5 HEAT TRANSPORT AND CONNECTED SYSTEMS

5.1 Introduction

The heat transport and connected systems will include those systems and boundaries that provide the necessary functions to safely remove and transport reactor heat to the steam generators under all plant operating conditions. The heat transport system (HTS) has been divided into five major areas that will provide for the removal of heat from the reactor under normal operations as well as under anticipated, unlikely, and extremely unlikely conditions. The division of the HTS into five systems as indicated below was done for ease of review.

(1) reactor vessel system
(2) primary heat transport system (PHTS)
(3) intermediate heat transport system (IHTS)
(4) steam generator system (SGS)
(5) shutdown heat removal system (SHRS)

The major components and subsystems of the HTS will be covered within each of the five divisions.

The overall cooling requirement of the HTS will be 975 MWt, which will be divided among three parallel, essentially identical cooling circuits. The major components of the HTS and the SGS have been sized on the basis of the HTS thermal hydraulic design conditions. A partial list is given in Table 5.1.

The preliminary heat balance of the nuclear steam supply system (NSSS) indicates that when 975 MWt is transferred from the NSSS to the balance of the plant, the required net reactor thermal power would be approximately 967 MWt. The difference between the design value of 975 MWt for the reactor and the 967 MWt reactor power represents a small margin (0.79%) in the rating of the reactor. This is due to fluid friction, decay of radioactive sodium, and PHTS and IHTS pump power in the loops. This is demonstrated in Figure 5.1 which shows the NSSS heat balance and balance-of-plant (BOP) interface.

5.2 Reactor Vessel System

The principal components that will make up the reactor vessel system are the reactor vessel, the closure head, and the guard vessel. The essential safety-related function of these components is to provide containment of coolant, cover gas, fuel, and associated thermal and nuclear activities under all anticipated, unlikely, and extremely unlikely conditions. These components will be designed and fabricated to appropriate standards that reflect the importance of this primary safety function. This system also will provide radiation shielding as well as access for surveillance material and inservice inspection and for controlling, monitoring, and servicing the core and its associated components and structures. The structural evaluation is discussed in Section 3.8.

The reactor vessel system materials have been selected to provide properties that will ensure nonbrittle behavior and avoid conditions that might result in rapidly propagating failures in the reactor vessel system. Corrosion by sodium will be minimized by controlling the oxygen concentration in the sodium. The materials are covered in greater detail in Section 5.11.

The reactor vessel system will be designed for an operating life of 30 years. The reactor vessel and closure head will be designed and manufactured in accordance with ASME Code, Section III, Division 1, "Nuclear Power Plant Components," with addenda through Winter 1974. The guard vessel will be constructed in accordance with ASME Code, Section III, Division 1, and addenda through Summer 1975.

The reactor vessel system will be designed to accommodate thermal-hydraulic design conditions of 730°F inlet temperatures and 995°F outlet temperatures and an inlet-to-outlet-nozzle pressure differential of approximately 126 psig.

5.2.1 Subsystem Description

5.2.1.1 Reactor Vessel and Support

The reactor vessel (Figure 5.2) will be a single-walled structure and will constitute the primary structural boundary that will encompass the reactor core, the core coolant, core cover gas, and the core auxiliary components and structures. The reactor vessel and support will be constructed mainly of austenitic stainless and low alloy steels and will consist of six basic sections: the support rings, the vessel flange, the barrel, the core support forging and cone, the inlet plenum, and the vessel thermal liner. As the central component in the primary heat removal system, the reactor vessel will distribute coolant flow to remove heat generated in the core, reflector, blanket, shield and other vessel internals including structural elements. The support ring will support the reactor vessel and internals and closure head. The vessel flange will be a second SA 508 Class 2 steel ring forging welded to an Inconel 600 transition section. The transition will be welded to the barrel. The barrel will comprise the upper cylindrical portion of the vessel. The lower end of the barrel will be joined to the core support forging and cone, which will provide support for the core support structure. The inlet plenum will be designed for 200 psig at 775°F and -15 psig at 600°F; the stainless steel portion of the outlet plenum will be designed for 15 psig plus head of sodium (~15 psi) at 900°F and -15 psig at 600°F.

The vessel thermal liner will protect the barrel from the high-temperature sodium in the outlet plenum and from excessive thermal transients. The annulus between the vessel and liner will be 2.5 in. Bypass coolant from the inlet plenum will be diverted to this annulus to maintain the vessel wall temperature at 900°F or below during normal operation. Two holes with special pressure reducers will be provided in the core support cone to vent gas from underneath the cone to the outlet plenum.

The bypass flow that will cool the vessel wall behind the thermal liner will enter the outlet plenum through bypass flow holes in the thermal liner located a few inches beneath the sodium pool free surface level. Sufficient mixing will occur away from metal surfaces so that excessive fatigue damage to the type 316 stainless steel should not occur.

Four auxiliary nozzles will be provided in the upper part of the barrel. Two of the nozzles will be for the auxiliary sodium system: one inlet nozzle for sodium makeup and one outlet nozzle for sodium overflow. The other two nozzles will be for the cover gas system: one gas inlet nozzle and one gas outlet nozzle. The nozzles will provide a loop for the control of the sodium level, for the removal of impurities from the sodium, and for the taking of samples of the coolant. The cover gas nozzles will provide for circulation of the cover gas to and from the radioactive argon processing system (RAPS) and for controlling the pressure of the primary coolant system.

Sodium coolant will enter the reactor vessel from the heat transport system through three 24-in. nozzles located 120° apart in the inlet plenum at the bottom of the reactor vessel. The coolant will be mixed within this region and distributed through the core component and liner cooling annulus. Core flow and bypass flows will be mixed in the outlet plenum above the core and discharged through three 36-in. nozzles. Sodium level during normal operation will be maintained approximately 14.5 ft above the top of the outlet nozzles.

In the event of a pipe break, sodium will be maintained at minimum safe level (approximately 20 in.) above the primary outlet nozzles to ensure continuity of the primary heat transport system (PHTS) loop syphon in undamaged loops for heat removal. This level will be approximately 9 ft above the top of the core. The excess coolant also will ameliorate differences in channel discharge temperatures and mitigate the severity of coolant temperature transients.

The reactor vessel design includes provision to accommodate inservice inspection. Capability to visually inspect the exterior surface of all critical pressure boundary welds will be provided. Accessibility to critical areas between the guard vessel and reactor vessel will be provided for inservice visual inspection by a remote surveillance system throughout the life of the vessel. Refer to Section 5.12 for evaluation of inservice inspection.

5.2.1.2 Closure Head

The closure head (Figure 5.3) will provide the closure for the reactor vessel and will be attached to the vessel flange and complete the vessel pressure boundary. Access to the reactor vessel interior will be provided through various penetrations in the closure head. One of the primary functions of the closure head is to provide positioning and support for the reactor control rods, instrumentation, and refueling equipment. Refueling equipment will be positioned by three eccentric rotating plugs that will provide access to all routinely removable elements in the reactor core. Each rotating plug will be provided with a mechanical lock and electrical interlocks, which will prevent plug rotation during reactor operation and during refueling operations when plug rotation is not desired. The closure head will have an average temperature of 400°F under both normal and refueling conditions. This will ensure that the liquid sodium will not freeze. Equipment and components requiring repair will be located in areas that will be accessible for maintenance.

A series of metal plates will extend beneath the head to provide thermal and radiation shielding and to suppress gas entrainment at the surface of the sodium coolant. Each plate will be supported by a central support column affixed to the lower shield plate. The support columns will be inserted in the lower

shield plate and bolted to the top surface of the lower plate. A single support will provide adequate support and reduce the thermal stresses by permitting the plates to flex freely under the expected thermal gradient.

5.2.1.3 Guard Vessel

The reactor vessel will be surrounded by a bottom-supported, right circular cylindrical guard vessel (Figure 5.4) fabricated from SA240 type 304 stainless steel. The purpose of the guard vessel is to ensure that the core and reactor vessel outlet nozzles will remain submerged in sodium in the event of a coolant leak in the piping connections or reactor vessel nozzles. The guard vessel will be sized so that it will extend approximately 6 ft above the minimum safe sodium level providing for sodium shrinkage and pumping head differential. The guard vessel also will provide a uniform annulus for inservice inspection of the reactor vessel, with clearances that will preclude contact with the reactor vessel and piping under accident conditions. Insulation will be provided on the outer surface to limit the heat load into the reactor cavity cell. A trace heating system will be mounted on the outer surface of the vessel for preheating and heating during prolonged plant shutdown.

The maximum and minimum widths of the radial gap between the guard vessel and the reactor vessel have been conservatively calculated. Thermal expansion, deviations in alignment of the two vessels, out-of-roundness, tolerances in the diameters of the two vessels, and possible deviations in straightness caused by manufacture and subsequent operation are some of the relevant factors that have been taken into account in the design.

Continuity detectors and aerosol sampling lines will be mounted inside the guard vessel to detect potential leaks in the reactor vessel or inlet or outlet piping and are discussed in greater detail in Section 5.7. The guard vessel will be sized to provide a nominal clearance of 8 in. between the walls of the reactor vessel and the guard vessel that would control the sodium lost in the event of a leak.

5.2.2 Evaluation

The extent of the review of the reactor vessel system included the design criteria, regulatory positions applied (see Section 3.9) design of the components, and development testing done in support of the design. On the basis of its review, the staff has concluded that the reactor vessel system design, design bases, and analyses will satisfy the intent of the following principal design criteria (PDC):

(1) PDC 4, "Protection Against Sodium and NaK Reactions," as it relates to the design and location of components and systems containing sodium or NaK to limit the reaction of Na/NaK spills. Fire control, inerted cells, and leak detection will be provided to limit and control these reactions and ensure that the components will perform their safety functions

(2) PDC 11, "Instrumentation and Control," as it relates to the monitoring of system variables such as temperature, pressure, flow, and system leakage under various operational conditions

(3) PDC 26, "Heat Transport System Design," as it relates to the system design and the ability to remove heat from the reactor and transport it to the turbine generator or some ultimate heat sink under all plant conditions. Provisions for diversity, redundancy, and independence were considered in this review

(4) PDC 27, "Assurance of Adequate Reactor Coolant Inventory," as it relates to the ability of the system to limit the loss of reactor coolant so that an adequate inventory will be maintained to remove reactor heat under various conditions

In addition to the above PDC, the system was reviewed to verify that provisions for the following were applied to the design:

(1) PDC 5, "Environmental and Missile Design Bases"
(2) PDC 8, "Reactor Design"
(3) PDC 12, "Reactor Coolant Boundary"
(4) PDC 13, "Reactor Coolant System Design"
(5) PDC 28, "Quality of Reactor Coolant Boundary"
(6) PDC 29, "Fracture Prevention of Reactor Coolant Boundary"
(7) PDC 30, "Inspection of Reactor Coolant Boundary"

These criteria are applied in the direct review in other sections and were evaluated in this section only to provide guidance for the review of the design.

In conjunction with the PDC, the Standard Review Plan (SRP) was used to review this system. Design details contained in Chapter 5 of the SRP for the reactor coolant system connected systems were applied to the materials and the structural integrity of the system and are found in Sections 3.9 and 5.11. The following SRP sections were used directly in the review of this system inasmuch as they apply to a reactor of this type. Although the SRP sections used in the review were not written specifically for a sodium-cooled reactor, the intent of the SRP was used as guidance.

(1) SRP Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," as it relates to periodic inspection and testing of important areas of the reactor coolant boundary covered in Section 5.12.

(2) SRP Section 5.2.5, "Reactor Coolant Boundary Leakage Detection," as it relates to the liquid-metal-to-gas leak detection system covered in Section 5.7.

The essential safety related function of these components is to provide containment of coolant, cover gas, fuel, and associated thermal and nuclear activities under anticipated, unlikely, and extremely unlikely conditions. These components will be designed and fabricated to appropriate standards that reflect the importance of this primary safety function.

The reactor vessel, closure head, and guard vessel will be designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code. Section III, Division 1, Nuclear Power Plant Components, with addenda as shown in Table 5.2-1 of the PSAR. Applicable ASME Code Cases for elevated temperature

components and their effective dates will also be imposed and are covered in Section 3.9. Access for surveillance materials and in-service inspection is provided and discussed in greater detail in Section 5.12.

Thermal and stress analyses of the reactor vessel, closure head, and guard vessel will account for all anticipated, unlikely, and extremely unlikely conditions that will be imposed on the plant. The analyses performed by the applicants cover steady-state and transient conditions for loads resulting from dead weights, temperature effects, seismic events, internal pressure, vibration, and fluid-flow forces.

Dimensions of the reactor vessel, closure head, and guard vessel will be measured and checked against the dimensions and tolerances specified on the manufacturing drawings. Any deviations will be documented by Nonconformance Reports. Deviations that do not meet the requirements of the ASME Code will not be permitted. Pressure tests will be performed on the completed reactor vessel and on the completed closure head as required by the ASME Code. The high-pressure inlet plenum portion of the reactor vessel has been pressure tested to a pressure of 250 psig. The pressure test also provided structural verification of the core support structure, although not required by the ASME Code. The closure head was pneumatically tested to a pressure of 18 psig.

To promote mixing of the three inlet streams in the reactor vessel inlet plenum and minimize thermal gradients in the pressure boundary of the inlet plenum, each inlet nozzle will be provided with an internal pipe elbow that will deflect the flow downward and away from the wall. In this manner, the mixing of the entering sodium will occur in the interior of the plenum, providing coolant uniform temperature to core components.

On the basis of technical input from its consultant (Brookhaven National Laboratory) and its evaluation of the reactor vessel system, the staff concludes that this system will meet the guidelines of SRP Section 5 as applied to this review, and the requirements of PDC 4, 5, 8, 11, 12, 13, 26, 27, 28, 29, and 30 and that the design will be acceptable for a construction permit (CP).

5.3 Primary Heat Transport System

5.3.1 General Description

The primary heat transport system (PHTS) will consist of three parallel circuits of piping and components, which will operate independently of one another, yet have common flow paths through the reactor vessel. The preliminary heat balance of the nuclear steam supply system is shown on Figure 5.1. Each circuit will contain essentially identical primary and secondary sodium cooling loops arranged to permit natural circulation for shutdown decay heat removal.

For separation purposes, the components in the primary loops will be located in separate cells in the reactor containment building so that a fault in one circuit can be isolated from the two remaining cirucits. For maintenance purposes, the components in the secondary loops will be located in equipment cells in a service building outside the reactor containment building.

5.3.2 Subsystem Description

5.3.2.1 Primary Sodium Loop

The PHTS loops will transport the radioactive sodium coolant from the reactor vessel to the intermediate heat exchangers, (IHXs) which will thermally link the primary and intermediate loops. Heated sodium will flow from the reactor vessel outlet nozzles, located above the reactor core, to the suction of a centrifugal, free surface pump. The sodium from the pump discharge will be circulated to the shell side of the IHX where the heat will be transferred to the secondary loops. The primary sodium flow will continue from the IHX through a check valve, magnetic flow meter, and through the cold-leg piping.

The primary sodium will enter the inlet nozzles located in the lower end of the reactor vessel and flow through the reactor core, thus completing the flow circuit. All three loops will be essentially equivalent and arranged to provide nearly identical sodium transport times. By varying the sodium flow rate essentially in proportion to the power produced in the reactor, transfer of power by the heat transport system will be achieved. The flow rate of primary sodium will be maintained and controlled by varying the pump speed. The flow will be monitored by a magnetic flow meter located in the cold-leg piping downstream of the IHX, between the check valve and the reactor vessel.

The primary system hot-leg piping from the reactor vessel outlet to the primary pump will have an outside diameter of 36 in. and a wall thickness of 0.500 in. The remaining hot-leg piping (pump discharge to IHX) and the cold-leg piping from the IHX to the reactor inlet nozzle will have an outside diameter of 24 in. and a wall thickness of 0.500 in. The following considerations were included in sizing the primary system piping:

(1) The sodium velocity will be less than 30 ft per second, at maximum flow, in the primary loop piping.

(2) An allocated pressure drop for the primary loop piping is consistent with pressure drop allocations for the system components, reactor, and the single-stage primary pump.

The primary system hot-leg piping will be type 316 stainless steel and the cold-leg piping will be type 304 stainless steel. Welded pipe connections will be used throughout the system. Pipe support will be provided by constant load hangers sized to support the insulated piping system when filled with sodium. Mechanical snubbers also will be provided to limit pipe deflections during seismic loading. The primary sodium pumps and intermediate heat exchangers will be rigidly supported to maintain fixed center line locations and will be encased by free-standing, structurally independent guard vessels. Differential thermal expansion will be accommodated by the inherent flexibility in the piping arrangement and by gaps existing between the major components and the guard vessels. The primary heat transport system (PHTS) piping and components will be located within the reactor containment building, which will be a seismic Category 1, tornado- and missile-hardened building. The three heat transport system loops will be located in three individual separate cells inside containment. The motors, speed controllers, and heating and seal assemblies for each pump drive system will be located in an air environment in containment above

the operating floor. This separation of equipment will provide the capability for independent access for inspection and maintenance.

