

**CLINCH RIVER
BREEDER REACTOR PROJECT**

**PRELIMINARY
SAFETY ANALYSIS
REPORT**

SUMMARY VOLUME

PROJECT MANAGEMENT CORPORATION

NOTICES

1. The Energy Reorganization Act of 1974 (Public Law 93-438) establishing the Energy Research and Development Administration (E.R.D.A.) and the Nuclear Regulatory Commission (N.R.C.) became effective on January 19, 1975.

Throughout this Preliminary Safety Analysis Report, appearance of or reference to the Atomic Energy Commission (A.E.C.) (with the exception of the Directorate of Regulation) will now mean the Energy Research and Development Administration.

Appearance of or reference to the Atomic Energy Commission (Directorate of Regulation) will now mean the Nuclear Regulatory Commission.

2. This PSAR Summary refers to the PSAR as submitted in April 1975. It is not intended that this be updated as the total PSAR evolves, since its function is to give an overview of PSAR content rather than detailed information.

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FOREWORD

The Clinch River Breeder Reactor Plant PSAR contains seventeen design oriented chapters in accordance with the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LMFBR Edition (prepared by the Regulatory Staff of the U. S. Atomic Energy Commission issued February 1974). In addition, five appendices accompany the seventeen design chapters to support various Project positions and provide insight to selected fallback positions.

This Summary Volume of the Clinch River Breeder Reactor Plant PSAR is a condensed guide to the content of the entire CRBRP PSAR. The intent of this volume is to provide an overview of the PSAR and direction as to where within the PSAR specific topics can be found.

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CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

The Clinch River Breeder Reactor Plant (CRBRP) will provide a vital step in the United States Reactor Development Program. The objective of the U.S. Energy Research and Development Administration (ERDA) Liquid Metal Fast Breeder Reactor (LMFBR) Program is to develop, on a broad, proven technological and engineering base, with joint utility and industry participation, a commercial breeder reactor industry.

In Chapter 1 of the CRBRP 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 designated construction and identifies key items of research and development work necessary.

Section 1.1 of the PSAR provides the objective of the LMFBR Program, general background information on the CRBRP, the basis for the application, the design safety approach and a brief description of the Reliability Program. Because of the central significance of the Reliability Program to this application, major portions of Section 1.1 and C.1 of the PSAR are included as Addenda A and B of this Summary. An introduction to the Reference Design, the rationale leading to a Parallel Design and the objectives and activities designed to support the eventual cessation of the parallel design are also presented in Section 1.1. Section 1.1 concludes with an assessment of the applicability of Regulatory Guides.

Section 1.2 is comprised of a general overview of the plant design including a list of the major operational parameters, a brief description of the site and principal plant systems including a compendium of general arrangement drawings of all the major structures.

Section 1.3 presents in tabular form a comparison of selected safety features of the CRBRP and those of other large fast reactors throughout the world. This Section also tabulates in detail the principal similarities and differences between the 975 MW thermal CRBRP and the 400 MW thermal Fast Flux Test Facility.

Section 1.4 details the various organizations participating in the Project (Energy Research and Development Administration, Project Management Corporation, Tennessee Valley Authority, Westinghouse Advanced Reactors Division, General Electric, Atomics International and Burns and Roe) and their inter-relationships.

As a first-of-a-kind plant, it is to be expected that there is a significant quantity of technical information which has yet to be established. Section 1.5 itemizes the safety related research and development programs. These programs are designed to obtain the necessary technical information required to give assurance of the capability of the safety features or components to perform as intended. Table 1.5-1

(reproduced below) from the PSAR, provides a listing of those areas requiring further technical information and the sections in the PSAR where a discussion on these topics can be found. For each of these programs a criterion of success is identified and potential fallback options discussed. The fallback options are presented in the event the program produces an unexpected result. Each program description contains a schedule, with milestones, indicating that the majority of the work will be completed before issuance of a Construction Permit and all of the programs will be completed in advance of issuance of an Operating License.

Chapter 1 also contains a Section 1-A which is a compendium of flow diagram symbols to assist the reader.

TABLE 1.5-1

FURTHER TECHNICAL INFORMATION REQUIRED

<u>PSAR Section</u>	<u>Section Heading and Tasks</u>
1.5	<u>Introduction</u>
1.5.1	<u>Information Concerning the Adequacy of a New Design</u>
1.5.1.1	Shutdown Systems Reliability
1.5.1.2	Shutdown Heat Removal Systems and 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	<u>Information Concerning Margin of Conservatism of Proven Design</u>
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	Failed Fuel Assembly Tests for Accident Conditions Duct Wall Behavior Test 217-Rod Instrumented Assembly
1.5.2.3	Reactor Thermal and Hydraulic Tests Large Bundle Partial Blockages Evaluations Inlet Plenum Bubble Dispersion Test Inlet Module Blockage Prevention Test Inlet Plenum Particle Mobility Test
1.5.2.4	Core Restraint System Tests Full-Core Restraint System Test
1.5.2.5	Critical Experiments for Reactivity Coefficients, Control Rod Worth and Fuel Assembly Movement
1.5.2.6	Source Range Flux Monitoring System Tests
1.5.2.7	Ex-Vessel Transfer Machine Heat Removal Tests

CHAPTER 2 SITE CHARACTERISTICS

This chapter of the SAR provides information on the geological, seismological, hydrological and meteorological characteristics of the site and vicinity, in conjunction with the site geography, demography and land use in the site vicinity. This chapter details the adequacy of the site from a safety viewpoint.

Section 2.1 and 2.2 detail the site geography, and demography and land uses. The geography of the site includes the site exclusion (control) area and the site boundaries as shown on the Figure 2-1. The low population zone (LPZ) is identified as 5.0 miles. The current population distribution within the area and projections for the population distribution through the year 2010 are detailed. Transient population in the area is discussed along with recreational use. Details of schools and hospitals are contained along with dairy use of the land and water supplies within the area. There is no military use of the land within a ten mile area.

Section 2.3 contains information derived from the Oak Ridge X-10 weather station. From the X-10 station data, tables of rainfall, severe weather, temperature, wind, humidity, fog, stability conditions, dilution factors and classifications (including building wake factors and X/Q values) are derived.

Section 2.4 contains a detailed study of the hydrology. This section includes a description of the plant relative to the topography. The site is located adjacent to TVA flood control watersheds. Tabulated in this section is a list of exterior accesses to Category I structures of the site, noting that all accesses to the buildings are located above the maximum flood level of 809 feet. The plant is located such that Surges, Seiches and Tsunamis are non-existent.

Attention is given to the environmental acceptance of the plant's effluents to the surrounding areas. Liquid effluent releases are detailed within this section and the minor consequences of such releases are described. The effect of low water on plant operations as well as the ground-water hydrology are presented. The ultimate heat sink for this plant is the emergency cooling towers. Approximately 100 tables and figures detail the hydrological description of the site.

Section 2.5 covers the geology and seismology characteristics of the site and the results of investigations dating back to early 1972. Studies of the site have been on-going. The study region for the geology and seismology includes an area of 200 mile radius with emphasis on the Valley and Ridge Physiographic Province.

Details of the vibratory ground motion, surface faulting, stability of subsurface materials and slope stability are also given. Some 70 tables and figures are presented to substantiate geologic and seismology site characteristics. It was concluded that in view of all site related considerations, the Clinch River site is suitable for locating the breeder reactor plant.

Supplement 1 of this Chapter contains additional meteorological data from a tower located at the site. The data are summarized and will be analyzed for future application to the site. Initially, the data appears to indicate that the use of the Oak Ridge X-10 Station data is conservative.

Supplement 2 of Chapter 2 provides responses to Nuclear Regulatory Commission questions pertinent to the subject matter contained in the chapter:

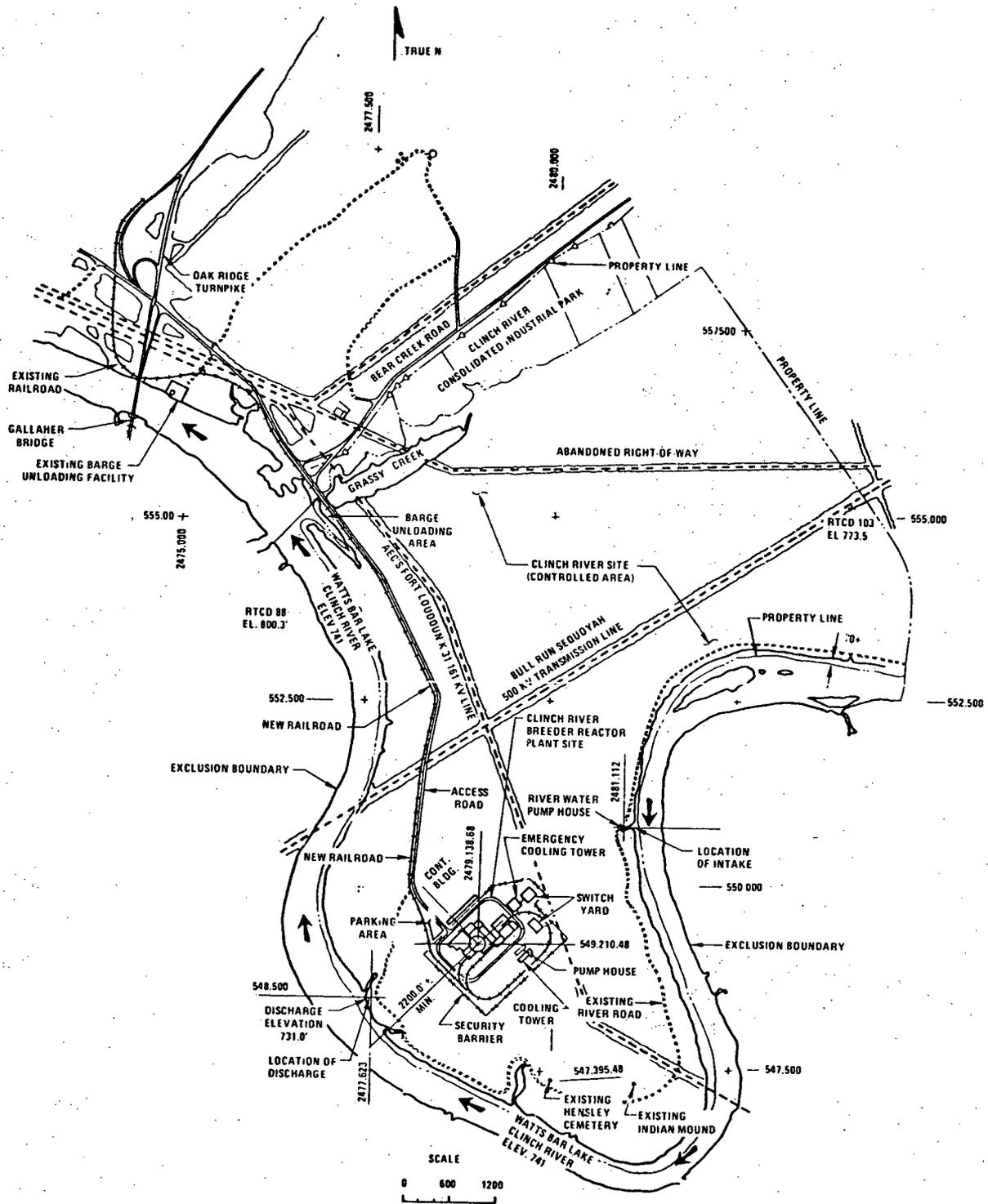


Figure 2-1. Site Location and CRBRP Layout

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CHAPTER 3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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 of the plant design to these criteria is discussed in detail. These criteria are based on the 10CFR50 criteria but are not identical to them, because of design differences between LMFBR's and LWR's.

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 specified. The design basis tornado is defined, in accordance with Regulatory Guide 1.76, as having 360 mph velocity (290 rotational and 70 translational). The radius of maximum rotational wind is specified as 150 ft. and a pressure drop of 3.0 psi at 2.0 psi/sec is specified. Tornado missiles are specified, identical to those used for the Sequoyah nuclear power station located in the same geographical area. The design basis wind is specified 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 design of enclosing structures. The structures themselves will be designed as being capable of withstanding the hydrostatic forces resulting from the flood as well as providing the watertightness as required.

A fairly detailed discussion of missile protection is included. This considers tornado and rotating component 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. Methods of analysis of missile effects (including equations used), and means of protection are discussed.

There is no potential for significant pipe whip for the PHTS or IHTS because of the low pressures in those systems. In the water/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 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 specified 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

CHAPTER 4 REACTOR

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 removable fuel, blanket, and control assemblies, removable 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 structure consists of the core support structure plate and cone, the core barrel, horizontal baffle, fixed radial shielding, and inlet and bypass 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 core support structure includes features to prevent large debris from completely blocking flow to any of the inlet modules.

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 shrouds 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 fueled 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 removable 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.

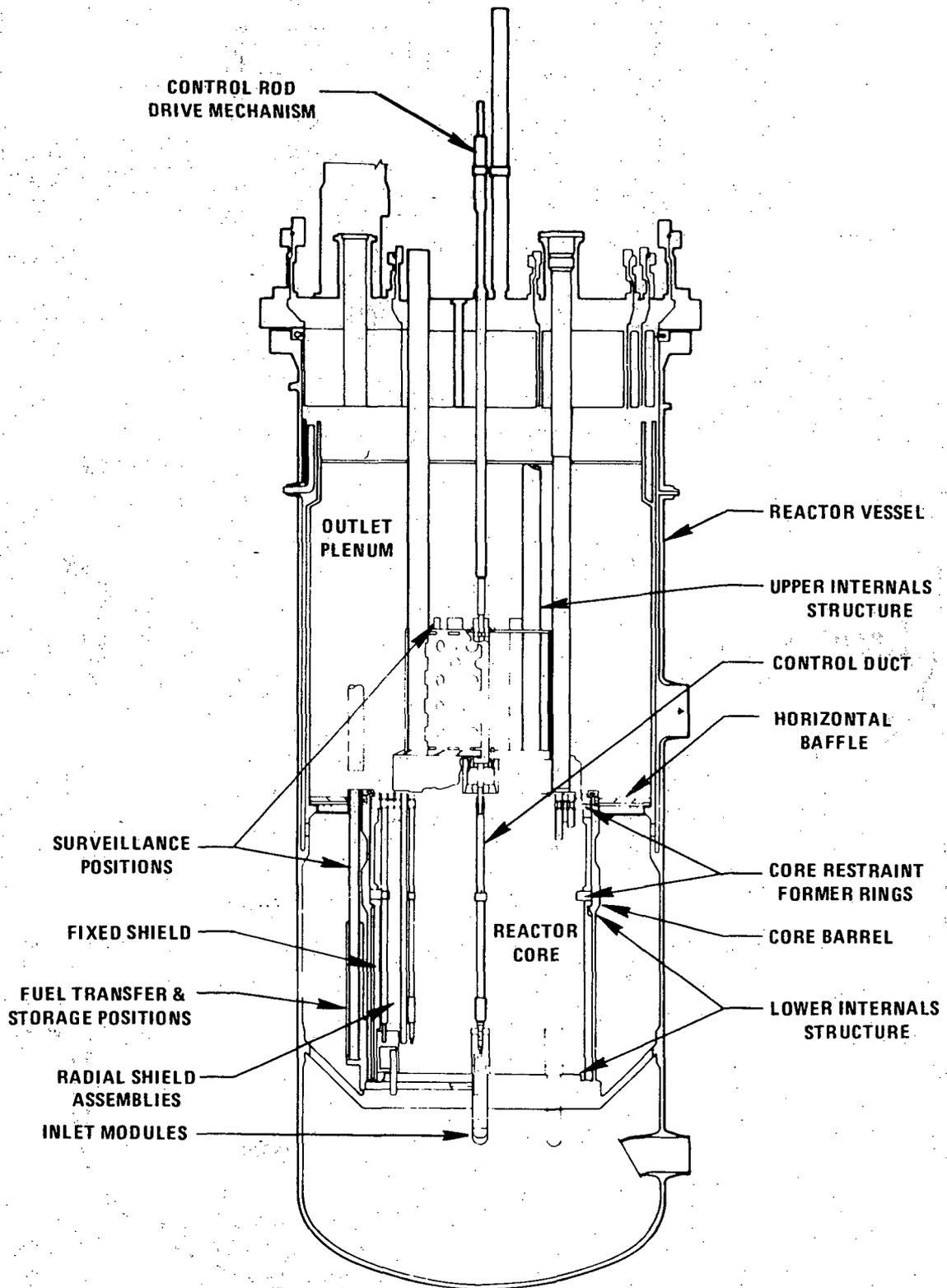


Figure 4-1. Reactor Elevation

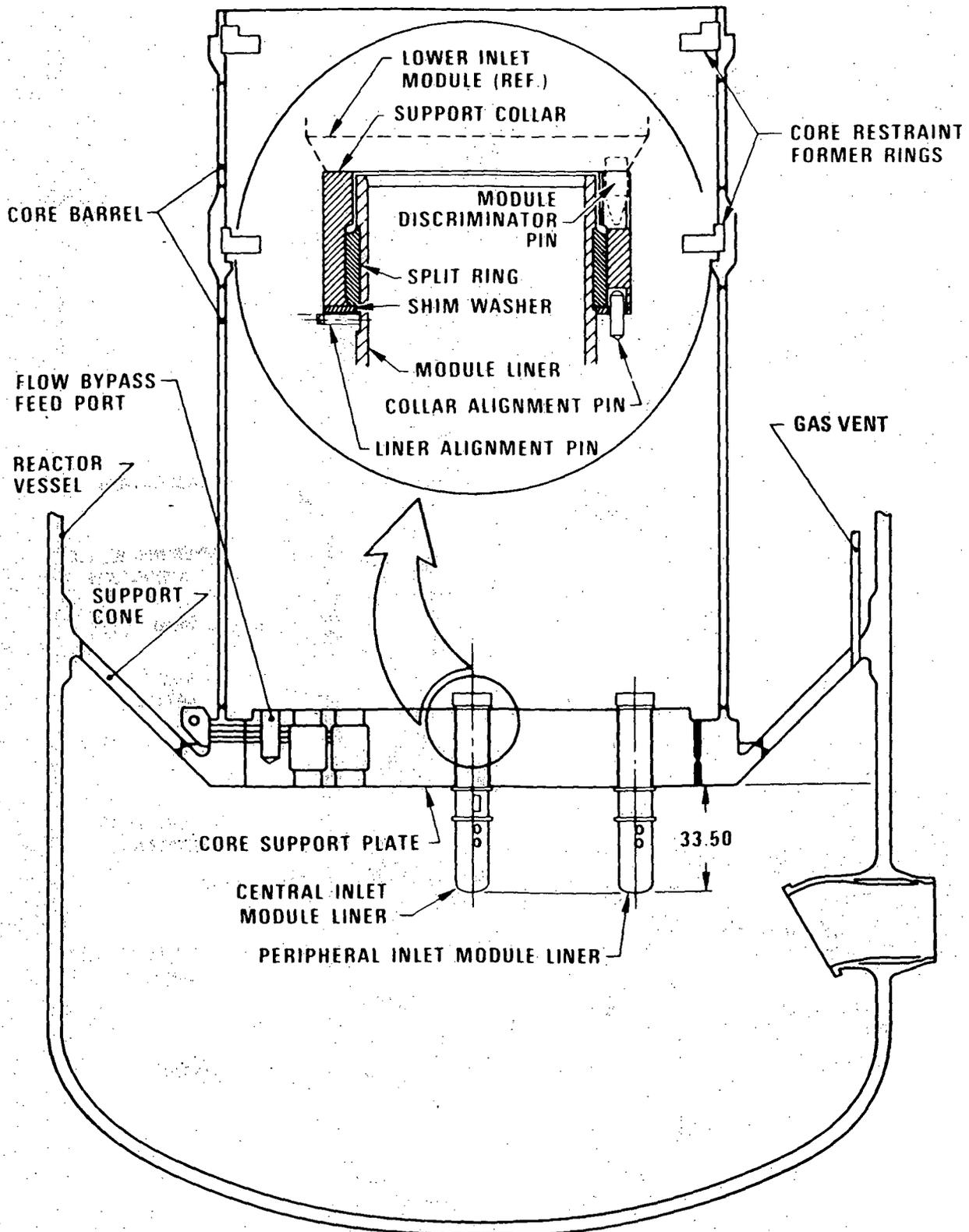


Figure 4-2. Lower Internals Structure

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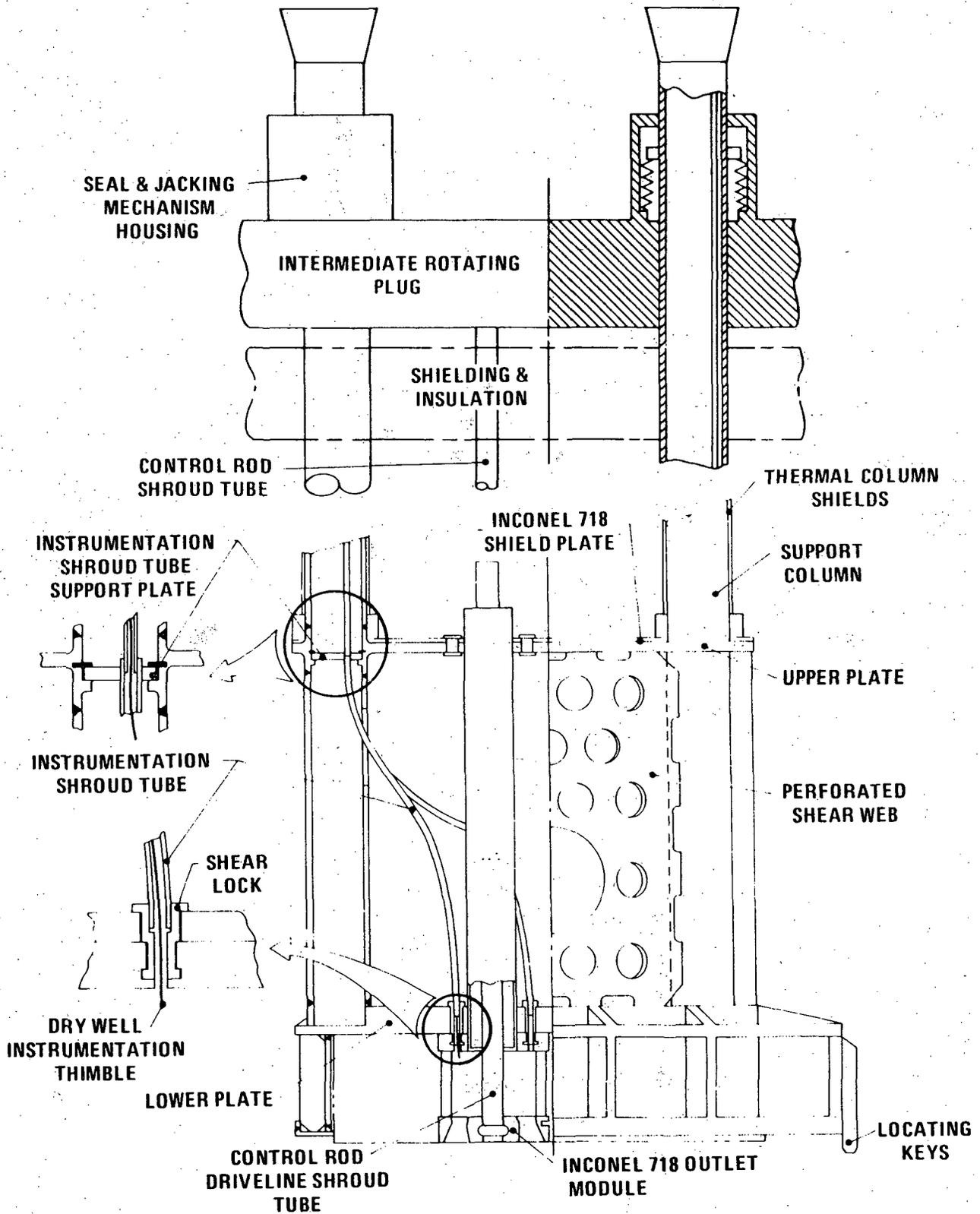
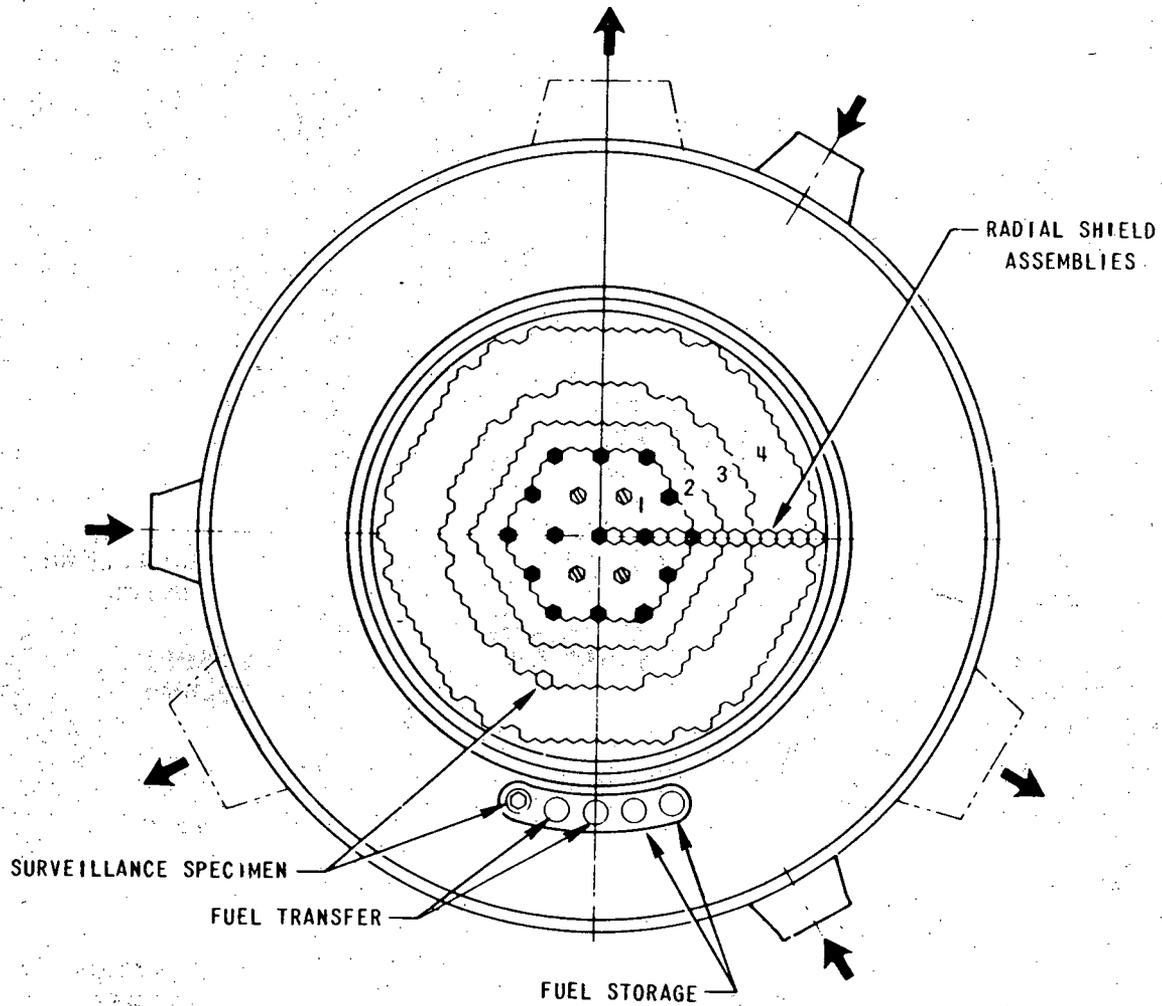


Figure 4-3 Upper Internals Structure

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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

7487-4

CHAPTER 5 HEAT TRANSPORT AND CONNECTED SYSTEMS

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

Section

- 5.2 Reactor Vessels, 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 (SGAHR)
 - Overflow Heat Removal Service (OHRS)

See Figures 5-1, 2 and 3.

The Reactor Vessel, PHTS, IHTS and the intermediate sodium boundary of the SGS will be designed and fabricated according to the ASME Code, Section III, Class I rules. The steam/water side of the sodium dump subsystem of the SGS will be ASME Section III, Class 2 or 3 as appropriate. Code cases 1592-1596 and RDT Standards E15-2 and F9-4 will be used as applicable. The design bases sections of the Chapter present these and other performance, materials, steady state, and transient system requirements.

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 Tube, 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.

Items of Special Interest

Areas of particular interest in Chapter 5 include: mitigation of the consequences of reactor coolant boundary leaks, the PHTS "leak-before-break" assumption, mitigation of the consequences of sodium water reactions and provisions for decay heat removal. A brief description of the PSAR treatment of each of these items follows.

a. Mitigation of the Consequences of Reactor Coolant Boundary Leaks

The role of guard vessels, check valves, low pony motor shut-off 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 of 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.

b. PHTS "Leak-Before Break" Characteristic

The characteristic 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 the low 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-the-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.

c. 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.

Chapter 5 describes the assumptions and analysis techniques used to determine the maximum credible 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 designed to withstand those pressure transients.

d. Decay Heat Removal

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 requirement. 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 objective 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. Natural circulation is induced by proper elevation of the PHTS, IHTS and SGS components (see Figure 5-4). 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 with one loop by forced or natural circulation under any postulated PHTS and IHTS operating mode. This includes performance during use of main condenser cooling, venting of steam through relief valves and Protected Air Cooled Condenser (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 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 heat load is assumed between one and 24 hours after reactor shutdown, the event is a faulted plant event.

ELEV. (INCHES)

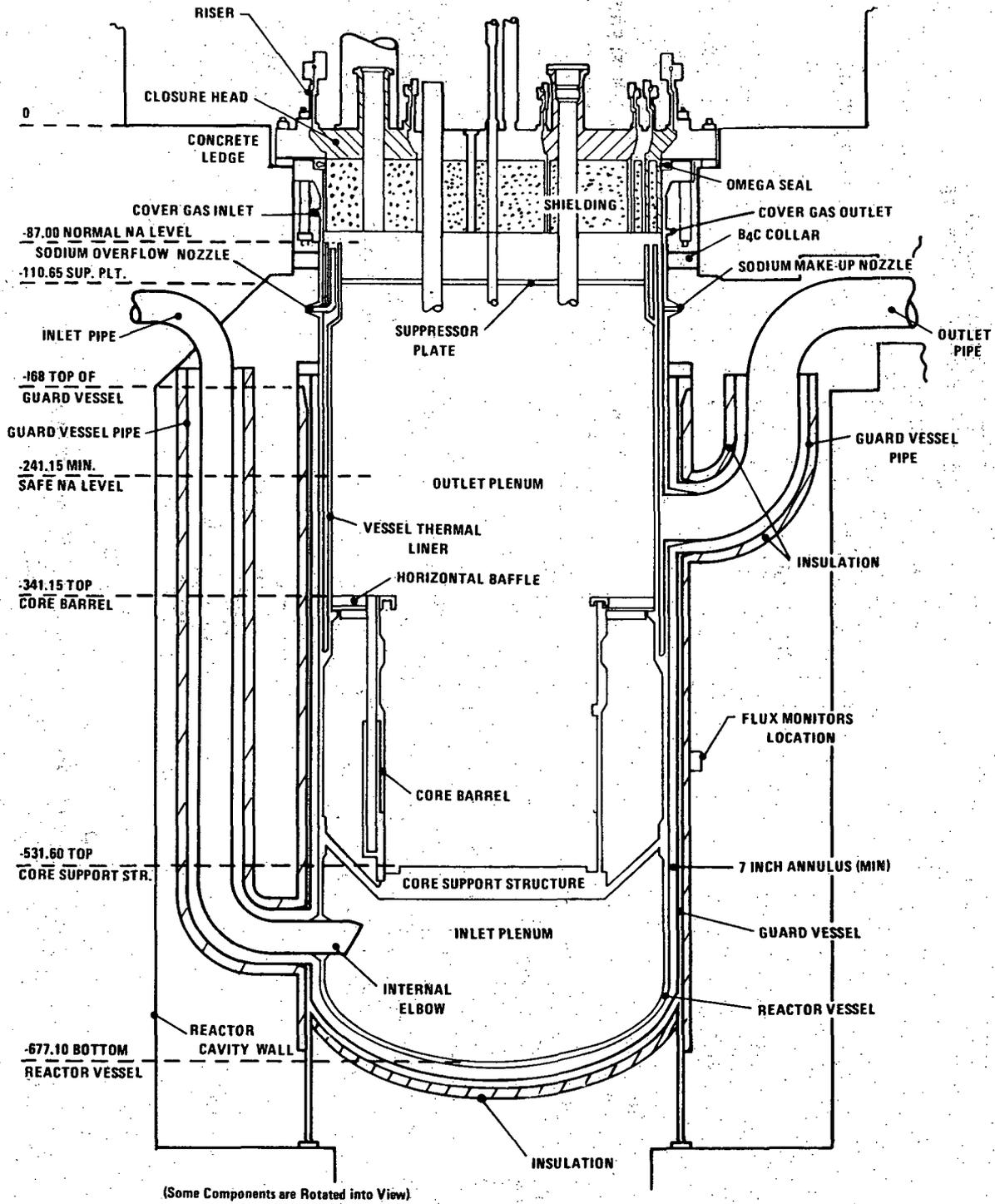


Figure 5-1 Reactor Enclosure

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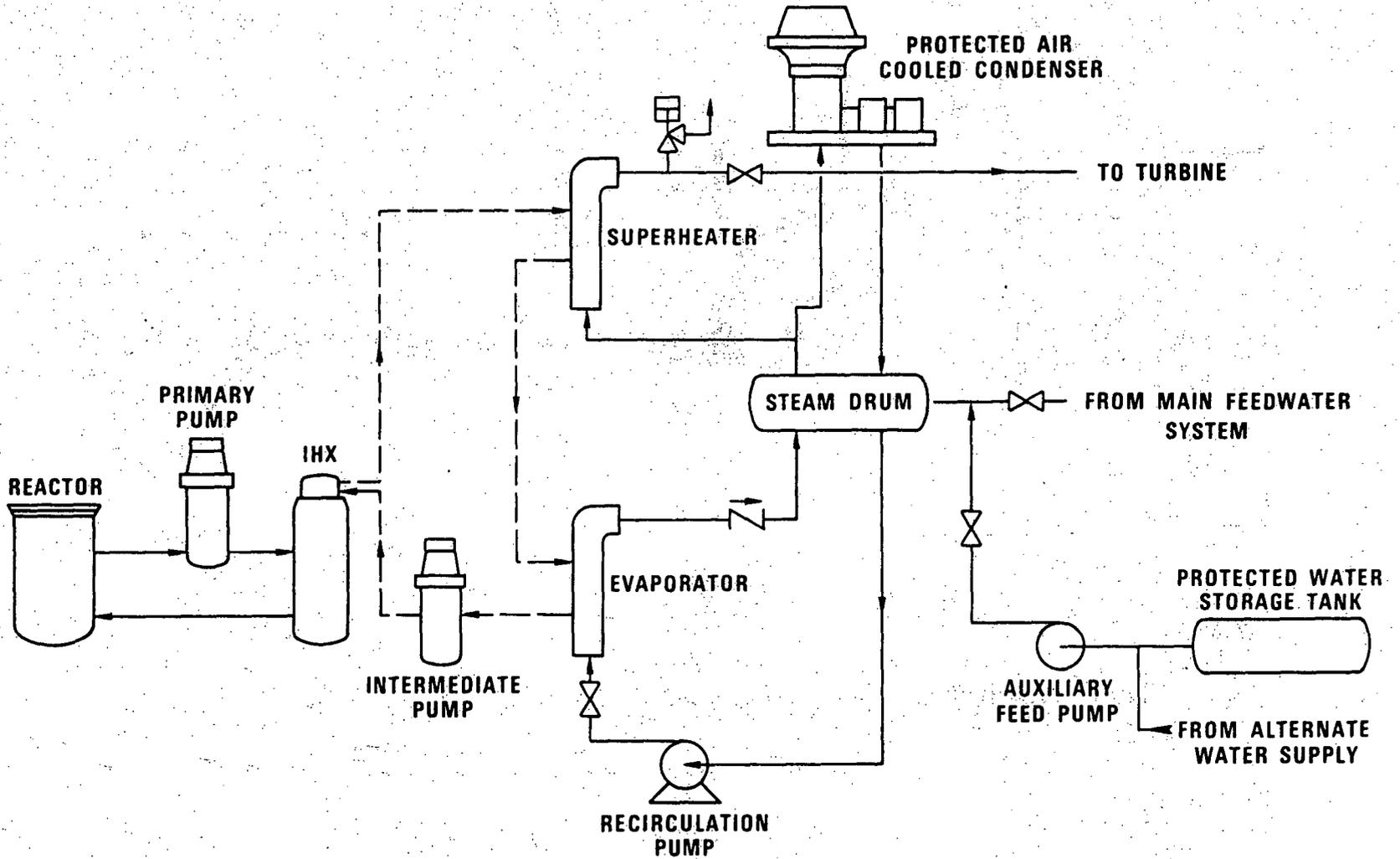


