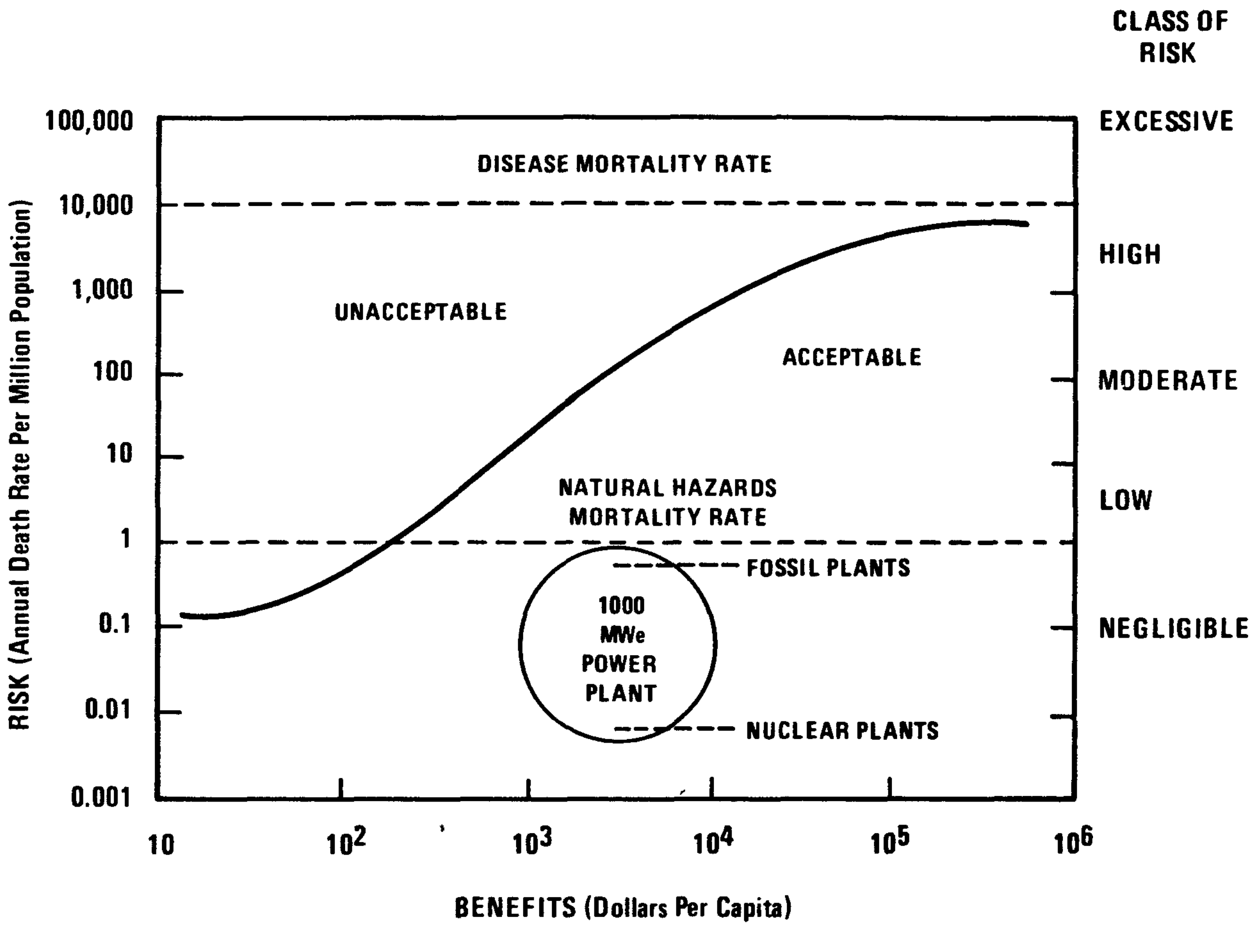

Figure D-2. Benefit-Risk Pattern Involuntary Exposure