The intermediate heat exchanger (IHX) tube bundle will provide a passive mechanical barrier isolating radioactive primary sodium from intermediate sodium and prevent the transport of primary sodium out of containment. The pressure on the intermediate heat transport system on the side of the IHX will be maintained at a minimun of 10 psi above the primary sodium (shell side) to ensure that in the event of any tube leaks, radioactive sodium will not leak into the intermediate system. A requirement for a positive Δp from intermediate to primary will be specified in the Techical Specifications (TS).

The primary loops will be arranged so that natural circulation will provide sufficient heat transport to remove decay heat in the event that all power to the pumps is lost. However, the pumps will be equipped with pony motors to provide forced circulation for normal decay heat removal during a loss of offsite power. The pony motors will be supplied with normal and emergency diesel power (see Section 8) and will be designed to provide approximately 7½% of the system design flow.

The subject of natural circulation is mentioned throughout this evaluation with respect to the HTS. The thermal-hydraulic analysis of a reactor is an important part in the design process as well as the operational analysis of the plant. Therefore, the topic of natural circulation will be discussed in greater detail in Section 5.6.3.

5.3.2.2 Primary Sodium Pump

The primary sodium pumps (Figure 5.5) will be free-surface, single-stage, vertically mounted, centrifugal pumps driven by a variable-speed, 5,000-hp, squirrel-cage induction motor. Low flow capability (7½%) for decay heat removal and other low power standby conditions will be provided by an auxiliary 75-hp pony motor. Motive power for the pump's main drive motor will be supplied by a variable frequency, fluid motor-generator (MG) set, which will produce the variable pump speeds. Each pump will be designed to deliver 33,700 gpm of 995°F sodium at a 458-ft head. Pump design will be based on sodium level control using a stand pipe bubbler system in which a continuous flow of argon gas will be supplied to the pump cover gas space above the sodium free surface. The standpipe essentially will be one leg of a manometer that automatically will balance the gas pressure in the pump tank. The standpipe bubbler will automatically perform the function of a pump cover gas vent and relief valve to the cover gas system and will not rely on the signal of a liquid level sensor for level control. Level sensors will be available for alarm.

It is a design requirement and has been demonstrated analytically that failure of any part of the pump rotating assembly will not affect the integrity of the pressure boundary. Additionally, failure of any part of the rotating assembly or the lower region of the inner structure will not reduce the flow path area below a required minimum for natural circulation under shutdown conditions. Other areas of design that were considered involve missiles resulting from a fracture of the pump rotating assembly at maximum overspeed; first bending critical speed of the shaft will be at least 125% of design speed, and the potential for amplification of impeller volute reaction pressures by acoustic

resonance within various portions of the pump will be minimized. These and other design factors have been substantiated by analytical methods and testing. Currently, pump tests are being performed at the Energy Technology Engineering Center (ETEC) in California on the CRBR pump. The pump tests include not only the CRBR unit using a fluid-drive coupling, but also those with solid-state drives. The results from these tests will be evaluated when the testing has been completed and the data reduced to a useful form. The results will then be used for validating computational tools.

5.3.2.3 Intermediate Heat Exchanger

The IHX (Figure 5.6) being used for the CRBR project basically will be a straight tube, flexible downcomer design consisting of an enclosure shell, tube bundle with replaceable bellows, and cylindrical hanging support. The main subassemblies and other design features, which include the shell assembly, tube bundle assembly, hanging support, and channel subassembly, are discussed in greater detail in Section 3.8 of the SER. The vendor of the IHX will perform hydraulic flow model testing to verify performance characteristics. These tests should establish uniformity of the flow distribution in the IHX as well as ensure flow stability and predictions of heat transfer performance. Additionally, a full range of vibrational tests will be performed to confirm the absence of damaging tube vibrations.

In evaluating the IHX with respect to the design-basis sodium-water reaction, the conditions selected by the applicants were judged to be more conservative than the steam generator design-basis sodium-water reaction (see Section 5.5). The PSAR presents a design basis event for the IHX design that consists of an instantaneous rupture of an evaporator or superheater tube followed by the rupture of six adjacent tubes, which results in actuation of the rupture disk, automatic isolation and blowdown of the evaporator modules and superheater in the affected loop, and manual actuation of the sodium dump system. The applicants have calculated the pressure pulse resulting from a sodium-water reaction (shown in Figure 5.7). The peak pressure experienced by the IHX for this event is approximately 330 psia. The IHX will be designed to a pressure of 700 psia. On the basis of this design, the IHX would have a margin of almost 370 psia for the design-basis sodium-water reaction. In addition, a trip of the reactor, turbine, and sodium pumps occurs. The intermediate sodium system experiences a pressure transient resulting from the reaction. Differential pressure between the primary and intermediate sides during this event was evaluated and determined to have adequate margin.

The IHX was also evaluated for check valve slam. This event occurs as a result of primary pump failure. The accident consequences on the reactor are discussed in Section 15.3. The applicants have concluded that the event will involve a postulated instantaneous stopping of the primary pump impeller while the system is operating at 100% power. A reactor trip will be initiated by the primary/intermediate flow ratio subsystem. The transient responses of temperature, flow, and pressure on both primary and intermediate sides of the IHX were satisfactory with adequate margin-to-design limits. Both the sodium-water reaction and check valve closure events are classified as unlikely events for the IHX. In evaluating the adequacy of the IHX with respect to these events, the dynamic nature of the intermediate sodium pressure history, which in turn is used to determine the pressure-induced primary stresses, and according to

the PSAR the fatigue damage associated with the cyclic nature of the pressure history were accounted for as required by ASME Code Case 1592.

5.3.2.4 Guard Vessels

The IHXs and primary sodium pumps will be located in structurally independent, free-standing guard vessels which will be fabricated from type 304 stainless steel.

The IHX and primary pump guard vessels will enclose the component and its primary system inlet and outlet piping up to a level which will preserve the reactor system minimum safe sodium level in the unlikely event of component leakage. The top elevation of each guard vessel will be sufficient to prevent spillage of sodium over the lip when the pump is running at pony motor speed and the sodium is at the minimum safe level in the reactor. The guard vessels will have segmented covers. These covers will not establish a pressure boundary. The covers will contain penetrations for a drain and purge pipe, a gas sample tube, and leak detector guide tubes. Additionally, inspection ports will be provided to facilitate periscope inspection of the component during refueling, maintenance, or other shutdown periods. The maximum pressure which can be experienced by the guard vessels will be equal to the sodium head of the filled vessel.

The volume of the guard vessel and the volume of sodium above the minimum safe level (MSL) in the reactor vessel and pump tank have been sized to ensure that the guard vessel volume will be less than or equal to the volume loss from the reactor and pump tanks for any leak condition assuming contraction and cooldown after the leak, assuming no reactor vessel makeup. At minimum operating sodium level the volume of sodium above MSL in the reactor vessel and pump tanks will be 4430 ft^3. The net volume of each PHTS guard vessel will be 2,700 ft^3. The net volume of the reactor guard vessel will be 3,100 ft^3.

Continued reactor cooling will be provided in the unlikely event of a pipe failure by the PHTS elevated piping arrangement. All PHTS piping will be routed at an elevation above the tops of the PHTS guard vessels thereby limiting the loss of coolant in the unlikely event of a pipe failure.

The combination of guard vessel elevation, guard vessel volume, reactor vessel and pump tank sodium inventory above the minimum safe level, pony motor shutdown head, and elevated piping ensures a limited loss of reactor coolant and continued reactor cooling capability.

5.3.2.5 Primary Heat Transport System Check Valves

The only valves located in the primary heat transport system will be the cold-leg check valves. The function of these valves will be to limit flow reversal in a loop which has a pump failure. The valve will be physically located in the 24-in.-diameter cold-leg piping on the horizontal section between the intermediate heat exchanger and the reactor inlet.

The hydraulic characteristics of the check valve will be similar to those of the FFTF cold-leg check valve (CLCV) on which the CRBRP valve design will be based. Tests performed on the FFTF valve showed that no accumulation of gas should occur in the valve body over the full 40% to 100% flow range.

The applicants have stated that additional evaluations will be made of the flow phenomena within the CLCV body to determine the likelihood of gas accumulation in the CLCV body during plant operation, although existing evaluations showed that there is no place in the PHTS where gas bubbles can accumulate (see Section 15.2.3.2, PSAR). Additionally, information will be provided in the FSAR either to confirm that gas accumulation in the CLCV will not be a safety concern, or to define the actions to be taken to maintain gas accumulation below a known safe level. This is not to preclude venting of the CLCV if found necessary.

5.3.3 Evaluation

The components of the reactor coolant pressure boundary, as required by 10 CFR 50.55a, have been properly identified and classified. These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and addenda required by 10 CFR 50 codes and standards (see Section 3.9).

On the basis of its review and evaluation completed by its consultant (Brookhaven National Laboratory), the staff concluded that the PHTS design, design bases, and analyses satisfy the intent and form a portion of the body of information that satisfies the following CRBR Principal Design Criteria (PDC):

(1) PDC 4,"Protection Against Sodium and NaK Reactions," as it relates to the design and location of components and systems containing sodium or NaK to limit the reaction of Na/NaK spills. Fire control, inerted cells, and leak detection will be provided to limit and control these reactions and ensure that the components will perform their safety functions

(2) PDC 26, "Heat Transport System Design," as it relates to the system design and the ability to remove heat from the reactor and transport it to the turbine generator or some ultimate heat sink under all plant conditions. Provisions for diversity, redundancy, and independence were considered in this review

(3) PDC 27,"Assurance of Adequate Reactor Coolant Inventory," as it relates to the ability of the system to limit the loss of reactor coolant so that an adequate inventory will be maintained to remove reactor heat under various conditions

(4) PDC 11,"Instrumentation and Control," as it relates to the monitoring of system variables such as temperature, pressure, flow, and system leakage under various operational conditions

In addition to the above PDC, the system was reviewed to verify that provisions for the following will be applied to the design:

(1) PDC 5, "Environmental and Missile Design Bases"
(2) PDC 8, "Reactor Design"
(3) PDC 12, "Reactor Coolant Boundary"
(4) PDC 13, "Reactor Coolant System Design"
(5) PDC 28, "Quality of Reactor Coolant Boundary"
(6) PDC 29, "Fracture Prevention of Reactor Coolant Boundary"
(7) PDC 30, "Inspection of Reactor Coolant Boundary"

These criteria are applied in the direct review in other sections and were evaluated in this section only to provide guidance for the review of the design.

In conjunction with the PDC, the Standard Review Plan (SRP) was used to review this system. Design details contained in Chapter 5 of the SRP for the reactor coolant system connected systems were applied to the materials and structural integrity of the system and are found in Sections 3.9 and 5.11. The following SRP sections were used directly in the review of this system inasmuch as they apply to a reactor of this type. Although the SRP sections used in the review were not written specifically for a sodium cooled reactor, the intent of the SRP was used as guidance.

(1) SRP Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," as it relates to periodic inspection and testing of important areas of the reactor coolant boundary covered in Section 5.12.

(2) SRP Section 5.2.5, "Reactor Coolant Boundary Leakage Detection," as it relates to the liquid-metal-to-gas leak detection system covered in Section 5.7.

(3) SRP Section 5.4.12," Reactor Coolant System High Point Vents," as it relates to the capability of the system to exhaust noncordensible gases from the primary system that could inhibit natural circulation cooling (IHX vent line).

The components and structures of the primary heat transport system (PHTS) will be designed to accommodate the thermal transients resulting from anticipated, unlikely, and extremely unlikely conditions. These events are discussed in Appendix B of the PSAR, and the staff evaluation is discussed in Section 3.9 of the SER. The staff has concluded that the system will be designed so that an anticipated condition will not adversely affect the useful life of any heat transport system component. During and following an unlikely event, the heat transport system will remain sufficiently intact in order to perform its decay heat removal function. Additionally, on the bais of analyses computed by the applicants, the PHTS will accommodate, without loss of decay heat removal capability, the pressures imposed on the intermediate system by a steam generator design-basis sodium-water reaction and the sodium hammer resulting from a primary loop check valve closure caused by an event such as a primary pump seizure.

The only active component that is considered part of the PHTS is the primary pump. In the event of pipe leaks, the primary pumps will be automatically reduced to pony motor flow following shutdown of the reactor manually or reactor scram resulting from to reactor vessel low sodium level. The PHTS guard vessels will be designed so that they will be at an elevation that will be approximately 9 ft above the top of the reactor vessel discharge nozzles. This level is based on the combination of the pony motor shutoff head of 5 ft and the minimum safe reactor vessel level, which is 2 ft above the top of the reactor discharge nozzle, plus an additional 2 ft to accommodate sodium shrinkage and hydraulic uncertainties. The pony motors will have insufficient head to pump sodium out of a leak located outside (and subsequently above) the guard vessels. These features will limit the loss of sodium inventory in the event of a leak in the

PHTS and ensure that the core will remain covered with sufficient sodium for the unaffected PHTS loops to operate.

Overpressure protection in the reactor coolant system will be performed in the cover gas system. Pressures in the system normally will be a function of the cover gas pressure and the pressure developed by the main circulation pumps. Since there will be no isolation valves in the system, it will not be possible to isolate individual loops or components. The cover gas system is discussed in Section 9.5. A discussion of overpressure protection of the cover gas distribution system is covered in Section 15.7.2.

Primary to intermediate coolant leakage is considered unlikely because the intermediate system will be required to operate at a higher pressure than the primary. In addition, maintaining a positive differential across the IHX is a limiting condition for operation of the plant and will be specified in Section 16. Leakage of primary sodium into the IHTS, should it occur, will be detected by radiation monitors provided on the IHTS piping.

The thermal and hydraulic (T/H) characteristics of the reactor coolant system were determined using a CRBRP steady-state T/H plant model. T/H design-basis parameters are given in Table 5.2. Component structural design pressures and temperatures are given in Table 5.3.

System response with respect to anticipated, unlikely, and extremely unlikely conditions has been evaluated, and on the basis of the information presented by the applicants, the criteria applied to the reactor coolant pressure boundary provides reasonable assurance that the resulting quality standards will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and are considered acceptable. The analytical methods, thermodynamic data, and hydrodynamic data used in the T/H analysis for the CRBR and their applications were confirmed by independent confirmatory analyses discussed in Section 4.4 of this SER. Some development tests remain to be completed and/or documented. The applicants have committed to performing these tests before the FSAR is issued and before the startup period (letter, Dec. 20, 1982, HQ:S:82:149). Primary sodium pump flow tests and pony motor flow characterization tests are being performed.

Variable speed performance curves for the CRBRP sodium pump based on data obtained during water tests were made available by the applicants to evaluate pump performance. These were projected performance curves for operation in sodium. Performance mapping with 1,000°F sodium is now being performed. The applicants have agreed to make these data available when testing is completed so that independent calculations for natural circulation at the operating license review stage can be made.

The staff concludes that the design of the PHTS meets the requirements of PDC 4, 5, 8, 11, 12, 13, 27, 28, 29, and 30. On the basis of this, the staff concludes that the design of the PHTS is acceptable for issuance of a construction permit.

5.4 Intermediate Heat Transport System

5.4.1 General Description

The intermediate heat transport system (IHTS) will consist of the components and piping required to transport heat generated in the reactor from the intermediate heat exchanger (IHX) to the steam generator system (SGS). The system will consist of three essentially equivalent, independent, cooling loops operating in parallel. Each loop will contain nonradioactive sodium that will be circulated by means of an intermediate sodium pump from the tube side of the IHX through the steam generator and back to the IHX. There will be no direct sodium interconnections between the three loops.

5.4.2 Subsystem Description

5.4.2.1 Intermediate Sodium Loop

The IHTS sodium piping from the IHX will be located inside a cell in the containment building except where it will leave it through a containment penetration. The remainder of the loop will be located within a seismic Category I, tornado-resistant, missile-hardened accessible cell within the steam generator building (SGB).

Heated intermediate sodium leaving the IHX will flow through the hot-leg piping and enter the SGS superheater module. At the superheater exit, the piping will consist of two parallel runs, and extend to each of the two evaporator inlets. Cooled sodium from the evaporators will flow through two parallel runs joined together at a tee and continue as a single run to the IHTS pump suction. From pump discharge, the sodium will flow through piping to complete the circuit through the IHX.

Each cold leg of the intermediate loop will contain a vertical, single-stage, free-surface centrifugal pump. The sodium flow rate will be essentially the same as for the primary pumps and will be controlled by a variable-speed pump drive. A pony motor will be used to provide 7½% of design flow for operation during startup, shutdown, and hot standby and following accident conditions.

A permanent magnetic flow meter will be installed in the cold leg downstream from the pump in loops 1 and 3 (west and east loops, respectively) and in the hot leg of loop 2 (middle loop). This flow meter will provide sodium flow rates for process control and plant protection functions. A Venturi flow meter will be provided to measure the inlet sodium flow to each evaporator module in one loop of the plant. This flow measurement is part of the diagnostic instrumentation for the steam generators.