Figure 5-2 Heat Transport Systems Configuration

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SV-21

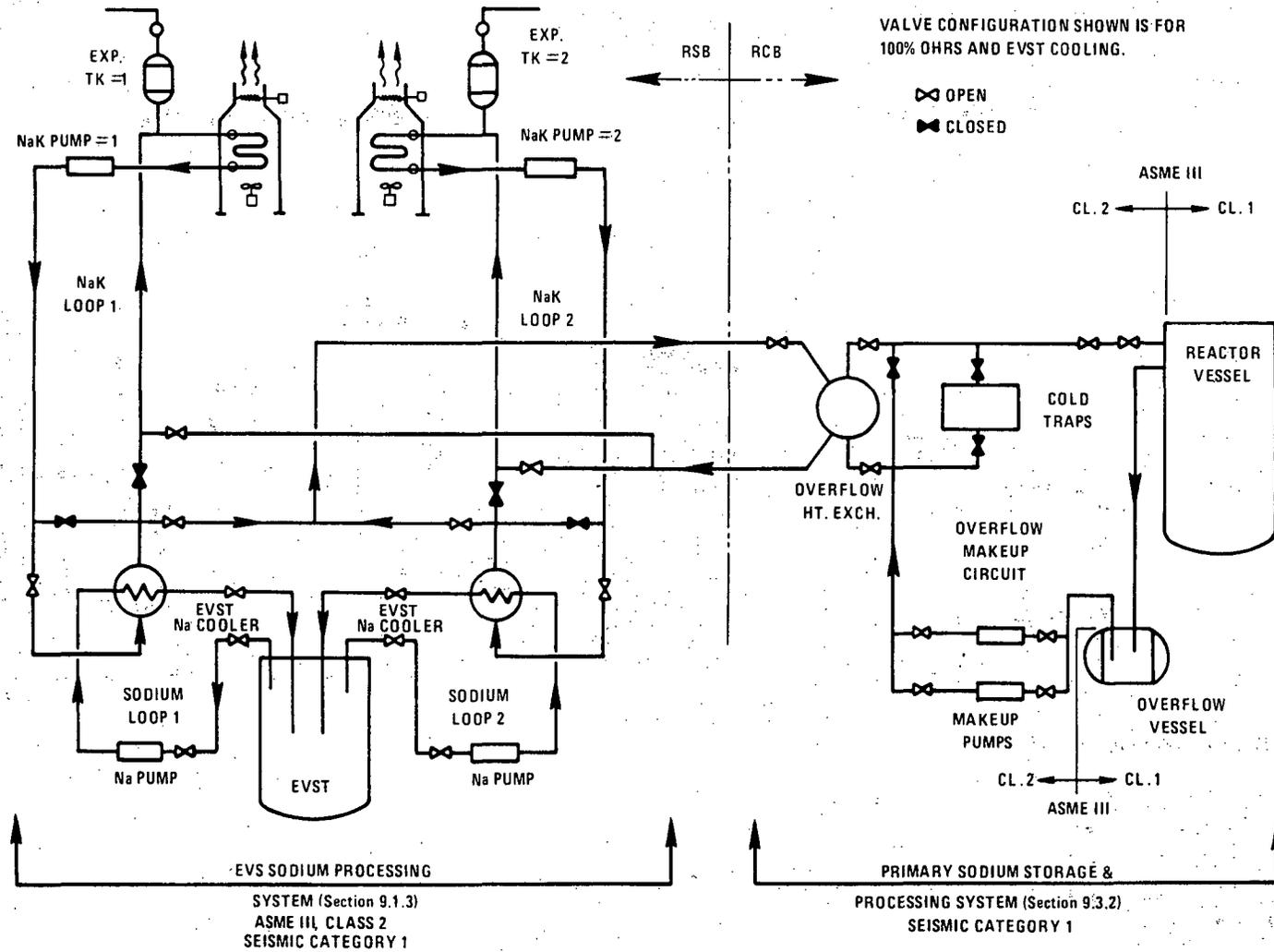


Figure 5-3 OHRs Configuration

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SV-22

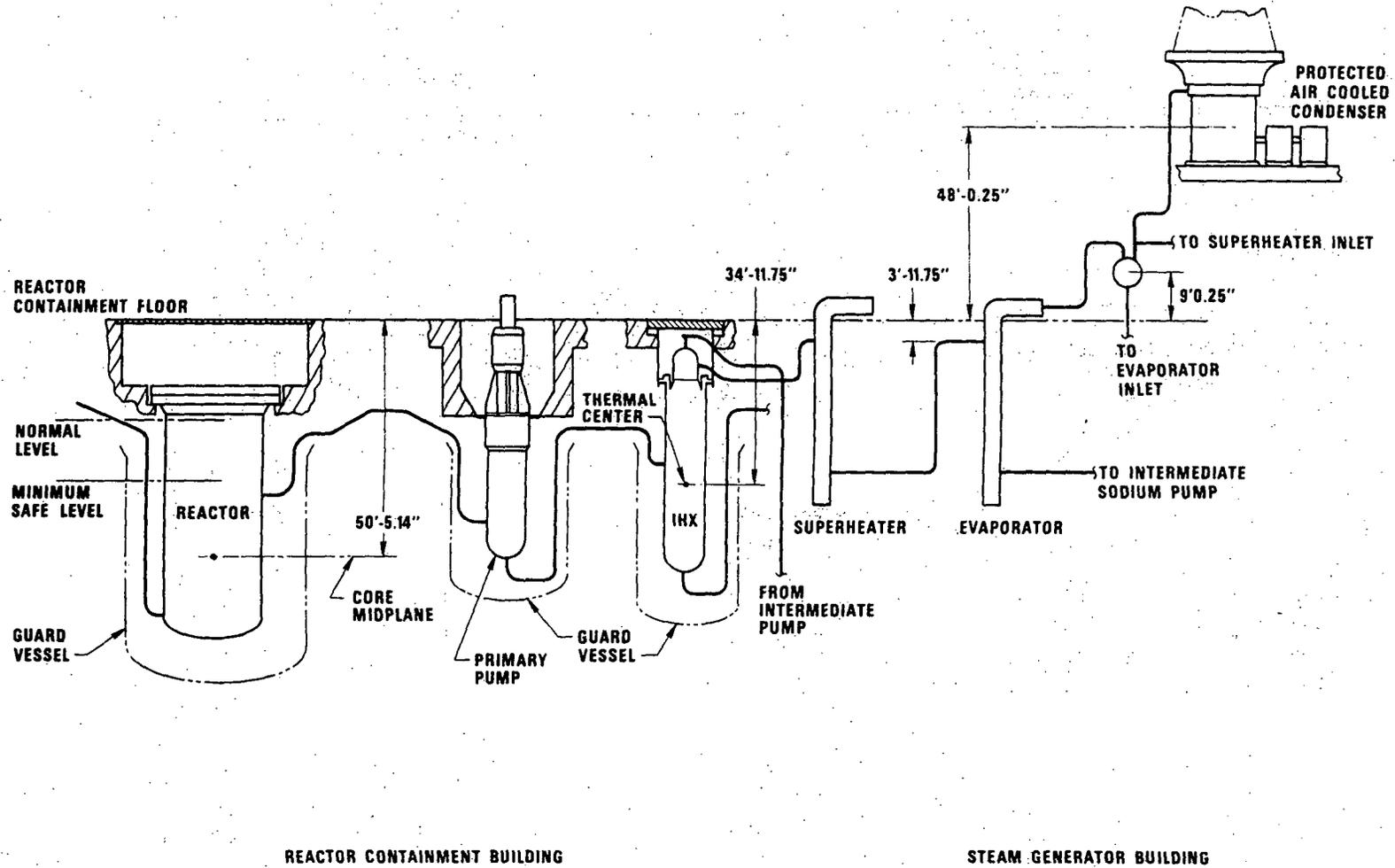


Figure 5-4 Hydraulic Profile

CHAPTER 6 ENGINEERED SAFETY FEATURES

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. Details of calculated site boundary doses are provided and are shown well below the 10CFR100 guideline exposures (See Table 6.2-1 reproduced 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 during plant normal operation 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.

Other Items of Interest

The containment design basis accident is a postulated accident which is extremely unlikely to occur. In addition, very conservative assumptions are used with regard to certain input parameters and the total disregard of protective actions which can be effected. Based upon this extremely conservative accident analysis, it has been established that there will be no need for a post accident containment atmosphere cleanup system, although the latter will be subject to continuing evaluation.

The design of the Containment Isolation System is in full conformance with the CRBR General Design Criteria (GDC). In those cases where the full and detailed design information of the systems involved is yet to be developed, design requirements more stringent than the GDC are used. This conservatism in the present design, if substantiated by later evaluations, may then be removed at a later date.

The design of the Habitability Systems of the Control Room is based upon arbitrary and conservative assumptions over and above the containment design basis accident requirements. Notwithstanding the use of such extremely conservative design bases, the design of the Habitability Systems fully meet the requirements of the NRC's regulations set forth in 10CFR20 and the interim GDC.

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

Organ	Dose (Rem)		
	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

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 (BOP) 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 analysis is included to demonstrate that even gross failure of these systems does not prevent Plant Protection System action. Analysis is 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 without exceeding the specified limits by either system, 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 (1) mechanical and electrical isolation of redundant components are provided, (2) functional or equipment diversity is included in the design of instrumentation and electronic equipment, and (3) the Primary Shutdown System uses a local coincidence configuration while the Secondary Shutdown System uses general coincidence. 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 RSS, 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, electric 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 BCP 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. Sensors 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.

The CRBRP instrumentation systems are important in providing the signal inputs to the Plant Protection System and as safety related display instrumentation. Major emphasis in Chapter 7 is placed on discussions and analyses of the following 6 instrumentation systems.

- (1) the flux Monitoring System which provides neutron level instrumentation for shutdown, startup and full power operation.
- (2) the Heat Transport Instrumentation System which provides pressure, temperature, flow, and other instrumentation in the Primary Heat Transport Loop, Intermediate Heat Transport Loop and Steam Generator.
- (3) the Reactor and Vessel Instrumentation System which includes in-vessel temperature, sodium level and vibration instrumentation.

- (4) the Fuel Failure Monitoring System which provides equipment used for the detection and location of potential fuel cladding failures.
- (5) the Leak Detection System which includes instrumentation used to detect and identify the location of sodium to gas leaks.
- (6) the Sodium-Water Reaction Pressure Relief Instrumentation System which detects the inception of a large sodium-to-water leak in any steam generator module.

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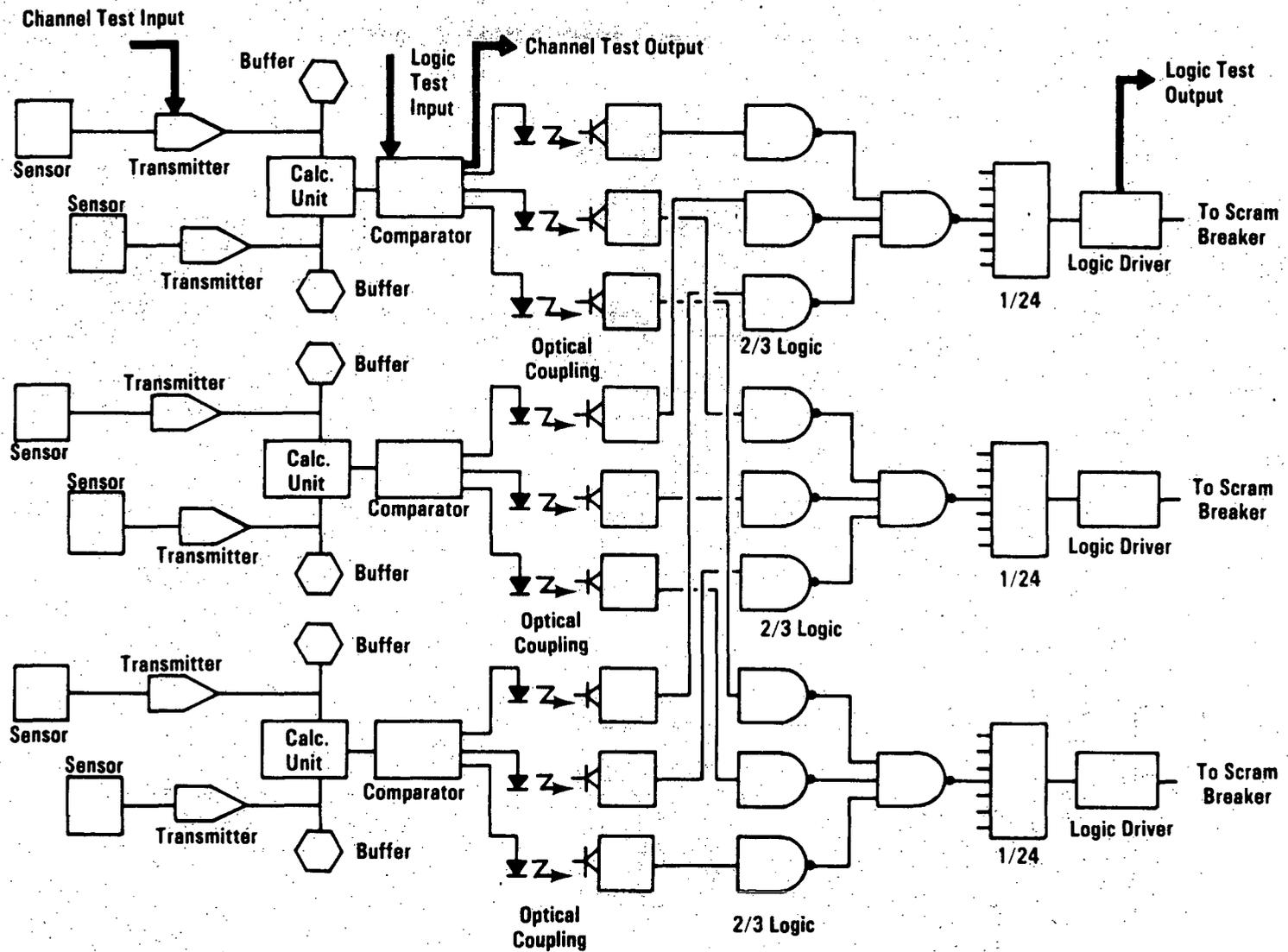


Figure 7-1 Typical Primary Reactor Shutdown System

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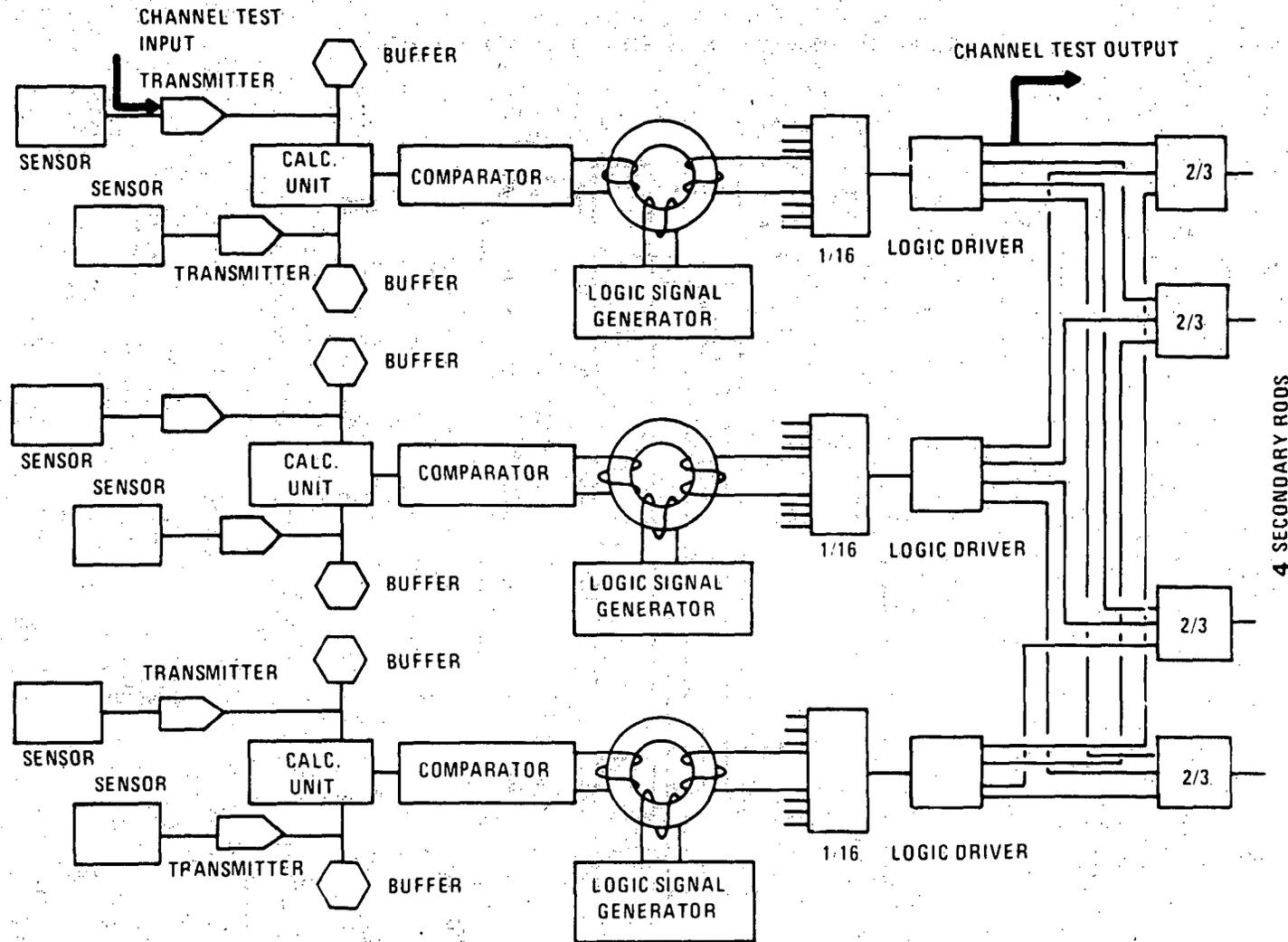


Figure 7-2 Typical Secondary Reactor Shutdown System

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SV-31

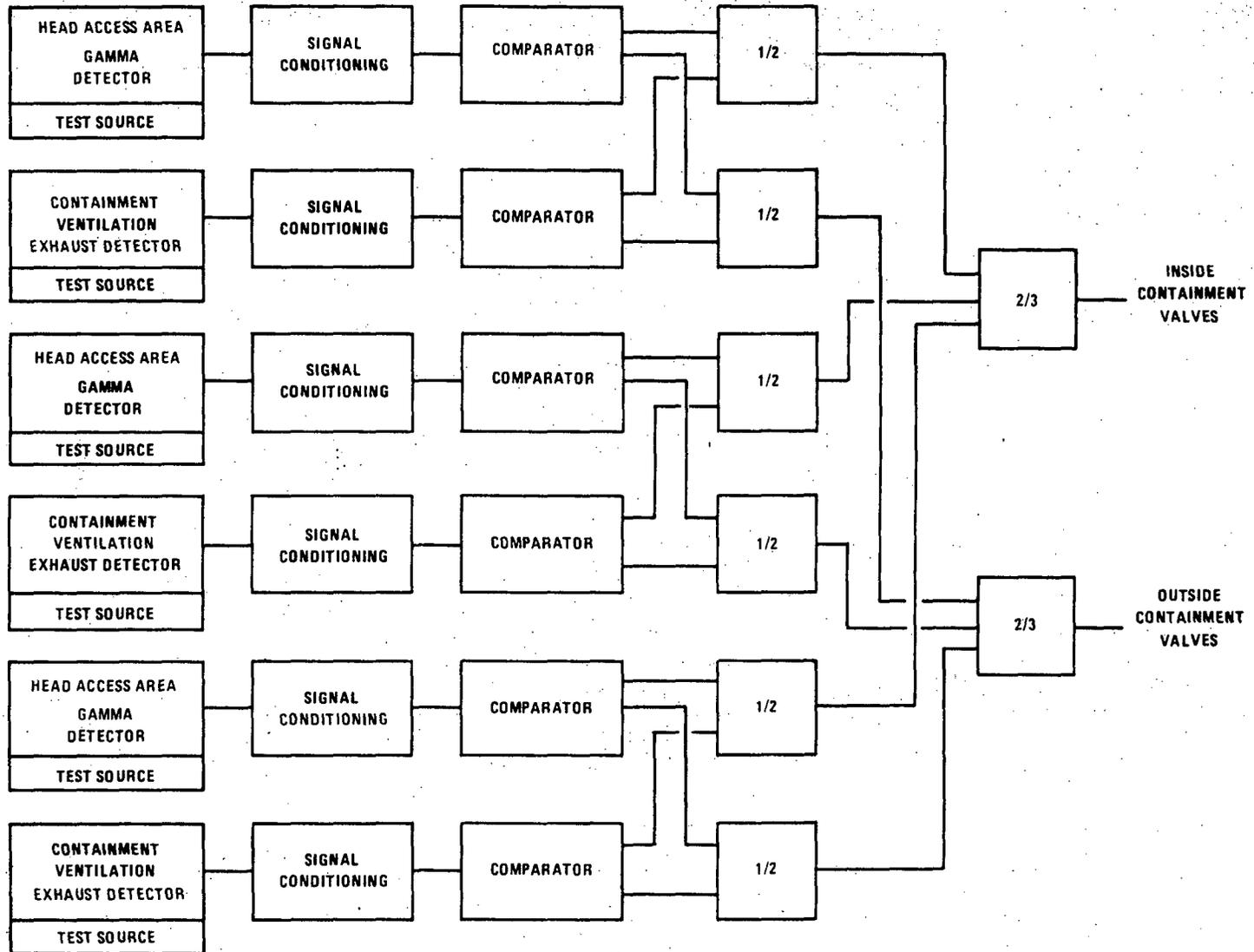


Figure 7-3 Containment Isolation System Block Diagram

CHAPTER 8 ELECTRICAL SYSTEMS

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

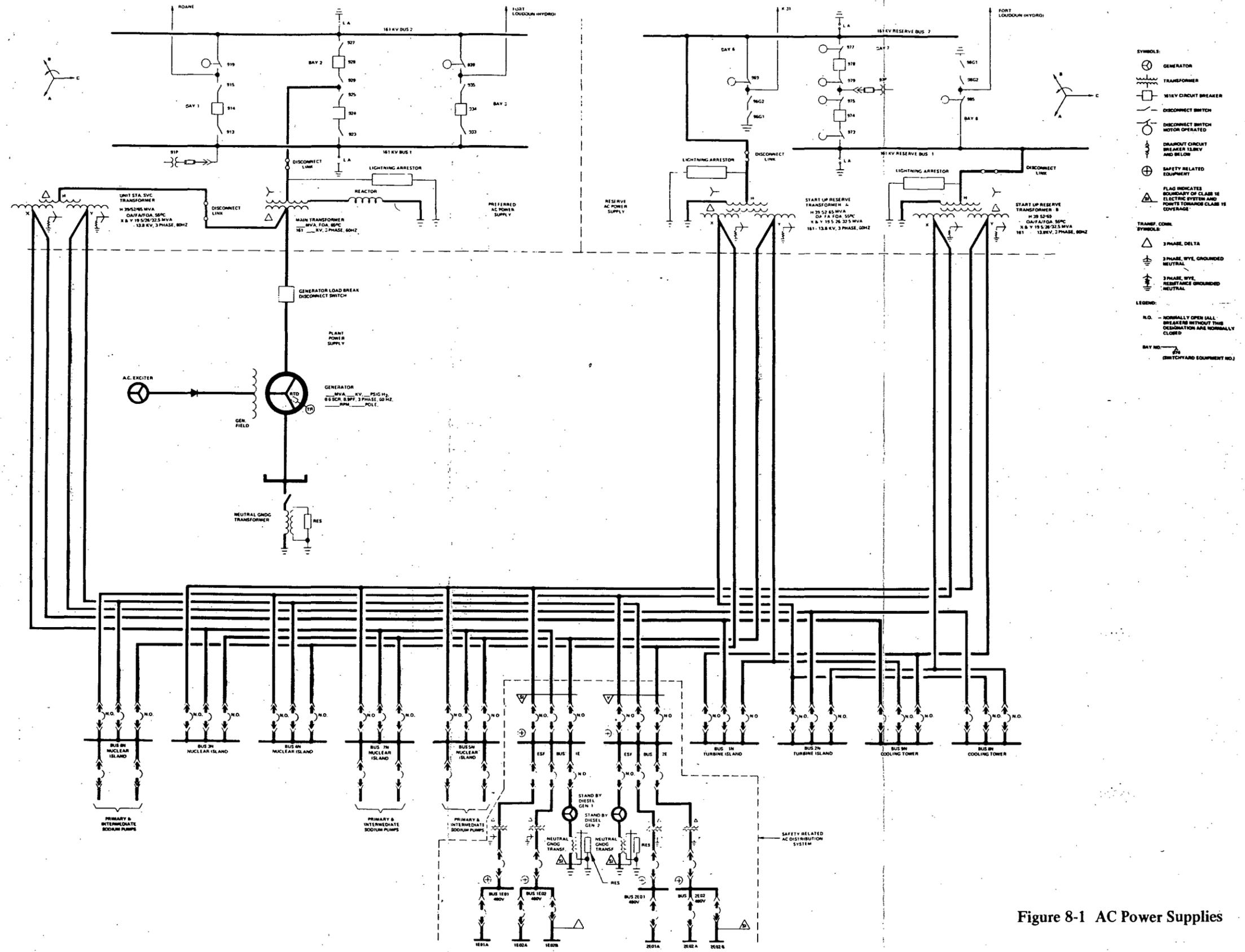
- a. The transmission lines and switchyards connecting the CRBRP to the TVA grid
- b. The normal AC distribution system
- c. Emergency AC and DC power supplies
- d. The safety related AC and DC distribution systems
- e. The safety related and non-safety related loads supplied by the emergency power supplies

See Figure 8-1 and 2.

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.

The Emergency AC and DC Power Supplies meet the AEC and IEEE requirements for separation and redundancy. The Emergency diesel generators are sized to support safe shutdown of the plant indefinitely and the DC power supplies are sized to start and supply all of their loads for two hours.

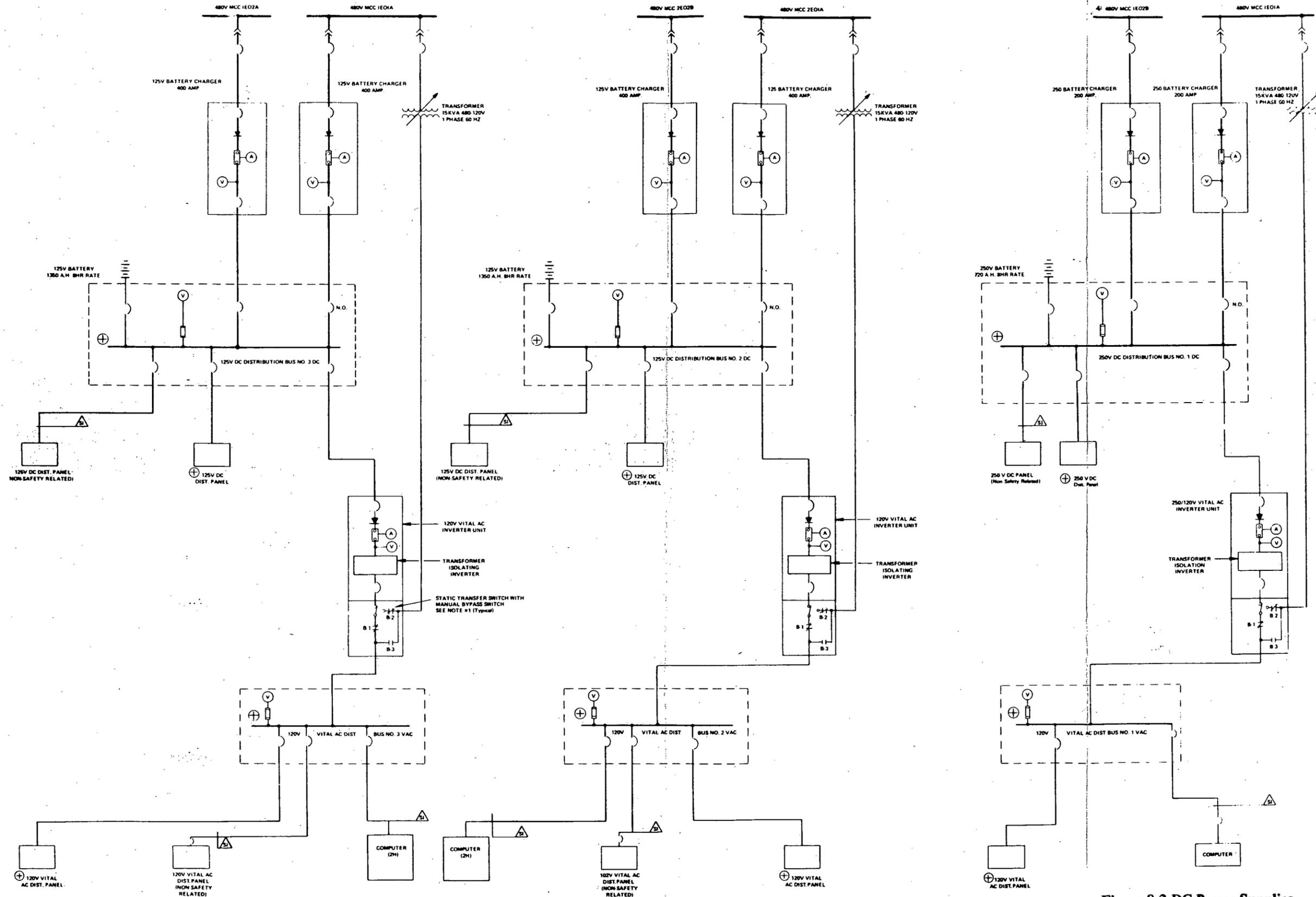
Chapter 8 includes a list of Class 1-E (safety related) and non-Class 1-E loads supplied by the Emergency Power Supplies, the power requirement of each and the sequence in which it will be automatically or manually connected to the diesel generators in the event of a loss of all offsite power supplies.



- SYMBOLS:**
- GENERATOR
 - TRANSFORMER
 - 181KV CIRCUIT BREAKER
 - DISCONNECT SWITCH
 - DISCONNECT SWITCH MOTOR OPERATED
 - DRAWOUT CIRCUIT BREAKER 181KV AND BELOW
 - SAFETY RELATED EQUIPMENT
 - FLAG INDICATES BOUNDARY OF CLASS 1E ELECTRIC SYSTEM AND POINTS TOWARDS CLASS 1E COVERAGE
- TRANSFORMER COVERS:**
- 3 PHASE DELTA
 - 3 PHASE WYE GROUNDING NEUTRAL
 - 3 PHASE WYE RESISTANCE GROUNDING NEUTRAL
- LEGEND:**
- N.O. - NORMALLY OPEN (ALL BREAKERS WITHOUT THIS DESIGNATION ARE NORMALLY CLOSED)
 - BAY NO. (SWITCHGEAR EQUIPMENT NO.)

Figure 8-1 AC Power Supplies

7483-1



GENERAL NOTES

- CONTACTS B 1, B 2, B 3 ARE MADE BEFORE BREAK CONTACTS OF THE MANUAL BY PASS SWITCH AND OPERATE AS FOLLOWS:
 - A) CONTACTS B 1 & B 2 ARE NORMALLY CLOSED
 - B) CONTACTS B 3 IS CLOSED AND CONTACTS B 1, B 2 ARE OPEN DURING TEST
- ALL BREAKERS ARE 2 POLE DEVICES UNLESS OTHERWISE INDICATED

SYMBOLS

- ⊕ SAFETY RELATED EQUIPMENT
- ⎓ CIRCUIT BREAKER
- ⎓ TRANSFORMER
- ⎓ MOTOR STARTER WITH OVERLOAD PROTECTION
- N.O. NORMALLY OPEN
- ⚠ FLAG INDICATES BOUNDARY OF CLASS I E ELECTRIC SYSTEM AND POINTS TOWARDS CLASS I E COVERAGE
- ⎓ DRAWOUT

Figure 8-2 DC Power Supplies
7483-2

CHAPTER 9 AUXILIARY SYSTEMS

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 of new and spent fuel assemblies is provided. Each sequential operation and the equipment or cells involved in those operations are described in detail.

The fuel handling sequence (see Figure 9-1) begins with receipt of new fuel at the site. Each assembly is inspected, heated, acclimated to a liquid sodium environment and stored in the Ex-Vessel Storage Tank (EVST), a large, two-tier sodium filled tank located in the Reactor Service Building (RSB). After shutdown for refueling, the reactor will be cooled to about 400°F and the Reactor Containment Building (RCB) hatch connecting the two buildings, is opened. The in-vessel transfer machine (IVTM) and ex-vessel transfer machine (EVTM) operate in conjunction to remove spent fuel from the reactor to the EVST and transfer new fuel from the EVST to the reactor on a one assembly at a time basis.

Spent fuel remains in the EVST storage for at least 100 days before being loaded into spent fuel shipping casks.

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.
3. Auxiliary Liquid Metal System - The auxiliary liquid metal systems provide for receipt, storage and purification of liquid metal used in the plant. The system also provides the capability for reactor sodium level control, accommodates primary sodium volumetric changes, and provides cooling for core components stored in the EVST.

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' operation. The overflow heat exchanger (OHX) provides such cooling capability. 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 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 heat exchanger. This method of decay heat removal is termed the Overflow Heat Removal Service (OHRS) (see Figure 5-2).

4. Piping and Equipment Electrical Heating - Discusses the design of electrical heaters, mountings and power controllers to heat sodium containing systems.
5. Inert Gas Receiving and Processing - The IGRP system supplies inert gases (Ar and N₂) and vacuum for plant systems. The system also supplies sodium cover gas, cell-inerting atmospheres, valve actuating gas in inerted cells, cooling gas, gas for seals, gas for fire control blanketing, gas for component cleaning and the vacuum subsystem.
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 - The Auxiliary Coolant Fluid System provides a means of removing waste heat from the Reactor Containment Building (RCB) and the Reactor Services Building (RSB). Dowtherm J is the cooling medium with the ultimate heat sink being provided by the treated water system.
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. Instrumentation requirements for each system are provided.
9. Compressed Air System - Various subsystems furnish instrument service and breathing air for the plant. Discussion is centered on the system design, operational testing and instrumentation requirements.
10. Communications: Lighting - Normal and Emergency systems are provided to support operation or shutdown of the Plant. The communications system includes provision for off-site communications.
11. Plant Fire Protection System - Means are supplied to fight conventional and sodium fires. The discussions include consideration of fire system arrangements throughout the plant.

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.

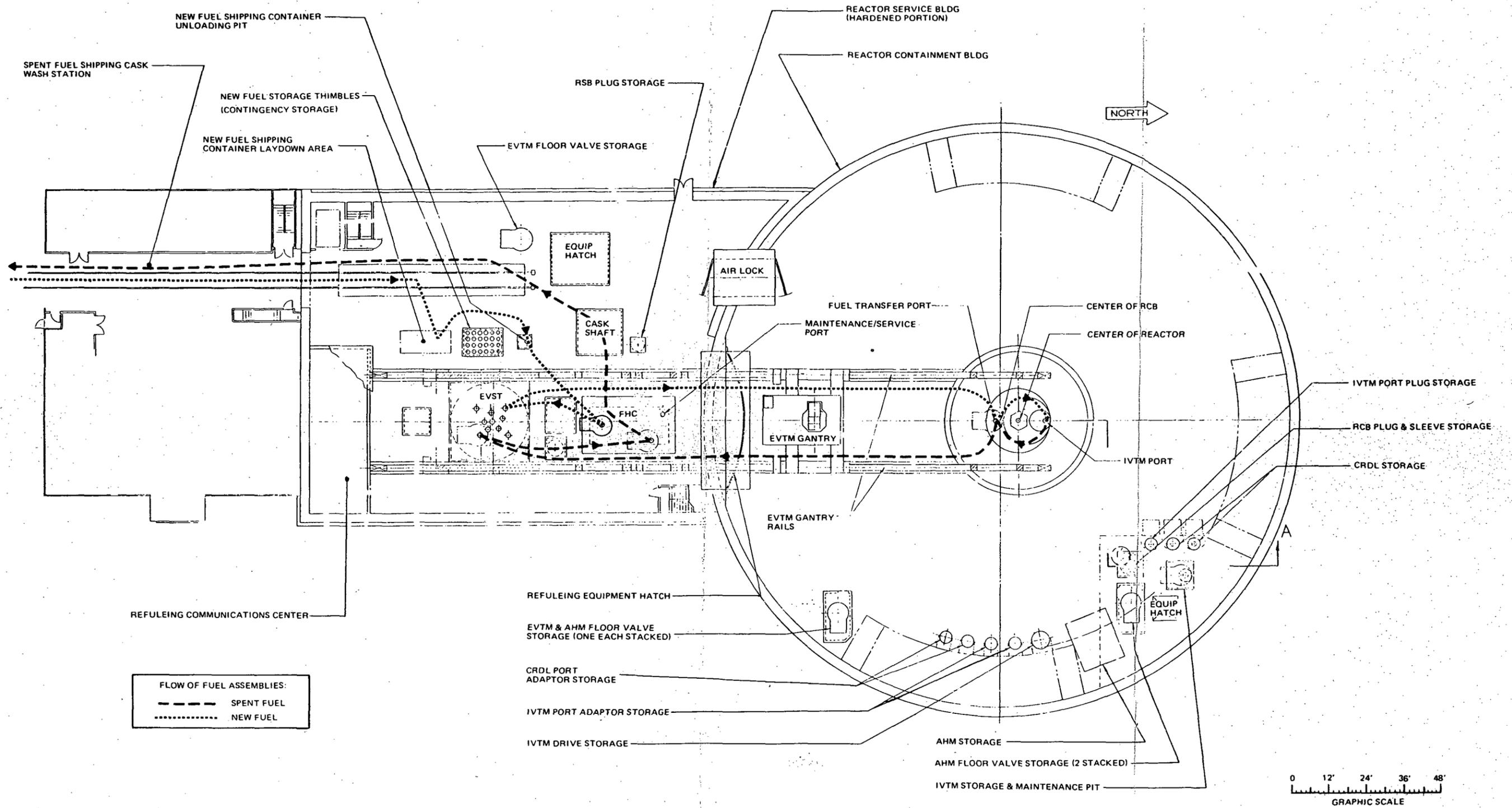


Figure 9-1. Refueling Operations

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CHAPTER 10 STEAM AND POWER CONVERSION SYSTEMS

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

See Figure 10-1.

The majority of the main steam supply and related systems are contained in the Turbine Generator Building which houses no safety related equipment. Thus a steam or feed line break there will not endanger such equipment. The requirements of ANSI B31.1 will be met in the design of that equipment. The portion of the Main Steam Supply and Feedwater Systems which are contained in the Steam Generator Building will be designed according to the appropriate ASME, Section III requirements and measures discussed in Chapter 3.0 will be used where necessary to protect safety related equipment from the effects of postulate pipe breaks.

Turbine-generator missile data has not yet been developed for the 3,600 rpm unit planned for the CRBRP. However, the potential missiles from this unit are comparable in energy, distribution and probability to the postulated missiles from 1,800 rpm units which have been shown acceptable and are in use at light water reactor plants.

The various locations where excessive concentrations of tritium can be postulated to occur are continuously or periodically monitored to ensure that releases from the systems do not exceed the appropriate limits.

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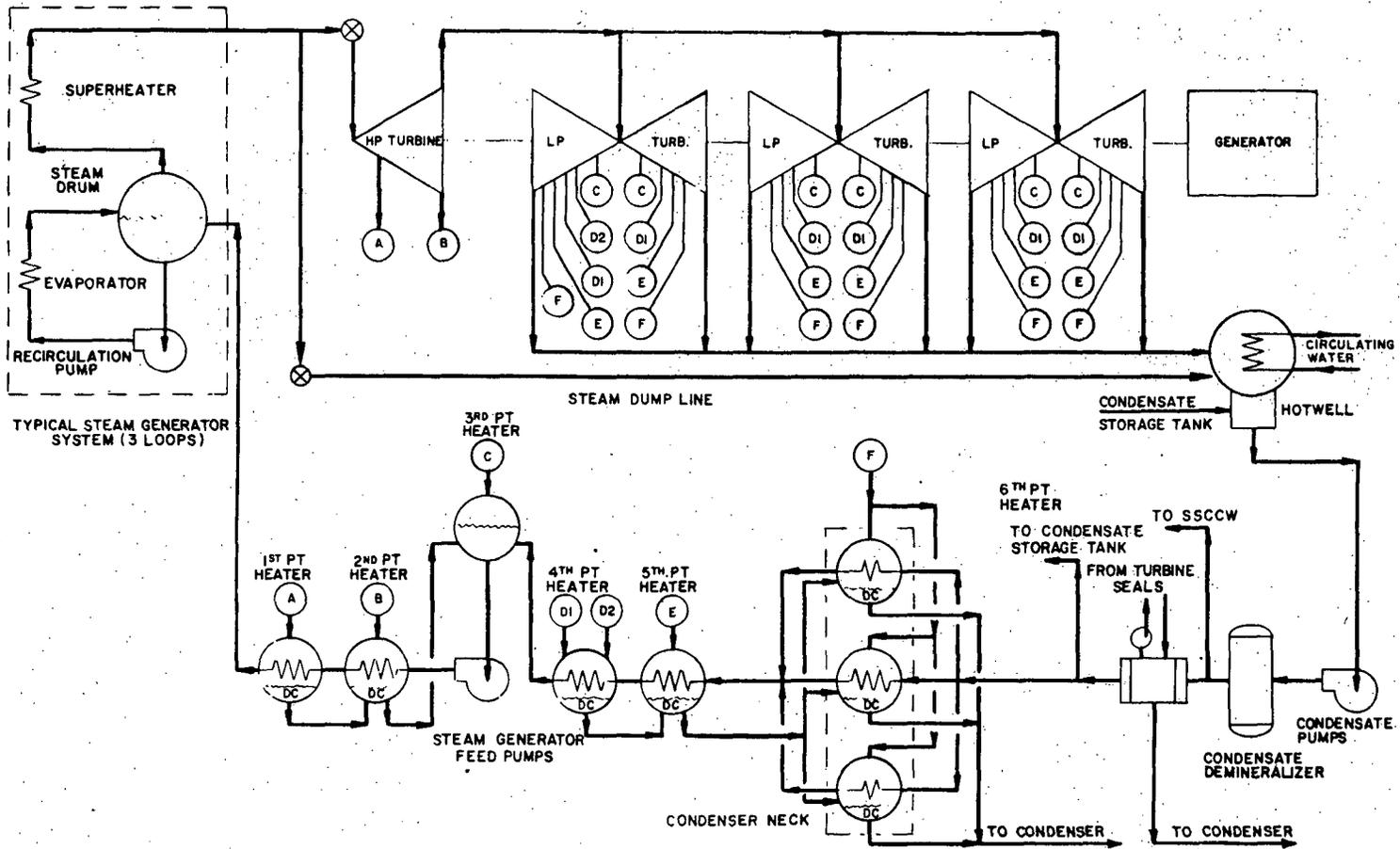


Figure 10-1 Steam And Power Conversion System

CHAPTER 11 RADIOACTIVE WASTE MANAGEMENT

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. 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, which is to 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 calculation. The sampling techniques, locations and frequency of sampling for the program are provided.

CHAPTER 12 RADIATION PROTECTION

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 bases 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.

The overall shield design objectives will perform a variety of functions under normal operating conditions. These functions include (a) permitting personnel access to required portions of the plant, (b) permitting refueling of the reactor, (c) permitting access 12 days after reactor shutdown to the high radiation portions of the restricted area, which will be maintained as exclusive areas during normal operation, (d) limiting neutron activation of intermediate sodium, such that the induced radiation dose rates will not require the establishment of a restricted area in the intermediate heat transport system areas, (e) maintaining all areas of the site outside of the reactor containment building, reactor service building and intermediate sodium piping penetration cells at the intermediate/reactor containment building interface as a continuous access area during normal operation, and (f), protecting structural components, equipment and nuclear instruments in order that required functions are safely provided throughout the lifetime of the plant.

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.

Selected plant locations provided with area monitors are discussed. The monitors are provided to continuously detect, measure, and indicate the radiation level and to initiate alarms for radiation levels above preset values. Locations, design dose rates, and ranges of sensitivities of the monitors are provided.

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.

A listing of plant monitoring for the CRBRP is presented. Fixed airborne radioactivity monitors will be provided in selected locations throughout the CRBRP design gaseous effluent release points to continuously detect, measure, indicate, and record airborne radioactivity. Mobile, continuous air monitors, will be provided to perform similar functions in areas not directly served by the fixed continuous air monitors, or when a check of radionuclide concentrations determined by the fixed air monitoring channel is desired. Sampling capabilities are also discussed and listed.

Health Physics Program objectives are stated, and facilities and equipment are discussed.

The health physics program and staff applies applicable radiation standards and procedures, reviews proposed methods of plant operation, participates in development of plant documents, and assists in the plant training program, providing specialized training in radiation protection. During preoperational tests and after plant startup, it provides health physics coverage for all operations including maintenance, fuel handling, waste disposal, and decontamination. It is responsible for personnel and inplant radiation monitoring, and maintains continuing records of personnel exposures, plant radiation and contamination levels. Through implementation of the program, plant personnel exposure will be maintained as low as practicable.

CHAPTER 13 CONDUCT OF OPERATIONS

This chapter describes the framework within which all phases of 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 organizations and various positions in the plant along with the required qualifications for the key positions. Figure 13.1-1 shows the CRBRP Organization Chart and includes expected staffing levels.

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.

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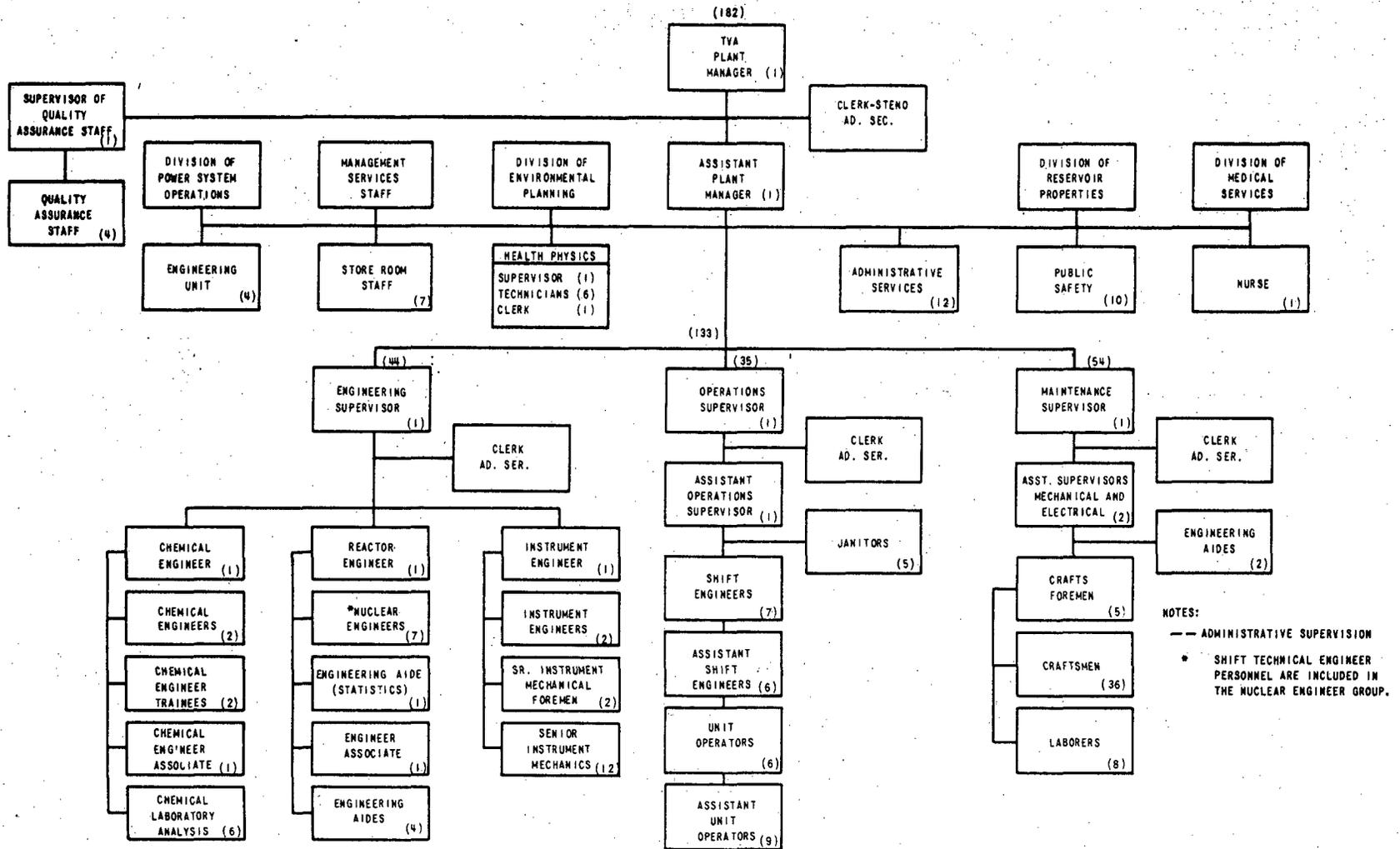


Figure 13-1 CRBRP Organization Chart

CHAPTER 14 INITIAL TESTS AND OPERATION

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, PMC has been assigned the responsibility for detailed planning, scheduling, coordination and conducting of plant testing with the assistance of the ARD technical staff. PMC 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. Acceptance testing has been divided into four distinct phases and four categories. The four phases include:

- Phase 1 - Pre-Operational Tests
- Phase 2 - System Operational Tests
- Phase 3 - Nuclear Startup Tests
- Phase 4 - Power Ascension Tests

The four categories have been divided according to effect on the plant and responsible organization for preparation of test procedure and specification as follows:

<u>Category</u>	<u>Effect on Plant</u>	<u>Responsible Organization For Preparation Of Test Specification And Procedure</u>
A	Direct	B&R, AI, GE
B	Direct	ARD, B&R, AI, GE
C	Limited	ARD, B&R, AI, GE
D	None	ARD, B&R, AI, GE

CHAPTER 15 - ACCIDENT ANALYSIS

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The overall Design of the CRBRP is based on a three level of design approach which establishes a defense-in-depth for the health and safety of the general public. The three levels of safety approach encompasses: 1.) The provision of a sound reliable plant, 2) The limitation of any accidental condition to acceptable values within the plant capability and 3) The protection of the public against certain extremely unlikely events by additional plant capabilities. Chapter 15 addresses a broad spectrum of accident events in which the efficacy of the three levels of design is demonstrated. The results of these analyses clearly demonstrates that none of these events result in a site boundary dose in excess of the 10CFR100 Guidelines.

Section 15.1.1 begins with a reiteration of the safety philosophy contained in Section 1.1 and provides an extensive discussion of first and second level design features. A collation of third level design margin requirements and a preliminary assessment of the feasibility of compliance with these requirements. A collection of tables and figures specifying the numerical requirements are also included. 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 or 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 valve.

These are only a small sample of the total listing of requirements, but are indicative of the level of detail provided. After the time of selection of the third level margin requirements which were based on FFTF experience and a measure of engineering judgement, the CRBRP HCDA parametric analysis were developed to a point at which it was considered they could give useful guidance on the acceptability of the third level margin requirements. PSAR Table 15.1.1-1, reproduced here, compares the primary system loadings with those from conservative CRBRP HCDA analyses.

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 would 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.

Section 15.1.3 discusses the plant protection system trip level design events and duty cycles. A table is provided (Table 15.1.3-1, not reproduced here) showing the applicable PPS substem trip levels or trip equations.

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 reactivity insertion or undercooling events presented in these sections.

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. In this section the events are again categorized as Anticipated, Unlikely, or

Extremely Unlikely. A summary of the results of these analyses appears in Table 15.5-1 of the PSAR and reproduced below. The results show that there are no adverse consequences associated with any of the fuel handling events.

Section 15.6 address a broad spectrum of potential sodium fires. The events discussed in this section have such low probability of occurrence that they have all been categorized as extremely unlikely events. A summary of the results of these analyses appears in Table 15.6-1 or the PSAR and reproduced below. As can be seen from the summary table no deleterious consequences are associated with any of the sodium fire events.

Section 15.7 addresses a group of other events that do not appear to fall under any of the preceeding categories. Again in this section the accident events are listed as Anticipated, Unlikely or Extremely Unlikely. A summary of the results of these analyses appears in Table 15.7-1 of the PSAR and reproduced below. As can be seen from these data there are no adverse consequences associated with any of these events.

TABLE 15.1.1-1

COMPARISON OF HCDA PRIMARY SYSTEM LOADINGS

	<u>Reference Case</u>	<u>Structural Evaluation Fuel Vapor Expansion Case</u>
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

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TABLE 15.2-1
REACTIVITY INSERTION DESIGN EVENTS

Section No.	Event	Max. Clad. Temp.* Primary Scram	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

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

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TABLE 15.2-1 Continued

Section No.	Event	Max. Clad. Temp.* Primary Scram	Secondary Scram	Comments
15.2.2.3	Mal-operation of reactor plant controllers	<1510°F	<1610°F	Less than limiting condition shown in 15.2.1.2-1
15.2.3	Extremely Unlikely Events			
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	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)

†Not regarded as credible, used for evaluation purposes only.

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TABLE 15.3-1
UNDERCOOLING EVENTS

Section No.	Event	Max. Clad Temp. *		Comments
		Primary Scram	Secondary Scram	
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. Temperature spike associated with secondary shutdown is considerably 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 module isolation valve	<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

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TABLE 15.3-1 (Continued)

Section No.	Event	Max. Clad Temp.*		Comments
		Primary Scram	Secondary Scram	
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 bypass system	<1365°F	<1365°F	Core sees only normal trip
15.3.3	Extremely unlikely events			
15.3.3.1	Steam or feed-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	Primary heat transport system pipe leak	no effect	no effect	No effect on reactor core or primary system temperatures or pressures
15.3.3.5	Intermediate heat transport+ system pipe leak	no effect	no effect	Core temperatures would not increase

*Fuel pin cladding midwall temperature (under wire wrap)

+Not regarded as credible, evaluations event only.

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TABLE 15.5-1
FUEL HANDLING AND STORAGE EVENTS

Section No.	Event	@ Site Boundary (2-hr)	@ Low Population Zone (30-day)	Comments
15.5	Fuel handling & storage events			
15.5.1	Anticipated events (None)			
15.5.2	Unlikely events			
15.5.2.1	Fuel assembly dropped within reactor vessel during refueling	<1.67 REM	<0.00084 REM	Consequences of this event are within the umbrella of Section 15.5.2.4
15.5.2.2	Damage of fuel assembly due to attempt to insert a fuel assembly into an occupied position.	<8.98x10 ⁻⁴ REM	<3.25x10 ⁻⁴ REM	This event is well within the suggested guideline dose rate.
15.5.2.3	Single fuel assembly cladding failure and subsequent fission gas release during refueling	8.98x10 ⁻⁴ REM	3.25x10 ⁻⁴ REM	This event is well within the suggested guideline dose limits.
15.5.2.4	Cover gas release during refueling	1.67 REM	0.00084 REM	This event is well within the suggested guideline dose limits
15.5.2.5	Heaviest crane load impacts reactor closure head	<1.67 REM	<0.00084 REM	Consequences of this event are within the umbrella of Section 15.5.2.4
15.5.3	Extremely unlikely event			
15.5.3.1	Collision of EVTMM with control rod drive mechanism	<1.67 REM	<0.00084 REM	Consequences of this event are within the umbrella of Section 15.5.2.4

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TABLE 15.6-1
SODIUM SPILL EVENTS

Section No.	Events	Sodium Spill		Location			Max. Off-Site Dose	Max. Press/Temp	Comment
		Gallons	Temp. (°F)	Atmosphere	Bldg.	Cell	% of 10CFR100		
15.6	Sodium Spills								
15.6.1	Extremely Unlikely								
15.6.1.1	Primary sodium in containment, storage tank failure during maintenance	32,000	400	Normal Air	RCB	Overflow Tank Cell	0.0004	1.8 psig/ 248°F*	Doses well within guideline limit containment pressure, less than 1/5 design. Temperatures within design limits.
15.6.1.2	Failure of ex-vessel sodium cooling system during operation	7,500	500	Inerted	RSB	Cooling Equip. Cell	0.35	1.5 psig/ 145°F**	Doses well within guideline limit. Cell pressure well within projected design pressure. Temp. within design limits.
15.6.1.3	Failure of ex-containment primary sodium storage tank	90,000	400	Inerted	SGB/ IB	Storage Tank Cell	2.20	4.0 psig/ 260°F**	Doses well within guideline limit cell pressure within projected design pressure, temp. within design limits.
15.6.1.4	Primary Heat Transport System piping leak	100 kg/min (~30 gal/min) for 10 min.	1015	Inerted	RCB	PHTS Cell	0.001	5.0 psig/ 300°F**	Doses well within guideline limit cell pressure within projected design pressure, temp. within design limits.
15.6.1.5	Intermediate Heat Transport System piping leak	400,000	800°F	Normal Air	SGB/ IB	IB	0.046	<3.0 psig/ 600°F	Doses well within guideline limit. Cell must be vented at a rate of 700,000 cfm to maintain 3 psig design pressure.

*RCB - Reactor Containment Building
RSB - Reactor Service Building
SGB/IB - Steam Generator Bldg/Intermediate Bay
PHTS - Primary Heat Transport System

*In Containment
**In Affected Cell

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TABLE 15.7-1

OTHER EVENTS

Section No.	Events	Potential Limiting Parameters	Comments
15.7	Other Events		
15.7.1	Anticipated Events		
15.7.1.1	Loss of One D.C. System	None	No adverse operating conditions have been identified with this event.
15.7.1.2	Loss of instrument or valve air system	None	Detailed description of failure effects or safety-related instrument air supplies, if any will be provided in the FSAR.
15.7.1.3	IHX Leak	None	Core sees normal shutdown.
15.7.1.4	Off-normal cover gas pressure in the reactor primary coolant boundary	None	No adverse operating conditions associated with this event.
15.7.1.5	Off-normal cover gas pressure in IHTS	None	No adverse operating conditions associated with this event.
15.7.2	Unlikely events		
15.7.2.1	Inadvertent release of oil through the pump seal (PHTS)	None	No adverse consequence identified at this time.
15.7.2.2	Inadvertent release of oil through the pump seal (IHTS)	None	No adverse consequence identified at this time.
15.7.2.3	Generator breaker failure to open at turbine trip	None	Core sees only normal shutdown.
15.7.2.4	Rupture of RAPS Surge Vessel	<2.5 REM (integrated 2-hr dose at the site boundary)	Consequences will be within suggested guideline doses.
15.7.2.5	Liquid rad-waste system failure	3.7x10 ⁻⁶ REM @ site boundary 3.05x10 ⁻⁷ REM @ LPZ	Consequences are well within the suggested guideline doses.
15.7.3	Extremely unlikely events		
15.7.3.1	Leak in a core component pot.	~3200°F Center Fuel Pin	Only slight cladding melting. Fission gas release within umbrella of Section 15.5.2.3.
15.7.3.2	Spent fuel shipping cask dropped from maximum possible height	5.95x10 ⁻⁵ REM Whole Body @ SB (2-hr) 4.78x10 ⁻⁶ REM Whole Body @ LPZ (30-day)	Doses are well within the suggested guidelines.
15.7.3.3	Maximum possible conventional fires, flood, and storms	None	None
15.7.3.4	Failure of plug seals and annuli	None	No adverse consequences associated with this event.
15.7.3.5	Fuel rod leakage combined with IHX and steam generator leakage	None	No adverse consequences associated with this event.

CHAPTER 16 TECHNICAL SPECIFICATIONS

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 specifications 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. However, these are presented as being preliminary only, the actual technical specifications will be provided in the FSAR, and may well differ from these.

Although it is customary in PSAR's to provide general information only, this chapter includes detailed information in an attempt to provide an insight to the expected operating characteristics of the plant.

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 Settings. 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.

CHAPTER 17 QUALITY ASSURANCE

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, which is shown in Figure 17-1, 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 of Components
16. Corrective Action
17. Quality Assurance Records
18. Audits

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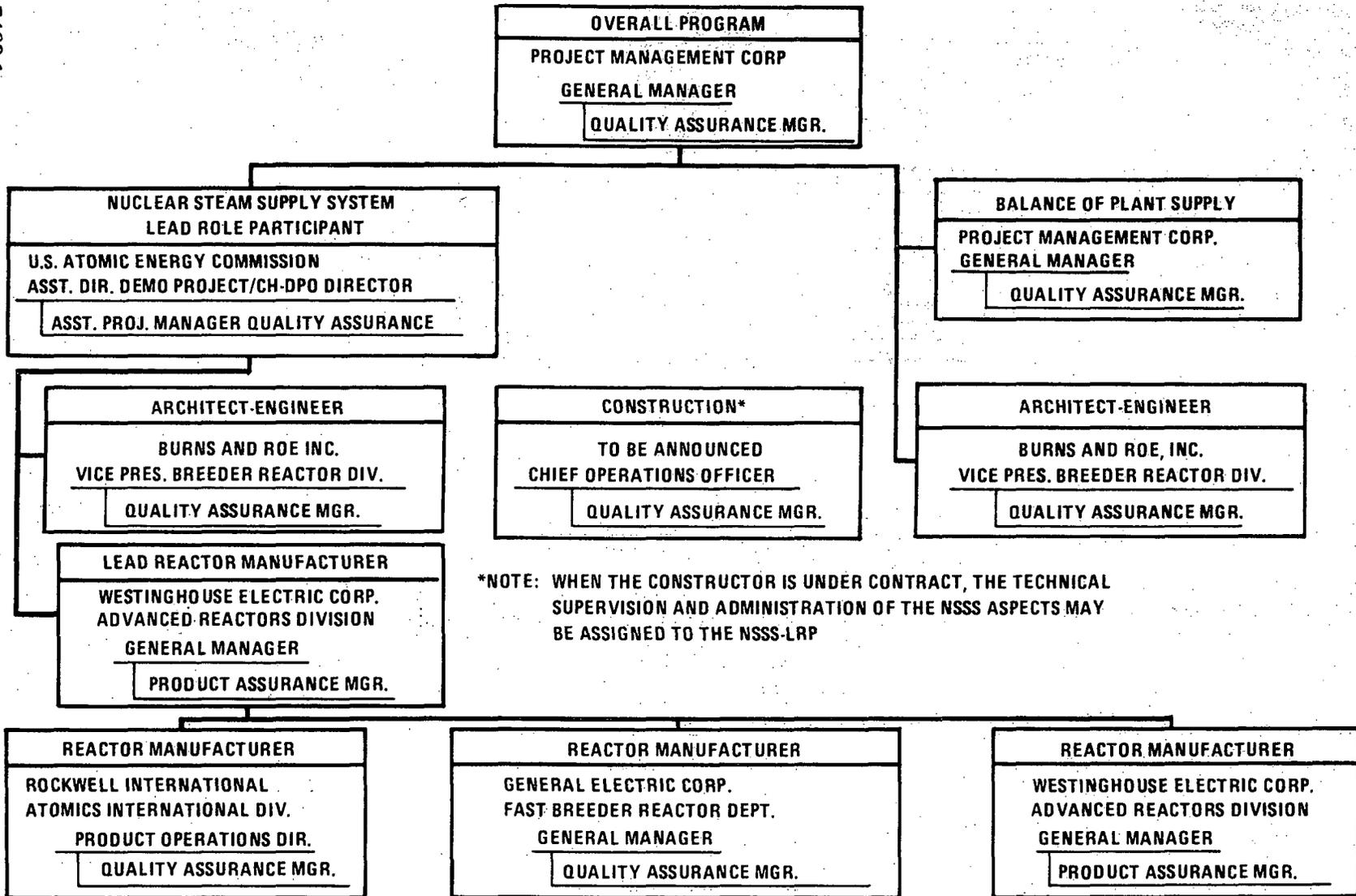


Figure 17-1. CRBRP Organizational Relationships

APPENDIX A COMPUTER CODES

Appendix A provides brief abstracts of the 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 70), references, available in the open literature or at the user's location, have been cited to provide a source for supplementary information. In the case of proprietary codes, the originating organization has been identified.

APPENDIX B GENERAL PLANT TRANSIENT DATA

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.

Example descriptions are provided below:

Normal Event 4a - Loading and Unloading

The plant design loading and unloading events are conservatively represented by a continuous and uniform ramp power change of 3% of rated power per minute through the load range of 40% to 100% of rated power. This load range is the maximum permissible consistent with the reactor control system, which is designed to accommodate automatic load following capability while maintaining rated steam conditions. Load changes in this region are accomplished by linearly varying primary and intermediate sodium flows with power while holding turbine inlet pressure constant.

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 pressure increases 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 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.

Faulted Event 1 - Safe Shutdown Earthquake

Requirements and load combinations of the SSE are defined in Section 3.7. The SSE loadings shall be considered to occur in conjunction with a reactor trip. Following the SSE, the intermediate heat transport system, steam generator system, and steam generator auxiliary heat removal system together must provide for removal of stored and decay heat.

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 Section 1.1 of the PSAR (reproduced in Addendum A of this Summary), and in Addendum B of this Summary (reproduced from Appendix C of the PSAR).

C.1 Rationale and Summary

This Section of the Appendix sets out the criteria for success of the programs in terms of reliability goals and their allocations and summarizes the results of the initial reliability assessment. It also describes the overall reliability program in summary form.

The goal, determined to acceptable address concerns of public risk from LMFBR's and hence to eliminate core disruptive accidents as a basis for design, is as follows:

- The probability of exceeding 10CFR100 guidelines shall be less than one chance in a million per reactor year.

For purposes of the reliability assessment described in the PSAR, this goal has been conservatively interpreted to mean that the probability of losing core coolable geometry will be less than one chance in a million per reactor year. This goal has been divided into three parts as follows:

Element	Goals (Failures per year)
Shutdown System	$<10^{-7}$
Shutdown Heat Removal	$<8 \times 10^{-7}$
Faults leading to LCG not sensed by PPS	$<10^{-7}$

A detailed treatment of the rationale for selection of the goals, is given in Section C.1 of Appendix C of the PSAR.

C.2 Current Reliability Assessment

The current reliability assessment is that the plant meets the overall objectives presented in Section C.1 of the PSAR. It is important to recognize that this conclusion is not based solely on the quantitative analysis presented. The basis for the conclusion of plant

safety adequacy comprises four major elements, namely:

1. The quantitative assessment based on available reliability methodology and hardware reliability information, as presented in Section B.1.3.1 of Addendum B of this summary.
2. The qualitative reliability activities within the project which impose a systematic and disciplined method of plant design. This approach serves to minimize the likelihood of design oversights and in particular to identify common-mode failure potential. These activities are described in Section B.1.3.2 of Addendum B of this summary.
3. The presence of redundancy and diversity in essential design features in the systems of interest. These elements of equipment design are described in Section B.1.3.3 of Addendum B of this summary.
4. Capability to incorporate design and procedural changes to enhance the reliability over and above that required to meet normal design practices.

C.3 Reliability Verification

The overall shutdown system and decay heat removal reliability programs are summarized in PSAR Section C.1 and described in some detail in PSAR Section C.3.

The major reliability verification tasks are:

- The Reliability Manual, as a guide to correct and consistent application of reliability methodology in the project. The manual covers the methods of assessment (FMEA, FTA, Monte Carlo simulation, Bayesian techniques, etc.) and the management procedures required for their implementation. Prime responsibility for manual preparation is with the shutdown system program. Supplemental procedures will be provided by the decay heat removal program for that program's special needs.
- Analyses beyond the initial assessment which are underway (rod worth requirements and uncertainties, speed of response requirements, conditions necessary to preservation of core coolable geometry, influence of component repairability, etc.).

- Development of a failure and repair data base. The first phase covers collection and reduction of existing relevant data and of near term available data (FFTF, CRBRP component tests, etc.); the second phase is collection and interpretation of CRBR early operational data. The shutdown system program has the lead responsibility for central data bank development. The decay heat removal effort will contribute pertinent data on thermo-hydraulic and structural components.
- Features of the test program. The discussion covers planned tests of components, subsystems, and systems, identification of major existing test facilities which will be used, and facility construction and/or modifications necessary to meet program needs. This section includes conceptual arrangement drawings of the proposed modified or additional facilities for the shutdown system testing.

Schedules for the reliability verification programs are shown in bar chart form, indicating the availability of data to support the plant operating license application.

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 transient overpower and loss of flow (TOP and LOF) events. The results of analyses of these events are summarized in Table D-1.

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.

Much of the Appendix 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 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. 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.

Finally, the radiological consequences are examined, and results quoted in Tables D-2 and D-3. 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 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 D-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 D-2

RADIOLOGICAL CONSEQUENCES OF
HYPOTHETICAL TOP-HCDA'S

	10 CFR 100	Expected	Off-Site Doses (REM)				
			Upper Bound				
			Base Parametric Cases				
			<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
<u>2 Hr S.B. (0.41 mi)</u>							
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
<u>30 Day LPZ (5 mi)</u>							
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	0.657
Whole Body**	25	0.0015	0.0035	0.014	0.034	0.063	0.131
<u>Head Leak Rate</u> (%/day at 20 atm.)			10	10 ²	10 ³	10 ⁴	∞
<u>RCB Pressure</u> (psig)		1	1	1	1	1	10

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

TABLE D-3

RADIOLOGICAL CONSEQUENCES OF
HYPOTHETICAL LOF-HCDA

	<u>10 CFR 100</u>	<u>Off-Site Doses (REM)</u>				
		<u>Representative</u>				
		<u>Parametric Cases</u>				
		<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>
<u>2 Hr S.B. (0.41 mi)</u>						
Bone	150	0.093	0.926	9.11	131.7	1893
Thyroid	300	0.0063	0.063	0.616	8.90	128.7
Lung	75	0.005	0.050	0.488	7.05	101.4
Whole Body	25	0.028	0.258	1.54	3.04	5.5
<u>30 Day LPZ (5 mi)</u>						
Bone	150	0.019	0.185	1.82	26.4	106.2
Thyroid	300	0.001	0.011	0.105	1.52	6.75
Lung	75	0.001	0.01	0.097	1.40	5.67
Whole Body**	25	0.0035	0.014	0.031	0.062	0.214
<u>Head Leak Rate</u> (%/day at 20 atm.)		10	10 ²	10 ³	10 ⁴	∞
<u>RCB Pressure</u> (psig)		1	1	1	1	10

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

APPENDIX E - PRIMARY PIPE RUPTURE FALLBACK POSITION

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. However a parallel design to mitigate the consequences of such a rupture is being pursued which could be incorporated into the CRBRP if necessary.

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 for all anticipated operating conditions.

Two 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 and shall be capable of being built and installed in the plant with minimum effect on the start-up schedule.

Discussion of Program

Current design studies indicate that the pipe sleeve concept is the most promising design option for mitigation 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 additional safety margin. In addition to the pipe sleeves other mitigating options are being evaluated. These options

include the use of pipe restraints and flow diodes.

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 will not cause failure of the sleeve.