An expansion tank will be provided in each IHTS loop cold leg to accommodate the sodium volume change resulting from thermal expansion over the operating range. Each tank will be located in the SGB near its corresponding sodium pump. The system cover gas operating pressure will be maintained by control of the argon gas supply. The cover gas volumes in the expansion tank and pump tank will be interconnected. A vent-sodium stream for the superheater and each evaporator will flow from the high point bundle of the module. This stream

will provide positive flow through the upper tube bundle of the module to prevent gas buildup and provide a source of sodium for sodium-water leak detection.

Although each loop will have similar components, the loop piping will differ in length and configuration because of the differences in distance between the IHX units and the piping penetrations into the steam generator cells. Piping configurations within the steam generator cells will be identical for the three loops. Within each loop, equal flow to the two evaporators will be maintained through the use of identical piping between the superheater discharge and the evaporator inlets. The piping from the evaporators to the IHTS mixing tee and from the mixing tee to the pump also will be identical in all loops.

5.4.2.2 Intermediate Sodium Pumps

The intermediate sodium pumps will be free-surface, single-stage, vertically mounted, drawdown-type centrifugal pumps driven by a variable-speed, 5,000-hp squirrel cage induction motor. An auxiliary 75-hp pony motor on each pump will provide low flow capability. Variable pump speed from 22½% to 100% design speed will be achieved by the main drive motor, which will be supplied with variable frequency power from a fluid-coupled motor generator.

The intermediate sodium coolant pump will be identical to the primary pump in design; however, there will be no radiation or thermal shielding required and the head and flow requirements will be different. The difference in head and flow requirements is based on hydraulic resistance at steam generator system thermal hydraulic conditions. Additionally, there is the requirement to maintain a positive pressure on the IHTS side of the IHX with respect to the PHTS side of the IHX.

5.4.2.3 Intermediate Heat Transport System Piping and Support

The intermediate heat transport system (IHTS) piping will conduct sodium in a continuous loop to transport reactor heat to the steam generator system. Each IHTS piping run will be provided with the necessary elbows, tees, and reducers to provide adequate loops for thermal expansion. Although each loop will contain the necessary appendage piping and associated fittings, there will be no valves in the main sodium-flow piping. Each loop will have similar components, although the loop piping will differ in length and configuration because of the differences in distance between the IHX units and steam generator modules.

The hot-leg piping will extend from the IHX outlet nozzle through the reactor containment penetration to the superheater inlet nozzle. From the superheater outlet nozzle, two parallel runs will extend to each of the two evaporator inlet nozzles. The cold-leg piping will begin at the evaporator outlet nozzles and will consist of two parallel runs joined by a mixing tee to the IHTS sodium pumps to the IHX.

5.4.3 Evaluation

The IHTS has no parallel in light-water reactors and, therefore, precise guidance in terms of general design criteria was not available. The IHTS was designed to satisfy Class 1 requirements. Therefore, guidance from the LMFBR Standard Format and Content (SFAC) (USAEC, Feb. 1974) which interprets the IHTS

as comprising part of the reactor coolant system, was used. The basic acceptance criteria for this section as in the PHTS were based on meeting PDC 26, 27, 31, 32, and 33. The design description and objectives described in the PSAR are adequate to assess the ability of the system to meet these objectives.

The intermediate pumps will be essentially the same as the primary pumps, and the minor differences mentioned earilier cause no design difficulties. As with the primary pumps, the availability of prototype performance will ensure that the design objectives will be met.

The use of ferritic steel and transition spool pieces in the IHTS piping at elevated temperatures is covered in Section 3.9 of this SER which addresses the issues of piping integrity and stress evaluation.

The IHTS will have no isolation valves within the normal circulation path so that isolation of individual system pipe sections or components will not be possible. If for some reason the system did become blocked, the intermediate pump would not overpressurize the system because the IHTS structural design will be sufficient to withstand the static head of the pump. In the event of an argon supply valve failure, the system would not be overpressurized because the combination of argon supply pressure of 115 psi and pump shutoff head would not exceed the IHTS design pressure. This is true even if the pump shutoff head associated with the PHTS pump were reached instead of the IHTS pump shutoff head.

The IHTS may be subjected to a pressure pulse in the event of a water or steam leak in the steam generation system. For large or intermediate sodium-water reactions, the resulting pressure increase resulting from the formation of reaction products in the faulted evaporator or superheater module will be relieved through rupture disks. This is covered in greater detail in Section 5.5.4, which discusses the steam generation system.

To provide sufficient intermediate system flow following a postulated loss of all pumping power, the IHTS piping and components will be arranged to promote natural circulation of the sodium. The hydraulic profile shown in Figure 5.8 shows relative elevations of piping and components.

The sodium inlet at the evaporators will be 35 ft above the geometric center of the IHX. For offnormal and transient conditions, the resulting thermal head as indicated in Figure 5.8 will be sufficient to provide the required postshutdown IHTS flow.

With loss of pumping power and start of natural circulation, the thermal center in the IHX will move. The plant transient computer program, DEMO, was used by the applicants to calculate the natural circulation flow rates. The IHX model is highly nodalized (45 nodes in the direction of flow path on each side). Heat transfer was calculated individually at each node. As a result, shifts in thermal center elevation were inherently calculated by this mode. Consequently, the transient results presented included changes in coolant used by circulation resulting from the shift of the thermal center.

Inservice inspection will be performed on the components of the IHTS to provide a continuing assurance that these components can perform their functions safely.

The examinations will include continuous monitoring by liquid metal-to-gas leak detection systems (covered in Section 5.7) and visual inspections of the IHTS components. Dissimilar metal welds will be inspected by volumetric methods. Hangers and snubbers will be visually inspected. Snubbers also will be functionally tested to ensure continued operability.

The IHTS design for the CRBR was reviewed by the staff and its consultants (Brookhaven National Laboratory). The scope of the review included the design criteria, analysis of thermal-hydraulic performance, and the thermal-hydraulic design. The review concentrated on the design of the components, the criteria the applicants, and the analysis methods and development testing done in support of the design. Some of the thermal-hydraulic design will have to be confirmed by final analysis and startup testing; however, the design criteria appear satisfactory and have led to a design that should operate as intended with adequate margin and conservatism to the design limits. No hydraulic instabilities are expected since this system is entirely one-phase flow. The applicants will establish a preoperational and initial startup test program to measure and confirm thermal-hydraulic design aspects. The staff concludes that the proposed design of the IHTS will meet the requirements of PDC 26, 27, 31, 32, and 33 and is capable of meeting its design objectives and, therefore, is acceptable for a construction permit.

5.5 Steam Generation System

5.5.1 General Description

In many ways the fast flux test facility (FFTF) is the prototype for CRBR. FFTF was, however, a test reactor to evaluate fuels and materials for the LMFBR program and, therefore, for simplicity of operation had no conversion equipment. For the demonstration plant, the steam generators have dual importance: (1) the steam generators/superheaters are the components that couple the conventional steam conversion part of the system and the sodium loops and (2) there is no FFTF experience to draw on for this component.

There have been some well publicized problems associated with foreign steam generator experience, for example, the Soviet BN-350 and the British Prototype Fast Reactor (PFR). Both of these systems had extensive and persistent problems with steam generator leaks. There has also been some positive experience with steam generators, perhaps not so well publicized. EBR-II has operated a sodium-water steam generator for 19 years without a water-to-sodium leak, and the French Phenix steam generator operated 10 years before it experienced its first leak.

When the superheater evaporator develops a leak, high-pressure water/steam is forced into the relatively low-pressure sodium. Because of the highly reactive nature of sodium when it comes in contact with water, the resulting jet of water/steam forms a reaction front where the two fluids are in quasi-equilibrium and reacting. This reaction front produces considerable heat and high temperatures (1,500°-2,000°F) and rapidly liberates a considerable volume of hydrogen.

The IHTS will see the hydrogen volume from a sodium-water reaction as a pressure pulse. The surrounding steam generator (SG) tubes will see the jet produced as a hot caustic jet that will cause wastage (corrosion/erosion) and thermally related failure if properly oriented. Knowing the bounding condition for a fault of the steam generator, the IHTS, especially the IHX, can be designed to survive this failure. Also, a mitigating system can be designed to limit the pressure pulse and put the SG in a condition that will ameliorate the event and protect the system from further damage. CRBR SGs will have such a system as discussed in subsequent sections. This system is known as the sodium-water reaction pressure relief system (SWRPRS).

5.5.2 Subsystem Description

5.5.2.1 Steam Generators

The three IHTSs each will have an independent steam generation system. This system will be composed of two identical and parallel connected evaporators that will supply steam to a single superheater unit. The evaporator and superheater modules will be of identical design. Therefore, there will be nine identical SG modules in CRBR. The applicants intend to have an onsite spare to cover contingencies.

In the IHTS, the superheater will receive 922°F sodium from the IHX and produce 906°F, 1,550 psia steam, which will go to a header and then to the turbine generator, which is discussed more fully in Section 10. The sodium will leave the superheater at 845°F. The sodium flow will then split into two essentially identical hydraulically parallel flow paths into the two evaporator modules. The temperature of sodium entering the evaporator modules will be 845°F, and 651°F as it emerges and returns to the IHX. In the evaporators, 1,889-psia steam will be produced in a recirculation mode with an exit quality of 50%. The steam/water mixture will flow to the steam drum where the steam will be separated and dried before being sent to the superheater.

The basic SG module is shown in Figure 5.9. The SG module will be a tube-and-shell hockey-stick design with the sodium on the shell side and the water/steam on the inside of the tube. The flow type will be counter current with the hot sodium entering the top and emerging at the bottom. Water/steam will enter the bottom of the unit by a steam dome that will serve as a plenum for the tube sheet and pass up the inside of the tubes and emerge from the tube sheet into the upper steam dome. The modules will be welded into the IHTS, but will have bolted access ports into the steam domes for inspection or maintenance.

Internally, there will be 739 tubes of 2¼% Cr-1% Mo 0.109 in. wall thickness. The tubes will have an outside diameter of 0.628 in. The tubes will be butt welded to tube stubs that will have been machined from the tube sheet and therefore, will be an integral part of the tube sheet for the first 1.75 in. The welds will fully undergo nondestructive testing and be heat treated. The shell side of the modules are also will be 2¼% Cr-1% Mo. The details of the design and structural evaluation are discussed in Section 5.11.

Leak detection for small water-to-sodium leaks is discussed in detail in Section 7.5.2. This system is provided so that small leaks can be sensed and appropriate action taken before they develop into large leaks. Intermediate to

large leaks will be mitigated by the SWRPRS mentioned earlier and discussed in Sections 5.5.2.2 and 7.5.3.

The design-basis water-to-sodium accident has been selected to be an equivalent double-ended guillotine (EDEG) break of a steam generator tube followed in 1 sec by a second EDEG followed, in turn, by a third EDEG. This three-EDEG series occurs after the small leak which triggers the sequence has slowly pressurized the system to just below the burst pressure of the protective rupture disc in the IHTS expansion tank. This assumption makes the pressure pulse most severe. The pressure pulse that occurs from this design-basis leak (DBL) in the steam generator module is 395 psia in the module occurring at 0.420 sec and 333 psia at the IHX occurring at 0.412 sec. The DBL for the steam generator is intended to present a conservative basis to compute design loads and is not intended to represent a realistic sequence.

The steam generator system also will include the water dump subsystem and the sodium dump subsystem.

5.5.2.2 Sodium-Water Reaction Pressure Relief System

The sodium-water reaction pressure relief system (SWRPRS) is an engineered safety system which is designed to accommodate the sodium-water reaction which would result from the design-basis leak in the SGS. The SWRPRS system will consist of rupture discs, associated piping, and separation tanks for each of the three SGS loops (see Figures 5.10 and 5.11). Dual rupture disc assemblies will be installed on the intermediate sodium piping adjacent to the steam generator units and in the IHTS expansion tank to sodium tank gas equalization line. For large sodium-water reactions, excessive pressure in the intermediate sodium system will burst the rupture discs, dumping sodium and reaction products to the SWRPRS separation tanks. Gaseous reaction products will be directed to the flare stack where an igniter will be installed. The dual rupture disc configuration will be used to avoid complete dumping of the intermediate sodium system into the SWRPRS in case of a leak in the first disc.

Intermediate size steam or water leaks in the SG modules up to approximately 2 lb per second would be accommodated without failure of the SWRPRS main rupture discs. This will be accomplished by relieving IHTS system pressure at the expansion tank through the expansion tank/dump tank gas equalization line rupture disc to the sodium dump tank.

The SWRPRS reaction products separator tanks (two per loop) will be used to separate most of the liquid sodium and solid sodium-water reaction products from the gases, hydrogen, and steam, which will be ejecting the sodium from the failed unit. The tanks will be sized to hold the total volume of sodium in the three steam generator modules, the volume of sodium in the intermediate sodium pump and expansion tank, and the volume of sodium which would be expected to spill from the remaining secondary sodium system into the superheater unit due to the momentum of the fluid in the loop.

Gases, which will be primarily nitrogen used to inert the SWRPRS and the hydrogen from the sodium-water reaction, will be piped to a stack where the hydrogen will be ignited and burned. Burning the hydrogen will prevent the creation of an atmosphere which could mix with oxygen and form a potentially explosive region

in the system. A nonreturn valve will be located in the line between the tanks and the stack to reduce backflow of air into the system.

5.5.2.2.1 Evaluation

The SWRPRS piping and equipment internal to the steam generator building (SGB) will be seismic Category I. The SWRPRS piping and equipment external to the SGB are designed to meet ANSI B31.1. If a seismic event should damage this portion of the SWRPRS, so that a vent path to the atmosphere were not available, overpressure protection would be maintained by the internal SWRPRS volume. Therefore this meets the requirements of PDC 1, "Quality Standards and Records," and PDC 2, "Design Bases for Protection Against Natural Phenomena." In addition, PDC 5, "Environmental and Missile Design Bases," was applied to this system.

The SWRPRS will normally be filled with an inert gas to avoid the possibility of a hydrogen explosion in the system after activation of the system following a sodium-water reaction. Also, the inert gas will be maintained above atmospheric pressure to prevent inleakage of oxygen and verify that the inert atmosphere is being maintained in the SWRPRS.

The steam generator development program is providing the information for assessing the reliability of rupture discs. According to the applicants, tests have been conducted by the manufacturer to confirm that the as-built rupture disc design will function within specified burst pressure tolerances for the large-size discs required in CRBR. Additionally, tests are being conducted in conjunction with steam generator component tests. Multiple disc assemblies similar in design to the CRBRP are being installed in the large leak test rig (LLTR). Performance of the double reverse buckling disc will be measured during these tests.

Instrumentation and control of the SWRPRS and leak detection are covered in Sections 7 and 5.7, respectively, in this SER. Therefore PDC 11, "Instrumentation and Control," and PDC 4, "Protection Against Sodium and NaK Reactions," apply and were evaluated with respect to this system.

A large sodium-water reaction could inject slugs of sodium into the piping from the rupture discs to the separator tanks. These slugs can cause high reaction forces in the piping because of hydrogen gas pressure resulting from the sodium-water reaction. The applicants have stated in the PSAR that the results of the TRANSWRAP analyses will be used to optimize the piping sizes and system from the rupture disc assemblies to the separator.

5.5.2.3 Sodium Dump Subsystem

The steam generator system will provide one sodium dump subsystem for each of the three parallel independent IHTS loops. Each dump subsystem will consist of a sodium dump tank located at the lowest building level under the evaporator and superheater modules (see Figure 5.10). The capacity of each dump tank will be approximately 9,480 ft^3 at 930°F which will be large enough to store all of the sodium from the IHTS and auxiliary systems within each loop. This subsystem will serve as a sodium sump for sodium at operating temperatures, intermediate sodium storage, and removal or cleanup of sodium-water reaction products. The sodium dump subsystem will be used for normal drainage or for draining the sodium from

the affected IHTS loop following a sodium-water reaction occurring in an evaporator or superheater module. This system will be manually initiated from the control room and will drain the IHTS sodium and sodium-water reaction products through lines which will be connected at five different locations in the IHTS loop. These piping lines will provide gravity drains from the sodium expansion tank, the superheater, the evaporators, and the low points in the hot and cold legs of the IHTS main piping to the dump tank, as shown in Figure 5.12. Drain valves will be located in five piping runs mentioned above. Each piping run will contain a pair of drain valves connected in series. Controls and indications for these valves will be located on the main control board.

According to the applicants there is no automatic action associated with the removal of sodium from the loops (letter, Feb. 15, 1983, HQ:S:83:216). Additionally, the applicants have stated that suitable interlocks will be provided to minimize the possibility of erroneous dumps from the sodium dump and the water dump subsystems (letter, Oct. 21, 1982, HQ:S:82:113).

5.5.2.3.1 Evaluation

The sodium dump subsystem components are designed in accordance with ASME Code, Section III, "Nuclear Power Plant Components." The sodium dump tank is designed to satisfy the requirements of seismic Category I requirements. Therefore this subsystem meets the requirements of PDC 1, "Quality Standards and Records," and PDC 2, "Design Bases for Protection Against Natural Phenomena."