In-service inspection will be visual in nature. 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.

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. For a break at the reactor inlet, representing the worst case location for pipe rupture, the hot channel 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 increase instantaneously 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 inlet 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 inlet plenum does not exist.

If a postulated double-ended rupture is required to be treated as a basis for design, venting capability may be required for the PHTS cells and reactor cavity. The PHTS cell design will accommodate the maximum temperature of 1015°F. The RC is currently being designed for 35 psig pressure and 1015°F temperature and is therefore adequate to ensure inerting capability and structural integrity in the event of primary pipe ruptures. From a radiological standpoint, preliminary analysis indicates that a large margin (greater than a factor of 10⁴) exists between the potential doses at the site boundary and low population zone and the applicable guideline limits.

APPENDIX F CORE DISRUPTIVE ACCIDENT ACCOMMODATION

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 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.

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 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 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 text covers 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 structures and some variants of these) In each case design drawings

are given, and a date of 12-1-75 is quoted for completion of preliminary design.

Part II - Ex-Vessel Core Catcher (EVCC)

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.

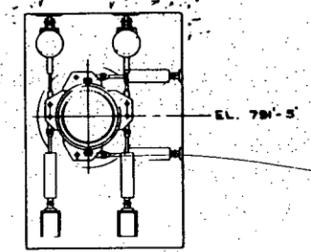
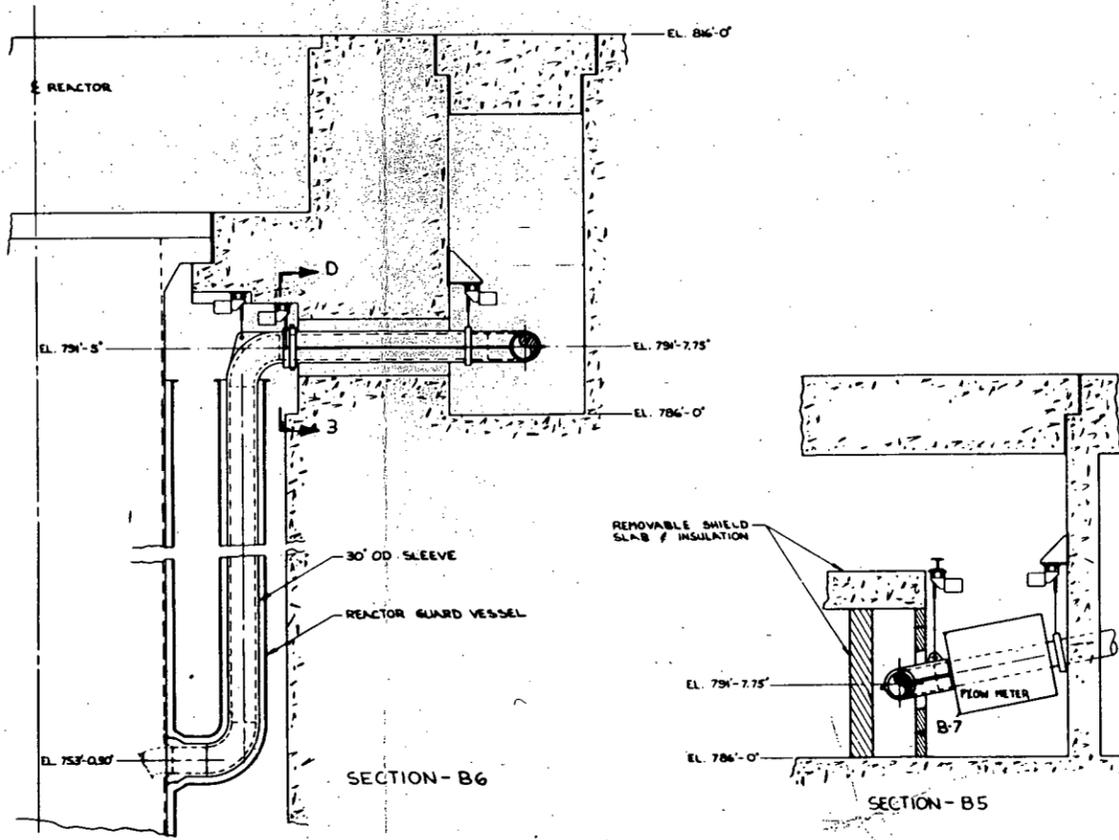
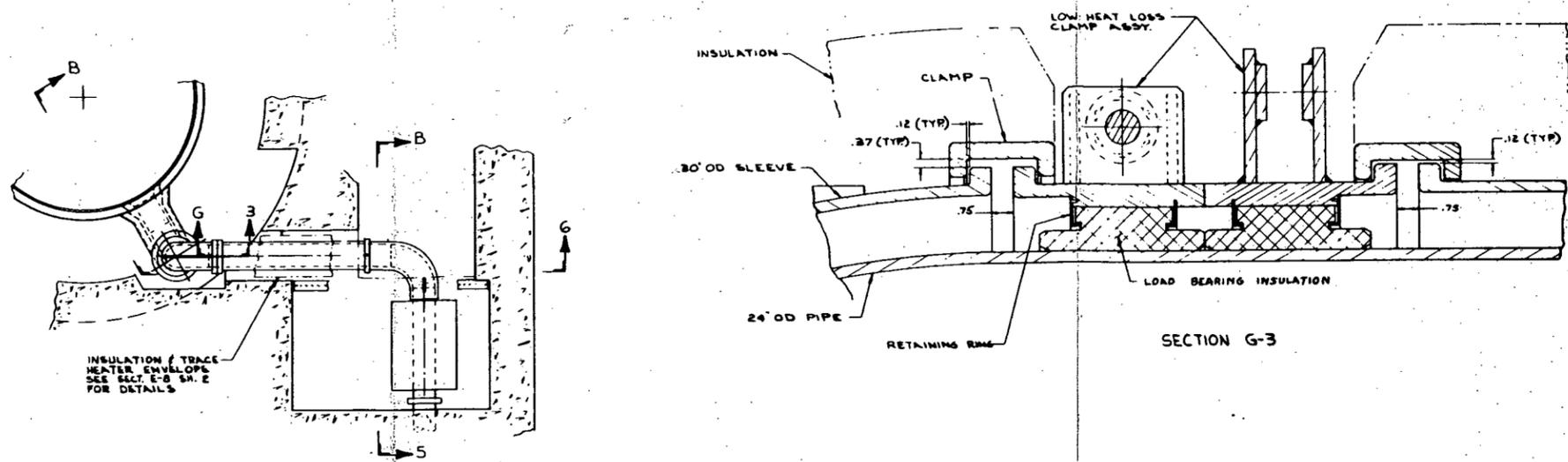


Figure E-1 CRBRP Primary Cold Leg Piping Sleeve Concept

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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
- A 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.

ADDENDUM A

INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

The Clinch River Breeder Reactor Plant (CRBRP) will provide a vital step in the United States' reactor development program. The objective of the U. S. Energy Research and Development Administration (ERDA) Liquid Metal Fast Breeder Reactor (LMFBR) program is to develop, on a broad, proven technological and engineering base, with joint utility and industry participation, a commercial breeder reactor industry.

In keeping with the objective of the LMFBR program, the objectives of the CRBRP are as follows:

1. To confirm and demonstrate the potential value and environmental desirability of the LMFBR concept as a practical and economic future option for generating electrical power;
2. To confirm the value of this concept for conserving important nonrenewable national resources;
3. To develop, for the benefit of government, industry and the public, important technological and economic data;
4. To provide a broad base of experience and information important for commercial and industrial application of the LMFBR concept; and
5. To verify certain key characteristics and capabilities of LMFBR plants for operation on utility systems such as licensability and safety, operability, reliability, availability, maintainability, flexibility and prospect for economy.

Since there is limited experience within the present-day licensing framework which is directly relatable to a first-of-a-kind demonstration plant such as the CRBRP, the information presented in this introductory section is more extensive than normally found in light water reactor PSAR's. For clarity of presentation, Section 1.1 has been subdivided as follows:

- 1.1.1--This subsection gives the information requested by the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LMFBR Edition (SFAC).
- 1.1.2--This subsection presents the basis for the application and details the manner in which the CRBRP design approach will assure compliance with applicable Commission requirements.
- 1.1.3--This subsection describes the applicability of the Regulatory Guides issued through June, 1974.

1.1.1 General Information

This PSAR is submitted in support of a joint application by Project Management Corporation (PMC) and the Tennessee Valley Authority (TVA) for a CP and Class 104(b) Operating License to construct and operate the Nation's first large-scale LMFBR Demonstration Plant.

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 United States Energy Research and Development Administration's (ERDA) Division of Reactor Research and Development. The General Electric Company and the Atomic International Division of Rockwell International Corporation have major subcontracts, related to the NSSS, from Westinghouse. Burns & 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 be operated by TVA. Further amplification of the relationship 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, approximately 25 miles west of Knoxville. The site is on a peninsula bounded on the north by ERDA's Oak Ridge Reservation and on the remaining sides by the Clinch River. 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 gross generation level of 380 MW(e). This power level is discussed under the terms "thermal/hydraulic" (T/H) conditions in various sections of the PSAR. It is this power level which forms the basis for the present application, and 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 1121 MW(th) corresponding to a gross generation level of 439 MW(e). These latter conditions are referred to as "stretch" conditions in this PSAR. 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, subsequent to issuance of the Construction Permit, a supplementary application may be made to increase the power level to these "stretch" conditions. However, for purposes of the CP review, the additional capability of permanent plant components should be treated as an inherent margin in the plant design.

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. Since operation with only two loops in service may be desirable for maintenance or other reasons, the plant is designed with the capability for two-loop operation. The power level appropriate to two-loop operation will be established before application for the plant Operating License is made. It is the Project objective for the CP review to establish that sufficient redundancy has been provided in the heat removal system to permit two-loop operation. Although specific analyses related to two-loop operation are not presented, sufficient information to show that no major impediments exist to eventual two-loop operation is presented in Chapter 5 and Appendix C.

The scheduled construction completion date for the Plant is September 1981; power operation is anticipated early in 1983.

1.1.2 Basis for the Application

The basis for the CRBRP application is to provide a plant which meets all applicable Federal regulations including those specified in 10 C.F.R. 100. This application follows the conventional course for licensing of a nuclear power plant, however, due to the lack of precedents, the CRBRP design approach utilizes more extensively reliability techniques to provide a systematic determination of events to be included in the plant design basis than would be normal in a single LWR application.

Recognizing that the CRBRP is a first-of-a-kind plant, that existing experience is therefore limited and that guidelines are not directly applicable in all cases, a dual design approach has been adopted. The primary approach excludes severe accidents from the plant design bases due to their low probability of occurrence. The secondary approach, or fallback position, assumes such severe accidents in the plant design bases even though the Project considers this secondary approach to be overly conservative. This dual approach is summarized in Table 1.1-1 and is described in the following paragraphs.

First (1.1.2.1), the overall philosophy of the design approach is discussed. This includes not only a systematic treatment of conditions and events to be considered in order to assure a reliable and safe plant, the primary approach, but also provides a second set of design considerations as a fallback position, the secondary approach. Second (1.1.2.2), the manner in which the primary approach is implemented into the plant design is presented. Third (1.1.2.3), the design efforts to implement fallback features are discussed. And fourth (1.1.2.4), the procedure is presented for judging that the primary approach is acceptable, and that fallback features need not be implemented in the design.

1.1.2.1 Design Safety Approach

The overall design of the CRBRP is based on the natural three levels of design which Regulatory uses to evaluate the adequacy of proposed nuclear power plants.

Overview

The first level consists of a technically sound design that results in high reliability and minimizes the occurrence of accidents. The second level provides protection against failures or malfunctions which might occur in spite of precautions taken in the design, construction and operation of the plant in a manner which minimizes plant damage and ensures safety to the public and the operating staff. Reliability tasks have been established to assure highly reliable design and performance of certain systems within both levels one and two. The third level provides assurance that the public is protected even in the event of extremely unlikely circumstances of failures or malfunctions. A systematic approach using reliability methodology is employed to select the limiting design basis. Also, margins are provided in the design for defense against events beyond those included in the Extremely Unlikely category. An overall perspective of the manner in which this multi-level design approach is being incorporated in the CRBRP is provided in Table 1.1-2. This brief foregoing description represents what is termed in this PSAR the "Reference Design".

Early in the Project life, it was decided that the design of certain features to accommodate the consequences of severe accidents would be pursued as a fallback position, in the unlikely event that the Project failed to satisfactorily support the selected design basis. This approach is termed the "Parallel Design" and comprises the features of the Reference Design plus the modifications and additions to the Reference Design to permit the plant to accommodate the consequences of more severe accidents.

The Reference Design is described and discussed in Chapters 1 through 17 and Appendices A through C; the additional efforts related to the Parallel Design are described and discussed in Appendices E and F. This dual approach for submittal is consistent with the understanding developed in Reference 1.

Level 1 Design

The first level of design provides reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic features of the design, such as quality assurance, redundancy, maintainability, testability, inspectability, and fail-safe characteristics. The plant is being designed not only to accommodate steady-state power conditions,

but also to have adequate tolerance for normal operating transients, such as start-up, shutdown, and load-following. As a basic part of the LMFBF development program, a number of large scale engineering proof tests are being performed to verify the design concepts. This testing process, in the first level, is to provide predictability of performance and, hence, safety through assurance of the use of proven methods, materials, and technology.

Extensive pre-operational test programs will be conducted in the plant to assure conformance of components and systems to the established performance requirements. Key parameters will be monitored continuously or routinely and a well-defined surveillance, in-service inspection, and preventive maintenance program will be carried out by a trained operating and maintenance staff to provide assurance that as-built quality is maintained throughout the life of the plant.

Level 2 Design

The second level of design provides protection against Anticipated and Unlikely Faults (such as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors - Faults are defined in Table 1.1-A) which might occur in spite of the care taken in design, construction, and operation of the plant. This additional level of defense for the public and the operating staff is provided by redundancy of critical components as well as by protection devices and systems designed to assure that such events will be prevented or arrested. The requirements for these protection systems are based on a spectrum of occurrences which could lead to off-normal operation which the plant design must safely accommodate. Conservative design practices, including redundant detecting and actuating equipment, are incorporated in the protection systems to assure both the effectiveness and reliability of this second level of design. These systems are designed to be routinely monitored and tested to provide full assurance that when they are required to operate, they will do so reliably.

Reliability Efforts in Levels 1 and 2

As part of the first and second level design efforts, tasks to assess and assure acceptably high reliability of certain plant systems have been established.

Level 3 Design

The third level of design supplements the first two levels by providing acceptable plant response to Extremely Unlikely Faults such as pipe leaks, large sodium fires, or large sodium-water reactions. Although these faults are of low probability, appropriate engineered safety features are incorporated into the CRBRP design to safely accommodate such events. Typically conservative assumptions and evaluation methods, such as assumed failure of any single active component, are used to develop adequate designs. In addition, conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant. These include such low probability events as severe earthquakes, tornadoes, and floods. These faults and natural phenomena combine to define the design basis envelope.

Reliability Program

As with LWR Plants, a broad spectrum of events which have potential for their consequences to exceed the 10 C.F.R. 100 guidelines, have been analyzed to determine if it is appropriate for such events to be part of the design basis envelope. For the CRBRP, this effort is considerably expanded in degree and extent of use because of the lack of precedents for establishing a design basis envelope for a plant of this kind. In this plant, an approach based on LWR precedent is taken for certain events; for example, it is considered that a large commercial airplane (B747 or DC10) striking the containment is an event that is so unlikely that it need not be considered as a design basis. The rationale by which this conclusion is drawn is that the plant is remote from commercial airports and from established FAA airways (see Section 2.2 of this PSAR). Further, the air traffic in this particular geographical area is so sparse that the likelihood of such an occurrence is low. This is the same rationale that has been applied to light water plants located in similarly remote areas. This approach has also been applied to LWR's to establish that pressure vessel rupture should not be a design basis event. The rationale by which this event is excluded as a design basis event has been presented by the Regulatory Staff in Reference 2. It shows that the statistical probability of such an event is so low that it need not be used as a design basis. This method is applied in the CRBRP and its use is expanded to assist the designer in achieving high reliability in specific plant systems.

Accidents with potential to exceed 10 C.F.R. 100 guidelines are either in the design basis envelope of the plant or excluded from it depending on the probability of the event which initiates the accident. A comprehensive Reliability Program has been established to determine which events should be included in the design basis envelope and to assure the high reliability of systems necessary to prevent the onset of accidents which are excluded from the design basis envelope. This program is described in Appendix C. As an initial step in the Reliability Program, a systematic assessment of a broad spectrum of accidents for CRBRP has been made. This assessment shows that

the only events which are not included in the design basis envelope or excluded by their inherent low probability (as in the case of the airplane striking the plant) are events which involve severe reactor core damage. The assessment further shows that a loss of in-place coolable (core) geometry* is the severe reactor core damage event that, assuming conservative calculations, has the potential to exceed 10 C.F.R. 100 guidelines. The Project's review of this event has shown that the probability of occurrence of its initiators can be controlled by design. Therefore, it has been taken as a Project position that the above described event (i.e. initiation of loss of in-place coolable geometry) will be shown to be of such low probability that they need not be taken as design bases.

The manner in which this acceptably low probability is shown involves the application of proven reliability methodology in the design and assessment of the systems relied upon to prevent initiation of loss of in-place coolable geometry. The rationale for the determination of the initiators to be so treated is presented in Appendix C to this PSAR.

Third Level Design Margins

Although the approach thus far described treats a broad spectrum of events, even including those so remote in probability that none is expected during the plant lifetime (Extremely Unlikely Faults), additional conservatism is provided in the form of design margins or features to accommodate even more severe and less probable unidentified events.

The Project has provided margin of protection against such accidents which result in design requirements for the plant to accommodate (for example).

Impact loadings on the vessel head.

Dynamic loadings within the primary system, principally on the vessel and the intermediate heat exchanger and coolant loop components.

Thermal, mechanical and geometric requirements in the core support structure to enhance post accident cooling capability.

Radiological protection for the control room under accident release conditions.

* Loss of in-place coolable geometry is broadly defined as the onset of clad melting. Amplification of this definition is given in Appendix C (C.1.3.3).

Parallel Design

In the event that initiators leading to loss of in-place coolable geometry could not be shown on a timely schedule to have an acceptably low probability, it was deemed appropriate to initiate a parallel design effort. The Parallel Design provides for the design and development of features to minimize accident consequences from events not used as design bases for the Reference Design.

The Parallel Design comprises additional protective features, such as a sealed area above the reactor head, an ex-vessel core retention and cooling system, and features to minimize the consequences of a large pipe rupture. The development of these features is on a schedule to permit them to be incorporated into the plant on the current construction schedule. The Parallel Design will be vigorously pursued until the Reference Design is shown to be a suitable basis for licensing. Section 1.1.2.4 delineates the activities which must be completed in order to permit termination of the Parallel Design efforts.

1.1.2.2 Reference Design

The CRBRP Reference Design is based on the areas of consideration (all except Parallel Design Options) discussed in the Design Safety Approach in the preceding subsection (1.1.2.1). The manner in which each of these considerations is implemented into the Reference Design is briefly discussed in this subsection.

Level One

The purpose of the first level of design is to assure that the plant is reliable, operable, inspectable, testable, and maintainable. These factors are of prime importance in implementing the first level of design into the CRBRP. Therefore, a number of plant design decisions were made to incorporate design features which by their very nature avoid the occurrence of accidents or mitigate accident effects should they occur. The following examples are related only to core characteristics, however, they are typical of the design features incorporated in this level:

- A core restraint system to control core position and assure an acceptable power coefficient which cannot be degraded by core movement.
- Features to assure that rapid outward motion of control rods is prevented.
- Reactor fuel assemblies with fuel pin spacing designed to reduce potential for reductions in coolant flow due to fuel swelling or plugging.

Level Two

It is recognized, however, that errors or malfunctions can occur despite the care and attention provided by implementation of the first level. Therefore, implementation of the second level of design includes in the design a number of protective systems and plant features provided to protect against malfunctions, and to limit their consequences to definable and acceptable levels.

Examples of these features are:

- Two diverse, redundant, and independent shutdown systems, each capable of shutting down the reactor with one rod inoperative.
- Three loop designs providing redundant heat removal capability such that core cooling is maintained even if an active component of one loop is disabled at the same time normal off-site power supply is lost.

- Extensive leak detection capability to provide assurance that any leaks in the coolant boundaries are detected promptly so that corrective action can be taken.
- Guard vessels and elevated piping to assure core coverage and continuity of core cooling as a further measure of inherent safety against coolant leakage.

Furthermore, testing and development programs are established to define clearly the nature and consequences of accidents which might result from certain malfunctions. These programs include:

- The characterization of corrosive effects of sodium leakage on external surfaces as a function of leak size, location, temperature, material, and local atmosphere.
- Studies of water to sodium leaks in the steam generators. This includes the validation of analytical techniques used to predict pressures in the IHX during a postulated sodium-water reaction.

Reliability Program

Primary emphasis is placed on a design adhering to the first two levels; part of this emphasis is carried by the Reliability Program which will be pursued and used (with design modifications if necessary) to demonstrate and confirm that initiators or events that can lead to loss of in-place coolable geometry have an acceptably low probability of occurrence and thus need not be used as a basis for design.

Based on a thorough review of existing Regulatory documentation, appropriate literature, and the application of reliability techniques to complex systems and preliminary analyses for the CRBRP, the project has established the following goal:

The probability of exceeding 10 C.F.R. 100 guidelines shall be less than one chance in one million per reactor year.

The Reliability Program and its schedule are detailed in Appendix C of this PSAR. It treats postulated severe events and establishes that loss of in-place coolable geometry is the event which is conservatively related to the potential to exceed 10 C.F.R. 100 guidelines. Paramount in this reliability program is the confirmation that the dual, independent, diverse, and redundant shutdown systems and the shutdown heat removal systems are highly reliable since these systems, as part of the second level of design, are most important in the protection of the plant. The Reliability Program includes for these systems both quantitative and qualitative analysis, design reliability requirements, continuous design reliability control, and extensive test programs over the next several years to confirm their reliability. In addition to these systems, the Reliability Program treats other possible initiators such as fuel element failure propagation potential and structural reliability of components such as support systems and piping.

Certain development programs are necessary to demonstrate and/or confirm high system reliability; these are described in Section 1.5 of the PSAR. In addition to the program descriptions in Section 1.5, material relevant to a full understanding of each of these areas is contained in other parts of the PSAR as shown in Tables 1.1-4 through 1.1-6. Furthermore, because of the importance of the Reliability Program, Appendix C must be regarded as one of the keystones of this application, and a fundamental factor in the design of the plant. The Reliability Program is treated in an appendix rather than the main part of the PSAR 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 the Standard Format, where development needs related to the Reliability Program are given, there is no place appropriate for presentation of this material.
- It is necessary for proper comprehension of this program that all relevant material be collated and presented in a coherent fashion.

Sufficient information is presented in this application to establish that there is reasonable assurance that initiators of events which may lead to loss of in-place coolable geometry are of such low probability that they need not be taken as design bases.

Level Three

Implementation of level three in the plant design envelope involves the consideration of Extremely Unlikely Faults (Table 1.1-3) in the design of the plant. Plant capability to accommodate such events is assured through the use of conservative assumptions and evaluations. Analyses relating to these faults, for example, pipe leaks, large sodium fires or large sodium-water reactions, are presented in Chapter 15 of the PSAR. The conservative design bases for such low probability natural events as earthquakes, tornadoes, and floods are presented in Chapter 2.

Third Level Design Margins

Specific design requirements have been placed on particular components, systems, and structures (Table 1.1-7) which are beyond those required by the plant design basis envelope. These requirements are termed "third level design margin requirements." Each of the design chapters of this PSAR (Chapters 3 through 12) contains a statement of the third level margin requirements applied to the area of the design covered by that chapter. In Section 15.1.1 the derivation of these requirements is provided. Also in Section 15.1.1 these requirements are collated to give visibility of the third level margin requirements for the plant as a whole, and to show by preliminary assessment that the design meets these requirements.

To establish the requirements for third level design margins, it is assumed that in-place coolable geometry is lost. This assumed event leads to a core disruptive accident. The third level design margin requirements shown in Section 15.1.1 are based on generic analysis of core disruptive accidents from previous experience from FFTF analyses and preliminary analyses for the CRBRP. Appendix D details analyses of core disruptive accidents. These analyses are being updated to include core disruptive accidents specific to the CRBRP. Section 15.1.1 also contains a comparison of the most significant third level design margin requirements and loadings derived from the latest analysis as reported in Appendix D to confirm that the margin requirements were chosen prudently and that they do not differ substantially from, and are in many cases more conservative than, loadings calculated in Appendix D.

1.1.2.3. Parallel Design

The rationale leading to the Parallel Design approach was discussed in Section 1.1.2.1 (see Table 1.1-1). The Parallel Design considers the use of a hypothetical core disruptive accident (HCDA) as a design basis, the possibility of sealing the head access area, the provision of an ex-vessel core catcher, and the provision of safety features to mitigate consequences of a loss of piping integrity. These activities are programmed, funded and scheduled, and pursued in the same depth of detail and hardware as the Reference Design. They are termed parallel design options, and when combined with the features of the Reference Design and appropriate modifications thereto comprise the Parallel Design.

The discussion of mitigation features for the loss of primary piping integrity is contained in Appendix E and the remaining activities are detailed in Appendix F.

Features to Accommodate Primary Pipe Rupture

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 under the service conditions imposed by CRBRP operation. A full understanding of the rationale behind this position can be gained from study of the PSAR sections listed in Table 1.1-6. There are certain programs identified in Section 1.5 of the PSAR which are expected to demonstrate and confirm the acceptably low probability of loss of piping integrity. Completion of these programs is expected by June 1976. Pending such completion, a Parallel Design effort is being maintained, as indicated below.

Appendix E treats a massive failure of the primary coolant boundary as a design basis and shows a preliminary design of features that could be incorporated into the CRBRP. With these features incorporated, Appendix E demonstrates that a massive failure of the primary coolant boundary will not result in loss of in-place 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, would convert it to a PSAR appropriate to the Parallel 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.

Features to Accommodate Hypothetical Core Disruptive Accidents

Appendix F presents that portion of the Parallel Design which takes HCDA events as design bases. The version of Appendix F submitted at this time is to be regarded as a report on the status of activities in relation to the Parallel Design, and will be replaced with the completed version of Appendix F in September, 1975. 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 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 will be demonstrated. A supplement to Appendix F will contain pages which, when inserted into the main body of the PSAR in place of the currently existing pages, will convert it to a PSAR appropriate to the Parallel Design, in which HCDA events form the basis for design. For clarity of presentation Appendix F, including the supplement, will be printed on yellow paper.

1.1.2.4 Cessation of Parallel Design Activities

It is the objective of the Project design efforts as enhanced by the Reliability Program to show that the Reference Design is a satisfactory basis for licensing the CRBRP. However, the Parallel Design will be carried forward until this position is accepted by Regulatory.

The Project plans to provide Regulatory with information by June 1976 to justify cessation of the parallel design options. Supplemental documents will detail the confirmatory tests and analyses performed in later years up to late 1978, and the shutdown system reliability program will be further reported through the system test period up to criticality and beyond, as noted in Appendix C.

This section provides the activities which will need to be completed and milestones by which the acceptability of the Reference Design will be judged. Regulatory and ACRS must, of course, be independently satisfied that such activities are completed and milestones are met before cessation of parallel design activities. Therefore, reviews would be scheduled with these organizations at significant milestones.

The activities in support of the judgement to cease parallel design activities are listed in Table 1.1-8 for piping integrity and 1.1-9 for absence of HCDA design bases. Satisfactory completion of these activities constitutes the criteria for cessation of the Parallel Design. The activities listed in these tables are further discussed below.

1.1.2.4.1 Activities to Support Cessation of Parallel Design for Loss of Piping Integrity

Massive failures of the primary coolant boundary are not considered appropriate as bases for the design of the plant, because of the properties of the stainless steel piping by CRBRP operating conditions. PSAR sections identified in Table 1.1-6 provide the detailed technical justification for this position. For clarity of presentation, however, the salient points are summarized below.

1.1.2.4.1.1 Identification and Implementation of Design Controls

The first stage of the rationale employs extensive control of the design and manufacture of the primary coolant boundary. This includes full observance of all applicable ASME Code, Section III, Class I and Section IX requirements, enhanced by RDT Standards. These RDT Standards upgrade the Code requirements in the following areas:

- Design specifications for welded pipe
- Weld filler material
- Welder qualification
- Welding procedures
- Manufacturing operations
- Finishing of joints
- Penetrant and radiographic examination of welds
- Elbow material specifications

A detailed discussion of these requirements, including specific rejection criteria, and a comparison of CRBRP and ASME Code Inspection Criteria is provided in Section 5.3.3.6. The implementation of these design controls and inspection procedures for the PHTS piping will be fully in place prior to the decision to cease primary piping safety features design activities.

1.1.2.4.1.2 Identification of Possible Initiating Mechanisms

An important aspect of establishing the acceptably low probability of loss of piping integrity is the identification of possible pipe rupture initiating mechanisms. Two basic tools from reliability methodology are utilized systematically to meet this objective, namely, Failure Mode and Effects Analysis and Fault Tree Analysis. These activities are primarily performed by the piping design engineer with guidance and monitoring by reliability engineering. As part of the initiating mechanism identification task, the probability of overlooking flaws of specific sizes will be assessed. The assessment will be based on a search for existing data on this subject, showing the probability of flaws going undetected in spite of required inspections and of flaws developing after the inspections from metallurgical/chemical phenomena. The schedule for completion of the sub-tasks to identify initiating mechanisms is:

Piping FMEA

Preliminary 6/75
Updated 5/76

Assessment of probability
of undetected flaws

3/76

Thus, this task will be fully completed prior to cessation date of June 1976.

1.1.2.4.1.3 Analysis of Pipe Fracture Mechanics Phenomenology

Reliability analyses are planned to show that the probability of a pipe break which might lead to loss of in-place coolable geometry is acceptably low. The analysis will consider flaw growth modes including the potential for wall penetration. The calculated growth parallel to the pipe wall will also be compared to the critical crack length. Other studies will cover the crack growth directions with time, to show that even if the crack did grow significantly, that piping wall penetration would occur before the crack lengthened to critical size, detectable leakage would signal a leak, and operator action would shutdown the reactor safely. The reliability analysis will be built on the most current piping stress analysis and on test data which measure flaw growth morphology with time; influence on fracture toughness of environment, stress level, thermal aging, and as-fabricated materials properties; and critical crack size. The availability of data to support analysis of piping fracture mechanisms phenomenology is as follows:

Stress Analysis	2nd interim stress report, 5/76 Final stress report, 1/78 (confirmatory)
Basic cyclic flaw growth data	Available
Critical crack size	Available
Scale model elbow fracture data	Available
Caustic environment effects	1/76

Thus, the quantitative analysis will be based on the second stress report and all of the necessary experimental data.

1.1.2.4.1.4 Test Confirmation of Analytical Results

Confirmatory testing is planned to substantiate the fracture mechanics analytic model described above. These confirmatory tests will be completed as follows:

Elbow burst tests	12/75
Cyclic crack growth data and crack growth morphology	12/75

Full scale welded elbow tests 3/78

Since the latter tests are confirmatory to the earlier analysis they are not considered vital to the decision to cease parallel design activities.

1.1.2.4.1.5 Reliability Assessment

The elements of the analysis and test information will be incorporated in the performance of the reliability analysis. These analyses will be performed as follows:

Reliability Analysis:	Preliminary	6/75
	Updated	5/76
	Final	5/78

From preliminary scoping analysis performed for FFTF it is clear that with the planned development tests and the expected data accumulation the reliability of the CRBRP primary pipework will be shown to meet the reliability target so that the parallel design activity may be terminated. Following the production of the final stress report and the collection of final failure data a confirmatory reliability analysis will be prepared. The associated methodology is discussed in Appendix C Section 3.3.

1.1.2.4.1.6 Leak Detection Capability

Notwithstanding the exceedingly low probability of crack propagation to a point at which core flow would be impaired, considerable attention has been paid to the provision of multiple methods of leak detection. Thus, even if a leak were postulated, there is ample assurance that it would be promptly detected.

A detailed discussion of the various candidate methods of leak detection is given in Section 7.5.5, and only a brief summary is given at this point. First, the multiple methods of leak detection are:

- Radiation monitoring
- Aerosol monitoring
- Continuity detectors (cables)
- Contact detectors (spark plugs)

Hard wired audible group alarms are sounded in the control room upon indication of a leak, and the approximate location of the leak is visually displayed to the operator. Thus, operator awareness is assured rapidly, following detection of a leak, and appropriate remedial action will be taken.

Prior to the decision to terminate parallel design options a firm design capability will have been demonstrated, and the sensitivity of the instruments to be provided will be established by the last test results for the aerosol monitors in feature tests in mid-1976. Analysis will be provided to show that leak detection can be obtained well in time for remedial action.

1.1.2.4.1.7 Consequences of Hypothetical Large Breaks

Even though large failures of the PHTS boundary are not considered appropriate as design bases, analyses have been conducted to determine the consequences of such a postulated break. These are presented in Appendix E of this PSAR, and are summarized below:

The analysis shows that unless the reference design is modified, a double-ended pipe break between the reactor inlet nozzle and the top of the inlet downcomer pipe, including the elbow can result in coolant temperatures exceeding the coolant saturation temperature in the hot channel for three-loop operation. Since a criterion used to demonstrate accommodation of postulated pipe ruptures is the prevention of coolant boiling in the core, design modifications are required to limit the leakage flow from the reactor vessel for a postulated rupture in the inlet downcomer.

One feasible approach is to provide pipe sleeves around the sections of the primary piping in which a double-ended rupture could cause coolant boiling in the core hot channel. This concept provides the capability of quickly building up a static head within the sleeve following a pipe rupture and increases the effective impedance to outflow from the reactor. The pipe sleeve concept has the additional advantage of not degrading plant performance under normal operating conditions.

In addition to the core temperature transients, pipe ruptures in the primary heat transport piping could cause pressure and temperature transients in the cells and structures. The design approach to accommodating such transients is to ensure that no failure will result that could lead to loss of safety function of any components within these areas or cause ingress of cell gas into the reactor inlet plenum.

1.1.2.4.2 Activities to Support Cessation of Parallel Design for HCDA Safety Features

1.1.2.4.2.1 Identification and Implementation of Design Controls

Major attention is paid to quality control in the design, fabrication and installation of the CRBRP components and systems. The detailed procedures by which such controls are implemented are given in Chapter 17 of the PSAR and amplified in the QA manuals of Project participants. From these it can be seen that a thorough and comprehensive process of qualification and approval exists at every stage.

Such procedures, however, would be of little value without stringent standards against which they must be assessed. These include not only the requirements of Section III of the ASME Code, but also appropriate application of a large number of RDT, ANSI, IEEE, MIL Spec and other standards. RDT Standard F2-9T will be appropriately applied to the establishment of reliability controls for each safety related system, the integration of the goal into the design requirement, and the demonstration that the goal has been met. The reliability program is described in much greater detail in Appendix C of this PSAR. It is one of the keystones of the CRBRP design safety approach, and is part of the basis for the position that accidents involving loss of in-place coolable geometry should not be regarded as design bases for the plant.

1.1.2.4.2 Identification of Possible Initiating Mechanism

A program has been established to identify potential initiating mechanisms for loss of in-place coolable geometry. The identification process is based on reliability analysis tools such as Failure Modes and Effects and Fault Tree Analysis (FMEA and FTA).

Because of the importance of the reactor shutdown system and shutdown heat removal system, preliminary FMEA's have already been completed for those systems and will be further updated prior to the decision to cease parallel design activities.

Initiators not addressed by the reactor shutdown system are currently being analyzed. These include structural failures (core support, reactor vessel, upper internals, etc.) and other low probability events such as fuel failure propagation (See Section 1.1.4.3.2.5). The FMEA analysis will be supplemented by a Fault Tree Analysis of the systems to assure that all possible failure chains and their consequences have been considered.

As seen in Tables 1.1-9 and 1.1-10, qualitative and quantitative analysis of all essential elements will have been performed in depth by May 1976, producing high confidence that all potential initiators of loss of in-place coolable geometry have been identified.

1.1.2.4.2.3 Reliability Confirmation to Acceptable Level of Probability

Each initiating mechanism identified will be evaluated to assure that its contribution to the overall probability is within the initial allocation provided. Reliability goals have been established and allocations of the goals to appropriate reactor systems have been made. Consistent with their overall importance in preventing core damage, reliability assessments of the shutdown and shutdown heat removal systems have already been made. The results of these analyses are presented in Appendix C and both systems are shown to meet their allocated reliability goals. Updated reliability assessments of these systems will be available by May 1976 and will reflect increased maturity in design, modeling and data. The current assessments are based on best estimates of failure data from available sources.

A large test program is in place to support the results of the reliability assessments. Initial reliability testing is directed at the component/part level to assure that failure mechanisms are well understood and margins to failure above expected conditions known. Substantial FFTF shutdown system testing already completed is applicable to the CRBR and has been factored into the initial assessments. Specific testing of prototypic CRBR units has begun. Table 1.1-10 itemizes significant tests that will be providing data by May 1976. Data from the component level tests will be used in detailed reliability models to provide high confidence in the resultant predictions. By May 1976 a thorough understanding of potential failure mechanisms will exist and numerical assessments will have demonstrated the reliability of the shutdown and residual heat removal systems. A test program will be underway to confirm that the failure rates do not exceed acceptable levels (defined by the overall goal). Test results will be employed in confirmatory analyses even beyond the decision date for the cessation of parallel design activities.

1.1.2.4.2.4 Common Mode Failure (CMF)

A substantial portion of the reliability confirmation program is directed at eliminating CMF or confirming acceptably low probabilities of CMF mechanisms that might result in loss of in place coolable geometry. The major thrust of the CMF effort is to identify potential mechanisms using the procedures described in Appendix C and then to address each item. Consideration will then be given to design modifications to eliminate their potential (this is possible since the identification procedure has been initiated sufficiently early in the design process that design changes can be made when necessary without unacceptable schedular impact). A preliminary CMF analysis will be completed by June 1975, with an updated analysis by May 1976.

In addition to the qualitative analysis approach for identifying shutdown system CMF mechanisms, many aspects of the test program are directed at uncovering potential common mode failure mechanisms or establishing the margins to failure beyond expected operating conditions. The test plans will be complete in detail and will provide the necessary confidence that all areas of concern will be addressed, and by May 1976 some test results pertinent to CMF will be available. In the long term a system test will be performed on the Shutdown System which integrates the electrical and mechanical systems, and checks out procedures (maintenance, operational, etc.). While this test will not be initiated until 1979 to assure as near prototypicality as possible, a detailed description of this test will be available by May 1976 which will provide additional information concerning the depth to which the search for common mode failure mechanisms is being made. Table 1.1-10 shows key milestones associated with the CMF effort in the total reliability program.

1.1.2.4.2.5 Analysis of Fuel Failure Phenomenology to Show Insignificant Probability

Section 15.4 of the PSAR discusses the potential for occurrence of stochastic fuel pin failures. It is shown that large margins exist in the design and operating conditions to assure that fuel failure propagation cannot occur as a result of a stochastic failure or local faults.

The possibility of fuel failure propagation resulting from an enrichment error has also been considered in Section 15.4 of the PSAR. It is shown that even if an enrichment error large enough to cause some molten fuel to be produced in a fuel pin is assumed, and the molten fuel is released from the fuel pin, a voiding transient would result but would not cause failure propagation.

The PSAR evaluations in Section 15.4 show that no sources of damaging blockages can be identified. The design has margin to withstand local blockages, even including planar blockages that extend over several contiguous flow subchannels. Propagation of local blockages is not anticipated based on all available data, but further detailed analyses are planned. Fault trees have been developed for fuel failure phenomena and these will be extended to provide more detailed information on potential initiators and potential progression paths. The fault trees will be used to judge the probability of a loss of in-place coolable geometry as a result of fuel failure propagation phenomena in a semi-quantitative manner. By June, 1976 adequate information will be available to confirm the very low probability of fuel failure propagation.

The probability for stochastic fuel pin failure and local faults has been minimized as a result of the following activities:

- a. The fuel pin and assembly has been designed and analyzed to an extremely conservative set of design criteria, material properties, etc. See Chapter 4, Section 4.1.
- b. An extensive Quality Assurance and Quality Control Program is in place to assure that during design, fabrication, and the operations, the conservatisms are maintained.
- c. Extensive development program and operating reactor experience have shown that LMFBR fuel elements do not experience rapid fuel failure propagation.

1.1.2.4.2.6 Structural Failure Phenomenology

Structural reliability analyses will be performed for key structural elements, namely, coolant boundary, and components which support the reactor vessel and which support the core within the vessel. Analysis and test for pipe fracture mechanics phenomenology in support of the PHTS piping integrity program are provided in Sections 1.1.2.4.1.3 and 1.1.2.4.1.4. The analyses will follow

the approach of the stress-strength overlap method at the location of failure-governing stress in the key components. The theories of failure which will be applied are fracture mechanics (unstable crack expansion) and ductile modes of component failure. Most failures could add to the probability of failure to scram the reactor in a timely way or in an inability to remove post-shutdown heat. The structural failure probabilities will be factored into the overall treatment of the probability of loss of in-place coolable geometry. The elements of the analysis are scheduled for completion as follows:

Lack of significant initial flaw growth during plant life	Prelim. 6/75 Update 5/76
Substantiation under worst parametric conditions, including all environmental effects and material properties considerations	3/76
Identification of expected large critical crack sizes (expected from scoping analysis to be large)	1/76
Assessment of ductile failure mechanisms	3/76

Thus, by mid 1976, these four elements will provide adequate information to confirm the absence of structural failure potential before the cessation of parallel design activities.

1.1.2.4.2.7 Test Confirmation of Analysis

Tests are scheduled to confirm under realistic loading conditions the characteristic structural margin of the structural components. One test will consist of a destructive cyclic test of a selected heat transport component nozzle which is the most critical in the plant based on a combination of importance and structural margins. A second test article will be one critical piping segment, an elbow in the primary inlet down-comer region; this will be subjected to a similar test. This test is scheduled for completion by 3/78 in confirmation of the analysis.

1.1.2.4.2.8 Analysis of Core Disruptive Accidents

The analyses of core disruptive accidents in Appendix D to the PSAR includes two classes of events: those initiated by a transient overpower (TOP) condition and those initiated by a loss-of-flow (LOF) condition; each of these is coupled with a hypothesized failure to shut down the reactor. Since the two classes of events are predicted to have different impacts on the plant, in terms of energetics, fuel damage and radiological releases, the probabilities of each class of events will be developed. Information being developed in the reliability programs will be used to assess the relative probabilities of these classes of events by

early 1976. The interdependence of the various phenomena required to produce a core disruptive accident must be considered. For example, shutdown system failure may be less probable for LOF initiators than for TOP initiators since the required response time for the shutdown system may be much longer in the case of an LOF event and different rod redundancies exist for shutdown, TOP having the greater redundancy.

The progression of a core disruptive accident can be postulated to follow any of several potential paths, including energetic prompt disassembly, delayed disassembly and slow progression meltdown. Since the consequences of the several paths can vary considerable, the relative probabilities of the paths must be estimated. The developmental work at ANL and the HCDA analyses within the project will be used to make such estimates in early 1976.

The capability of the core support structure to cool and contain fuel in a subcritical configuration is an important factor in determining the consequences of core disruptive accidents in the reference design. The project is investigating ways to enhance the capability of the core support structure and a decision on possible changes to the reference design will be made by the end of 1975. This information along with the information on relative probabilities of different classes of HCDAs, and different progression paths will be used to assess the ability of the reference design, with its third level design margins, to accommodate a significant subset from the spectrum of core disruptive accidents. A preliminary assessment will be provided by mid-1976 and will be updated through 1977.

1.1.2.5 References to Section 1.1.2

1. Letter from L. Manning Muntzing (Director of Regulation) to John A. Erlewine (USAEC General Manager) "CRBRP Licensing Review," January 2, 1975.
2. WASH-1318. "Technical Report on Analysis of Pressure Vessel Statistics from Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service." May 1974. Regulation Staff, U.S. AEC

TABLE 1.1-1

SUMMARY OF DESIGN SAFETY APPROACH FOR THE CRBRP

This table represents the CRBRP Project Design Safety Approach.

1. The following CRBRP Design Safety Approach is generally consistent with the three levels of safety concept used by Regulatory 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 1.1-1A) 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 1.1-1A) and their inclusion in the design basis. Table 1.1-3 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 below, 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 margins and 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 in-place coolable 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
 - 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 will 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. The Project will also design features to mitigate such consequences (for example, a sealed head access area, which may or may not be inerted, an ex-vessel core catcher and other consequence limiting features). In the event that the R&D programs, discussed in 2 above, should be unsuccessful in demonstrating acceptably low probability for an event leading to loss of in-place coolable geometry, 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.

TABLE 1.1-1A

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 once or more during the life of the plant.

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 1.1-2
MULTI-LEVEL DESIGN OF CRBRP

<u>LEVEL</u>	<u>OBJECTIVE</u>	<u>OPERATING CATEGORY</u>	<u>TYPICAL EVENTS WITHIN OPERATING CATEGORY</u>	<u>OCCURRENCE PROBABILITY PER YEAR OVER 30-YR PLANT LIFE</u>	<u>TYPICAL DESIGN FEATURES</u>
1	Provision of simple, reliable and functional design free of defects, with inherent safe performance, fabricated and operated to the highest proven standards.	Normal Operation	Events which will normally occur: <ul style="list-style-type: none"> ● Full power operation ● Startup & Shutdown ● Random fuel pin failure ● Refueling 	1	<ul style="list-style-type: none"> ● Fuel assembly designed to prevent flow blockage ● Fuel of proven performance through life ● Core restraint to provide negative power coefficient ● Adequate Doppler coefficient ● Low pressure coolant systems with wide margin to boiling ● Maximum use of proven technology and hands-on maintenance ● Radioactive waste treatment system
		Anticipated Faults*	Events which, based on experience, are expected to occur at least once in the life of the plant: <ul style="list-style-type: none"> ● Loss of off-site power ● Loss of power to one pump ● Operator Error ● Spurious scrams 	$1 - 3 \times 10^{-2}$	<ul style="list-style-type: none"> ● Multiple reactor coolant loops ● Decay heat removal redundancy ● Battery power supplies for vital services ● Guard vessels for leak protection ● Inert atmosphere in Sodium Cells
		Unlikely Faults*	Events which are not expected to occur individually, but which might, based on the total list of such events, occur once during the life of the plant: <ul style="list-style-type: none"> ● Pump seizure ● Failure of dump system ● Steam Generator leak 	$3 \times 10^{-2} - \sim 10^{-4}$	<ul style="list-style-type: none"> ● Two independent shutdown systems ● Sodium water reaction protection system
		Extremely Unlikely Faults*	Events never expected to occur: <ul style="list-style-type: none"> ● Maximum flood or earthquake or tornado ● Large sodium-fire ● Large sodium-water reaction 	$\sim 10^{-4} - \sim 10^{-6}$	<ul style="list-style-type: none"> ● Site selection ● Flood barriers ● Containment isolation ● Low leakage containment
3	Provision of extra capability to cope with extremely unlikely events which are never expected to occur, and additional design requirements to provide prudent margin for unforeseen events.	Hypothetical Events	Design margins to provide for unforeseen events	10^{-6}	<ul style="list-style-type: none"> ● Capability to accept extra thermal loads in the core support structure ● Capability to accept dynamic loads in vessel and primary system components ● Geometric requirements in and around the vessel ● Control room radiological protection

* Defined in Table 1.1-1A

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TABLE 1.1-3

LIST OF EXTREMELY UNLIKELY FAULTS USED AS DESIGN BASES

Design Basis Earthquake, Flood or Tornado

Large Steam System Pipe Rupture

Sodium Fire Above the Operating Floor

Large Sodium Spills Inside and Outside
Containment

Large Na-H₂O Reactions in the Steam Generator

TABLE 1.1-4
SCRAM SYSTEM DESIGN AND RELIABILITY IN THE PSAR

<u>Section</u>	<u>Item(s) Discussed</u>
1.2.6	General Discussion of the Design
1.3	Comparison With Other Designs, in Particular, the FFIF
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 1.1-5

DECAY HEAT REMOVAL SYSTEM DESIGN AND RELIABILITY IN THE PSAR

<u>Section</u>	<u>Item(s) Discussed</u>
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 1.1-6

PRIMARY COOLANT BOUNDARY INTEGRITY TREATMENT IN THE PSAR

<u>Section</u>	<u>Item(s) Discussed</u>
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.

TABLE 1.1-7

**PRIMARY SYSTEM COMPONENTS DESIGNED TO ACCOMMODATE
ABOVE NORMAL DYNAMIC LOADINGS**

(Level 3 Design Margins)

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

TABLE 1.1-8

ACTIVITIES TO CONFIRM THE ACCEPTABILITY OF THE REFERENCE DESIGN AGAINST
LOSS OF PIPING INTEGRITY

Activity	Matl for Acceptability of Reference Design Available by 6/76	Confirmatory Material
<p>1. Identification and implementation of <u>design controls</u> including:</p> <ul style="list-style-type: none"> ● Applicable ASME, RDT design codes (F9-4, etc.) ● Applicable material standards (M5-4, M3-7T, M2-5T, M1-1,2, etc.) ● QA requirements for design, fabrication, installation and inspection (F3-6, 37, etc.) ● Operating procedures 	<p>✓ indicates material complete</p> <p style="text-align: center;">✓</p> <p style="text-align: center;">✓</p> <p style="text-align: center;">✓</p> <p>- SDD preliminary</p>	<p>Final 1/80</p>
<p>2. Identification of possible <u>initiating mechanisms</u> by:</p> <ul style="list-style-type: none"> ● FTA/FMEA ● Likelihood of undetected flaws 	<p style="text-align: center;">✓</p> <p style="text-align: center;">✓ (3/76)</p>	
<p>3. Analysis of pipe <u>fracture mechanics</u> phenomenology to show:</p> <ul style="list-style-type: none"> ● Lack of growth of initial flaw ● Penetration of flaw rather than extension ● Substantiation under worst parametric conditions of environment, stress, thermal aging, materials, loadings ● Large values of critical crack sizes 	<p>- Based on 2nd stress report 5/76**</p> <p style="text-align: center;">✓</p> <p>- Caustic environment tests 10/75</p> <p style="text-align: center;">✓</p> <p style="text-align: center;">✓</p>	<p>Final stress report 1/78</p>
<p>4. <u>Test confirmation</u> of analytical results (Based on FFTF work available and topical report by 10/75)</p>	<p>- Partial elbow tests final tests 12/75</p> <p>- Crack growth morphology 12/75</p>	<p>Welded elbow tests 3/78</p>
<p>5. <u>Reliability assessment</u> of probability less than 10^{-8}/yr</p>	<p>- Prelim 6/75 Updated 5/76</p>	<p>Final assessment of margins 5/78</p>
<p>6. <u>Leak detection</u> capability confirmed to show</p> <ul style="list-style-type: none"> ● Firm design capability ● Aerosol detection sensitivity tests ● Adequate time for remedial action 	<p style="text-align: center;">✓ } Prelim 6/75</p> <p style="text-align: center;">✓ } Tests 6/76</p>	

ACTIVITIES TO CONFIRM THE ACCEPTABILITY OF THE REFERENCE DESIGN AGAINST
LOSS OF PIPING INTEGRITY

Activity	Matl for Acceptability of Reference Design Available by 6/76	Confirmatory Material
<p>7. Safety analysis of <u>hypothetical consequences</u> to show that:</p> <ul style="list-style-type: none"> ● Leaks for large range of piping and leaks at critical point of less than 1 sq. ft. are within the core capability. ● Leaks over a certain size in limited locations are within the plant capability. <p><u>Documentation:</u></p> <ul style="list-style-type: none"> ● Preliminary report ● Final report ● Confirmatory supplement <p><u>** limiting criteria for June 1976 date.</u></p>	<p style="text-align: center;">✓</p> <p>- Preliminary rationale**</p> <p>10/75 6/76</p>	<p>Final assessment by ANL</p> <p>8/78</p>

TABLE 1.1-9

ACTIVITIES TO CONFIRM THE ACCEPTABILITY OF THE REFERENCE DESIGN AGAINST
LOSS OF IN-PLACE COOLABLE GEOMETRY

Activity	Material for Acceptability of Reference Design Available by 6/76	Confirmatory Material
<p>1. Identification and implementation of <u>design controls</u> including:</p> <ul style="list-style-type: none"> ● Applicable ASME, RDT, IEEE design codes ● Applicable material standards (M5-4, M3-7T, M2-5T, M1-1, 2, etc.) ● QA requirements for design, fabrication, installation, and inspection (F2-2, F3-6, etc.) ● Reliability requirements (F2-9T) <p>2. Identification of possible <u>initiating mechanisms</u> (FTA/FMEA) by:</p> <ul style="list-style-type: none"> ● Electronics ● Shutdown system failure modes (mechanical systems) ● Decay heat removal system failure modes ● Fuel failure propagation potential paths ● Structural failures ● Other transients <p>3. Reliability confirmation of absence of initiating mechanisms to <u>acceptable level of probability</u> (random independent failure rates) by:</p> <ul style="list-style-type: none"> ● Established goals & allocations ● Numerical assessments ● Component Tests ● Sub-system tests (FFTF scram tests completed 7/75, 6/76) ● Detailed requirements for operation, repair, monitoring, replacement and other controls necessary for reliability. 	<p>✓ indicates material is complete</p> <p style="text-align: right;">FTA 2/76</p> <p>- prelim. 3/75, update 5/76 ✓ - prelim. 4/75, update 1, 5/76 ✓</p> <p style="text-align: right;">FTA 2/76</p> <p>- prelim. 6/75, update 5/76 ✓</p> <p style="text-align: right;">✓</p> <p>- prelim. 12/74 - update 6/75</p> <p>Electronics on test 50% ongoing SHRS test plans 5/76</p> <p>- prelim. needs identified 1/76</p>	<p>Final FMEA 3-6/78</p> <p>Final of several 6/79 Electronics 12/79 SHRS tests 8/78 750/500 scrams comp. 4/78 for PCRS, SCRS - complete tech. specs 1/79 - complete operating procedures 1/80</p>

TABLE 1.1-9 (Continued)

ACTIVITIES TO CONFIRM THE ACCEPTABILITY OF THE REFERENCE DESIGN AGAINST
LOSS OF IN-PLACE COOLABLE GEOMETRY

Activity	Material for Acceptability of Reference Design Available by 6/76	Confirmatory Material
<p>4. Resolution of <u>common mode failure</u> mechanisms of significance by :</p> <ul style="list-style-type: none"> ● CMFEA within systems ● CMFEA between systems ● Identification of single failure points and their resolution or control ● Definition of margins against common mode failures by component tests ● System tests 	<ul style="list-style-type: none"> - prelim. 4/75 update 5/76 - prelim. 4/75 update 5/76 - prelim. 12/75 <p>Varied during 1976</p>	<p>Complete 12/78</p> <p>- /82</p>
<p>5. Analysis of fuel failure <u>phenomenology</u> to show insignificant likelihood of:</p> <ul style="list-style-type: none"> ● Manufacturing defects ● Propagation from stochastic failure mechanisms ● Propagation from an overenrichment error ● Propagation from local blockage mechanisms 	<ul style="list-style-type: none"> - 15.4 (PSAR) 3/75 <p>FTA with ANL review 9/75 - semi quantitative judgement 11/75</p>	<p>Review of ongoing ANL tests 1/79</p>
<p>6. Analysis of <u>structural failure phenomenology</u> for coolant boundary and support systems (vessels and core) to show:</p> <ul style="list-style-type: none"> ● Lack of growth of initial flaw ● Substantiation under worst parametric conditions of environment, thermal aging, materials, loadings ● Large values of critical crack sizes ● Assessment of ductile failure mechanisms 	<ul style="list-style-type: none"> - prelim. 6/75 update 5/76 3/76 <p>1/76</p>	<p>3/78</p>
<p>7. <u>Test confirmation</u> of analytical results (refer to 6)</p>		

TABLE 1.1-9 (Continued)

ACTIVITIES TO CONFIRM THE ACCEPTABILITY OF THE REFERENCE DESIGN AGAINST
LOSS OF IN-PLACE COOLABLE GEOMETRY

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Activity	Material for Acceptability of Reference Design Available by 6/76	Confirmatory Material
<p>8. Analysis of <u>core disruptive accident</u> phenomenology to show:</p> <ul style="list-style-type: none"> ● Relative probabilities of potential initiators (LOF versus TOP, etc.) ● Estimated relative likelihoods of accident mechanistic paths leading to energetic disassembly or slow meltdowns ● Assessment of capability of reference design to accommodate a significant proportion of all CDAs <ul style="list-style-type: none"> - Enhancement of core support structure thermal capability ● Assessment of consequences of limited number of transients beyond plant capability 	<p>- Prelim. (Appendix D, PSAR 3/75)</p> <p>1/76</p> <p>3/76</p> <p>6/76**</p> <p>12/75</p> <p>6/76**</p>	<p>Update 12/76</p> <p>} Update 12/76-12/77</p>
<p><u>Documentation</u></p> <ul style="list-style-type: none"> ● Preliminary report ● Final report ● Confirmatory test supplements 	<p>5/76</p>	<p>7/78</p> <p>6/82</p>

** limiting criteria for May 1976 date

TABLE 1.1-10

ACTIVITIES FOR: IDENTIFICATION OF FAILURE MODES
RELIABILITY CONFIRMATION
CMF ABSENCE

Identification of Failure Modes

Shutdown System

Preliminary FMEA	3/75
FMEA Update	5/76
FTA	2/76
Preliminary CMFA	6/75
Updated CMFA	5/76

Decay Heat Removal System

Overall System FMEA	1/75
Individual System and Selected Component FMEA	- Prelim. 9/75
	Updated 5/76
Individual & Selected Component FTA	- Prelim. 2/76
Individual & Selected Component CMFA & SPFA	- Prelim. 5/76

Structural Failures

Selected Structural Component FMEA	- Prelim. 6/75
	Updated 5/76

Reliability Confirmation

Shutdown System	- Prelim. 10/74
	Updated 6/75

Decay Heat Removal/Structural	- Prelim. 12/74
	Updated 5/76

Testing

Shutdown System	
Begin Electrical System Test	3/76
FFTF System Test Complete	7/75
Secondary Flow & Latch Test Complete	3/76
Decay Heat Removal/Structural Component Tests	- Plan 3/76
	Tests 8/78
Detailed Operating Requirements	5/76

CMF Absence

Shutdown System	- Prelim. 6/75
	Updated 5/76

Decay Heat Removal /Structural Events not addressed by PPS	- Prelim. 5/76
	- Prelim. 5/76

1.1.3 Applicability of Regulatory Guides

This section 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 was 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.

A percentage-rating scale has been 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.

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 CRBRP)
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 (Caldwell) 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%		The intent of this Guide is consistent with GDC 55 and GDC 56 of the Proposed CRBRP GDC. However, for the current design selections of the CRBRP, there are no instrument lines penetrating the containment.
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		REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE)
		INTENT	DETAILED PROVISIONS	
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)	100%	0.0	For the bottom liner in the concrete base, ASME-III, Division 2 provisions will be followed. (Note: This is so in order to be consistent with E-Spec)
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, 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%	0.0	This Guide is mainly to concur on the requirements as set forth in ANSI H45.2.11 (Draft No. 3, Rev. 1, July 1973). The intent is applicable. For the detailed provisions the CRBRP QA program will be followed.
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 SFAC.
1.30	Quality Assurance Requirements for the Installation, Inspection, & Testing of Instrumentation & Electric Equipment (formerly Safety Guide 30)	100%	0.0	The intent is applicable. For the detailed provisions, the CRBRP QA Program will be followed.
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%	0.0	The intent of this Guide is applicable. For the detailed provisions, the CRBRP QA Program will be followed.

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%	<p>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.</p> <p>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.</p>
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	<p>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.</p> <p>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.</p> <p>Because of the above reasons, this Guide is not rated.</p>
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, & Handling of Items for Water-Cooled Nuclear Power Plants (3/16/73)	100%	0.0	<p>The intent of this Guide is consistent with Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".</p> <p>For the detailed provisions, the CRBRP QA Program will be followed.</p>

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%	0.0	The intent of this Guide is consistent with Appendix B to 10 CFR 50. For the detailed provisions, the CRBRP QA Program will be followed.
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.

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	
1.45	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.
1.46	Protection Against Pipe Whip Inside Containment (5/73)	100%	0.0	The detailed provisions of this Guide are largely not applicable to an LMFBF plant. 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%	0.0	The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed.
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%	0.0	The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed.
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)	100%	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%	0.0	The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed.

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%	0.0	The intent of this Guide is applicable.
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 operability 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 Pressurized Water Reactor Steam Generator Tubes	0.0	0.0	This Guide applies only to PWRs. It is not applicable to the CRBRP.

ADDENDUM B

RELIABILITY PROGRAM

C.1.0 Purpose

The purpose of this Appendix is to describe the the CRBRP Reliability Program, which provides the means for assisting in the determination of which events should be included or excluded as CRBRP design basis events. Additionally, the program provides continual assessment, verification, and design control to assure that the CRBRP Reference Design is a satisfactory basis for licensing.

C.1.1 Introduction

Appendix C contains three basic sections. Section C.1 provides a summary of the material presented in this appendix. This section also provides the development of the overall reliability criterion and goals as well as the essentials of the plan by which technical, and scheduler objectives are achieved.

Section C.2 outlines the reliability methodology utilized for this program and provides the current reliability assessments.

Section C.3 presents the basic aspects of the planned verification process which includes additional development activity, a confirmatory test program and key milestones to be met, as well as a description of available test facilities.

C.1.1.1 Definitions of Terms Used in this Appendix

Reliability

Characteristic of an item expressed by the probability that it will perform a required function under stated conditions for a stated period of time.

Unreliability

Numerical compliment of reliability.

Availability

Characteristic of an item expressed by the probability that it will be operational at a selected future instant in time.

Unavailability

Numerical complement of availability.

Common Mode Failures

Multiple failures which result from a single initiating independent random cause such as a common property, common process, common environment, or common external event. As such, they form an important sub-set of the range of independent random failures.

Redundancy

The performance of the overall function by two or more independent means.

Diversity

Performance of the same function by two or more different and independent means.

Random Independent Failure Rate

The expected number of failures of a given type in a given time interval, wherein each failure is mutually independent of the remainder of the failures.

Safe/Unsafe/Failures

Safe failures are those failure events which do not affect the ability to perform the safety function when required. Unsafe failures are those failure events which can degrade the safety function.

Mean Time to Failure

Arithmetic mean of the times to failure.

C.1.2 Summary

The overall design safety approach for CRBRP is described in Section 1.1 of the PSAR. This approach will assure that the plant meets the requirements set forth in applicable Federal Regulations. The elements of the overall approach addressed in this appendix are:

- To identify those extremely unlikely events having the potential to exceed 10 C.F.R. 100 guidelines
- To confirm through assessment design and confirmatory analysis and testing that all such events are of sufficiently low probability to justify exclusion from the CRBRP design bases.

Applying this approach has resulted in the identification of events which should be included in the CRBRP design basis and those which should be excluded. The events included in the design basis are identified in Chapter 15 of the PSAR. Analysis of those events and the assurance that they are conservatively accommodated by the CRBRP design is provided in the appropriate chapters.

Those events identified as having the potential to exceed 10 C.F.R. 100 guidelines and which have not been included in the plant design bases are treated in this Appendix. The basis for exclusion consists of a logical process of establishing criteria, allocating goals, performing conservative assessments and implementing testing, design and analysis activities as required. These activities are pursued using traditional reliability methodology to achieve a high level of confidence that predetermined goals are met.

The overall goal established for the Reliability Program is:

- The probability of exceeding guideline values shall be less than one chance in a million per reactor year.

Initial reliability assessments indicate that the CRBRP will meet this goal. The details of and bases behind this goal, its allocation and the assessments are presented in Section C.2 of this appendix.

The Reliability Program furnishes design verification and confirmatory data through a test program which utilizes accepted reliability engineering methods to assure: proper tests are conducted; proper selection of test articles; and identification of those components and systems whose failure modes are most critical to plant reliability and safety. The program thus assures that when built, the plant meets the objectives of the overall design approach described in Section 1.1 of the PSAR.

C.1.3 Reliability Program

This section provides a description of the Reliability Program and associated activities necessary to successfully complete its mission. The criterion to judge which events are to be treated by the Project and indeed to judge success and failure of the program is discussed in C.1.3.1. A discussion regarding the determination of the events which may have the potential to exceed dose guidelines and therefore are of interest to the program is presented in C.1.3.2. Having established a success criterion and the events to be measured against it, reliability allocations are made which must be achieved by each system contributing to success. This allocation is discussed in C.1.3.3. The Reliability Program plan which implements the methodology necessary to assure that the overall reliability goal is achieved is discussed in C.1.3.4.

The basic objectives of the Reliability Program are:

- To identify those extremely unlikely events having the potential to exceed 10 C.F.R. 100 guidelines.

- To confirm through assessment, design and confirmatory analyses and testing that all such events are of sufficiently low probability to justify exclusion from the CRBRP design bases.

The Reliability Program elements are oriented heavily toward the first two levels of design to ensure that adequate reliability is included as an integral part of the plant design. In addition, it is an established CRBRP design philosophy to utilize to the maximum extent possible proven components and subsystems to minimize developmental reliability problems. For example, the shutdown system monitor and control electronics are essentially duplicates of the FFTF system in the areas of component types, subsystems, and suppliers. Where new components or subsystems must be used, applicable experience and data from closely comparable elements will be used to the maximum extent possible.

C.1.3.1 Reliability Criterion

In the absence of specific guidelines for the assignment of events into an LMFBR design basis envelope, the literature on reliability safety assessment was researched. Several sources were consulted, including principal References 1-6. It was recognized that the applicability of these and other documents to LMFBR's and, in particular, to the CRBRP varied. Each was reviewed to provide guidance in the selection of a reliability criteria, which could be used to meet the objective of the Reliability Program. The most comprehensive and relevant treatment of the subject was presented in Section II of Reference 1.

Based on the evaluation of this literature and the first-of-a-kind nature of this plant, the Project concluded that assurance against low probability accidents for the CRBRP should be as stringent as other commercial power reactors. Accordingly, the Project reliability criterion is based upon the overall safety objective that:

The likelihood of exceeding guideline values should not be greater than one chance in one million per year.

C.1.3.2 Event Analysis

The second stage of the Reliability Program was to identify events which appear to have the potential to exceed 10CFR100 guidelines, and to compare the probability of such occurrences with the reliability criterion established above (C.1.3.1). Based on the results of this comparison, judgment was made on those events which should be included in the CRBRP design basis.

Events which have the potential to exceed 10CFR100 guidelines involve those components and systems of the plant which have large inventories of fission products and transuranium elements. These sources were identified and they include: the reactor core, the refueling machine, the waste gas storage system, the liquid waste storage system, primary sodium itself and the ex-vessel fuel storage tank. Of the events involving these sources, those associated with the refueling machine, the waste gas storage system, the liquid waste storage system and the primary sodium were determined not to exceed the guidelines of 10CFR100. Analyses to confirm this conclusion are presented in Sections 15.5, 15.6, and 15.7 of this PSAR.

The remaining events are those associated either with the reactor core, or the ex-vessel fuel storage tank. An initial assessment indicated that the events associated with the ex-vessel storage tank, e.g., a leak or loss of cooling, would not result in consequences that are in excess of 10CFR100 and furthermore, such events are very improbable. This initial evaluation is being updated to confirm that the design can accommodate such events or to confirm their extremely low probability. The initial analyses for these events is presented in Section 15.6.

The final source having the potential to exceed 10CFR100 is the reactor core. In order for these events to lead to excessive releases of radioactivity, it would be necessary to violate the three containment barriers; i.e., the fuel cladding, the reactor vessel and the reactor containment. There are only two identifiable events which could lead to sequential violation of these three barriers. They are either an inadvertent energy release in the primary system, or a sustained temperature within the primary system higher than that which ensures primary boundary integrity. Both of these events are consequences of a loss of in-place coolable geometry which could extend to widespread sodium boiling and fuel melting. A detailed Fault Tree Analysis (Figure C.141) has been constructed to identify initiators which potentially lead to loss of in-place coolable geometry of the core. This initial analysis identified four initiators:

- Total loss of Heat Removal Capability to the Core following scram
- Transient without Scram
- Transient beyond the capability of plant protection system
- Assembly-to-Assembly Failure Propagation.

These initiators were thus included for evaluation by the Reliability Program.

C.1.3.3 Initial Goals and Allocation Model

Based on the evaluation of initiators discussed above, it has been determined that the event which has potential to exceed 10CFR100 guidelines is loss of in-place coolable geometry.

The criterion which was established for the probability of exceeding 10CFR100 guidelines can be related to the probability of loss of in-place coolable geometry, considering the combination of all the following probabilities:

probability of loss of in-place coolable geometry, and

probability that loss of in-place coolable geometry leads to breach of the PHTS and the containment (either by physical damage or by excessive leakage), and

probability of excessive activity release to the environs, and

probability that this release leads to a radiological dose at site boundary greater than 10CFR100 guidelines

While it is certain that the last three of these probabilities are each less than one, credit for this has not been taken at this time. Thus, the criterion established in Section C.1.3.1 is conservatively modified to relate directly to loss of in-place coolable geometry rather than the potential to exceed guideline values and becomes:

The probability of loss of in-place coolable geometry in the core shall be less than one chance in a million per reactor year.

The remainder of Appendix C uses this as the reliability criterion.

Selection of test articles in quantities for reliability confirmation is based on the initial evaluation and identification of those components and subsystems, and their attendant failure modes which are the most critical to plant reliability and safety and for which the least data is available. Test modifications or additions will be recommended as necessary such that specific potential failure modes will be fully explored during the test. In addition, the designer will make specific recommendations concerning the test of certain operational characteristics to add to the verification of specified design criteria. This is the basic philosophy under which the CRBRP test planning is done, thus assuring maximum results from any test to be conducted.

It is expected that the test results will confirm that the pre-established goals can readily be met, particularly since the first assessment is quite conservative and is based on a first case consideration of all contributors to the goals. Testing will also provide initial information on the variation of failure rate with lifetime ("bath tub" characteristics of failure) for the selected components. In particular, it will add to the evaluation of component and subsystem failures which occur at the onset of the "bath tub" characteristic such that provisions will be made to assure that essential component and subsystem failure rates will have reached constant value. Projection of wear-out failures will be included in maintenance and replacement planning to ensure replacement or preventive maintenance before actual failure can occur. This approach to test program utilization provides a high level of confidence in the assessments for CRBRP.

Use of Reliability Engineering in the confirmation process maximizes the assurance that when built, the plant will meet the objective associated with the three levels of design and accompanying assurance of a maximum level of public safety. The approach described above, involving analysis and testing, is based upon accepted practice in the aerospace industry and sensitive industrial activities. Development of the applied methodologies occurred in defense and space programs and have been "tried and proven" for these kinds of programs. Thus, maximum advantage is being taken of developed methodologies to assure reliable and safe operation of CRBRP.

Common Mode Failure

The potential for common mode failures will be identified by a detailed Common Mode Failure Analysis together with rigorous Failure Mode and Effects Analysis (FMEA) and Fault Tree Analysis (FTA). Determination of common mode failures will be by considering functional dependency, parts of similar manufacture, environmental causes, operating and maintenance errors, input and interface parameters, and failures induced by a preceding failure. The analyses program is also designed to provide design changes for identified common mode failures. These include design diversity, diversity in component fabrication-procurement sources, enhanced testability, reduction in the conditional probability of common mode failures after a casual event has occurred, and stringent procedures designed to eliminate human error in the design, analysis, operation, and maintenance of the CRBRP. The subsystem and system tests within the Reliability Program permit an empirical search for common mode and single random failures to complement the failure mode and effects and fault tree analyses. Special tests will be performed under abnormal conditions to verify the absence of common mode failure mechanisms. These tests will be designed to assist in the identification of potential common-mode failures related to:

- a) internal and external environments
- b) design deficiencies
- c) functional deficiencies
- d) operating and maintenance deficiencies

The components test program will verify that common mode failures will not be introduced via design deficiencies at the component level.