In the event that a sodium-water reaction occurs, sodium contaminated with sodium-water reaction products would be dumped into the dump tank and maintained in a molten state by trace heaters on the tank. The trace heat system would also maintain the sodium from an IHTS loop in the molten state in the event a loop were drained to the dump tank. This meets the requirements of PDC 7, "Sodium Heating System."

The sodium dump tank will be designed so that all normal inspection, maintenance, and repair can be performed during normal shutdown periods, or after the removal of sodium and/or sodium-water reaction products.

This subsystem will be provided with instrumentation for sodium level and temperature indication displayed in the control room. Additionaly, each piping run of the sodium dump system will contain a pair of isolation valves in series, with independent controls and contact-type liquid metal-gas leak detectors in each loop. A more detailed discussion is covered in Section 5.7 of the SER. This meets the requirements of PDC 4, "Protection Against Sodium and NaK Reactions," and PDC 11, "Instrumentation and Control."

The staff has reviewed the proposed engineering criteria, functional requirements, and general arrangement of this subsystem and concludes that the design meets the applicable requirements of the principal design criteria referenced above. The Standard Review Plan was not used in the evaluation since this system is only applicable to sodium systems and there is no analogy to LWRs. The staff also concludes that the proposed seismic and safety (quality group) classifications for the sodium dump system are consistent with anticipated service conditions, safety requirements, and the conclusions drawn in Section 3.2 of this SER. Therefore, the applicants' proposed preliminary design and associated design criteria can be used with reasonable assurance that the final

design will be acceptable from the safety standpoint, and that sufficient information has been presented to ensure that the applicant will meet all applicable principal design criteria in the design, construction, and operation of the sodium dump subsystem.

The staff review has included the applicants' proposed design criteria and design bases from the standpoint of compliance with those principal design criteria specifically applicable to the SWRPRS and falling within the scope covered in this section of the SER. On the basis of that review, the staff concludes that the design meets the applicable requirements of PDC 1, 2, 4, 5, and 11. This conclusion is based on the material presented in the PSAR and that the proposed seismic and safety (quality group) classification for the subsystem are consistent with anticipated service conditions, safety requirements, and conclusions drawn in Section 3.2 of this SER. Additionally, the staff concludes that (1) the applicants' proposed preliminary design and associated design criteria can be used with reasonable assurance that the final design will be acceptable from the safety standpoint, and (2) sufficient information has been presented to ensure that the applicants will meet all applicable principal design criteria in the design, construction, and operation of the CRBRP sodium-water reaction pressure relief system.

5.5.2.4 Water Dump Subsystem

One water dump subsystem will be provided for each of the three parallel and independent steam generation systems (SGSs). Each water dump subsystem will consist of water dump piping and a water dump tank. The water dump tank for each loop will be located in the steam generator building near the SGS recirculation pumps (see Figure 5.10). The dump tank will be provided in each loop to accept and store the water from the evaporator when rapid depressurization of the evaporator modules is required.

Blowdown of an evaporator module will be accelerated by the quick dump valves, which are part of the steam generator and water steam system. The quick dump valves will be located between the recirculation pump discharge and the inlet of each evaporator module. Additional blowdown capacity will be provided by the power relief/safety valves located at the outlet of each evaporator module. The water dump valves will exhaust to the water dump piping which will direct the water/steam mixture to the water dump tank, where the water will be temporarily stored and the flashed steam will be vented to the atmosphere (see Figure 5.13).

Pressure relief in the water dump tank will be achieved through an open pipe, which vents to the atmosphere to limit the maximum internal pressure and ensure the integrity of its components.

5.5.2.4.1 Evaluation

The water dump subsystem components are designed in accordance with ASME Code, Section III, "Nuclear Power Plant Components." The dump tank and its associated piping are designed to satisfy the requirements of Quality Group C according to RG 1.26.

The water dump tank is designed so that all normal inspection, maintenance, and repair functions can be performed during normal shutdown periods or after the removal of water following an evaporator blowdown.

Each water dump subsystem will have sufficient capacity to store the water dumped after isolation of both evaporators in one loop. The water dump subsystem will be designed to reduce the evaporator operating pressure to approximately 300 psig within 30 sec, working in conjunction with the power relief valves. Additionally, no single failure of the isolation and dump equipment shall cause the loss of shutdown heat removal capability. Also, no single failure should cause the water dump valves in the same dump path to open. The applicants will have to provide an analysis to show that in the event the system fails to operate as intended, the consequences of a sodium-water reaction will not cause the loss of shutdown heat removal capability. The analysis should also evaluate what changes, if any, would occur in the design-basis sodium-water reaction as a result of misoperation of the water dump subsystem during the event. This analysis can be completed during final design before the OL review.

The instrumentation and control of the water dump subsystem is discussed in Section 7 of this SER. The applicants have stated that interlocks will be provided to minimize the possibility of erroneous dumps from this system (letter, Oct. 21, 1982, HQ:S:82:113). Based on this and the conclusions in Section 7, this subsystem meets the requirements of PDC 11.

Because there is no parallel to LWRs, the Standard Review Plan was not used in the evaluation of the water dump subsystem. The seismic and safety classifications prepared and the conditions outlined are acceptable and consistent with the conclusions drawn in Section 3.2 of this SER.

With the exception of those matters identified above for resolution at the operating licensing stage of review the staff concludes that (1) the applicants' proposed preliminary design and associated design criteria can be used with reasonable assurance that the final design will be acceptable from the safety standpoint, and (2) sufficient information has been presented to ensure that the applicants will meet all the applicable principal design criteria in the design, construction, and operation of the CRBR water dump subsystem.

5.5.3 Evaluation

Because of the three-loop design, each of which will be capable of removing the full residual heat load, the steam generator system (SGS) will be is inherently protected against most single passive component failures. A failure in the turbine header region, which will be common to all three loops, will lead to closure of the main steam isolation valves (superheater outlets), thus restoring the independence of the loops. With nine heat exchangers in the SGS (6 evaporators, 3 superheaters), failure of one will be acceptable. The thermal impact on the core will be minimal as shown in Section 15.3. Redundant rupture discs in the SWRPRS subsystem will reduce the significance of a single rupture disc failing. The three-loop factor automatically will lead to a degree of redundancy and diversity. Further, there will be several safety/relief valves in the SGS providing a large margin of safety for valves failing to open.

The largest safety system in the SGS actually will be a combination of subsystems designed to respond to a potential sodium-water reaction in a heat exchanger. The consequences of a sodium-water reaction will be limited by the sodium water reaction pressure relief subsystem (SWRPRS). The objectives of the SWRPRS will be to:

(1) prevent the pressure generated by a sodium-water reaction within a steam generator module from reaching a value that would violate the integrity of the IHX primary to intermediate boundary

(2) control the solid, liquid, and gaseous products of a large sodium-water reaction so that the solids and liquids will be contained in appropriate vessels and the gas (hydrogen) will be released to the atmosphere in a safe manner

The SWRPRS is to be designed to accommodate the sodium-water reaction that would result from the design-basis leak for a steam generator module. On the basis of testing and some operational data from foreign reactors and the models used for their design basis and licensing, the applicants have proposed a design basis that would entail the following:

(1) a precursor leak that would allow the sodium pressure to reach 325 psig

(2) a primary failure of one tube of one equivalent double-ended guillotine (EDEG) break at time equal zero

(3) a secondary tube failure of one EDEG at time equal 1 sec

(4) a tertiary tube failure of one EDEG at time equal 2 sec

Water and/or steam flow will be injected into the sodium mass from both ends of the tubes, causing high pressures and acoustic waves in the intermediate sodium system. The pressures will burst the rupture discs to the SWRPRS, permitting the ejection of sodium, sodium-water reaction products, and steam and water into the SWRPRS.

The PSAR analysis has been reviewed for the design-basis sodium-water reaction, and it has been concluded that the basic response of the SWRPRS will have the capability to prevent design limits from being exceeded as well as to limit the radiological consequences to conform to 10 CFR 20 limits at the site boundary.

The staff concludes that the proposed design of the steam generation system will meet the requirements of PDC 2, 3, 4, 5, 7, 11, 26, 35, 36, and 37 as applied to this system. On the basis of the evaluation of this system and conformance with the above PDC, the staff concludes the steam generation system will be capable of meeting its design objectives and, therefore, will be acceptable for issuance of a construction permit.

5.6 Shutdown Heat Removal System

5.6.1 General Description

Plant shutdown may be planned or may occur because of some offnormal condition. Examples of the latter include anticipated transients and a range of unlikely events such as the loss of offsite power, station blackout, and simultaneous loss of all three PHTS loops. Residual reactor heat must be removed with high reliability in such circumstances to ensure that fuel damage does not occur. The details of which fuel limits must be maintained to prevent fuel damage are discussed in Section 4, and the description of the dynamics of some of the transients and unlikely events are provided in Section 15. The staff's evaluation of the design of systems to remove residual heat under various conditions follows.

The CRBR shutdown heat removal system (SHRS) will have three modes of operation:

(1) For normal shutdown operation, decay heat will be transferred from the core through the IHTS to the SGS and eventually to the main condenser and the cooling towers (i.e., the normal heat sink).

(2) If the PHTS and IHTS are operational but the normal heat sink is not, decay heat will be transferred from the core through the IHTS to the steam generator auxiliary heat removal system (SGAHRS), but heat removal from the SGAHRS will be through steam vent valves directly to the atmosphere in conjunction with the protected air-cooled condensers (PACCS). The PACC will use forced air cooling provided by a fan and discharge heat to the ultimate heat sink, the atmosphere. These systems are shown in Figure 5.14.

(3) If the IHTS is not available, decay heat from the core will be rejected through a direct heat removal service (DHRS), which will include a heat exchanger between the primary coolant sodium and a secondary sodium-potassium (NaK) loop with heat removal from the secondary loop achieved by air blast heat exchangers. This system is discussed in greater detail in Section 5.6.4.

The demand for the functioning of the SHRS is considerably high. More detailed description of the modes of operation and the associated subsystems is provided below.

Every time the reactor is scrammed, the SGAHRS portion of this system becomes operational. It is common for other LMFBRs, operating, under construction, or under design, to normally discharge the decay heat through the water/steam system, at least at the beginning of the transient. For CRBRP, the decay heat will be transferred from the core to the primary sodium through the IHX into the secondary sodium, through the steam generator into the water/steam system.

The purpose of the SHRS is to remove all sensible and decay heat from the reactor following rated power operation for all anticipated, unlikely, and extremely unlikely conditions. Each steam generator loop, regardless of recirculation pump operation (main feedwater or auxiliary feedwater), will be capable of removing all short-term and long-term decay heat from the IHTS,

provided either the main condenser/feedwater train or the SGAHRS is available to that loop.

The normal path for heat removal from the SGS following a normal plant shutdown will be provided by the main condenser/feedwater train. The pony motors will provide primary and intermediate sodium flow. Feedwater flow will be provided from the main feedwater system. Steam flow will be bypassed to the condenser through the turbine bypass to maintain pressure at the turbine admission valve at 1,450 psig to avoid lifting of safety or power relief valves and provide for the controlled dumping of steam. The main cooling towers will provide the ultimate heat sink.

This normal heat removal path will consist of nonsafety class equipment and its operation will not be required for safe shutdown of the plant. These systems are discussed in greater detail in Section 10.

Section 5.6.1 of the PSAR addresses the steam generator auxiliary heat removal system (SGAHRS). The SGAHRS will consist of two major subsystems: the auxiliary feedwater system (AFWS) and the protected air-cooled condensers (PACCs), which will provide short- and long-term heat sinks, respectively. Section 5.6.2 of the PSAR addresses the direct heat removal service (DHRS). The DHRS will be designed to provide a diverse heat removal path to the ultimate heat sink and represents a fourth path to reject shutdown decay heat. Initiation of DHRS operation is affected by operator action and is established only if the normal or SGAHRS heat removal systems are not operable.

5.6.2 Subsystem Description

5.6.2.1 Steam Generator Auxiliary Heat Removal System

The SGAHRS has been divided into four systems and are treated separately in order to facilitate the review of this heat removal path with respect to the SHRS. The systems within the SGAHRS are as follows:

(1) auxiliary feedwater system (AFWS)
(2) protected water storage tank (PWST)
(3) protected air-cooled condensers (PACC)
(4) SGAHRS vent valves

The SGAHRS is a safety-related system whose main function will be to provide redundant shutdown decay heat removal capability when the main nuclear steam supply system heat sink or main feedwater supply is unavailable. It also will function to remove shutdown heat loads for refueling and other long-term outages. The SGAHRS hydraulic profile is shown in Figure 5.15.

Feedwater flow will be maintained by the auxiliary feedwater subsystem (AFWS). This subsystem will draw water from a protected water storage tank (PWST) and pump it to the steam drum. Three pumps will be provided to supply the total auxiliary feedwater flow required for all three loops. Two pumps will be electric driven and powered by normal and emergency plant ac power. The third pump will be steam turbine driven. The steam-turbine-driven pump will use steam bled from the steam drums. The steam-turbine-driven pump can deliver 100% of the total auxiliary feedwater demand requirements. The electrical

driven pumps can each deliver 50% of the total auxiliary feedwater demand requirements.

In reviewing the design, pipe breaks to the AFWS were considered both upstream and downstream of the isolation valves during normal plant operation and after an auxiliary feedwater (AFW) initiating event. Continued heat removal capability will be ensured by the multiple loop feature of the SGAHRS and heat transport system. The PWST will be sized to retain sufficient water for at least 30 days of SGAHRS operation in addition to allowing for 10 min to isolate the AFWS from a large break. The PSAR states that before depletion of the PWST supply, an alternate water supply will be provided by a separate header connecting the feedwater pumps to the 250,000-gal condensate storage tank (CST). This alternate supply line, although only Class 3 piping, will provide the capability for the AFW pumps to take suction from the CST in the unlikely event a pipe break should occur between the pump suction isolation valve and the PWST, or PWST depletion does occur. However, the applicants have not taken credit for this subsystem in the event of a safe shutdown earthquake, since it is not seismic Category I and cannot be depended on in this event.

The applicants have committed to performing a reliability analysis (PSAR Appendix C, Section C.6.4, Amendment 71, Sept. 1982) for the AFWS, including all active components and systems needed for short-term (steam vents) and long-term (PACC) decay heat removal. The analysis must be in accordance with Item II.E.1.1 of NUREG-0737 using the methodology defined in NUREG-0611 and NUREG-0635 to determine the system reliability and major contributors to AFWS failure under various loss-of-main-feedwater transients. The applicants have also committed to performing a failure mode effect analysis (FMEA) for the systems involving shutdown heat removal and natural circulation events, including the PHTS and IHTS (PSAR, Amendment 71). The AFWS is discussed in greater detail in Section 5.6.2.2.

5.6.2.2 Auxiliary Feedwater System

The auxiliary feedwater system (AFWS) is a part of the steam generator auxiliary heat removal system (SGAHRS); it will remove decay heat from the steam drums when the normal path for heat removal through the main condenser/feedwater train is not available. It is a safety-related system designed to provide sufficent water from the protected water storage tank to the three loops of the steam generator system to remove all short-term decay heat by means of steam venting. The protected air-cooled condensers (PACCs) are a second part of the SGAHRS, which will operate to remove long-term decay heat below 45 MWt. The SGAHRS is described in Section 5.6.1 of the PSAR.

The AFWS design criteria and bases and preliminary design were reviewed in accordance with SRP Section 10.4.9 (NUREG-0800).

An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system design criteria and bases and

preliminary design will meet applicable portions of the following CRBR Principal Design Criteria (PDC).

(1) PDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to structures housing the system and the system itself being capable of withstanding the effects of earthquakes.

Acceptance is based on meeting the guidance of RG 1.29, "Seismic Design Classification," Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.

(2) PDC 5, "Environmental and Missile Design Bases," as it relates to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks. The basis for acceptance for meeting this criterion is the SRP Sections 3.5 and 3.6 series.

PDC 5 does not include loss-of-coolant accidents as a condition to be protected against, but includes anticipated operational occurrences and the effects of Na and NaK and their aerosols and combustion products.

(3) PDC 6, "Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

(4) PDC 17, "Control Room," as it relates to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown. Acceptance is based on meeting BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System," with regard to cold shutdown from the control room using only safety-related equipment.

PDC 17 does not include loss-of-coolant accidents as a condition to be protected against, and cold shutdown has been replaced with conditions applicable to CRBR.

(5) PDC 35, "Reactor Residual Heat Extraction System," and PDC 38, "Additional Cooling Systems," to ensure

(a) the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions

(b) redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.) BTP ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactions," as it relates to AFW pump drive and power supply diversity shall be used in meeting these criteria

(c) the capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained

In meeting these criteria, the recommendations of NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents Westinghouse-Designed Operating Plants," and NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," shall also be met. An acceptable AFWS should have an unreliability in the range of 10^{-4} to 10^{-5} per demand based on an analysis using methods and data presented in NUREG-0611 and NUREG-0635.

Although this reliability criterion was derived from an analysis of LWR accident sequences, the staff is using it to evaluate the CRBRP AFWS on the basis of its experience that it represents an acceptable range that can only be met by a well-designed (from a reliability standpoint) AFWS.

PDC 35 includes the availability of at least two flow paths for residual heat removal.

PDC 36 and 37 apply for inspection and testing. The residual heat extraction system is excluded from PDC 38, 39, and 40; however, the principals of redundancy, single failures, inspection and testability are covered in PDC 35, 36, and 37.

SRP Section 10.4.9 is considered applicable to the review of the CRBR auxiliary feedwater system with PDC 35, 36, and 37 replacing GDC 44, 45, and 46. Conformance with the acceptance criteria of SRP Section 10.4.9 provides the bases for concluding that the AFWS design criteria and basis and preliminary design are acceptable.

The safety-related AFWS will provide auxiliary feedwater (AFW) from a protected water source or from an alternate source to the steam drums in the event of loss of the main heat sink and/or the normal feedwater. The system will consist of two 50% capacity motor-driven pumps, one 100% capacity steam-driven pump, one protected water storage tank, and piping and valves. The two motor-driven AFW pumps will be powered from emergency ac buses, each supplied from a separate diesel generator. The turbine-driven pump will receive steam from any of the three steam drums. There will be two parallel AFW flow paths to each drum, one from the turbine-driven pump through a control valve and one from the motor-driven pumps (with headers) through another control valve. The level setpoints for the steam drum are such that normal flow will be supplied from the motor-driven pumps only with the turbine driven pump in a recirculating mode. If the drum levels should fall below the turbine pump setpoint, the control valve downstream of that pump would open and begin supplying flow.

The AFWS will be capable of removing plant short-term decay and sensible heat through the use of only one heat transfer loop.

The protected water storage tank (PWST) is designed to provide sufficient water so that the heat removal function can be accomplished by venting of steam until the heat load is reduced to a level that permits using the PACCs as the heat sink. It will include an additional amount of water to ensure that valve leakage and losses resulting from postulated pipe failures do not result in a premature depletion. Water from the PWST will be

piped through three separate lines to the suction side of the AFW pumps. The condensate storage tank will provide alternate water supply to AFW pumps. Water from the PWST cannot drain to the condensate tank because of the elevation differences.

The AFW will be not shared between any other units and, therefore, the requirements of PDC 6, "Sharing of Structures, Systems, and Components," are not applicable.

(1) The staff has reviewed the AFWS design criteria and bases and preliminary design to verify their acceptability with respect to its classification and operating characteristics.

(a) Minimum performance requirements for the AFWS have been identified and are found sufficient for the various functions of the system. The AFW pumps will be sized to deliver enough flow to the steam drums to prevent drum dryout under the conditions of a pipe break in one loop and venting from the superheater and the steam drum from the other two loops. To meet these requirements, the turbine-driven AFW pump will be sized to deliver 1,432 gpm. The two motor-driven pumps will each be sized to deliver one-half this flow rate. Assuming a single active component failure, the system will be capable of providing sufficient water for removing plant decay and sensible heat. Following a plant trip with SGAHRS initiation, the steam drum pressure will be controlled to 1,475 psig by the superheater vent control valve. If this normal means of venting is not available, the steam drum pressure will be controlled to 1,550 psig by the steam drum vent control valve. The AFW pumps will be designed to supply water at 1,700 psi discharge pressure to overcome friction and elevation losses while supplying AFW to the steam drum at 1,550 psig.

The PWST will be designed to hold 9,111 ft^3 of usable water. This volume is based on the most severe safety design-basis event that can destroy the normal heat sink, the alternate water supply from the condensate storage tank, and offsite power sources. It will include an allowance for a pipe rupture (on one loop) in the normally pressurized region of the AFWS. Allowance will be made for leakage from the various valves and pumps to ensure adequate feedwater makeup for a 30-day period. Operator action will be necessary to isolate the pipe rupture in 10 min. On the basis of the above, the staff concludes the AFW system meets the requirements of PDC 35.

(b) Adequate isolation of the AFW from other nonessential systems will be included in the system design. The entire SGAHRS, which includes the AFWS, will be designed to seismic Category I requirements. The steam generator building will provide the interface between seismic Category I and nonseismic Category I piping for the line from the condensate storage tank to the PWST. The protected water storage tank, AFW pumps, and AFW piping and valves will all be located in the steam generator building.

Therefore, the staff concludes that the AFWS will meet the requirements of PDC 2 and 38 and the guidelines of Position C.2 of RG 1.29.

(c) Provisions of AFWS testing and inspection will be included in the design. Each AFW pump will be equipped with a minimum flow recirculation line with a fixed orifice and isolating valves. An AFW pump test line will be provided which will return flow to the PWST during periodic testing. This line will branch off downstream of the tee where the motor-driven and turbine-driven pump supply lines join at the loop #1 valve station. This line will contain redundant automatic valves for isolating the AFW supply from the PWST should SGARS be initiated during testing.

The AFWS components will be located in areas that will be accessible during normal plant operation to permit periodic inservice inspection. Therefore, the staff concludes that the AFWS will meet the requirements of PDC 36 and 37 with respect to functional testing and inservice inspection. The applicants should be asked to address the NUREG-0611 requirements for functional testing and inspections in the FSAR.

(2) The staff has reviewed the AFWS design criteria and bases and preliminary design for protection against the effects of natural phenomena, pipe breaks or cracks in fluid systems outside containment, single system component failures, loss of an onsite motive power source, or loss of offsite power.

(a) All AFWS components will be located in seismic Category I structures. Protection against failure of nonseismic Category I system features of nonseismic Category I system features will be provided. Failure of nonseismic Category I system, components, or structures will not adversely affect the functioning of the AFWS because adequate isolation or separation will be provided. Water from the protected water storage tank cannot drain to the nonseismic condensate storage tank because of the elevation difference and layout of the piping. Therefore, the staff concludes that the essential portions of the AFWS will be adequately protected from earthquake and the requirements of PDC 2 and recommendations of NUREG-0611 are met.

(b) Protection against missiles, tornadoes, and floods will be provided. The AFWS will be located in the tornado-missile and flood-protected steam generator building and will be designed to seismic Category I requirements.

Motor- and turbine-driven AFW pumps will be located in separate compartments within the steam generator building, which will be provided with adequate drainage and will provide protection against internally generated missiles and flooding. (Refer to Sections 3.4.1, 3.5.1.1, and 3.5.1.2 of this SER for further discussion of flood protection and internal missile and tornado-missile protection). The staff, therefore, concludes that on

the basis of its review the AFWS will be protected from floods, tornadoes, and missiles and that it will meet the requirements of PDC 2 and 5 and the recommendations of NUREG-0611.

(c) The AFW trains will not be used during startup and shutdown; therefore, they will not be designed as high-energy lines (except piping from the AFW isolation valves to the steam drum and from the steam drum to the AFW pump turbine isolating valves, which will be designed as high-energy lines and will be evaluated for postulated ruptures). Protection against moderate-energy pipe cracks in the AFWS will be provided by separation and redundancy of equipment. The AFWS will be separated from the effects of high- and moderate-energy line breaks in other systems. These include the effects of pipe whip, jet impingement, and flooding. High-energy piping systems will be properly separated and protected so that the effects of a rupture in one loop will not affect operation of another loop in the AFWS. Therefore, the staff concludes that the AFWS will be protected against the effects of pipe whip, jet impingement, and flooding associated with pipe breaks and that it will meet the requirements of PDC 5 with respect to pipe breaks outside containment. Environmental qualification of AFWS components with respect to pipe breaks is discussed in Section 3.11 of this SER.

(d) The essential AFWS trains can function automatically as required in the event of a loss of offsite power. The heat transfer path from the steam drums under this condition will be to the atmosphere through the SGAHRS vent pressure control valves. This will be supplemented by steam condensation in the PACC until the PACC can carry the entire heat load. The turbine-driven pump will function independently of any ac power and thus will not be affected by a loss of offsite power. Power for the motor-driven pumps normally will be provided by two onsite emergency diesel generators. The power supply train for each pump will be physically separated from that of the other pump. Driving steam for the turbine-driven pump will be provided from any of the three steam drum main steamlines upstream of the main steam isolation valves and will be discharged to the atmosphere from the turbine. Each of these three steamlines will contain two locked-open, manual isolation valves, an electrically operated normally closed isolation valve, and a check valve to prevent flow reversal.

Downstream of the final isolation valves, the three lines will be headered together and and then pass through a normally closed electrohydraulically operated pressure control valve before connecting to the turbine-driven pump. Redundant control systems will be provided to ensure the opening of the control valve on a turbine-driven pump start signal. Any power failure will result in the isolation valves failing in the open position. The AFW supply lines from both the turbine and motor-driven pump

discharge headers will each contain a manual locked-open isolation valve, a normally open electrohydraulic control valve, a normally closed electrical isolation valve, and a manual locked-open isolation valve. After the final isolation valves, the two supply lines, one from the motor-driven pump and one from the turbine-driven pump, will be joined and the resulting supply line will contain two check valves and a manual isolation valve before joining the steam drum. The discharge from each AFW pump will have a minimum flow recirculation line with a failed-open isolation valve, a restricted orifice, and a manually locked-open isolation valve.

Therefore, the staff concludes that the requirements of PDC 35 and the recommendations of NUREG-0611 with respect to the ability of the AFWS to transfer decay heat from the reactor coolant system under a loss of offsite power will be satisfied.

(e) The AFWS will be designed to accommodate a single failure in any active system component without loss of function. The AFWS will consist of two redundant, 100% capacity trains. Both the trains will be capable of supplying any steam drum and will be powered from separate sources. Diverse power sources will be provided; electric and steam AFWS pumps will be provided with three suction supply connections from the protected water storage tank through separate, normally open, electrically operated valves. A redundant alternate water supply from the condensate water storage tank will be provided with motor-operated normally closed valves. Thus, adequate feedwater will be ensured to the three steam drums in the event of a postulated design-basis accident concurrent with a single failure of an active component. Adequate isolation will be provided for the AFWS (see Item (1)(b) above). Therefore, the staff concludes that the AFWS will meet the requirements of PDC 35 and the recommendations of NUREG-0611 with respect to single failure.

(f) Adequate AFW flow will be ensured to the steam drums in the event of the loss of offsite and all emergency onsite ac power by the safety-related turbine-driven pump subsystem, which can perform its safty function independent of ac power. Loss of all ac power will not adversely affect the position of motor-operated valves in the turbine-driven pump subsystem. All electrically operated isolation valves in the normal turbine-driven pump suction and discharge path to the steam drums will normally open or fail open on loss of ac power. Therefore, the staff concludes that the AFWS will meet the requirements of PDC 35, the guidelines of BTP ASB 10-1, and the recommendations of NUREG-0611 with regard to AFWS power diversity.

(g) The AFWS will be designed with adequate instrumentation to automatically initiate AFW flow to the steam drums upon receipt of an actuation signal. A low steam drum level, or high steam to feedwater flow ratio trip signal will start all AFW pumps, align the AFW sources, and open flow paths from the AFW pumps to the steam drums. The design will allow flow to any combination of

steam drums and minimize the effects of pipe failure, active component failures, power supply failure, and control system failure. Should a break occur in the high-energy AFW line to the steam drum, the flow limiting capability of the control valve would limit break flow to approximately 110% of rated flow to ensure supply to the other operating loops. Should the flow limiting function fail, the AFW isolation valve would automatically close when the AFW flow remains above 150% for 5 sec.

Isolation valves will also close automatically on detection of sodium-water reaction, high steam drum level, and a steam drum pressure less than 200 psig. The automatic closure will occur only in the affected loop. Valves that will close because of a high drum level will reopen automatically when the drum level falls to the low drum level trip point. The outlet steam isolation subsystem will provide isolation of a steamline break.

Further discussion of automatic AFWS initiation and flow indication including compliance with the recommendations of Item II.E.1.2 of NUREG-0737 is contained in Section 7.4.1 of this SER. Manual capability to initiate and control the AFWS pumps/valves and isolate the AFWS train will be provided in the control room and local control panels for SGAHRS. Therefore, the staff concludes that AFWS will provide adequate instrumentation and control for prompt initiation of shutdown using safety-related equipment in accordance with the requirements of PDC 17, the guidelines of BTP RSB 5-1, and the recommendations of NUREG-0611.

The applicants should address the guidelines of BTP ASB 10-2, "Design Guidelines for Water Hammers in Steam Generators With Top Feedring Design."

(h) The AFWS will be designed with sufficient flow capacity so that the system can remove the decay heat from the steam drums over the entire range of reactor operation. In Item (1)(a) above, the applicants' design-basis flow capacity is noted. It is based on providing enough flow to the steam drums to prevent dryout under the conditions of a pipe break in one loop and venting from the superheater and steam drum on the other two loops. The staff finds this flow capacity acceptable.

The PWST will be designed to hold 9,111 ft^3 of usable water. This volume will ensure a cooldown of the reactor coolant system to the point at which the PACCs alone can handle the decay heat loads.

Therefore, the staff concludes that the AFWS will meet the decay heat removal requirements of PDC 35 and the recommendations of NUREG-0611.

(3) The generic recommendations of NUREG-0611 as they relate to improvements in the AFW design have been discussed in the preceding paragraphs. The applicants have to address the additional short-term recommendation No. 2 regarding the AFW pump endurance test and other inservice inspection and functional testing of system components important to safety. This concern will be reviewed at the operating license review stage.

The applicants have addressed the conditions under which one of the AFWS pumps and its associated flow train can be taken out of service for maintenance in PSAR Section 16.3.4.2 (Technical Specifications for SGAHRS) but have not addressed the time limit for these outages. The applicants should commit to providing these outages according to the recommendations of NUREG-0611 at the operating license review stage.

(4) AFWS reliability analysis has not been performed in accordance with Item II.E.1.1 of NUREG-0737. No reliability analysis is required at the construction permit stage, but such an analysis will be required at the operating license stage.

On the basis of its review, the staff concludes that the auxiliary feedwater system will meet the requirements of PDC 2, 5, 6, 17, 35, 36, and 37 with respect to protection against natural phenomena and environmental effects, shared systems, operational capability from the control room and the remote shutdown panel, decay heat removal and cooling water capability, inservice inspection and functional testing capability; the guidelines of RG 1.29 and BTPs ASB-10-1 and RSB-5-1 concerning seismic classification, power diversity, and design of decay heat removal systems; and the recommendations of NUREG-0611 concerning generic improvements to the AFWS design.

5.6.2.3 Protected Water Storage Tank

The protected water storage tank (PWST) is a safety-related, seismic Category I system that will provide sufficient water (approximately 9,111 ft^3) so that the heat removal function can be accomplished by venting steam until the heat load is reduced to a level that will permit the use of the PACCs as the heat sink.

Sufficient water will be provided to accomplish venting, plus an additional amount to ensure that valve leakage and losses resulting from postulated pipe failures will not result in the premature depletion of the water supply. There will be three low-pressure, low-temperature lines from the PWST to the three AFW pump inlets. The PWST will be sized to hold sufficient water for at least 30 days of SGAHRS operation and to allow for 10 min to isolate the AFWS from a large break. PWST volume requirements are shown in Table 5.4.

5.6.2.4 Protected Air-Cooled Condensers

The long-term heat removal subsystem, protected air-cooled condensers (PACCs), also will function while steam is vented for short-term decay heat rejection and will be located at a higher elevation than the steam drum. The PACC will reject heat to the atmosphere. Saturated steam will be supplied to the PACC

from the steam drum, and saturated water will be returned by gravity flow. The unit will utilize a fan to force air across the condenser tubes for peak performance. The PACC system will provide the long-term heat removal capability to remove reactor decay heat when the SGAHRS is activated following a reactor shutdown. The capacity of the PACC on a single loop will be sufficient to assume the total decay heat load before the PWST water is depleted.

The PACC is a tube-type steam condenser constructed of carbon steel. Heat will be rejected to the atmosphere by condensing the saturated steam from the steam drums by forced circulation of air over the tube bundles. Each unit will be sized to reject 15 MWt under conditions of forced convection on the air side and natural circulation flow on the steam/water side. Each PACC will have two half-size tube bundles, two variable blade pitch fans, and two sets of variable position louvers to control air flow and, therefore, heat rejection. The electrical power supplies and instrument and control (I&C) circuits for the PACCs will be Class 1E. Refer to Sections 7 and 8 for information on I&C and power sources, respectively. The design parameters for the PACCs are given in Table 5.5.

5.6.2.5 Steam Generator Auxiliary Heat Removal System Vent Valves

There will be two lines, one branching from the steam drum to superheater piping and the other branching from the superheater to the main turbine line, which will contain a locked-open, manual isolation valve and a normally closed electrohydraulic operated pressure control valve. Both lines will be used to vent steam from the system to release heat from the plant and maintain the steam drum at a pressure below the design head of the auxiliary feedwater pumps. The superheater vent valve and vent line will be made of 1¼% Cr-½% Mo steel; the steam drum vent valve and vent line will be carbon steel. Following the plant trip and the initial pressure-reducing transient, these valves will normally be used as the only safety-related means for venting steam during SGAHRS operation. Safety relief valves located at the superheater outlet will serve as a backup should both the SGAHRS superheater and steam drum power vent valves be unavailable. These steam generator system valves will be set to open at a higher pressure. The advantage of separate SGAHRS vent valves is a controlled steam drum pressure by venting through valves designed for low erosion rather than the on/off operation of the safety valves. The superheater vent valve and the steam drum vent valves in each of the three HTS loops are powered from separate battery supplies, thus ensuring a vent capability should a battery system fail.