Testing within the SHRS Reliability Program will have the same objective, namely to help define potential common mode failures. However, practical limitations within the heat removal program prevent common mode failure exploration in the laboratory to the same extent as planned for the shutdown system. In addition, the Steam Generator Development Program (PSAR Section 1.5) will yield important common mode failure data relating to steam generator tubes.

Additional details concerning the methods to be employed in the area of common mode failures are provided in Section C.3.1.1.2.

Confidence Limits

The method of computing confidence limits in the reliability programs will vary from subtask to subtask. At one end of the spectrum are the analyses, like the total shutdown heat removal reliability assessment, which are based on data of varying degrees of accuracy from many sources. In these cases, the situation is not amenable to large sample tests to determine confidence interval and require more innovative treatment. At the other end of the spectrum are certain subassembly tests in the shutdown system program in which enough large sample test data will be available for conventional statistical reduction. Between the two extremes are situations, e.g., major assembly evaluations within the shutdown systems portion of the reliability program, in which confidence limits will be determined using applicable statistical techniques such as Bayesian, (See Ref. 7 and 8) selected prior information and partially from data generated within the present program. More detail regarding confidence limit determination is provided below, with emphasis on the approach being initiated in the Shutdown Heat Removal System reliability analysis.

The current assessment is a point estimate. A point estimate does provide a valid engineering assessment of the quality of the system and it identifies areas which require special attention. For more refined future assessments, a confidence interval will be calculated. The method of calculation will be based on uncertainties about the mean failure rates. Appropriate techniques will be applied to develop a probability density function where the probability density function directly yields a confidence interval or probability band.

It must be recognized that a definition of confidence limits for components/systems with very high reliability objectives does involve some practical difficulties, as acknowledged in Ref. (8). However, confidence limits have an obvious usefulness in decision making. Therefore, confidence limits will be attached to the experimental and analytical results from the reliability program by best engineering utilization of the methods available from probabilistic analysis state-of-the-art and the data which is available.

C.1.4. Goals (Numerical)

Initial numerical reliability allocations have been made which are consistent with the overall goal and which provide the designer with a realistic but 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.

C.1.4.1 Shutdown System Goal

For the shutdown systems the success criteria is addressed by a goal to provide a design such that the unavailability of the two systems is less than:

Primary	$< 10^{-4}$
Secondary	$< 5 \times 10^{-4}$

for transients which, without scram, could result in loss of in-place coolable geometry.

The above goal is based on the objective of less than 10^{-7} SDS failures per year that could lead to loss of in-place coolable geometry. For the ideal case of complete elimination of common mode failures between the Primary and Secondary Shutdown Systems, the above goals would provide an unavailability of less than 5×10^{-8} . However, it must be recognized that unavailability is associated with the probability of failure per challenge. Thus to determine the probability of failure per year one would have to account for the number of challenges expected.

An estimate of the number of times that the protection system is challenged to prevent loss of in-place coolable geometry per year is determined by:

- Defining which duty cycle events cause the hot channel sodium temperature to exceed 1700°F in 10 minutes or less without protection;
- Summing the number of occurrences of these events over the plant lifetime; and
- Dividing the total by 30 to determine the average number of Challenges per year.

This approach is based on the following considerations:

First, the onset of sodium boiling in the hot channel is not a sufficient condition for loss of in-place coolable geometry; further, all primary flow events which result in pressure decreases that permit sodium boiling below 1800°F are included in the "must scram" category. (Note that the sodium boiling temperature at pony motor flow conditions is >1720°F). For these events, the 1700°F criterion is conservative for determining scram speed of response adequacy, since primary pressures are sufficiently high to prevent boiling until after the sodium temperature has exceeded 1750°F. For all events involving power changes and intermediate or steam side perturbations, the use of 1700°F as the temperature for onset of boiling provides significant margin since the boiling temperature in the core exceeds 1800°F for all of these events.

Second, operator reaction to significant disturbances which affect multiple critical parameters would be expected in less than five minutes. However, to assure conservatism in the analysis, a delay of ten minutes has been assumed. The only operator action necessary is to manually depress the scram button which is located in the control room. Operator action required does not include finding a means for inserting stuck rods. The probability of all rods (in both systems) failing to insert after a manual scram will be calculated as part of the common mode failure analysis. If this probability is not sufficiently small, corrective action will be initiated. Minimum combined system rod insertion requirements for manual scram in the above category will be addressed in the next iteration of the reliability assessment.

Third, the duty cycle overstates the total number of events to assure adequate thermal transient design.

Based on this approach, ~2 challenges per year are specified for the shutdown system. These two challenges arise from the following two sources:

- (1) The upset events listed in Table C.1-1.
- (2) Five emergency events over the life of the plant (provided in Appendix B).

Of the upset events in Table C.1-1, the following events result in overcooling of the core prior to scram and do not result in a potential challenge to the Shutdown System from the standpoint of losing coolable core geometry: U1, U2a, U2e, U7, U13, U15, U20 and U21b. No further

consideration of these events is required because the event plus postulated failure of the Shutdown Systems does not result in reaching a 1700°F hot channel sodium temperature. Events U8, U9, and U17, all assume scram action is part of the initiating sequence and therefore pose no challenge to the Shutdown Systems. Event U5a does not affect the reactor temperatures because the redundant feed pump is automatically started which prevents significant sodium temperature changes. Event U2d and U2f are defined to terminate without reaching full core ΔT .

The following events result in an increasing primary cold leg temperature in one loop due to a partial or complete loss of heat removal capability through one loop: U4a, U4b, U10, U11, U12, U14, U19, U21a, U22 and U23. Any heat removal loss in the Steam Generator System requires ~ 20 seconds for the resultant temperature wave to travel from the steam generator to the IHX in addition to the time required for the event to cause changes in the intermediate cold leg temperature. Since the unmitigated intermediate pump coastdown event causes essentially the same magnitude of change as primary hot leg temperature without the 20 second delay, all of these events are enveloped by the coastdown event. For the intermediate pump coastdown, the vessel inlet temperature rises one third of the loop ΔT in approximately 1 - 1.5 minutes due to the increase of one primary cold leg temperature to approximately hot leg temperature conditions after the pump has coasted down. This results in a vessel outlet temperature rise of ~ 70 degrees (and a 70°F hot channel sodium temperature rise). After 10 minutes the inlet temperature has risen less than 150°F. This results in a hot channel sodium temperature of less than 1500°F at approximately 10 minutes. Since an intermediate flow, three separate primary cold leg, and three separate primary hot leg temperature alarms and all precision meters have been over limits for at least three minutes, operator action in 5 minutes or less is reasonable. Further, no reactor control action has been postulated which would significantly reduce the temperature reached and whose action is independent of the pump failure. Therefore, these events do not exceed the sodium boiling limit within 10 minutes.

Event U3a involving a partial loss of primary flow in one loop results in a maximum hot channel sodium temperature of less than 1450°F since the core flow is only reduced to 90%. The other events (U2c, U2b (with unlimited power increase), U3b, U5b, U6, U16 (with unlimited power increase) and U18) may result in exceeding a 1700°F hot channel sodium temperature in less than 5 minutes.

The rationale for including five emergency events in the plant duty cycle is provided in Appendix B. The sum of the frequencies of the events as given in Table C.1-1 plus the additional five emergency events is sixty-six over the thirty year plant life. Since the individual event frequencies are conservative, a conservative estimate of the frequency per year is ~ 2 .

The Secondary Shutdown System availability goal has been made less stringent than the Primary for the following reasons:

- To provide design flexibility for increased diversity in order to minimize common-mode failures while accepting a potential reduction in random failure availability. The provision of diversity involves use of a design concept which may lie at a different point on the experience curve than the concept employed in the Primary System.
- The Secondary Shutdown System does not have an increasing rod redundancy over an operating cycle (i.e., does not have an increasing number of rods withdrawn at full power - hence more to insert during scram) as occurs in the Primary System. Consequently, the system availability over a given period between tests, for a fixed individual rod availability will be lower for the Secondary System.

In (Ref. 1), the Regulatory Staff discusses at length the reliability of current shutdown systems based on available data. It is shown that on the basis of operating experience from 228 reactors around the world, an unavailability of $\sim 10^{-4}$ (based on monthly testing) can be deduced for state-of-the-art single shutdown system reactors. This figure is based on common mode failures within the system. Since this level of reliability has already been estimated for reactors with single fast acting shutdown system reactors, similar results can be achieved for each of the two independent CRBRP shutdown systems using good design and reliability engineering practices. It is recognized that common mode failure between the two systems must be addressed. However, because of the diversity, redundancy, and physical separation of the two systems (both electrical and mechanical), one can reasonably expect that the potential for common mode failure of both systems is significantly less than that for common mode failures within single systems already shown to have unavailabilities of less than 10^{-4} .

To provide confidence beyond that gained from light water reactor field experience that the CRBRP shutdown system can achieve the stated goals, a preliminary estimate of the failure probability has been performed for the dual shutdown system of CRBRP, and is described in Section 3.1.

The numerical goals specified are consistent with the philosophy presented in this Section. Reliability predictions using the failure mode and effect and fault tree analyses coupled with an extensive confirmatory testing program at the component, subsystem, and system level will provide data necessary to substantiate achievement of the specified goals.

An allocation of the Primary and Secondary Shutdown System random independent failure goals has been made for the electrical and mechanical subsystem.

	<u>Unavailability Goal</u>
Primary Shutdown System	1×10^{-4}
Mechanical	7.5×10^{-5}
Electrical	2.5×10^{-5}
Secondary Shutdown System	5×10^{-4}
Mechanical	3.8×10^{-4}
Electrical	1.2×10^{-4}

Suballocations within the subsystems will be made by the responsible design groups to assure achievement of these goals.

C.1.4.2 Shutdown Heat Removal System Goal

The goal of the Shutdown Heat Removal System is to confirm that the probability of loss of in-place core coolable geometry due to failure to remove post-shutdown heat is less than $\sim 8 \times 10^{-7}$ per reactor year. In achieving this goal, analysis and testing of the Shutdown Heat Removal portion of the Reliability Program will cover those components whose failure would lead to lack of adequate core cooling following shutdown and for which failure-related data are judged to be most critically needed.

The heat removal systems reliability assessment based on analyses to date is presented in Section C.2.2. This assessment provides reasonable assurance that the reference design meets the stated safety objective. The activities of analysis and testing will be described in detail in Section C.3.2.

Three examples of development programs external to the reliability program which will provide data of direct interest are the steam generator development program, the bellows testing portion of the IHX development program, and the reactor vessel outlet plenum mixing test which will assure adequate cooling capability of the overflow heat removal service. Data from these activities will be incorporated into the reliability analysis as it becomes available.

C.1.5 Current Reliability Assessment

The current reliability assessment concludes that the plant meets the overall objectives presented in Section C.1.1. It is important to recognize that this conclusion is not based solely on quantitative analysis. The basis for the conclusion comprises four major elements:

1. The quantitative assessment based on available reliability methodology and hardware reliability information, as presented in Section C.1.4.1
2. The qualitative reliability activities within the Project which impose a systematic and disciplined method of plant design. This approach serves to minimize the likelihood of design oversights, and in particular to identify common mode failure potential. These activities are discussed in Section C.1.4.2.
3. The presence of redundancy and diversity in essential design features in the systems of interest. These aspects of the design are described in Section C.1.4.3.
4. Capability to incorporate design and procedural changes to enhance reliability. The integration of reliability in the design decision process is detailed in Section C.1.4.4.

C.1.5.1 Numerical Assessment

In this section, the results of the initial numerical assessments are presented. The current assessments indicate that the goals will be achieved in the CRBRP.

C.1.5.1.1 Shutdown System

An initial reliability assessment of the Shutdown Systems has been completed and is described in detail in Section C.2.1. Results of this analysis provide confidence that the SDS meets the numerical reliability goals associated with the prevention of loss of in-place coolable geometry, which in the analysis is conservatively represented by prevention of sodium boiling.

This assessment shows that the combination of both primary and secondary shutdown systems will more than adequately meet the goal of $<10^{-7}$.

Table C.1-2 summarizes the results of the analysis and compares the allocations to the various subsystems with the current numerical assessments for those subsystems.

C.1.5.1.2 Shutdown Heat Removal System

The results of the initial reliability assessment, presented in detail in Section C.2.2, of the shutdown heat removal system is that the probability of loss of in-place coolable geometry due to failure of that system is 4×10^{-7} per reactor year. This result is consistent with the preliminary allocation discussed earlier of 8×10^{-7} for shutdown heat removal. For purposes of the shutdown heat removal reliability, a criterion of failure more conservative than loss of in-place coolable geometry was used as a limit. The criterion which was applied was that sodium bulk temperature within the reactor vessel should not exceed 1250°F. This temperature would not produce a loss of in-place coolable geometry, but it represents a lower bound of a temperature above which long term integrity of the primary system is not assured. Analysis is continuing to confirm the acceptability of this limit. However, work to date establishes that the 1250°F limit is technically acceptable for the time required for shutdown heat removal. Analysis is underway which is expected to justify a higher temperature limit. Operation of portions of the heat removal equipment at this temperature is, of course, treated as a faulted condition, therefore some of the equipment may not be reusable after being exposed to the 1250°F environment during a given shutdown.

C.1.5.2 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 design oversights and consequences. These activities guide the designer 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. These analyses will be utilized to indicate areas of special concern (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.

C.1.5.3 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.

C.1.5.3.1 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 failures will not cause degradation of protection provided by that system. 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 not 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. Table C.1-3 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.

C.1.5.3.2 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 service [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 diverse in their functioning are available. The sinks are the main condenser, the safety relief valves and stored water 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. Complete redundancy does not exist among these components because of such things as common piping runs.

- b) Within the steam/water system, two sources of stored feedwater are available, as well as a main and an auxiliary feedwater pumping system. The auxiliary feedwater system has both motor-driven and steam turbine-driven pumps which add diversity.

A redundant and diverse path to the heat transport loops for decay heat removal is the OHRS. The OHRS utilizes a liquid metal-to-air heat exchanger and is therefore diverse in this important regard from the sodium/water interface in the normal heat transport paths.

- a) Within OHRS, all pumping is by electromagnetic pumps, diverse from the mechanical pumps in the heat transport system loops.
- b) At identifiable times after scram, the OHRS 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.

C.1.5.4 Implementation of Reliability Requirements into Design

Design, fabrication, assembly, and operation are the controlling factors in attaining the desired level of safety and reliability in any complex technical undertaking. Since these factors have a major impact on final system reliability, safety and reliability principles must be included at each step and at each level of detail. This section describes the measures being implemented within the project to provide total reliability assurance (availability aspects of reliability as well as the safety-related aspects being emphasized in this appendix) consistent with the appropriate provisions of RDT Standard F2-9T, "Reliability Assurance".

To ensure that the objectives and goals of the Reliability Program together with the foregoing design/reliability interaction tenets are implemented within the design process, the CRBR Project has evolved design and reliability procedures which:

- 1) Focus engineering and management attention on the requirements of reliability,
- 2) Ensure that reliability is treated as a design factor of equal importance with other performance factors by a close collaborative effort between design and reliability engineering personnel on a day-to-day basis,
- 3) Alert management, as well as designers, throughout the program to all reliability discrepancies that may require management decisions.

Within the project, an effective monitoring program exists which assures specific management cognizance and approvals to prevent an inadequate design proceeding into development, test, and production.

The specific means by which these objectives are realized and reliability requirements implemented in design are described below.

Two major functions are involved in implementing the Reliability Program:

- 1) a verification function to assure the program is effectively implemented
- 2) integral design engineering activity to assure reliability is incorporated into the product during the course of component design, fabrication, assembly, and operation.

In implementing both functions, maximum use is made of existing Quality Assurance procedures and organizations. The intent is to avoid duplication of prior effort or of personnel required to perform similar functions, and thus to minimize the administrative burden of the program. Specific implementation is being carried out as identified below.

- 1) Reliability Assurance activities are conducted in parallel with Quality Assurance activities during the project design, development, and testing phases.
- 2) Reliability Assurance documentation requirements are specified and incorporated into existing Quality Assurance documentation wherever feasible.
- 3) Engineering holds are required for reliability reasons and are incorporated into the existing engineering hold procedures.
- 4) Reliability Assurance audits are performed in a manner similar to Quality Assurance audits.
- 5) Reliability design review requirements are part of existing design review procedures.

C.1.5.4.1 Component Reliability Control Policy

The objective of the component reliability control policy is to make certain that each component developed for the CRBRP, in all stages from conceptual design through operation in the plant, meets established reliability requirements. A central principle in defining the practical details of the policy is that virtually all quantitative reliability analysis be performed under the direct control of the principal organizations participating in the project. The assurance that CRBRP mechanical components meet quantitative reliability requirements will be based on effort controlled by reliability engineering personnel within the principal organizations participating in the design of the plant. Some electrical components have traditionally been specified to meet analytically determined numerical reliability requirements and this practice will continue. The responsibility for ensuring that the equipment meets reliability requirements and conforms to the equipment as modeled in system reliability assessments resides with the design organization.

Suppliers of electrical and mechanical equipment designated on the reliability critical items list will be required in some cases to provide quantitative failure modes and effects analyses (FMEAs). Furthermore, suppliers may, on a selective basis, be requested in separate negotiations to provide other reliability analyses, qualitative or quantitative, depending on the nature of the component and reliability engineering skills within the supplier's organization. However, the bulk of the quantitative reliability analyses will be performed by the major plant design organizations.

C.1.5.4.2 Administrative Controls

The administrative controls which will be implemented to assure reliability goals are met by each piece of equipment and therefore by the plant are:

- 1) Preparation and implementation of program requirements which appropriately assign reliability assurance responsibilities to guide project management in reliability-related decisions.
- 2) Participation of reliability personnel in all design reviews to determine if specific criteria established for the system/component are met (with the level of conformance demonstration appropriate to the progress to date on the design, analysis, and testing of the component under review). These criteria, which must be incorporated into appropriate specifications and/or SDDs, may include completion of a preliminary FMEA, proper specification of reliability requirements (including definition of any faulted conditions which are presumed to be acceptable in safety-related reliability analyses), adequate forethought in the design and the component's maintenance/inspection plan, and appropriateness of test content and planning for support of the reliability objectives.
- 3) Reliability review and approval of Type 1 submittals (defined as equipment specifications, designs, test documents requiring RRD approval) for reliability critical equipment. Points of emphasis will be those listed for design reviews under Item 2 above.
- 4) Indoctrination and training to provide an introduction and guide to reliability objectives, standards, and practices. This tutorial function will include management product assurance, engineering, and equipment supplier personnel as appropriate.
- 5) Development of reliability analysis procedures for uniform application within the project (see Section C.3.1.1.1).
- 6) Preparation of a Reliability-Critical Items List of those items whose failure could directly affect loss of in-place coolable geometry. This task includes preparation of criteria for entries in this list. Priorities are established to focus management attention on a few entries which could have the greatest influence on loss of in-place coolable geometry.

- 7) Enforcement, through project management, of timely reporting and adequate analysis of component failures in test or operation. Corrective actions to prevent like failures are recommended. These may include item redesign, modification of the reliability assessment as influenced by the component failure rate, definition of additional testing, or modification of operation and maintenance procedures. Completion of the corrective actions are monitored.
- 8) Review of test planning to assure inclusion of support for reliability objectives. Tests beyond those already planned as development, qualification, and acceptance tests will be recommended only if additional tests specifically formulated for obtaining reliability or maintainability information are essential.
- 9) Performance of reliability audits to verify the implementation of a Reliability Program Plan.

C.1.5.4.3 Content of Specifications

The content of specifications regarding reliability will be as follows:

- 1) A requirement that well-defined supplier organization responsibilities for reliability assurance are assigned wherever design and analysis are the responsibility of the supplier.
- 2) Emphasis on design provisions for equipment access, inspection, and repair.
- 3) Preparation of an FMEA according to an established project procedure. When component design and analysis are not the responsibility of the supplier, this requirement may not be in the specification but will be assigned to the appropriate agency.
- 4) Provisions for supplier completion of other requested reliability qualitative and quantitative analyses to be specified and negotiated separately.
- 5) Timely and complete reporting of equipment failures in test or operation, including analysis of the failure and recommendations for corrective action.

C.1.5.4.4 Implementing Documentation

The top level implementing document is the Reliability Assurance portion of the Management Policies and Requirements document, which is presently in the process of review and approval. This Reliability Assurance portion reflects the elements of the Project Component Reliability Control Policy spelled out in Section C.1.4.4.1.

The next level document is the Reliability Program Plan. This plan describes the organizational structure, functional responsibilities, and lines of communication for the effective management and execution of the Reliability Assurance task. The plan covers all activities to include

the means for management visibility of the effectiveness of these activities, which are necessary to assure that reliability objectives are met throughout all phases of the CRBRP contract performance. The plan covers the following items:

- Management tools (Reliability Assurance Index, Reliability-Critical Items List, Engineering Hold Plan)
- Specific concentrated reliability programs (Shutdown System and Shutdown Heat Removal System Reliability Program)
- More broadly applicable reliability design and analysis tasks
- Reliability testing
- Specification controls (SDD and engineering specification content)
- Failure analysis and reporting
- Monitoring procedures

The next lower level of implementation is the generation of detailed engineering or product assurance procedures. Existing project procedures will be modified or augmented as required to specify all necessary reliability/maintainability aspects.

The lowest level implementing document is the Reliability Manual for Liquid Metal Fast Breeder Safety Programs (hereafter called Reliability Manual - see Section C.3.1). This document when complete will include detailed instructions on all aspects of reliability engineering methodology, such as reliability apportionment, failure modes and effects analysis, fault tree analysis, common mode failure analysis, Bayesian priors, testing planning (including accelerated testing and data sensing), and structural reliability analysis. This document, for which GE-FBRD has lead and coordinating responsibility, is already available in the form of a first draft. The first formal issue is scheduled for mid-1975. A designated activity at each design contractor will be to issue and maintain control copies of the Reliability Manual for the use of design groups.

C.1.5.4.5 Reliability Engineering Planning

The program for initiating and conducting reliability engineering activities to assure that the safety objective can be met will include the following:

- 1) Evaluate the criticality of each plant system and select those items for which reliability engineering activities are appropriate.
- 2) Perform studies or organize existing data for each selected item to identify probable types of failure and their effects.
- 3) Select those items meeting the definition of reliability critical items and determine if redesign of the components or systems or if a maintenance program can reduce the level of criticality.

C.1.5.4.6 Integration of Reliability Planning and Scheduling with Design

This section describes the interaction between reliability and design activities. In addition, schedules are shown which provide perspective between reliability output and design decisions. This interaction occurs in two situations: that in which input from reliability allocation, assessment, testing, and data collection impacts the normal design evolutionary process together with a feedback effect; and that in which input from reliability assessment, testing, etc. is used in guiding a project design decision. Treatment in this section is limited to components whose failures could directly affect plant safety.

Figure C.1-2 illustrates how the reliability program interfaces with the design activities, using the primary shutdown system as an example. Complementing (not shown on Figure C.1-2) are the interfaces from the design activities back into the reliability program.

Of particular importance to the project is the necessity to phase the reliability program milestones to support major design milestones and thereby to demonstrate progress toward the attainment of the overall CRBRP reliability goals.

Figure C.1-3 presents this phasing of the reliability and design activities of the systems. The relation of the Reliability Program milestones with significant decision points on the Project Schedule is also shown. To assure that progress is being made toward the overall reliability goals, decision criteria are identified as one phase of the reliability program.

C.1.6 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 will meet the reliability goals allocated to them, comprehensive programs have been established for the shutdown and heat removal systems to confirm the reliabilities with increased confidence and to improve specific elements of the design, which will increase the margin relative to the goals.

C.1.6.1 Shutdown System

The Shutdown System Reliability Program is described in Section C.3.1 and in Section 1.5 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 any plant transient is of sufficiently low probability 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 CRBRP testing 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 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.

The data bank 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 data necessary to define the overall CRBRP shutdown systems reliability when this data is integrated by the reliability analysis with information from other sources. These sources include component, part, FFTF, and CRBRP design verification test data. The test plan includes testing at component, subsystem, and system levels.

C.1.6.2 Shutdown Heat Removal System

The Shutdown Heat Removal Reliability Activity 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 a 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 reliability analysis task. Failure mode, fault tree, common mode failure, and single point failure analysis 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 in support of the reliability objectives includes testing of key components in the heat removal systems. The following components are included: 1) the steam generator tubes, 2) sodium leak detectors, 3) intermediate loop pressure relief rupture discs, 4) the power pressure relief valves, 5) the steam generator auxiliary heat removal system (SGAHR) instrumentation and controls, 6) the protected air-cooled condenser (PACC) louver actuators, 7) the turbine bypass valve, 8) isolation and control valves in the steam generator auxiliary heat removal system, 9) the sodium pump bearings and pony motors, 10) a segment of welded main loop sodium piping, and 11) the most critical sodium component nozzle.

C.1.6.3 Piping Integrity

Piping integrity is one identified source which fits into the category of other faults which could lead to loss of in-place coolable geometry. Reliability studies to establish the integrity of primary loop piping are under way. This work, which emphasizes piping integrity under reactor power operation, is being performed in parallel with fracture mechanics and pipe corrosion testing as described in Section 1.5.2.1 of the PSAR. The reliability analysis draws on the results of this testing and the piping stress analysis. Probabilistic analyses are being performed using structural reliability methodology with the objective of demonstrating that the probability of pipe rupture is within the combined goal of $<10^{-7}$ for other sources that could lead to loss of in-place coolable geometry. Details of the piping integrity reliability analysis are presented in Section C.3.3.

C.1.7 Reliability Program Support of Goal Achievement

C.1.7.1 Shutdown System Goal Program

Achievement of the reliability goals for the shutdown systems will have increasing support from the reliability development program for the following reasons:

The reliability confirmation test program to be implemented for the mechanical shutdown system is based upon confirmation of an individual rod unavailability of at least 0.01 for both the primary or secondary system. As is shown in Section C.2.1, an individual rod unavailability of 0.01 provides more than the necessary system level availability because of rod redundancy considerations in each system. Individual rod unavailability will be confirmed by a combination of large sample tests and supplemental analysis and testing. The basic reliability confirmation tests will be accomplished by performing large numbers of successful scrams under near prototypic conditions to provide the required confidence in achieving the reliability objective. Supplemental testing together with analytical efforts will be utilized to confirm shutdown system reliability for potential problem areas not totally represented in the large sample tests. The latter effort will emphasize verification of the absence of failure mechanisms due to interfaces, real time dependent effects, common mode failures, and environmental conditions not incorporated in the large sample tests.

Bayesian statistics will be used in the numerical interpretation of data associated with variables to which the shutdown system performance has low sensitivity. Consequently, component level testing in these areas can be used to focus on providing qualitative information to support the engineering analysis used to develop Bayesian priors and failure rates for final system reliability estimates. Typical variables that will be treated this way in the subsystem tests are irradiation effects, interfacing components performance, non-wear related real time effects, potential common mode effects, and acceleration multiplication factors.

Section C.3.0 gives the detail of tests currently planned to support the reliability confirmation effort. Tables C.1-4 through 10 describe the impact of the test and analytical input upon the various reliability assessments planned in the program. It should be noted that since details of the test plan are still being formulated some modifications to schedule and items tested may occur. Until the subsystem test program is under way, the numerical assessments will be obtained from a detailed reliability model using data obtained from component level testing combined with analysis.

The initial assessment described in Section C.3.1 predicts achievement of the shutdown system goal. Each succeeding assessment described in Table C.1-4 to 10 will improve the accuracy of that prediction. Once subsystem testing begins, confidence in the predicted numerical reliability will improve. Very few failures, if any, are expected in the subsystem test because of the broad base development program that preceded this portion of the program. FFTF testing of a system very similar to that used in the primary shutdown system already has achieved in excess of 400 scrams without failure. An

additional 400 scrams will be completed by the middle of 1975. For the secondary shutdown system, preliminary results on the coil-cord test and latch test will be available; the damper test will be completed by mid-1975. Both the primary and secondary systems will also have performed development tests up to and including subsystem testing in a prototypic environment providing additional confidence that the reliability confirmation will be successful prior to its initiation.

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. Before the time of decision on a Construction Permit, 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. 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.

As part of the detailed test plan now in preparation (see Figure C.1-2), the role each test plays in the overall numerical reliability confirmation will be provided. This plan will contain an initial reliability growth curve showing confirmed reliability as a function of time in the program and provide the basis for that assessment. An updated version of this curve will be available prior to a decision on the C.P.

The test activities beyond the Construction Permit decision point will serve to provide an additional margin in the assessment confidence. In conjunction, with an estimate of the degree of conservatism provided for in the analysis it is expected that the follow-on testing will provide a significant increase in the assessment level of confidence.

Certain aspects of the electrical SDS tests may still be in progress, but until their completion, options such as a reduction in the periodic 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 Initial Criticality.

C.1.7.2 Shutdown Heat Removal System

The Shutdown Heat Removal System Reliability Program will provide significant input relative to plant reliability to support the major plant construction decision points, namely, the Construction Permit, FSAR, and Initial Criticality. Progress of the program in providing data for decision-making and generally advancing toward achieving final program goals will follow the pattern of:

1. Explicit reliability analysis refined in sequential updates to include more precise modeling and failure rates.
2. Injection into the design process of visible and explicit treatment of reliability matters with formalized, systematic reviews of component and system design, and
3. Testing of components judged to be most crucial in the reliability analysis to provide an improved basis for failure rates used in the analysis, with significant potential for the added benefit of identifying weak features of the tested components.

The first element of the program, the explicit reliability analysis, is being carried out according to the best available analytic tools of current reliability methodology and utilizing the best sources of failure and repair data. The initial assessment was summarized in Section C.1.4.1.2 and is described in detail in Section C.2.2. The analysis as presented is judged to provide adequate assurance that reliability objectives are met with the present shutdown heat removal system design. However, certain elements of the analysis require confirmation. Some important areas of analysis refinement which are underway and will be reflected in updated analyses are a more precise representation of component repairability, more detailed component failure analysis to set a more firmly grounded sodium temperature limit as the shutdown heat removal practical failure criterion, and incorporation of more certain failure rates. The improvement in failure data will be based on existing failure data sources not yet fully exploited, on failure data to be generated by plant operation (e.g., FFTF) and FFTF and CRBRP development testing, and on failure data to be generated within the shutdown heat removal system reliability program itself. Unquestionably, other applicable test data will also be available, e.g., from development testing in this country and from European operating experience. The first major update is scheduled for early 1976, well prior to the Construction Permit, and will reflect better justified failure and repair data, better repairability representation, and more detailed analyses to define a realistic temperature limit. The next major update is scheduled for early 1978. This update is timed to benefit from most of the test data to be collected within this program. A final assessment will be available before the Final Safety Analysis Report.

The second general area of the program covers measures to inject more disciplined component design development into the engineering process. The measures include analytic tools developed during the course of reliability engineering experience which supplement the conventional engineering review of a design to meet its specified functions with disciplined and documented feature-by-feature review for failure potential and failure effect severity. In other words, the traditional engineering approach, with its good record of success, is refined by reliability analysis methods to an even higher level of effectiveness. These analyses consist of Failure Mode and Effects Analysis, which will be performed on all the systems of interest, namely,

the primary and intermediate heat transport systems, the steam generator system, the steam generator auxiliary heat removal system, the feedwater and condenser system, the main and auxiliary feedwater system, and reactor enclosure systems, and the overflow heat removal service. These FMEA's will treat those aspects which are related to the post-shutdown heat removal function. Within these systems, reliability critical components will be subjects of individual FMEAs. This list will include key components, such as the steam generator units and the intermediate loop sodium rupture discs. An overall heat removal FMEA is complete and available as Table C.2.2-2. The first complete set of individual FMEAs will be completed in early 1976, supporting the Construction Permit. Several updatings of the FMEAs are scheduled, with final issue in time for FSAR. Fault Tree, Single Point Failure, and Common Mode Failure Analyses will be built on the FMEAs and cover the same systems and components. Preliminary results of the FTA, SPFA, and CMFA will be completed for CP. Completed reliability analyses of this kind will be available as part of the FSAR. These qualitative and partially quantitative tools are elements of a more organized engineering design review.

The third element of program contribution toward the reliability objectives is the test program. The test program as presently planned is based on current best knowledge of component test data priorities. The test program is subject to change as dictated by progress with the qualitative and quantitative analyses of the heat removal systems. It will be seen that all the data will add to the certainty of the failure rates used in the analysis. Statistical interpretation of the data from the planned testing will be maximized, including combining the collected data with prior understanding of the phenomena of concern. The firm identification of test data needs and the intermediate test plans will be available for CP. All test data will be available for the final reliability assessment and for the FSAR. Specific items planned for tests are addressed in Section C.3.2.2.

C.1.7.3 Piping Integrity

The piping integrity reliability analysis is planned in such a way as to provide pertinent input for both the pipe integrity fallback design decision points as well as for CRBRP licensing milestones.

The first analysis will be complete to support the pipe integrity fallback conceptual design selection. The input will consist of the first interim piping stress report and piping failure test as collected through May, 1975. The two following refinements in the analysis, along with their success criteria, their timing, and the elements of refinement over the preceding analysis are described below. Note that this schedule is based on the pipe sleeve concept for pipe rupture accommodation. Should the preferred concept be a design feature other than a pipe sleeve, this schedule will be modified accordingly.

1. Initial Reliability Assessment

- a. Success criterion - Probability of pipe rupture occurrence is within the goal of $<10^{-7}$ for other faults leading to loss of in-place coolable geometry. This is based on a realistic

probabilistic analysis using best available input data and methodology.

- b. Timing - Pipe sleeve concept selection (see Appendix E)

2. Updated Reliability Assessment

- a. Success criterion - Confirmation of pipe rupture within the goal set in 1.a. above.
- b. Timing - Construction Permit and pipe sleeve preliminary design.
- c. Refinements.
 - 1) Stress analysis results utilizing more near final loads.
 - 2) Improved critical (unstable) crack length definition in actual elbow geometry.
 - 3) Expanded data on cyclic crack growth (S/N data) and growth morphology (through-thickness growth versus tangential extension) for actual elbow geometry.
 - 4) Conclusions on degradation of fracture toughness by caustic environment.
 - 5) Assessment of likelihood of not detecting flaws of specific sizes.

The Updated Reliability Analysis may predict a rupture probability lower than initially predicted, which would increase confidence in the piping integrity.

3. Final Reliability Assessment

- a. Success criterion - Confirmation of pipe rupture within the $<10^{-7}$ goal with improved confidence over the Updated Reliability Assessment as indicated in c.
- b. Timing - FSAR and sleeve installation decision.
- c. Refinements:
 - 1) Final stress analysis results with final loads.
 - 2) Testing to failure of a full size welded PHTS piping elbow and a PHTS nozzle in the Multi-Loading Test Facility at operating temperature. The measurements

will principally confirm the adequacy of the stress analysis used in the reliability assessment.

The final assessment likewise has the potential of a prediction of even higher reliability.