5.6.3 Natural Circulation

The normal mode of plant operation will be forced circulation with heat transfer from the reactor through the primary heat transport System (PHTS), intermediate heat transport system (IHTS), and steam generator system (SGS) to the balance of plant (BOP). Following a loss-of-offsite power (LOOP) event, forced sodium circulation in the PHTS and IHTS will be maintained by the main sodium pumps driven by pony motors, powered from onsite diesels, and final heat rejection will be through the steam generator auxiliary heat removal system (SGAHRS). In the extremely unlikely event of a loss of offsite and onsite ac power, the system can reject decay heat via natural circulation through the SGAHRS using the vent valves and the PACC with makeup water to the steam drum from the AFWS. The elevational arrangement of the intermediate heat exchanger (IHX) with

respect to the reactor core, the steam generator evaporators with respect to the IHX, and the steam drum and recirculation piping with respect to the evaporators will provide the natural thermal driving heads in the PHTS, IHTS, and SGS to ensure adequate flows for decay heat removal. The hydraulic profile for the CRBRP, shown in Figure 5.16, illustrates these elevational differences. Feedwater will be supplied by the auxiliary turbine-driven pump to the steam drums and final heat rejection will be achieved by venting steam to the atmosphere through the SGHARS vent valves on the steam drum and superheater outlet and through the auxiliary feed pump turbine. Figure 5.17 shows a schematic of the natural circulation decay heat removal train. The normal postshutdown decay heat removal train has been shown with the inoperative portions in phantom.

Natural circulation capability is an important inherent safety feature of CRBR. In case of a loss of all ac electrical power, the reliance will be on natural circulation to transport the heat from the reactor to an ultimate heat sink. The phenomenon of interest in natural circulation analysis is prediction of the buoyancy driving force available to cause the sodium to circulate and the sodium flows in a flow regime different from full-flow conditions. In addition, the buoyancy forces and flows are dependent on each other. Currently, there is no unified analytical tool, that is, single computer code, to predict loop flows, the flow pattern in the vessel, and detailed temperature profiles in the core. The approach being proposed for the CRBRP by the applicants is a coupling of three computer codes (COBRA-WC, DEMO-IV, FORE-2M) to predict natural circulation and core thermal performance. This is covered in more detail in Section 4.4.

The criteria proposed by the applicants for acceptable natural circulation are flows that will limit the maximum core hot-spot temperatures (fuel and blanket assemblies) to less than the sodium saturation temperature, which will be approximately 1,720°F at the top of the active core.

The staff has completed independent calculations to assess the natural circulation capability of CRBRP (discussed in greater detail in Section 4.4). These calculations have shown good agreement with the studies done by the applicants (Severson, Feb. 1982). The acceptability of the plant's response to the transition to and operation in the natural circulation decay heat removal mode will be verified as part of the operational testing during CRBR startup (see Section 4.4).

5.6.3.1 Evaluation

The review of the natural circulation capability of CRBRP incorporated such areas as the design of components, design criteria and how they were applied to the design, detailed evaluation done in support of the design, and development testing and natural circulation verification tests. Specifically, the review incorporated the SHRS design, design bases, and the analyses that satisfy the intent and form a portion of the body of information that satisfies the following CRBR Principal Design Criteria (PDC):

(1) PDC 11, "Instrumentation and Control," as it relates to the monitoring of system variables such as pressure, flow, temperature, power source, and system leakage under various operational conditions.

(2) PDC 15, "Electrical Power Systems," as it relates to the onsite and offsite electrical power systems provided to permit the functioning of systems and components important to safety. The onsite power system was reviewed to verify that independence and redundancy, assuming a single failure, were taken into account. A detailed review of electrical power systems is discussed in Section 8.

(3) PDC 18, "Protection System Functions," as it relates to the ability of the subsystems to initiate automatically, and to sense accident conditions and initiate the systems and components important to safety.

(4) PDC 35, "Reactor Residual Heat Extraction System," as it relates to the ability of the systems to remove heat from the reactor core to ultimate heat sinks under shutdown conditions, including accident conditions. Redundancy, independence, and diversity also were considered, as well as the loss of either onsite or offsite power, including station blackout.

In addition to the above PDC, the system was reviewed to verify that provisions for the following will be applied to the design:

(1) PDC 5, "Environmental and Missile Design Bases"
(2) PDC 6, "Sharing of Structures, Systems, and Components"
(3) PDC 36, "Inspection of Reactor Residual Heat Extraction System"
(4) PDC 37, "Testing of Reactor Residual Heat Extraction System"

These criteria are applied in the direct review in other sections and were evaluated in this section only to provide guidance for the review of the design.

In conjunction with the PDC, the Standard Review Plan was used to review this system. Design details contained in SRP Section 5 for the shutdown heat removal system (SHRS) which were applied to materials and structural integrity are found in Sections 3.9 and 5.11. SRP Section 10.4.9 was used to evaluate the AFWS. Sections 5.4.6, 5.4.7, and 6.3 of the SRP were used directly in the review of this system insofar as they apply to a reactor of this type. Although the sections of the SRP used in the review were not written specifically for a sodium-cooled reactor, the intent of the SRP was used as guidance in the evaluation as described below:

(1) SRP Section 5.4.6, "Reactor Core Isolation Cooling System," as it relates to a safety system (AFWS) which serves as a standby source of cooling water to provide decay heat removal capability whenever the main feedwater system is isolated. The piping and instrumentation diagrams (P&IDs) were reviewed to verify that the system will be capable of performing its intended function and will be both preoperationally and operationally tested.

(2) SRP Section 5.4.7, "Residual Heat Removal (RHR) System," as it relates to its use in conjunction with the main steam and feedwater systems, or auxiliary feedwater system in conjunction with the SGAHRS vent valves and the PACC subsystem to cool down the reactor coolant system following shutdown.

(3) SRP Section 6.3, "Emergency Core Cooling System," as it relates to the ability of the SHRS to withstand a single failure and have alternate sources of electric power.

The CRBRP has been designed to remove decay heat from the reactor core through the heat transport systems to the steam generator system via natural circulation of the sodium systems and within the steam generator system (the steam drum/evaporator recirculation loop). A preliminary assessment (Lowrie and Severson, March 1976) of CRBRP capability to provide adequate flow for decay heat removal under natural circulation conditions was made in 1976 by the applicants. Core design changes coupled with improved methods for calculating natural circulation have resulted in modifications to the initial assessment. The current assessment of the CRBRP capability to remove decay heat by natural circulation accounts for (1) the core design change from homogeneous to the heterogeneous core, (2) plant system and model changes made since the first assessment, and (3) updated input data that include decay power, core pressure drop, and test data from the CRBR prototype pump water tests.

The initial conditions used for the analysis of the natural circulation event are basically the thermal-hydraulic design values (THDVs), except that the cold- and hot-leg temperatures were set approximately 20°F higher to account for instrument error and possible operational uncertainties. THVD conditions are considerably higher than plant expected conditions and are discussed in greater detail in Section 4.4.

The reactor is assumed to be operating at full power (975 MWt) when it is postulated that there is a loss of all offsite ac power and that the emergency ac power system fails to operate. Emergency dc power is assumed to be available. This results in loss of power to the main feedwater pumps, steam generator system recirculation pumps, and the sodium coolant pumps (including power to the pony motors on both the primary and intermediate pumps). A reactor scram would occur approximately 0.6 sec after the loss of power.

The main coolant pumps would coast down and stop. Flow in all sodium loops would then be maintained exclusively by the thermal driving heads. Recirculation flow in the steam generator system would likewise be maintained by the natural circulation thermal driving head in the recirculation (evaporator/drum) loop. The steam drums in all three loops would be furnished with auxiliary feedwater from the turbine-driven auxiliary feedwater pump.

The final heat sink would be supplied by steam venting through the steam generator auxiliary heat removal system (SGAHRS) vent valves, and natural circulation flow thorough the protected air-cooled condensers (PACCs) controlled on steam steam drum pressure. The SGAHRS vent valves would remove the majority of the decay heat early in the event until the PACCs could handle the heat rejection load and control the steam drum pressure without the assistance of the vent valves.

The transient conditions for the reactor and heat transport systems during the first 30 sec would be very much like those of an ordinary plant trip. Following that time, the flows would be substantially lower than those produced by the pony motor-driven pumps, which are normally operating following plant trips.

The critical period for the natural circulation decay heat removal mode is the first 2-5 min after scram when the flows are low (because the pumps have stopped) and the decay powers are still relatively high.

The results of the analysis (Severson et al., Feb. 1982) showed that the time interval of interest for the event is the first few minutes, based on the criterion of no boiling in the core for acceptable natural circulation decay heat removal. Thus, the results of the analysis, presented by the applicants, covered the first 8 min, the critical period of the event. However, the applicants evaluated this event for an extended time period (2 hours) at the staff's request. The long-term natural circulation analysis was performed to address the transition from steam venting as the primary (short-term) heat sink to the PACCs as the ultimate (long-term) heat sink that is required for plant cooling. It was evident from the results that the inherent behavior of the present control system would ensure plant safety for well over 2 hours. Three redundant heat transport system loops should further ensure plant safety even in the event of a failure in any one loop.

Although station blackout is not a design-basis event for CRBRP, it is evaluated here to illustrate its features to prevent station blackout and evaluate the plant's ability to accommodate such an event. This unique reactor plant will have several redundant and diverse power sources to preclude station blackout. Specifically, there will be four connections to the TVA grid, three transformers with diversity, and a flexible bus connection for offsite power availability. For onsite power, the plant will have three seismic Category I diesel generators (one of different size and manufacture) and three groups of safety-related loads. Each load group will be adequate to achieve and maintain safe shutdown. A detailed review of the power systems is presented in Section 8.

A station blackout event can be accommodated by CRBRP using natural circulation heat transport system flows and the SGAHRS. The reactor shutdown system (RSS) will automatically shut down the reactor on loss of offsite power. PHTS, IHTS, and SGS natural circulation is implied by the design and no operator action is required. The SGAHRS power vent valves will automatically provide the short-term heat sinks necessary. Additionally, feedwater will be provided by the automatic operation of the turbine-driven auxiliary feedwater PWST pump, which will draw from the protected water storage tank (PWST). In conjunction with this, the PACC system will be automatically initiated and will provide the long-term heat sinks using water and air-side natural circulation. The turbine pressure control valve, the control governor, and the flow control valve of the turbine-driven auxiliary feedwater pump will be controlled from the Class 1E station battery system. The electrical power supplies and instrumentation and control circuits for the PACC system also will be Class 1E.

In evaluating this event, the staff has concluded that the offsite and onsite power supplies will ensure that station blackout will be an event with very low likelihood of occurring. However, in the event of a station blackout, the heat transport systems would naturally circulate and transport adequate heat from the reactor core to the SGAHRS; and the SGAHRS, in turn, would automatically provide multiple short- and long-term heat sinks using batteries for the necessary control power.

The arrangement of the components, valves, and piping of the SGAHRS in the auxiliary bay of the steam generator building (SGB) incorporates full safety considerations for environmental and operational accidents including flooding, missiles, jet impingment, and fire. The redundancy of loops in the SGAHRS will be maintained by routing the loops through separate cells in the SGB to protect adjacent loops from any accident in an adjoining loop. Auxiliary feedwater supply for the SGAHRS will be common to all three heat transport loops; therefore, complete separation cannot be maintained as in the case of the intermediate heat transport system (IHTS) and steam generator systems (SGSs). Because of this, special consideration will be given to flooding, missiles, jet impingement, and fires within the SGAHRS cells.

The SGAHRS is a safety-related system designed to provide the final heat sink for all postulated loss-of-feedwater or loss-of-normal-heat-sink events. As described earlier the SGAHRS will consist of two subsystems: the auxiliary feedwater system and the PACC system, which provide short- and long-term heat sinks, respectively. In the short term, decay heat will be removed by the main condenser whenever it is available. Long-term heat removal through the PACC system may be initiated at any time; however, because of the limited heat removal capability of the PACC system (45 MWt total for all three units with fans on), either the main turbine generator condenser or the short-term steam venting and auxiliary feedwater subsystem must function until the steam generator heat load falls below the capability of the PACC system. Under normal three-loop shutdown conditions, without using the main condenser, steam venting is expected to terminate within 1 hour after plant trip.

Whenever the normal heat removal path is not available in the short term, activation of the SGAHRS will occur automatically with both the auxiliary feedwater and PACC subsystems brought into service. The two subsystems will continue to function concurrently until the heat load is reduced to such a level that the PACC will remove the entire heat load. Operator action will be required only to shut off the AFWS pumps once the venting and feedwater supply requirements have ended. This is not critical, because failure to do so would only result in gradual heating of the remaining PWST water as it is recirculated from the AFW pump back to the PWST. The increase in temperature in the PWST water would be due to pump heat as it is recirculated. The amount of heat gained by the recirculation of auxiliary feedwater would be minimized by the bulk of the remaining PWST water. This is, therefore, not considered to be a problem and it is expected that the operator will have sufficient time to notice that the AFW pumps are running and shut them off.

The PACC system will be designed to operate on natural circulation in all cooling modes, that is, louvers closed, louvers open, and fan on/off. The rate of natural circulation will depend on how rapidly the steam is condensed. Three PACC units will be provided, one for each loop, each capable of removing the total decay heat (15 MWt) approximately 1 hour after shutdown. The design of these units has taken into account single active failures, seismic Category I requirements, and Class 1E power supplies and instrumentation and control. No auxiliary cooling is required for equipment to remove decay heat via forced circulation or natural circulation.

The applicants have committed to conduct full-scale PACC tests before the FSAR is submitted with low-pressure (165-250 psia) steam providing the means for

simulating PACC operation at CRBRP refueling conditions. This condition will be included as part of the PACC duty cycle, which will occur during plant startup from and plant shutdown to the refueling temperature of 400°F. The full-scale PACC tests that will be performed on the lead unit will address the thermal performance and stability of the design. The applicants also have committed to conduct a whole-plant test program of CRBR natural circulation and DHRS performance during plant startup testing. This program also will include the PACC units with the objective of confirming the thermal-hydraulic computer codes that have been used to predict the plant natural circulation behavior as well as the plant response under DHRS flow conditions. This will ensure that the predictions of the effects on temperature and flow of the variations in the available heat sinks are adequate.

The staff has evaluated Item No. II.F.2 of Appendix H of the PSAR, "NUREG-0718 Information Requirements for TMI-2 Action Plan Items Applicable to CRBRP." The design includes instrumentation which will provide for indication of inadequate core cooling in the control room. This instrumentation will include reactor coolant level, reactor core outlet temperatures, and intermediate heat exchanger outlet temperatures. The applicants have committed to describing the program for the development and implementation of procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling. These procedures can be submitted for review during the operating license stage of review. Additionally, natural circulation verification testing during the startup phase will provide adequate means to evaluate the plant's ability for indication of inadequate core cooling and will verify the procedures to be used.

The staff concludes that the proposed design of the SHRS will meet the requirements of PDC 2, 4, 5, 7, 11, 15, 18, 35, 36, and 37 as applied to this system. On the basis of the review conducted for this system and conformance with the above PDC the staff concludes the SHRS will be capable of meeting its design objectives and, therefore, the design will be acceptable for issuance of a construction permit.

5.6.4 Direct Heat Removal Service

5.6.4.1 General Description

There are a number of factors which will provide verification that SGAHRS will remove residual heat with a high level of reliability. Among these are the planned component and initial startup testing and the Reliability Program, discussed in Appendix C of the PSAR. Therefore, the staff believes that the steam and feed trains backed by SGAHRS for offnormal events will satisfy the requirement to reliably remove residual heat. However, to provide margin and diversity in the decay heat removal capability it was considered prudent to provide a supplementary means of removing long-term decay heat. The direct heat removal service (DHRS) was added to enchance the overall shutdown heat removal reliability by providing a fourth, diverse, heat removal path and heat sink.

According to the applicants the DHRS safety function is to transfer fission product decay heat and other residual heat from the reactor core at such a rate that fuel element and reactor coolant boundaries are maintained. The DHRS will function to remove decay heat following a reactor shutdown in which all IHTS/SGS heat sinks are lost.

For this event DHRS will be manually initiated remotely within the first ½ hour after shutdown. The PHTS pony motors will provide core flow. For initiation after 24 hours, DHRS will have sufficient capacity to limit the maximum overflow sodium temperature below 900°F. DHRS will be called into service only after two main heat transport loops have been removed from decay heat service for any reason and will carry the total heat load only if three main loops have been removed from decay heat service for any reason.

5.6.4.2 Subsystems Description

A schematic diagram for the DHRS is shown in Figure 5.18. DHRS will be provided by the auxiliary liquid metal system overflow and makeup circuit acting in conjunction with the spent fuel cooling and cleaning system. Hot primary sodium will overflow from the reactor vessel to the overflow vessel. The primary sodium electromagnetic (EM) makeup pumps will pump the hot sodium through the overflow heat exchanger and back to the reactor vessel. Operation of the primary pump pony motors will provide sodium flow through the core. The ex-vessel storage tank (EVST) NaK systems will remove the primary sodium in the overflow heat exchanger. The heated EVST NaK will be pumped by the NaK EM pumps to the EVST NaK air blast heat exchangers where heat will be transferred to the atmosphere (see Section 9.3).

The DHRS will be designed to increase the overall post shutdown heat removal reliability by providing long-term decay heat removal capability in addition to that provided in the three redundant heat transport system loops and SGAHRS.

The DHRS will be consist of one heat removal path capable of removing approximately 11 MWt from the primary coolant. That capability will be adequate to prevent loss of coolable geometry assuming the SGS heat sink is lost immediately upon shutdown from rated power and active operation of DHRS is initiated ½ hour after shutdown.

DHRS will be manually activated and remotely controlled from the control room. Instrumentation required to monitor the condition of the DHRS will consist of thermocouples on the EVST sodium outlet lines (three loops) and level indicators in the EVST and the reactor vessel (RV). These instruments will confirm that the sodium levels in the RV remain above the loop outlet nozzles and that temperatures remain below design limits. Other DHRS diagnostic instrumentation will not be essential for DHRS operation because the pumps and air blast heat exchanger will be operated at maximum design rates. When the reactor decay heat load has dropped sufficiently, the cooling capacity of the system may manually be reduced by lowering flow rates or fan speed, or by shutting down one of the NaK cooling trains.

The equipment from the overflow and makeup circuit and the spent fuel cooling and cleanup system which is cross connected to provide the direct heat removal service is schematically shown in Figure 5.18. The EVST air blast heat exchanger fans and louvers, pumps, and certain valves are classified as active components in DHRS. The air blast fans and louvers normally will be operated (for EVST cooling) at partial speed and partially open. During changeover to DHRS duty, the fans and louvers will be set manually at full speed and full-open position. The operating modes of DHRS valves are given in Table 5.6.

All active valves will be manually operable remotely from the control room. The operating modes of DHRS pumps are given in Table 5.7.

The active DHRS pumps will be electromagnetic pumps similar in design to that used for the fast flux test facility (FFTF). Some changes have been incorporated into the CRBRP design as a result of the tests of the FFTF pumps. Water flow tests have been conducted on the revised throat section of the CRBRP pump. These test results have been factored into the design of the pumps. A prototype pump will be built and sodium tested with the results being factored into the final design of the CRBRP plant units. The first plant unit also will be sodium tested with its associated control consoles to verify performance and capacity.

5.6.5 Evaluation

The DHRS will not be designed to provide heat removal by natural circulation. Electrical power must be available to supply the primary sodium makeup and EVST NaK pumps in addition to at least two PHTS pony motor when the DHRS is removing decay heat. DHRS will have no natural circulation heat removal capability since the overflow concept requires pumping, and will not be designed to operate when the PHTS loops are naturally circulating.

The direct heat removal service will provide a separate, diverse, fourth heat removal path that can adequately cool the reactor core after a shutdown. The DHRS equipment will have adequate heat removal capacity to maintain the maximum average PHTS coolant temperatures at or below 1,140°F for reactor trip from three-loop thermal-hydraulic design power level. Complete loss of heat transfer through the IHXs is assumed at the time of the reactor trip. DHRS will be manually initiated remotely from the control room and will be fully operational 30 min after the reactor trip. Manual operation will be accomplished by activating six switches on the DHRS startup panel. A sequencer will then automatically take over and ramp the sodium makeup pumps up to speed, bring the NaK loops into operation, and open and close the appropriate valves. This operation will take approximately 12 min to complete its cycle. It is assumed that all PHTS loops will be intact with sodium circulated by two or three pony motors.

Plant equipment will be designed to accommodate the event according to criteria that ensure that (1) the coolant and DHRS boundaries remain intact, (2) the primary pumps retain their ability to function at pony motor speed, and (3) active DHRS components retain their ability to perform as intended. The requirement for the reactor is that coolable core geometry be maintained.

The applicants completed a series of sensitivity studies to evaluate the ability of the DHRS to accommodate the failure of a single active component and not result in complete loss of DHRS heat removal capability (i.e., adequate heat removal capability to maintain coolable core geometry based on analyses where conservatisms are selectively applied). These sensitivity studies were initiated at the staff's request to address the single-failure capability, single-failure analysis, and the details of the base case F-2 analysis. The F-2 transient is described as "DHRS activation without SGS cooldown."

This event will occur after the postulated loss of all heat transfer through the IHXs at the time of reactor trip from three-loop rated power operation. One-half hour after scram, with the sodium in the reactor and heat transport system at approximately 1,050°F, the DHRS will be fully operational and will assume the decay heat load. Primary flow from three main coolant pumps at pony motor speed and maximum decay power is assumed. Primary system temperatures are based on the heat capacity of the reactor, overflow tank as appropriate for the overflow and return flow rates, and all three primary loops. The most limiting criterion for accommodation of this event is that the coolant and DHRS boundaries remain intact and that the primary pumps retain the ability to function at pony motor speed. Active DHRS components must also retain their capability to perform their intended DHRS functions. With this criterion met, Na boiling in the core will not occur.

Table 5.8 reflects the operating cases and sensitivity analyses completed by the applicants. The updated design case that is described above considers revised decay heat values that reflect current conservative design values. The results indicate that, with adiabatic IHX at time zero, the peak hot-leg temperature is 1,052°F, as shown in Figure 5.19. The peak temperatures for the single-failure assessment slightly exceed (by 7°F) the structural peak temperature analyzed for the F-2 transient. This is labeled "Single-failure evaluation #2" in Table 5.8 in which the following parameters are considered:

(1) two PHTS loops operating
(2) no IHTS heat capacity
(3) no SGS heat capacity
(4) one sodium makeup pump at 600 gpm
(5) one NaK pump at 600 gpm
(6) one air blast heat exchanger
(7) conservative decay heat values used
(8) no EVST heat load
(9) heat losses through insulation included

Although the results presented by the applicants for the single-failure analysis (see Figure 5.20) exceeded the structural peak temperature analyzed for the F-2 transient, the detailed transient curves for the preliminary assessment peaked at 1,130°F and were included in the equipment specifications. The detailed structural analysis is not planned to be redone, since according to the applicants, scoping analysis indicates that the capability of the primary boundary to maintain structural integrity, based on creep rupture considerations of Code Case 1592, at temperatures of approximately 1,200°F is approximately 100 hours. It is judged that there is considerable margin and conservatism within this system that the peak hot-leg temperature of 1,137°F would not impact the overall objective of the DHRS. Under these conditions, the DHRS design will provide sufficient cooling to ensure primary coolant boundary integrity and prevent loss of in-place coolable geometry of the core.

The consequences of protected loss-of-heat-sink (LOHS) events in the CRBRP have been studied for a number of accident conditions with and without activation of the DHRS. Confirmatory analyses have been conducted by Brookhaven National Laboratories (BNL) and Argonne National Laboratory (ANL) for the DHRS as an independent evaluation of the applicants' data. Parametric studies were done

to determine the sensitivity of reactor temperatures to DHRS performance using the SSC computer code developed at BNL and COMMIX code at ANL (see Section 4.4).

The calculated results were in excellent agreement with the applicants' predictions for the design-base case. The maximum hot-leg temperature was calculated to reach approximately 1,054°F using the SSC code as compared with 1,052°F using the DEMO code.

It can, therefore, be concluded that (despite calculational uncertainties and somewhat hypothetical assumptions on the accident conditions), the DHRS should perform as expected and that if the procedures following a LOHS are executed as designed, the impact of this accident will be minimal and the DHRS will perform its intended function.

As part of the staff's OL review, farther confirmatory analyses will be conducted at BNL to assess how the DHRS functions under the conditions of the F-2 transient with a single failure. In addition, calculations will be performed at Argonne National Laboratory (ANL) using the COMMIX computer code to evaluate the DHRS performance with one and two pony motors in operation in addition to the design-base case. This is expected to yield a balanced evaluation of DHRS performance under varying conditions.

The DHRS was reviewed for independence of electrical power sources. Each train of the DHRS and its supporting auxiliary cooling systems will be powered from separate Class 1E diesels. Initial findings indicated that both trains of DHRS will receive emergency ac power from the same source as the two motor-driven auxiliary feedwater pumps.

Considering the complexity of five heat removal paths and three emergency ac power sources, the staff has asked the applicants to perform a study to evaluate the load arrangement for heat removal capability. The study should not only consider a permanent splitting of the trains of DHRS to diesels 1 and 3, but also should evaluate the possibility for manual transfer of part of the DHRS emergency ac power from one diesel to another. In this way at least one train of DHRS would be available to remove decay heat in the event both diesel 1 and 2 should not be available. The ability to transfer part of the load has the potential to enhance the capabilities of the DHRS to remove decay heat. The staff will evaluate this approach at the operating license stage of review.

The components of the DHRS shall be designed, fabricated, erected, constructed, tested, and inspected to the standards of Section III of the ASME Code, 1974 Edition through the Summer 1974 Addenda, Class 1 or 2.

Additionally, the review incorporated the DHRS design, design bases, and the analyses that satisfy the intent and form a portion of the body of information that satisfies the following CRBR Principal Design Criteria (PDC):

(1) PDC 11, "Instrumentation and Control," as it relates to the monitoring of system variables such as pressure, temperature, flow, power source, and system leakage under various operational conditions.

(2) PDC 15, "Electrical Power Systems," as it relates to the onsite and offsite electrical power systems provided which are necessary for the

functioning of systems and components important to safety. The onsit power system was reviewed to verify that independence and redundancy, assuming a single failure, were taken into account. A detailed review of electrical power systems is discussed in Section 8.

(3) PDC 35, "Reactor Residual Heat Extraction System," as it relates to the ability of the systems involved to remove heat from the reactor core to ultimate heat sinks under shutdown and accident conditions. Redundancy, independence, and diversity were also considered in the review, as well as the loss of either onsite or offsite power. Station blackout does not apply to this system as it cannot operate without ac power. In addition, this system does not operate in the natural circulation mode; therefore, no credit was taken for this.

In addition to the above PDC, the systems were reviewed to verify that provisions for the following were applied to the design:

(1) PDC 5, "Environmental and Missile Design Bases"
(2) PDC 6, "Sharing of Structures, Systems, and Components"
(3) PDC 36, "Inspection of Reactor Residual Heat Extraction Systems"
(4) PDC 37, "Testing of Reactor Residual Heat Extraction Systems"
(5) PDC 38, "Additional Cooling Systems"

These criteria are applied in the direct review in other sections and were evaluated in this section only to provide guidance for the review of the design.

In conjunction with the PDC, the sections of Chapter 5 of the SRP which were integrated into the review of the DHRS system are listed below inasmuch as they apply and relate to structures, systems, and components important to safety. Because there is no parallel of this system in LWRs, the intent, rather than the direct application, of these sections of the SRP was used to evaluate the DHRS and its ability to perform its intended safety function. The DHRS is designed to accommodate a single-active failure and retain the capability to maintain coolable core geometry.

(1) SRP Section 5.4.6, "Reactor Core Isolation Cooling System," as it relates to a safety system that serves as a standby source of cooling to provide decay heat removal capability whenever the main system is isolated. The P&IDs and the functional description in the PSAR were reviewed to verify that the system will be capable of performing its intended safety function.

(2) SRP Section 6.3, "Emergency Core Cooling System," as it relates to the DHRS ability to withstand a single-active failure and the capability for alternate sources of electrical power. The DHRS has no parallel in LWRs with respect to this SRP section. This system does not function in the same manner as the ECCS in LWR plants.

It is the opinion of the staff and its consultants (BNL) that the DHRS will provide the supplementary capability for decay heat removal and will meet the design objectives outlined in Section 5.6.2 of the PSAR. The staff concludes that the proposed design of the DHRS will meet the requirements of the PDC as

applied to this system. Based on the evaluation of this, the DHRS, and conformance with the PDC, the staff concludes that the applicants' proposed preliminary design and associated design criteria can be use with reasonable assurance that the final design will be acceptable from the safety standpoint and is acceptable for issuance of a construction permit.

5.7 Liquid Metal-to-Gas Leak Detection System

5.7.1 General Description

The coolant leak detection system in a liquid metal cooled reactor performs a different role than the leak detection system in the conventional LWR. Leakage from valve stems and packing glands in an LWR is fairly common and expected. In a sodium-cooled system, the concerns are quite different. Although the sodium is hundreds of degrees below its saturation temperature and does not vaporize when inadvertently released from the system, sodium readily combines with any available oxygen and releases energy (burns) and combines with water or water vapor to form corrosive reaction products. Because of these concerns coupled with the implicit indication of failure of the integrity of the reactor coolant system boundary when leakage is detected, the detection of even the smallest leakage is important.

The purpose of the coolant boundary leakage detection system is to provide the means to detect coolant leakage and to the extent practical, identify the source of reactor coolant leakage. The staff has reviewed the applicants' liquid metal-to-gas leak detection system for purpose of determining its adequacy to perform the detection and monitoring function in conformance with the CRBR Principal Design Criteria, applicable regulatory guides, as well as appropriate applicants' performance design and test specifications. An additional description of this system is provided in Section 7.3.2.6.

Description of Applicants' Liquid Metal-to-Gas Leak Detection System

This section does not treat the leak detection of the steam generator system.

The steam generator leak detection system is treated in Section 7 of this SER. The leak detection system monitors the following CRBR systems.

(1) reactor enclosure
(2) primary heat transport system
(3) intermediate heat transport, and steam generator system
(4) impurity monitoring and analysis, reactor refueling and auxiliary liquid metal system.

The applicants' design bases are comprised of design criteria and quantitative and qualitative performance requirements. The salient design criteria are:

(1) The leak detection system must provide means for detecting and locating liquid metal-to-gas leaks throughout the plant at normal operating conditions.

(2) The leak detection system must operate at lower temperature conditions (<700°F), though its sensitivity may be reduced. The liquid metal-to-gas leak detection system shall detect and locate liquid metal-to-gas leaks

throughout the plant between the temperatures of 375°F to 1,000°F as required to fulfill continuous monitoring requirements of Appendix G the PSAR, "CRBR Plan for Inservice and Preservice Inspections."

(3) The leak detection system must perform its function during and after an OBE, even in the event of loss of offsite power.

The applicants' performance requirements for the PHTS are:

(1) The leak detection system must be able to detect "weeping" leaks which may have potential for long term growth. This leak is defined as a rate $\geq$ 100 g per hour in pipes and components at temperatures greater than 700°F.

(2) The leak detection system must be capable of identifying within which cells the leak has occurred. Also, detection must be provided within each of the reactor vessels, IHX, and pump guard vessels.

(3) Each cell (and contained guard vessels) shall be monitored for leaks by at least two diverse methods capable of providing the defined sensitivity. A confirmation method shall also be available in the event that conflicting information is provided by these systems.

(4) Each of the two diverse methods shall have an individual response time of less than 250 hours for the reference 100 g per hour leak size. Leaks of 30 gpm or larger shall be detected in less than 5 minutes. Leaks smaller than 30 gpm and larger than 100 g per hour shall be detected before either a total spill volume of 150 gal or 250 hours, which ever is less.

(5) Indicators and alarms for leakage detection shall be provided in the main control room.

(6) The leak detectors shall be equipped with provisions to readily permit testing for operability and calibration during plant operation.

5.7.2 Evaluation

In Westinghouse report WARD-D-0185, the applicants address their compliance with RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." The staff considers the intent and certain of the criteria of RG 1.45 to be applicable to CRBR. The staff believes that, consistent with the nature of the coolant and discussions in this section, the applicants are likely to meet the intent of RG 1.45.

Briefly, the type of detectors employed and locations chosen to implement RG 1.45 are as follows:

Contact detectors (spark plug) are designated for use in the space between the bellows and the stem packing of the bellows sealed sodium valves. The applicants plan to install 210 contact detectors. These detectors will be installed between the bellows and stem packing collar on bellows sealed valves. Contact detectors will consist of two open contact wires butted against the valve stem. If liquid sodium were to leak up the stem, it would contact the detector, causing a signal produced by reduction in resistance by making contact with both wires. The wires are gold plated, sheathed in stainless steel, insulated with mineral

oxide, and the sheath and insulation will be closed off by a glass seal, beyond the open contact butt ends.

Cable detectors will be placed in the bottom of guard vessels and below large tanks. The applicants plan to install 62 cable detectors. These detectors will consist of a loop of wire contained in a stainless steel sheath insulated with magnesium oxide. The sheath and insulation will be perforated at one end of the assembly to expose the wire loop. Contact with the wire loop by leaking liquid sodium will cause a low resistance path to ground, increasing the voltage drop across the alarm circuit, sounding the alarm.