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TABLE C.1-1

UPSET EVENT FREQUENCIES AND MAXIMUM TEMPERATURES

<u>Event</u>		<u>Frequency (30 years)</u>	<u>Max. Temperature within 10 Minutes</u>
<u>Upset Events</u>			
U-1a	Reactor trip from full power with normal decay heat	180	<1400
U-1b	Reactor trip from full power with minimum decay heat	0	<1400
U-1c	Reactor trip from partial power with minimum decay heat	0	<1400
U-2a	Uncontrolled rod insertion	10	<1400
U-2b	Uncontrolled rod withdrawal from 100% power	10	>1700
U-2c	Uncontrolled rod withdrawal from start-up with automatic trip	10	>1700
SVB-35 U-2d	Uncontrolled rod withdrawal from start-up to trip point with delayed manual trip	10	<1500
U-2e	Plant loading at max. rod withdrawal rate	10	<1700
U-2f	Reactor start-up with excessive step power change	50	<1700
U-3a	Partial loss of primary pump	2 per loop	<1500
U-3b	Loss of power to one primary pump	5 per loop	>1700
U-4a	Partial loss of one intermediate pump	2 per loop	<1500
U-4b	Loss of power to one intermediate pump	5 per loop	<1500
U-5a	Loss of AC power to one feedwater pump motor	10	<1500
U-5b	Loss of feedwater to all steam generators	5	>1700
U-6	Loss of flow in two sodium loops	10	>1700

TABLE C.1.1 (continued)

<u>Event</u>		<u>Frequency (30 years)</u>	<u>Max. Temperature within 10 Minutes</u>
<u>Upset Events (continued)</u>			
U-7a	Primary pump speed increase	5	<1400
U-7b	Intermediate pump speed increase	5	<1400
U-8	Primary pump pony motor failure	5 per pump	NA
U-9	Intermediate pump pony motor failure	5 per pump	NA
U-10a	Evaporator module inlet isolation valve closure	4 per loop	<1500
U-10b	Superheater module inlet isolation valve closure	2 per loop	<1500
U-10c	Evaporator module outlet isolation valve	4 per loop	<1500
U-10d	Superheater module outlet isolation valve closure	2 per loop	<1500
U-11a	Water side isolation and dump of both evaporators and the superheater	6/loop	<1500
U-11b	Water side isolation and dump of evaporator module	6/loop	<1500
U-11c	Water side isolation and dump of superheater	3 per loop	<1500
U-12	Loss of feedwater flow to one steam generator	3 per loop	<1500
U-13	Feedwater throttle valve failed open	6 per loop	<1500
U-14	Loss of one recirculation pump	8 per loop	<1500
U-15a	Turbine trip (without reactor trip)	50	<1500

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TABLE C.1.1 (continued)

Event		Frequency (30 years)	Maximum Temperature within 10 minutes
<u>Upset Events(continued)</u>			
U-15b	Turbine trip with reactor trip (loss of main condenser or similar problem)	10	<1500
U-16	Operating basis earthquake	5 with 10 cycles each	>1700
U-17	Three loop natural circulation	10	NA
U-18	Loss of preferred and alternate preferred power	6	>1700
U-19a	Small sodium-water leak: faulty evaporator module identified	3/loop	<1500
U-19b	Small sodium-water leak: unable to identify which module is faulty	3/loop	<1500
U-19c	Small steam-sodium leak: identified as super-heater leak	3/loop	<1500
U-20a	Inadvertent opening of one turbine bypass valve	5	<1500
U-20b	Turbine bypass valve fails open following reactor trip	5	<1500
U-21a	Inadvertent opening of evaporator outlet safety/power relief valves	5/loop	<1500
U-21b	Inadvertent opening superheater outlet safety/power relief valves	3/loop	<1500
U-22	Inadvertent opening of drum valve (blowdown valve, safety valve or SGAHRS valve)	3/loop	<1500
U-23	Inadvertent opening of evaporator inlet dump valve	3/loop	<1500

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TABLE C.1-2

RESULTS OF CURRENT UNAVAILABILITY ASSESSMENT
FOR SHUT DOWN SYSTEMS

		<u>Electrical</u>	<u>Mechanical</u>	<u>System</u>
Primary System	Allocated Unavailability	2.5×10^{-5}	7.5×10^{-5}	1.0×10^{-4}
	Current Assessment Unavailability	5×10^{-5}	$2 \times 10^{-5*}$	7.0×10^{-5}
Secondary System	Allocated Unavailability	1.2×10^{-4}	3.8×10^{-4}	5×10^{-4}
	Current Assessment Unavailability	6×10^{-5}	$2 \times 10^{-5*}$	8×10^{-5}

* Based on individual rod unavailability of 0.01.

TABLE C.1-3

SHUTDOWN SYSTEM DIVERSITY OF DESIGN

Control Assembly (CA)

Control Rod
Guide Geometry
No. of Control Rods

Primary

37 pin bundle
Hexagonal
15

Secondary

19 pin bundle
Cylindrical
4

Control Rod Driveline (CRD)

Coupling to CA
Connection to CRDM

Rigid Coupling
CRD Leadscrew to CRDM Roller
Nuts

Flexible Collet Latch
CRD Attached to CRDM
Carriage with Pneumatic
Activation of CRD Latch
through Slender Rod
Automatic

Disconnect from CA for Refueling

Manual

Control Rod Drive Mechanism

Type of Mechanisms

Collapsible Rotor-Roller Nut

Twin Ball Screw with Trans-
lating Carriage

Overall Mechanisms 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

TABLE C.1-4

ASSESSMENT NUMBER 1, PSAR SUBMITTAL

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

For details and summary of Assessment, see Section C.3.1.

TABLE C.1-5

ASSESSMENT NUMBER 2 -- JULY, 1975

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

System

Preliminary System Level FMEA Completed for Common Mode Failure Identification.

Primary Control Rod System

Increased design and configuration detail

Preliminary FMEA completed

First calculation of individual rod reliability completed

Preliminary PCRS Reliability model completed

Preliminary vendor assessment of PCRD reliability completed

* FFTF subsystem test completed - subsystem alignment, misalignment, and gross misalignment tests completed

Secondary Control Rod System

Increased design and configuration detail

Preliminary FMEA completed

First calculation of individual rod reliability completed

Preliminary SCRS reliability model completed

Preliminary fault tree analysis completed

Design review completed for CRDM/CRD/CA first prototype (preliminary)

Primary and Secondary Electrical System

FFTF electronic component (vendor) demonstration test completed

MIL-HDBK-217B prediction completed

Preliminary component FMEA completed

Preliminary design completed

* FFTF mechanical subsystem very similar to CRBRP

TABLE C.1-6

ASSESSMENT NUMBER 3 FIRST QUARTER, 1976

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

Primary Control Rod System

Design reviews completed for:

Upper Internals (final)*
Core Former Ring (final)*
CA and CRDM/CRD (preliminary)

Fault Tree Analysis completed
FMEA and CMFA updated

Secondary Control Rod System

The following Design Verification Tests will be completed:

Secondary Control Assembly Static Flow Test
Latch Test

FMEA and CMFA updated
FTA updated

Primary and Secondary Electrical System

Detailed FMEA completed
Initial group of electronic components begins Reliability Confirmation
test

* Interfacing components to PCRS and SCRS

TABLE C.1-7

ASSESSMENT NUMBER 4 FIRST QUARTER, 1977

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

Primary Control Rod System

Complete FFTF failed bellows test
Information will be available from the following Reliability Component Tests:

CRDM scram assist spring test (complete)
CRDM bellows test (complete)
Dynamic friction test (preliminary)
Bowed Duct drag force test (complete)

Information will be available from the following Design Verification Tests:

CRDM life test (preliminary)
CRD dashpot (complete)
Static friction test (preliminary)

Continuing FMEA results

Secondary Control Rod System

Design Verification Test
The secondary flow test under scram conditions will be completed
Component Reliability Test
The upper driveline bellows will be completed
The updated fault tree analysis will be completed

Primary and Secondary Electrical System

Final design review will be completed
60% of electronic components will have entered test
Initial data will be available from reliability confirmation test

TABLE C.1-8

ASSESSMENT NUMBER 5 FIRST QUARTER, 1978

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

Primary Control Rod System

Information will be available from the following Component Reliability Tests:

CRDM life tests (completed)
Pin rupture test (completed)
Duct impact test (preliminary)

Final FMEA completed
Final CRD/CRDM design review completed
Design Verification Tests

Dynamic Friction (completed)
Prototype subsystem test (preliminary)
CRDM performance (completed)
CA flow (completed)

Reliability Subsystem Tests
500 scrams completed in subsystem reliability confirmation

Secondary Control Rod System

Final FMEA will be completed
The following Design Verification Tests will be completed:

Nose piece and shield flow
Prototype secondary control rod subsystem test for component interference problems and overall subsystem effects

The following Design Limit Tests will be complete:

Contaminated argon cylinder valve actuation
Driveline crushing limit

(Cont.)

TABLE C.1-8 (continued)

The following Reliability Tests will be complete:

Pneumatic valve and cylinder
Seals and bushing
Lower driveline bellows
Latch-collet scram (500 component level scrams)
Argon sub-system test
Subsystem test 500 accelerated scrams* (400 scorable)

Primary and Secondary Electrical System

All electronic components entered into Reliability Confirmation Testing
Reliability Confirmation Testing 25% complete

* Scorable scrams are included in the statistical reliability confirmation; non-scorable scrams are performed at conditions near and exceeding design limits and, hence, are not included in the statistical confirmation phase but instead yield qualitative information relating to common mode failures and other conditions not totally represented in the subsystem test.

TABLE C.1-9

ASSESSMENT NUMBER 6 FIRST QUARTER, 1979

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

Primary Control Rod System

Final FMEA completed
Reliability Confirmation tests

* 1200 scrams completed (see Table C.1-4)

Preliminary data will be available from misalignment subsystem tests
Design verification tests

Prototype life (completed)

Secondary Control Rod System

The CRDM/CRD/CA design review will be complete
The following Design Limit Tests will be complete:

Gross failure of main shaft bellows including testing with no
bellows
Misalignment limits

The following Reliability Tests will be complete:

Latch-collet, 1500 additional component level scrams (2000
cumulative scrams)
Sub-system test - shortened driveline (the equivalent of
approximately seven years of real time testing will
have accumulated at this point)
Accelerated subsystem test, 200 additional scorable scrams
and 300 additional non scorable scrams (600 cumulative
scorable scrams and 400 cumulative non scorable scrams)

Primary and Secondary Electrical System

Reliability confirmation test 60% complete

* representative of early wear SVB-46

TABLE C.1-10

ASSESSMENT NUMBER 7 FIRST QUARTER, 1980

FACTORS LEADING TO INCREASED CONFIDENCE IN RELIABILITY ASSESSMENT:

Primary Control Rod System

Final CA design review completed
1500 scrams complete (see Table C.1-4)
(Subsystem common mode failure and misalignment tests will be completed in 1980)
Design verification tests:

- B₄C Irradiation (completed)
- FFTF Irradiation (preliminary)

Secondary Control Rod System

The guide tube deformation and bowing Design Limit Test will be complete.
The following Reliability Test will be complete:

Latch-collet real time test (3 additional equivalent years for a cumulative total of 10 equivalent years)

Accelerated subsystem test, 50 additional scorable scrams and 250 additional non-scorable scrams (650 cumulative scorable scrams and 650 cumulative non-scorable scrams)

60 real time subsystem scrams will be completed in 1981

Primary and Secondary Electrical System

Reliability Confirmation Test 80% completed

* Electrical subsystem reliability goal confirmed to an estimated 70% confidence

* The option of a decreased test interval will allow the system to be operated in a manner for which the reliability goal would be confirmed at a higher confidence level. This option will not be necessary when the confirmation test is complete.

7229-67

SVB-48

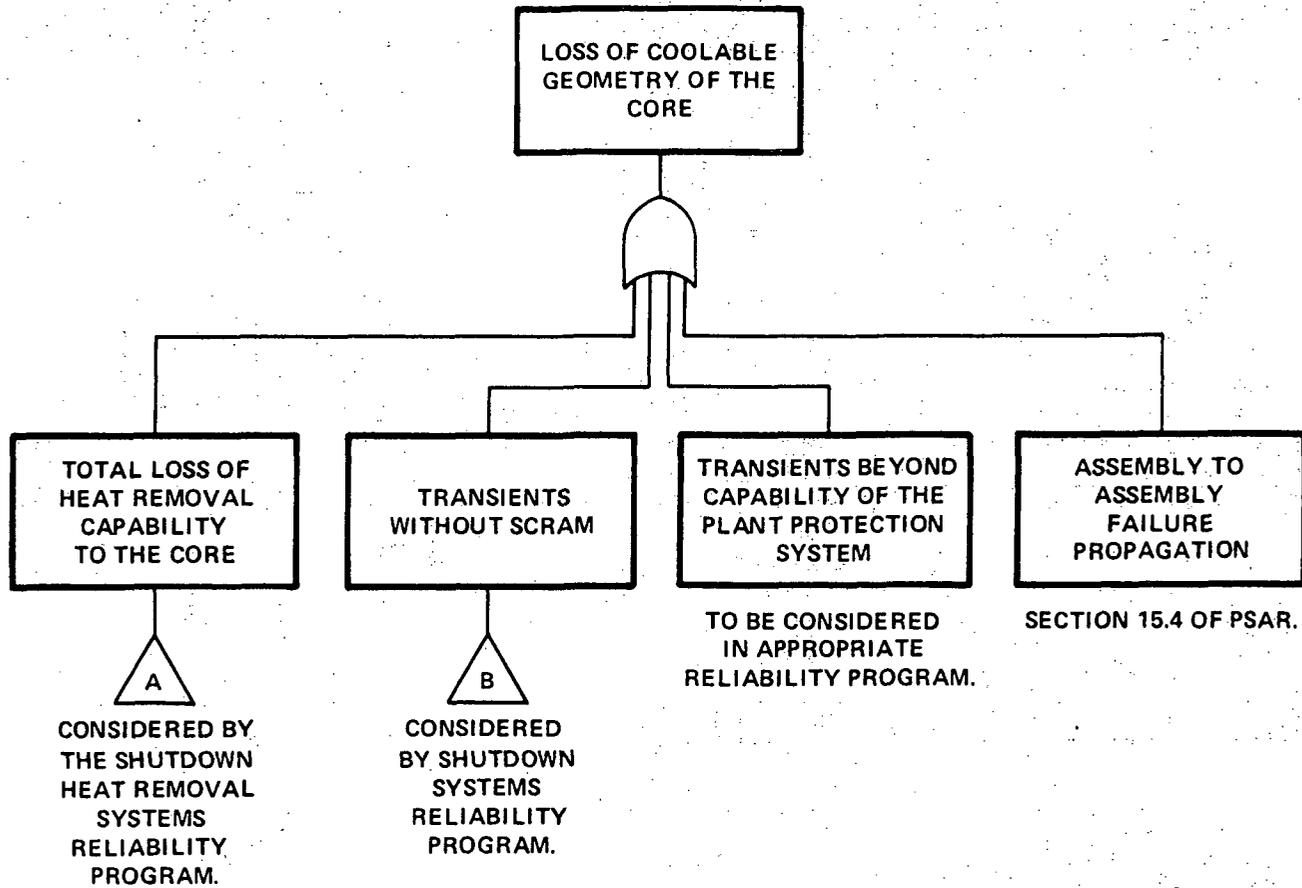


Figure C.1-1. Fault Tree

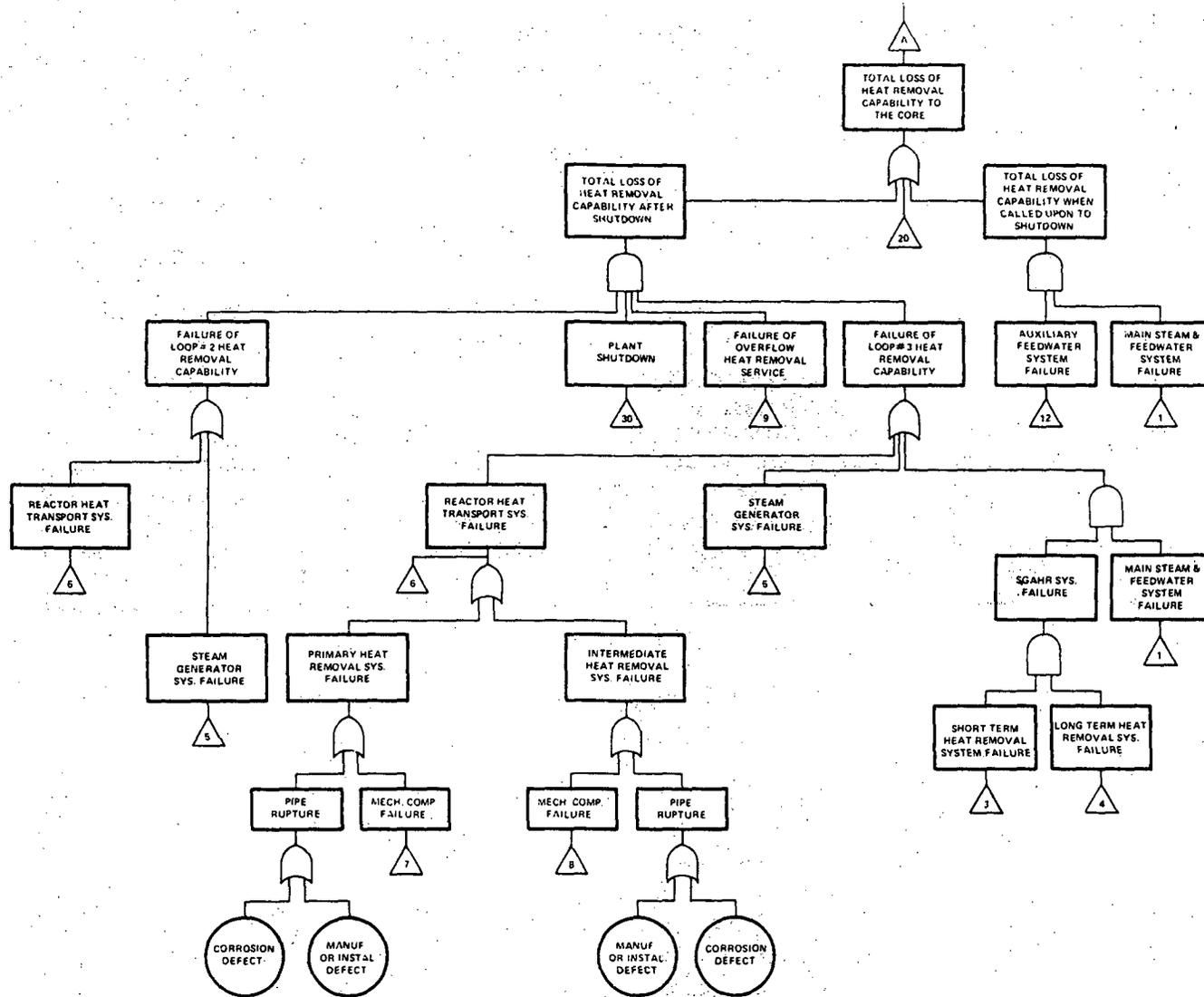


Figure C.1-1. (Cont.)

7229-S3

SVB-50

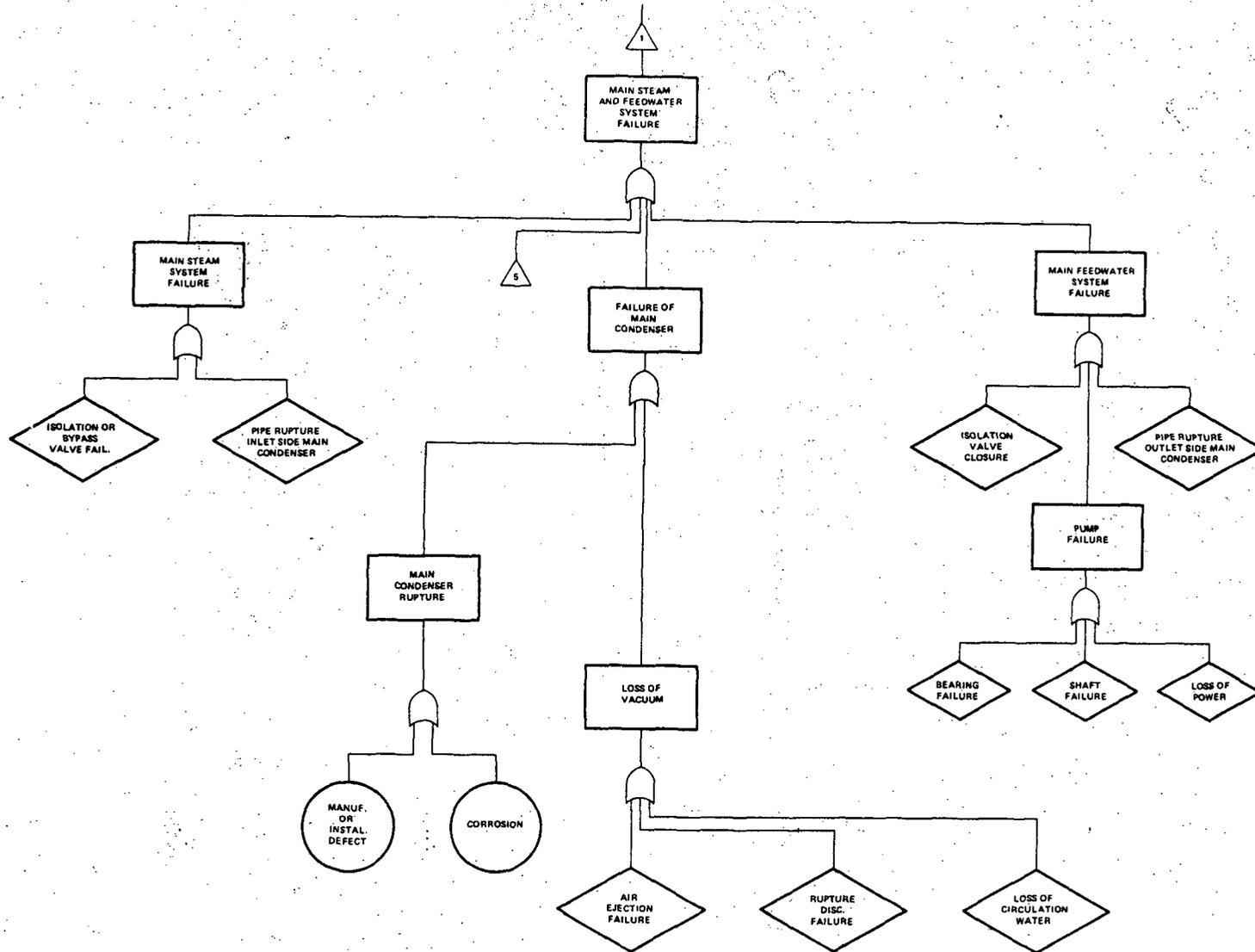


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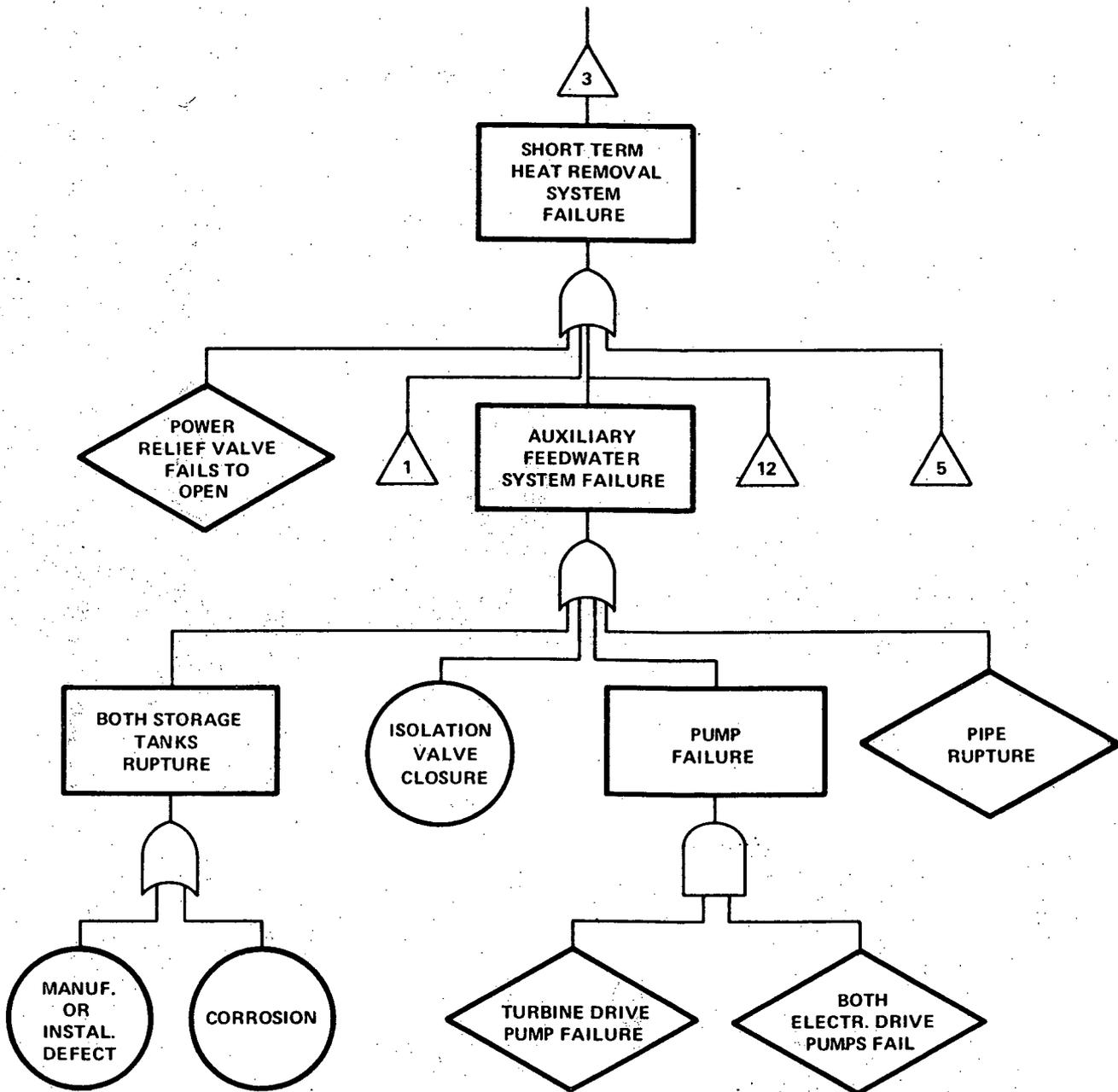


Figure C.1-1. (Cont.)

7229-54

SVB-52

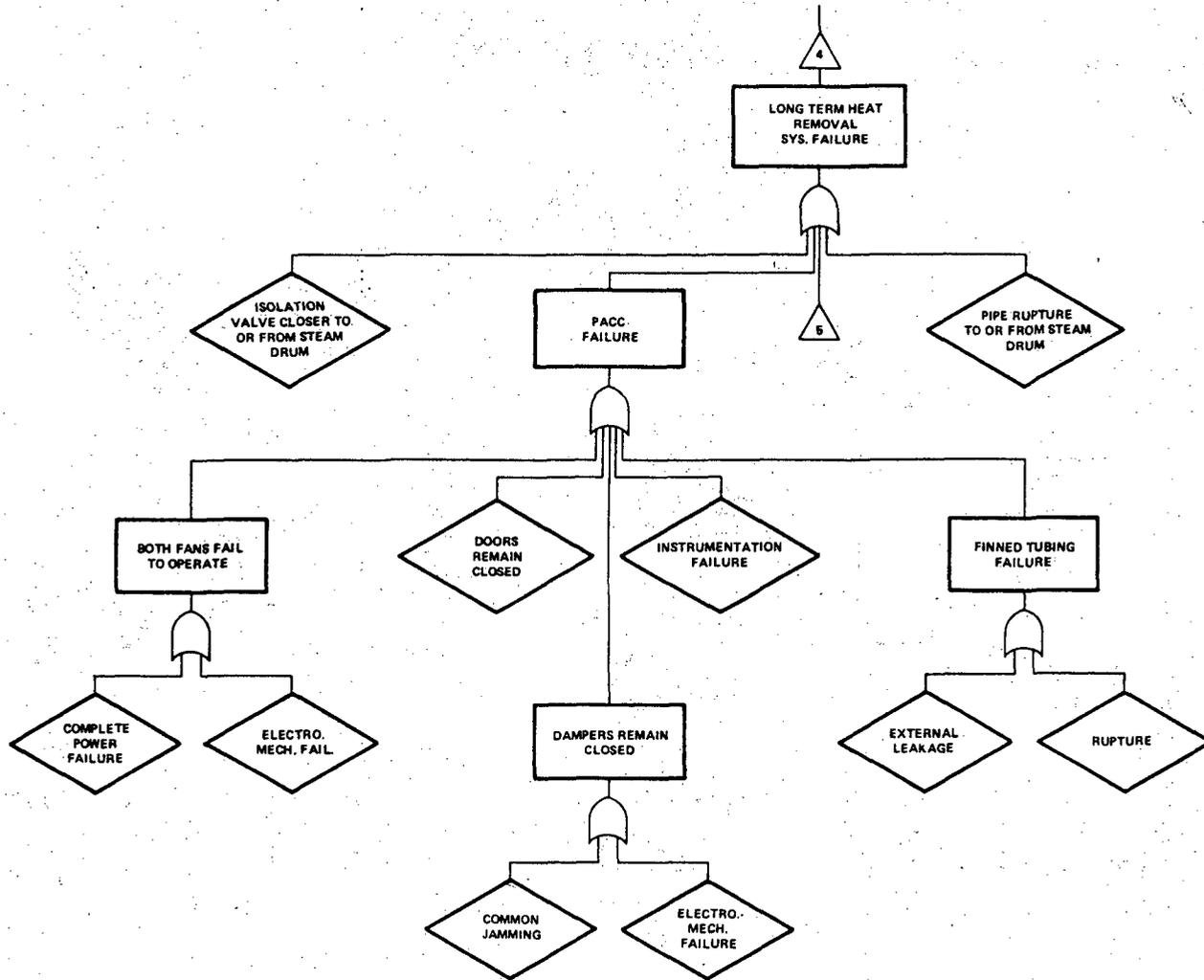


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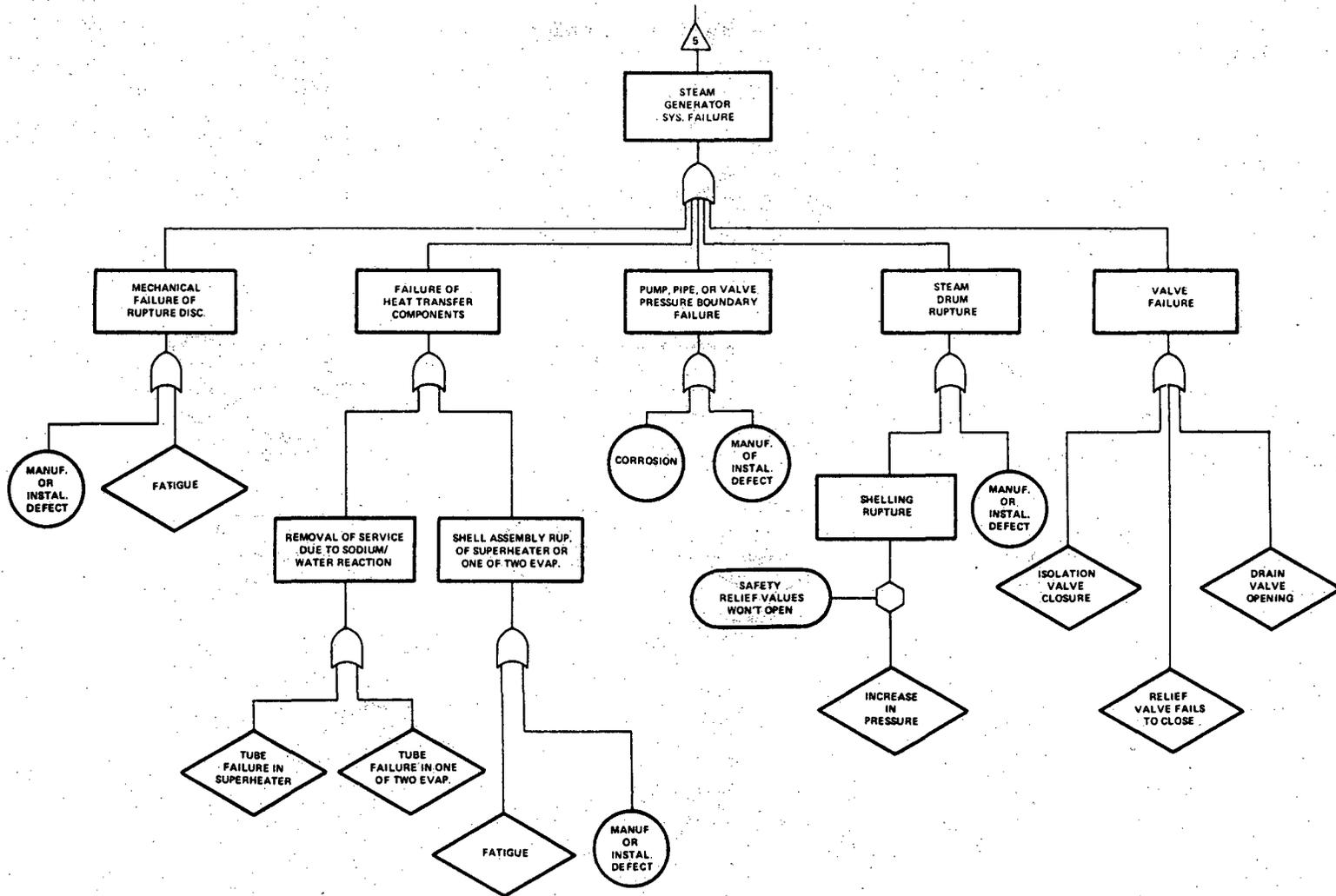


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7229-55

SVB-54

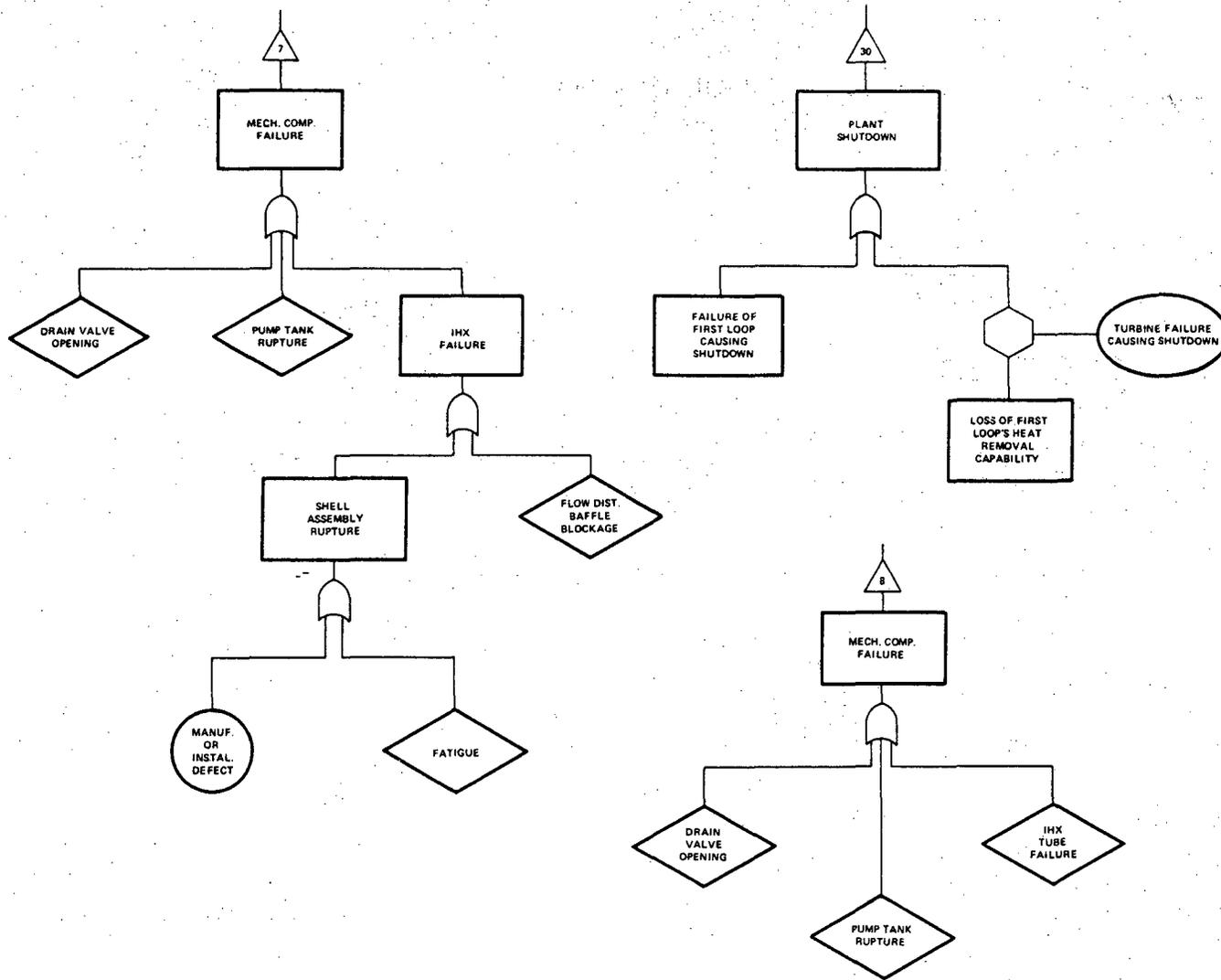
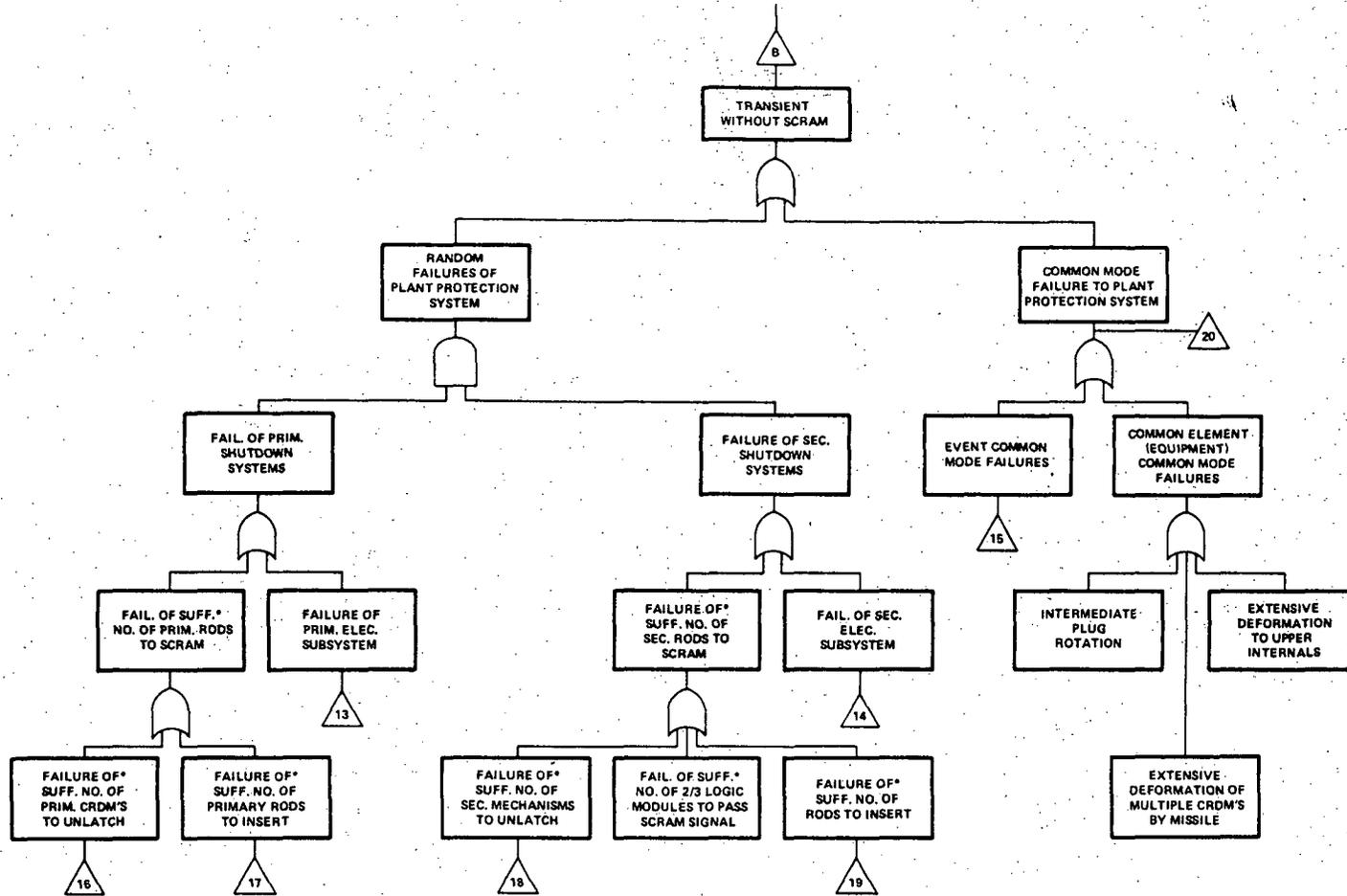


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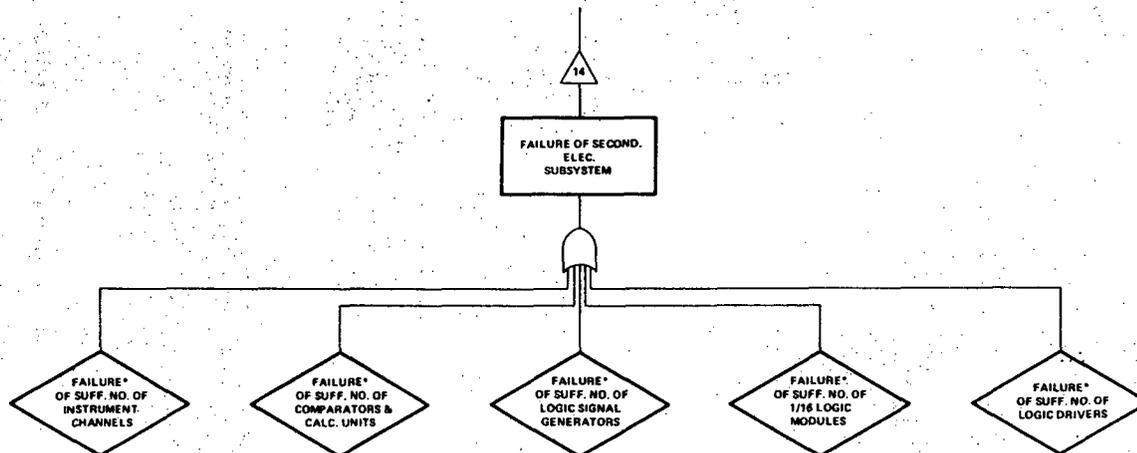
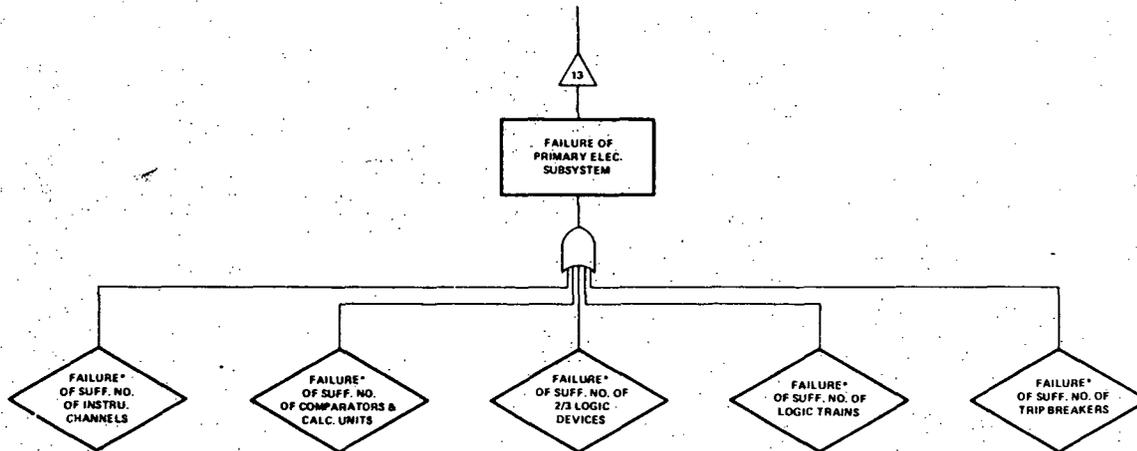
7229-56

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*IT IS RECOGNIZED THAT DIFFERENT RODS MAY FAIL SIMULTANEOUSLY FOR DIFFERENT REASONS, THE TREE SHOWS GENERAL COMBINATIONS WHICH CAN RESULT IN FAILURE OF ONE OR MORE RODS, TO FAIL THE PRIMARY OR SECONDARY SYSTEM WOULD REQUIRE SUCH COMBINATIONS OCCURRING SIMULTANEOUSLY TO THE SUFFICIENT NUMBER OF RODS.

Figure C.1-1. (Cont.)



*IT IS RECOGNIZED THAT DIFFERENT ROOS MAY FAIL SIMULTANEOUSLY FOR DIFFERENT REASONS, THE TREE SHOWS GENERAL COMBINATIONS WHICH CAN RESULT IN FAILURE OF ONE OR MORE ROOS, TO FAIL THE PRIMARY OR SECONDARY SYSTEM WOULD REQUIRE SUCH COMBINATIONS OCCURRING SIMULTANEOUSLY TO THE SUFFICIENT NUMBER OF ROOS.

Figure C.1-1. (Cont.)

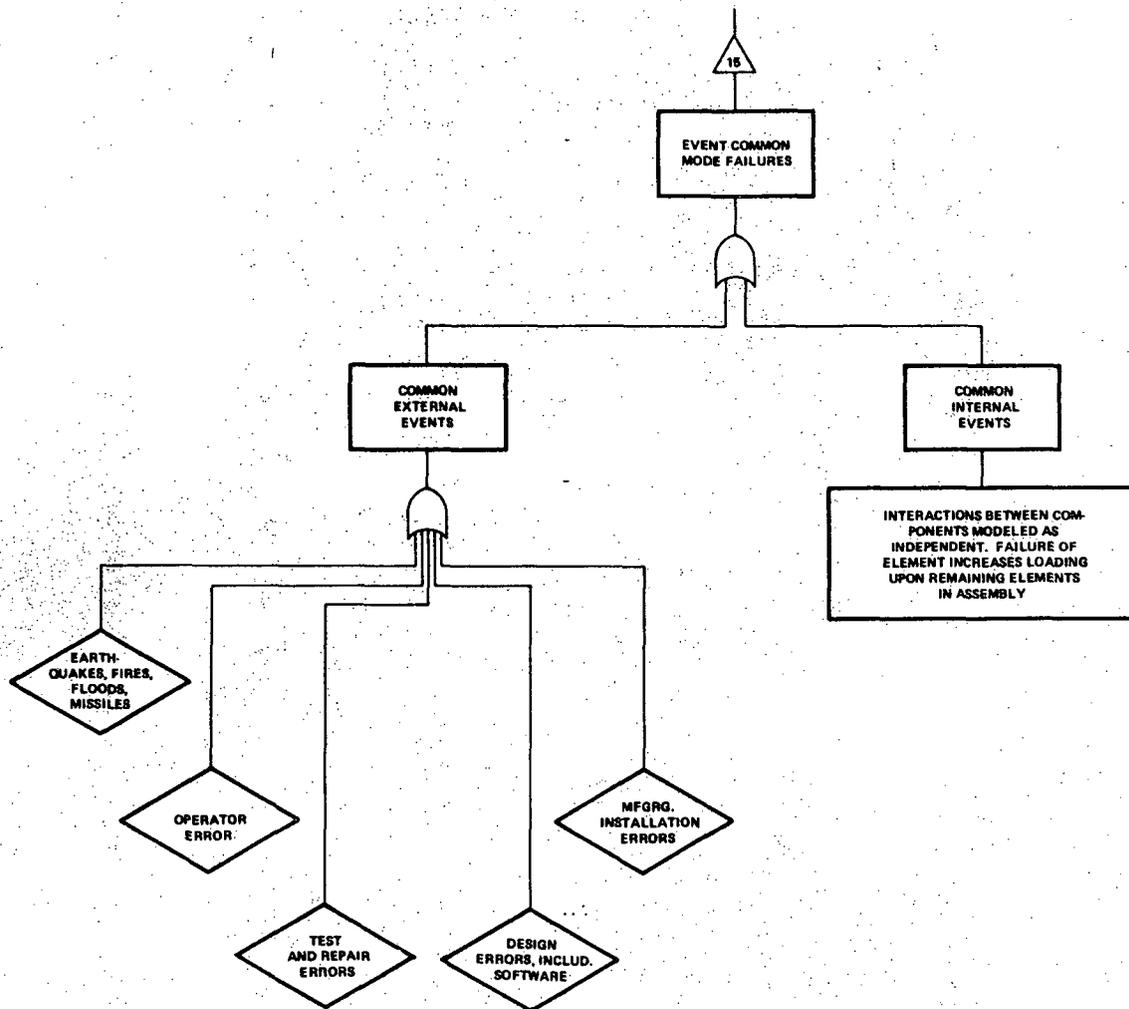


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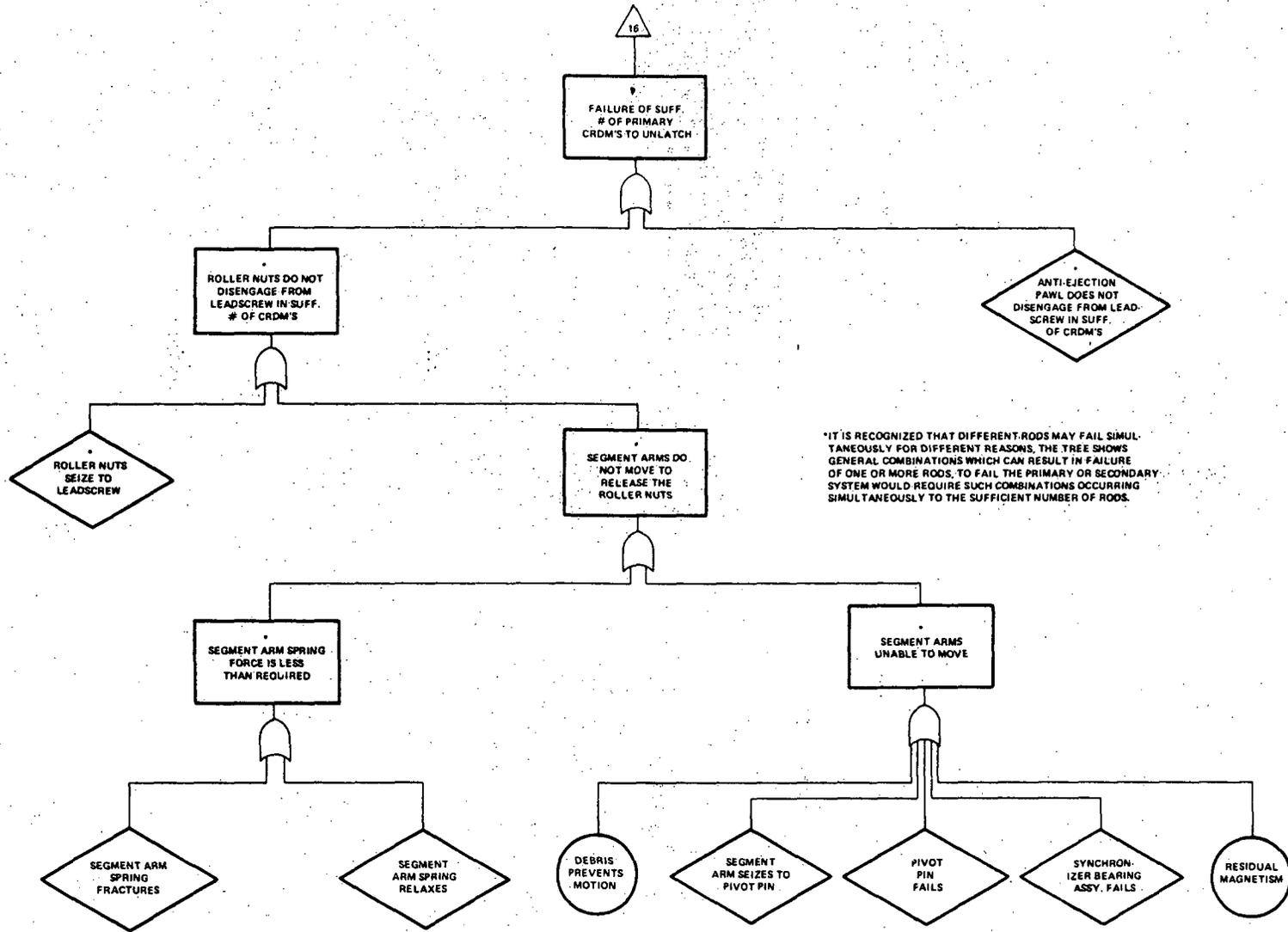
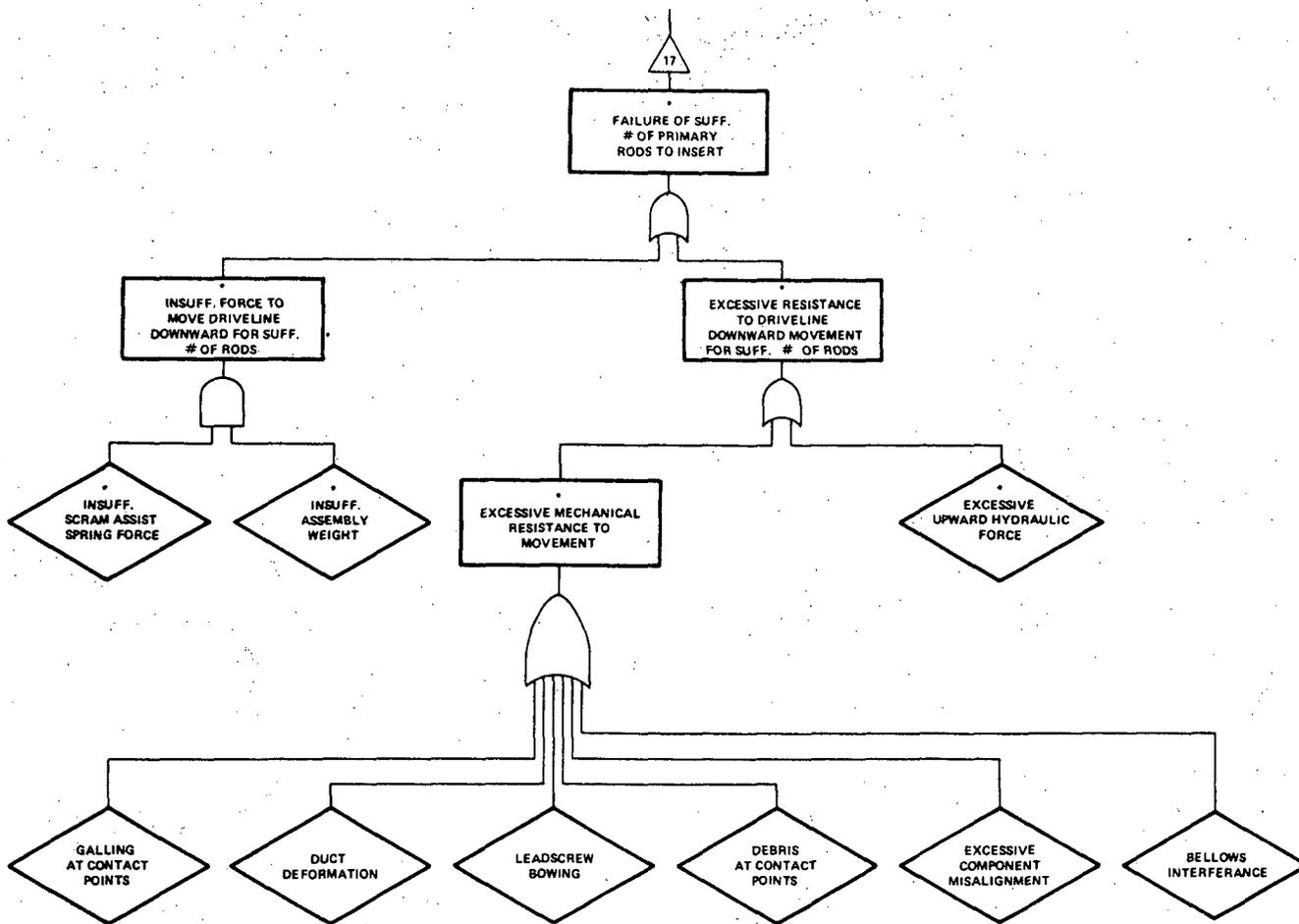


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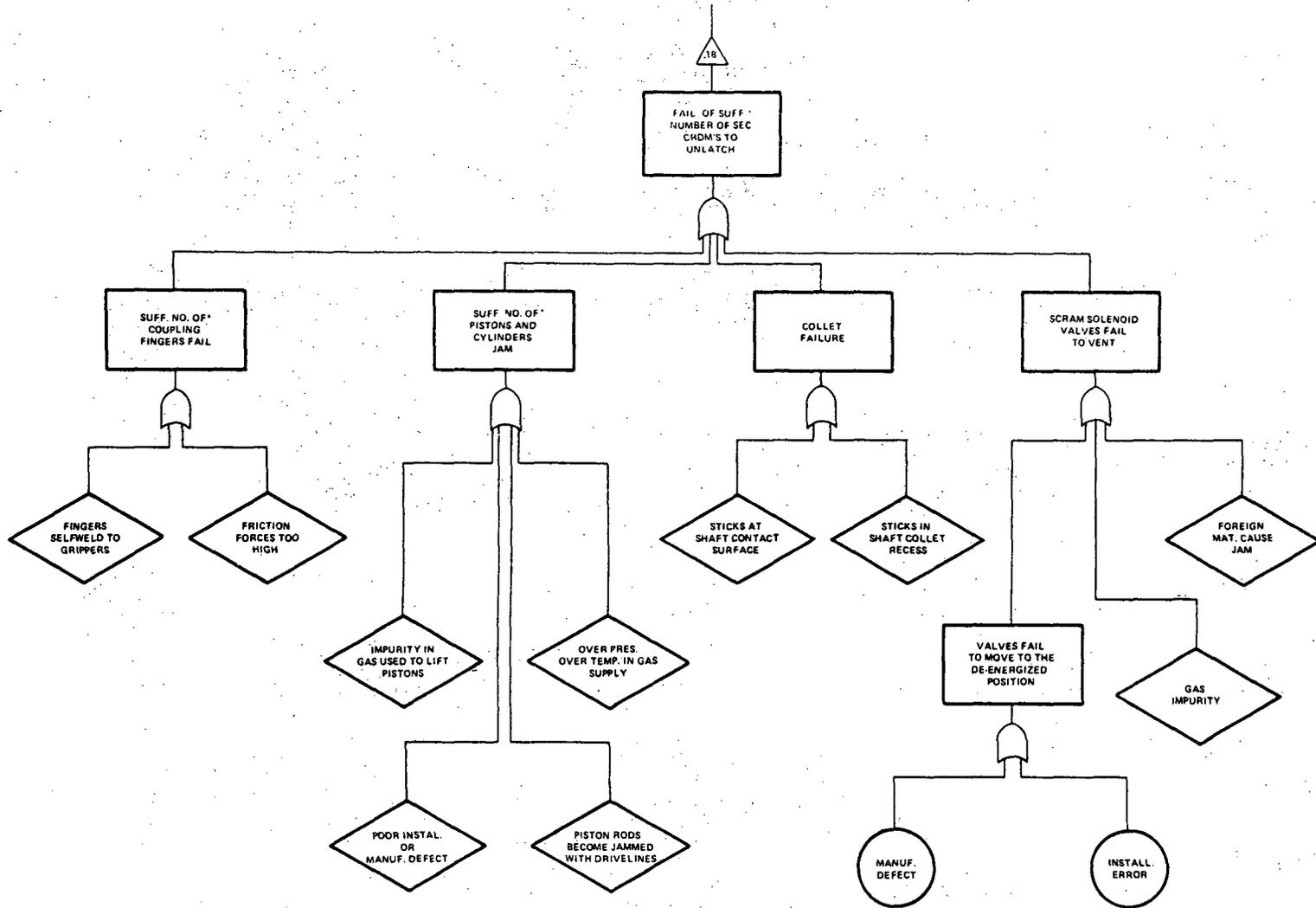
7229-60

SVB-60



*IT IS RECOGNIZED THAT DIFFERENT RODS MAY FAIL SIMULTANEOUSLY FOR DIFFERENT REASONS. THE TREE SHOWS GENERAL COMBINATIONS WHICH CAN RESULT IN FAILURE OF ONE OR MORE RODS. TO FAIL THE PRIMARY OR SECONDARY SYSTEM WOULD REQUIRE SUCH COMBINATIONS OCCURRING SIMULTANEOUSLY TO THE SUFFICIENT NUMBER OF RODS.

Figure C.1-1. (Cont.)



*IT IS RECOGNIZED THAT DIFFERENT RODS MAY FAIL SIMULTANEOUSLY FOR DIFFERENT REASONS. THE TREE SHOWS GENERAL COMBINATIONS WHICH CAN RESULT IN FAILURE OF ONE OR MORE RODS, TO FAIL THE PRIMARY OR SECONDARY SYSTEM WOULD REQUIRE SUCH COMBINATIONS OCCURRING SIMULTANEOUSLY TO THE SUFFICIENT NUMBER OF RODS.

Figure C.1-1. (Cont.)

7229-62

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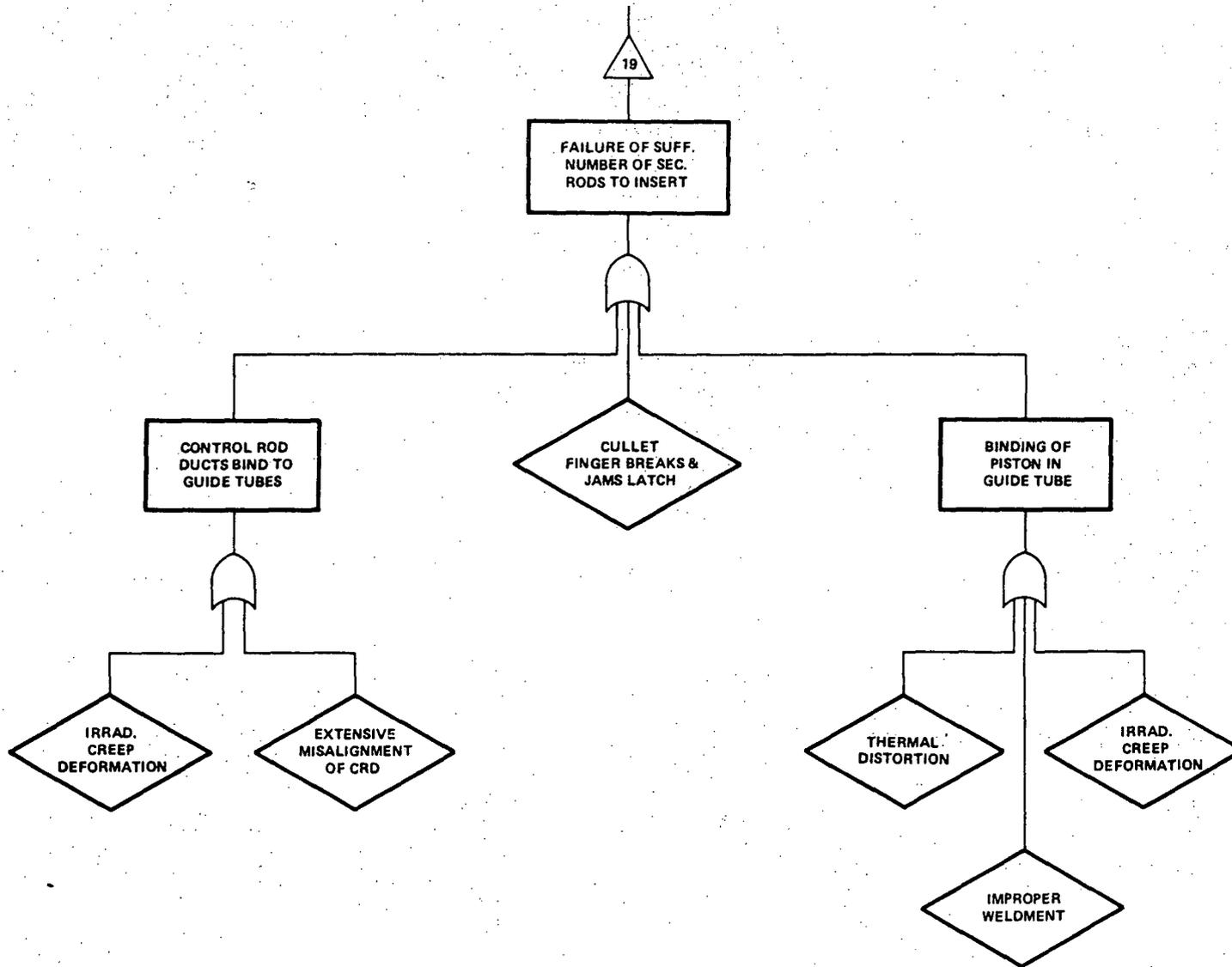


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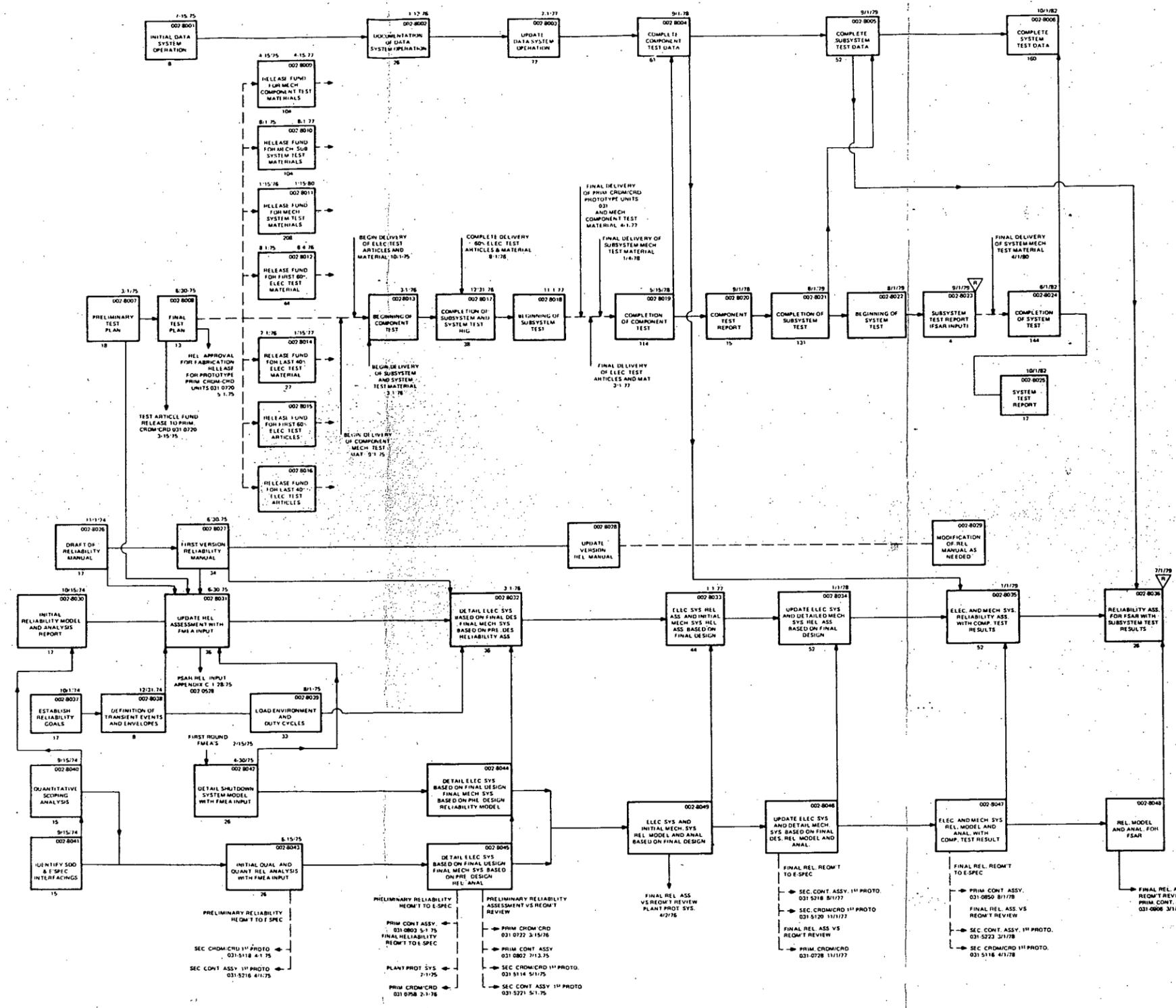


Figure C.1-2. Reliability Interface With Design

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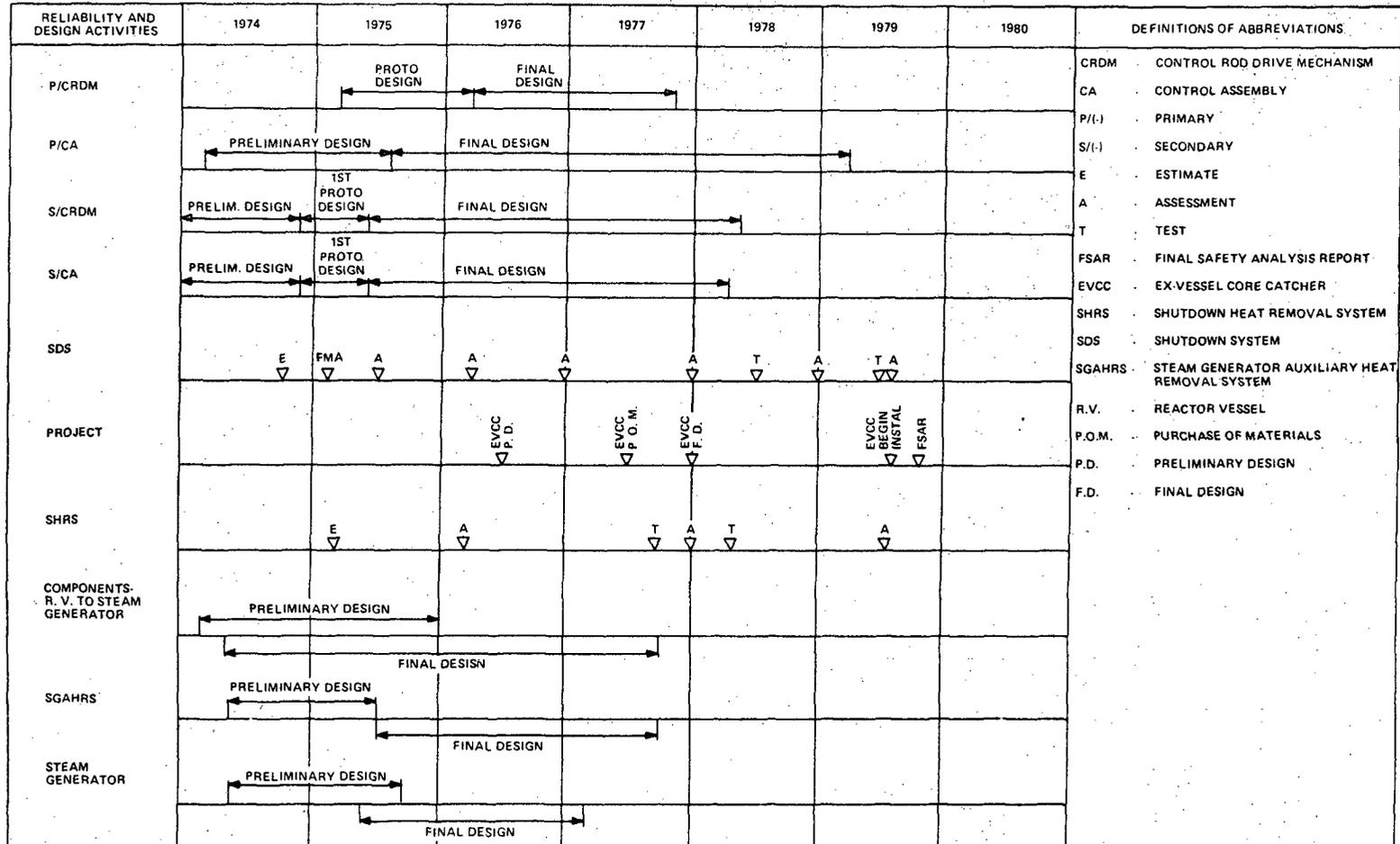


DIAGRAM 2.4B

Figure C.1-3. Phasing Of Reliability And Design Activities