Aerosol monitoring is performed by plugging filter aerosol detectors (PFADs). The applicants plan to install a number of PFADs. Aerosol monitoring will be performed by measuring the pressure drop across a membrane filter with a constant flow of gas sampled from, for example, the annular space between major piping and its insulation. The applicants plan to install sampling probes in the space within guard vessels and in cells containing liquid metal systems. Specifically, PFADs are to be installed in the main heat transport system piping and associated auxiliary piping, guard vessels, and major components.

Sodium ionization detectors (SIDs) are aerosol detectors which are to be used for cell atmosphere monitoring. CRBRP will have a number of SIDs installed in cells containing liquid sodium equipment and piping. The SID consists of a detector housing through which sampled gas is channeled. The sample gas will be ionized by means of a hot filament. Sodium ions will be picked up on a collector plate located opposite the ionizing fillament. Sodium ion current at the plate will be displayed and monitored.

In the PHTS, leak detection will be accomplished by monitoring the annuli between the reactor, IHX, pump, and their respective guard vessels. Cell atmospheres will also be monitored. Figure 5.21 (plan view) shows potential placement of PHTS leak detectors. The number of detectors and type and location will be determined during the FSAR stage of design.

In the inerted PHTS cells, aerosols will be monitored by particulate radiation monitoring equipment to pick up Na-24 or Na-22 resulting from the leaked coolant. Radiation particulate monitoring will be established for the following CRBRP cell groups.

Cell group	Equipment
1	Reactor guard vessel and cavity
2	PHTS loop 1 and components
3	PHTS loop 2 and components
4	PHTS loop 3 and components
5	Primary cold-trap sodium sampling package Primary cold-trap NaK loop equipment
6	Primary overflow vessel, primary makeup pump A, plugging temperature indicator
7	Primary makeup pump B, sodium valve gallery

Liquid sodium level sensors in the reactors, the EVST, the IHTS expansion k, and sodium storage tanks also will be monitored for indications of large l ks.

In addressing the possibility of loss-of-coolant accidents, the applicants have contended that the nature of the system and the properties of the materials used in the coolant system boundaries are such that small detectable leaks will occur rather than or before a large catastrophic break would occur. This has been deemed the "leak-before-break" concept. It is extensively discussed by the applicants in Westinghouse report WARD-D-0185. The staff's acceptence of this argument which is more fully discussed in Section 3.9, "Mechanical Systems and Components," Section 15.3, "Undercooling Events," and 15.6, "Sodium Fires," is predicated on a reliable, redundant diverse, leak detection system. The leak detection system will be, therefore, an essential factor in accepting the leak-before-break argument and the ability of the leak detection system to detect very small leaks becomes of greater importance. The staff reviewed the applicants' design and performance criteria, test results, and other information such as placement and diversity to determine if these elements reflected the ability to detect leaks consistent with the leak-before-break concept.

Tables 5.9 illustrates the development testing for liquid metal-gas leak detection systems. Tables 5.10, 5.11, and 5.12 tabulate the data which represent the leak detector types and locations in, respectively, the reactor and PHTS, IHTS and steam generators, auxiliary liquid metal systems, and other types of inferential measurements such as sodium level or cell pressures. Development test data were reviewed for cable, contact, SIDs and PFADs tested with sodium temperatures ranging from 350°F to 1,000°F in nitrogen atmospheres with oxygen content ranging from 1% to 21% and moisture contents ranging from 300 to 30,000 ppm. The sodium leak rates ranged from 0.4 to 243,000 g per hour. Westinghouse report WARD-D-0185 and Yule's Multi-Test Evaluations Report encompass some of the evaluated test data. Other data were presented in meetings between the applicants and staff. Figure 5.22, the staff feels, fairly represents a distillation of the data of particular interest.

5.7.3 Summary

In Westinghouse report WARD-D-0185, the applicants address their compliance with RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." The staff considers the intent and certain of the criteria of RG 1.45 to be applicable to CRBR. The staff believes that, consistent with the nature of the coolant and discussions in this section, the applicants are likely to meet the intent and pertinent criteria of RG 1.45. The staff believes that consistent with its importance to the leak-before-break concept, the leak detection system should be seismic Category I to satisfy the need for diversity after an SSE. The applicants agreed to make the leak-detection system in the PHTS, DHRS, and EVST seismic Category I and include them on Class 1E power in a meeting held December 21, 1983 (letter, Dec. 21, 1982, from J.R. Longenecker to P.S. Check). In addition to including an IHTS small-break-detection requirement in the PSAR and a technical specification pertaining to the operability

of the leak-detection system in the IHTs, the staff felt these commitments acceptably addressed their concerns. Refer also to Section 7.5.2.5.

The staff concludes from the applicants' proposed design and performance criteria, proposed location redundancy, diversity and independence and the test results summarized in Figure 5.22 that the proposed sodium to air leak-detection system is acceptable.

The proposed liquid metal to gas leak-detection system satisfies the intent and forms part of the body of information that satisfies the following principal design criteria: PDC 2, "Design Basis for Protection Against Natural Phenomena"; PDC 3, "Fire Protection"; PDC 4, "Protection Against Sodium and NaK Reactions"; PDC 5, "Environmental and Missile Design Bases"; PDC 8, "Reactor Design"; PDC 11, "Instrumentation and Control"; PDC 13, "Reactor Coolant System Design"; PDC 27, "Assurance of Adequate Reactor Coolant Inventory"; PDC 28, "Quality of Reactor Coolant Boundary"; PDC 29, "Fracture Prevention of Reactor Coolant Boundary"; PDC 30, "Inspection of Reactor Coolant Boundary"; PDC 32, "Fracture Prevention of Intermediate Coolant Boundary"; PDC 36, "Inspection of Reactor Residual Heat Extraction Systems"; and PDC 45, "Piping Systems Penetrating Containment."

5.8 Applicability of ASME Code Edition and Addenda and of ASME and ANSI Code Cases

This subject is covered in Section 3.9 of this SER.

5.9 Materials Surveillance Program for the Reactor Vessel and Internals

This review focused on PSAR Section 5.2, "Reactor Vessel, Closure Head and Guard Vessel" and specifically on Paragraph 5.2.4.5.1, "Surveillance," as given in Amendment 58, November 1980. The materials of the reactor vessel and internals that are of concern are types 304 and 316 austenitic stainless steel plates, forging, and welds.

The vessel beltline will be 2.38 in. thick. There will be a conical core support, which will be integral with the shell of the vessel. The core barrel and some other internals will have structural functions and are therefore of concern from the standpoint of neutron embrittlement. The closure head and guard vessel were not considered in this review, because fluence levels at those components are low.

Paragraph 5.2.4.5.1 states, "The requirements of Appendix H to 10 CFR 50 will not necessarily be followed since they were generated for ferritic material and the CRBRP reactor vessel will be austenitic." No alternative guidance is cited. The PSAR simply states:

> The philosophy of the CRBRP Materials Surveillance Program is to demonstrate continuing safe operation of permanent (30 year) reactor components by monitoring radiation induced property changes in the component materials. Monitoring of change is accomplished by testing materials specimens irradiated in the reactor, at intervals during the life of the plant and by comparison of the test data with data from unirradiated materials.

In this evaluation, Appendix H and ASTM E-185-7 were used for general guidan on determining the objective of a surveillance program and the selecting materials, specimen types, dosimetry, and withdrawal and testing schedules. For specific guidance, the staff used the fast flux test facility surveillance program as a model, keeping in mind that the two facilities are to be operated for different purposes. The PSAR also contains the following:

(1) Paragraph 4.3.2.9, "Vessel Irradiation," discusses the changes in fluence and energy spectrum to be expected with core life, and notes:

> One application of this flux data is in determining the total and fast fluence received by both replaceable and non-replaceable reactor components. The neutron fluence must be limited so that the end-of-life ductility for structural materials exceeds the specified minimum requirements.

(2) Values for fluence at the end of life (30 calendar years, 22.5 effective full-power years) derived from Table 4.3-25 are shown in Table 5.13.

(3) From Paragraph 4.2.2.1.1.4, "Fixed Radial Shielding," and Paragraph 4.2.2.1.1.9, "Removable Radial Shielding," comes the implementation of the concept that the core barrel and reactor vessel are to receive neutron fluences below that expected to reduce tensile ductility to 10% elongation at the end of design life. This fluence limit is stated to be 1.3×10^{22} n/cm^2 total fluence. If the predicted fluence values given in Table 5.1 are correct, the limit will not be exceeded for the core barrel or reactor vessel.

Paragraph 5.2.4.5.1 states:

> 2. Materials surveillance coupons, to monitor radiation effects in the materials of permanent reactor system components, shall be required if the predicted fluence is greater than 1×10^{21} n/cm^2, $E > 0.0$, in the component material.

If followed to the letter, this criterion would exclude reactor vessel materials from the surveillance program. This paragraph also states:

> 5. The test specimens shall be sub-size tension specimens as indicated in ASTM E8, having a gage diameter of 1/4 inch and a gage length of 1 inch and an overall length of 2 5/8 inch.

Thus, the criterion for vessel integrity is clearly tensile elongation. Yet, the modes of failure that are made more probable by neutron embrittlement (see Section 3.1, "Criterion 12--Reactor Coolant Boundary," of this report) include ductile rupture, creep rupture, and creep fatigue.

Evaluation

The staff review of the CRBR surveillance program has focused on the most probable modes of failure which are ductile rupture, creep rupture, and creep fatigue.

The staff in its evaluation has required additional information from the applicants about the types of specimens to be placed in surveillance, plans for dosimetry and thermal monitors, the location of surveillance capsules, and the withdrawal schedule. The staff also required more detail concerning the materials to be placed in surveillance, which components will be represented, and so forth. The information provided by the applicants has been reviewed and found sufficient to resolve the staff concerns. The staff finds the surveillance program and supplemental information acceptable for the purposes of this CP review.

5.10 Liquid Metal/Materials Compatibility

This subject is covered in Section 4.2.3 of the SER.

5.11 Heat Transport System Materials

5.11.1 General

The cooling capability of the heat transport system (HTS) will be 975 MWt and the reactor will be operated with three parallel independent primary loops of piping to transport heat from the reactor to the intermediate heat exchangers (IHXs). The IHX will separate radioactive from nonradioactive sodium and then will transport the heat to the steam generator system. The primary sodium will flow through the cold-leg piping through a check valve to the inlet nozzle of the reactor vessel located near the lower end of the vessel. The three loops will be essentially identical and will provide equal sodium transport.

The intermediate heat transport system (IHTS) will consist of piping and components to transport reactor heat from IHX to the steam generator system (SGS). The heated intermediate sodium will leave the IHX, flow through 24-in. (OD) hot-leg piping, and enter the SGS superheater module. Two 18-in. parallel pipings will carry the intermediate sodium from the superheater exit to two evaporator inlets. Cooled sodium from the evaporators will flow through two parallel 18-in. runs, joined together at a tee to a single 24-in. (OD) piping to complete the circuit through IHX.

There will be no piping connection between the primary and intermediate sodium loops. The IHX tube bundle will isolate radioactive primary sodium from the intermediate sodium and will provide a mechanical barrier to the transport of radioactive sodium out of containment. The sodium pressure within IHX will be maintained at a minimum of 10 psi above the shell side (primary) sodium so that any tube leaks will not result in a leakage of radioactive sodium into the intermediate system.

The heat transport system design conditions are similar to those of the fast flux test facility (FFTF), which is in operation, and the structural adequacy of the system was determined under a number of variable loading conditions, both normal and off normal, including a load histogram for the anticipated

events during 30 years of design life. In addition, an extensive developme and test program was conducted to verify design and component performance capability. Before startup, the applicants are planning extensive construction and elevated temperature acceptance testing to further prove a confirm the structural integrity of the components.

Each cold leg of IHTS will contain a single-stage centrifugal pump with a variable drive, which will control the sodium flow rate; a pony motor, which will provide 7½% of the design flow under accidental conditions, startup, shutdown, and the hot standby; and an expansion tank, which will accommodate sodium volume changes under operating conditions.

5.11.2 Primary Sodium Loop

The components of the primary heat transport system (PHTS) will be located inside the containment building and could be virtually isolated so that a fault in one circuit will not jeopardize the operational capability of the other two loops.

The PHTS components will be designed, analyzed, and fabricated in accordance with the ASME Code, Section III, Division I, 1974 Edition, Addenda to Summer 1975 for piping and check valve and Addenda to Summer 1974 for IHX, primary pump guard vessel, and IHX guard vessel, and supplemented by Code Cases (1592, 1593, 1594, 1596) for elevated temperature application and appropriate Department of Energy standards, for example, RDT E15-2NB-T, RDT F9-4T, and RDT F9-5T.

10 CFR 50, Appendix A, was modified where necessary to provide an adequate basis for the safety evaluation of FFTF. NRC is using similar criteria for CRBRP. CRBR PDC 12 requires that the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, or of gross rupture.

The materials for PHTS components shall conform to the ASME Code and RDT E15-2NB-T, October 1975. The reactor coolant pressure-retaining boundaries shall be fabricated from either type 304 stainless steel or type 316 austenitic stainless steel, with 0.055% minimum carbon for service above 800°F. In order to avoid the possibility of accelerated erosion, the coolant velocity will be limited to 30 ft per second. The use of austenitic stainless steel will also minimize the corrosion. The weld filler material for PHTS components and piping shall conform to the requirements of the RDT standards.

The maximum design temperature for the PHTS will be 1,015°F; the corresponding expected operating temperature will be approximately 50°F lower. Since the temperature will be above 800°F, the time-dependent behavior (or creep) of materials must be considered. Long-term creep rupture tests (~6 years) at 1,100°F and 1,200°F have revealed no discontinuities in stress rupture curves for annealed type 304 stainless steel; discontinuities in stress rupture curves do occur frequently in cold-worked materials. Extrapolations for creep rupture times of up to 2×10^5 hours are given for both types 304 and 316 stainless steels, based on analytical procedures that do not take into account possible changes in fracture mode at long times influencing the gradients of the stress-rupture curve at times approaching the lifetime of the HTS. The current design

assumes that the stress-rupture curves given in ASME Code Case 1592 are a reasonable extrapolation to long times.

The evaluation of creep fatigue interaction will follow the requirements of ASME Code Case 1592. The creep strength in the hotter regions of the HTS will be affected by sodium exposure; the available data indicate that fatigue life under strain-controlled conditions will be increased by high-temperature sodium exposure when compared with tests in air environment.

The applicability of Charpy V-notch toughness measurements to austenitic materials and weldments has not been demonstrated. Instead, J-integral tests and analysis procedures are currently utilized. Typical J-integral values for type 304 base metal are between 1,680 and 4,000 in.-lb/in^2 at the temperature range from room temperature (RT) to 830°F. A typical value for type 316 stainless steel is 3,500 in.-lb/in^2 at RT.

All allowable design stress values specified in the ASME Code were obtained from creep tests in air at 1,000°F to 1,100°F, and procedures are available to account for aging effects from stress rupture behavior in nonoxidizing sodium environments by a possible decrease in rupture strength. The change in rupture strength results from carbon and nitrogen transfer.

Several investigators studied crack growth in the weldments of types 308 and 316 filler metal that resulted from elevated temperature fatigue. The available J-integral values are 660 and 640 in.-lb/in^2 for types 308 and 316 submerged-arc weld deposits, respectively. The effects of test temperature, thermal aging, crack orientation relative to the weld, welding process, and irradiation are examined. The data from the Hanford Engineering Development Laboratory (HEDL), foreign test data, and the weld data show the same resistance to fatigue crack extension as the data on the base metal at high temperature.

Comprehensive test programs are currently under way for high-temperature test data applicable to CRBRP design procedures. Research is being conducted at both HEDL and NRL (Naval Research Laboratory) to investigate the toughness of stainless steel weldments and J-integral applications. The present fracture-mechanics calculation shows that the critical crack length is more than 22 in. The 24-in. (OD) pipe has a thickness of 0.5 in. Hence, any inside crack which escapes detection will surface ultimately if it grows 3 to 4 in. The recommended ASME aspect ratio is 1 to 6 (depth to total length). A 1-in.-long through-crack will leak profusely and can be detected within 2 hours.

The delta ferrite level in a weld composition will conform to the requirements of the ASME Code, Section III, 1974 and the Code Case 1592. The CRBRP had adopted the range of 5 to 9% delta ferrite for reducing the susceptibility of stainless steel welds to hot cracking.

For protection against stress corrosion, sensitization should be minimized by various measures, for example, the welding with low fluoride fluxes and heat treatments above 1,600°F (followed by rapid cooling). In case of a leakage, sodium will react with oxygen in the air environment so that the general corrosion on HTS cold-leg piping surface can be expected to be from 1 to 3 mils per month with occasional maximums to 20 mils per month.

Rules governing the high-temperature Class 1 components are as follows: (1) the rules for materials in NB-2000 shall apply except as modified by Code Case 1592 design rules, replacing NB-3000 and (2) DOE standards RDT F9-4T, January 1976, and RDT F9-5T, September 1974, shall be used to supplement the rules of Code Case 1592.

The PHTS boundary sections will be made from unstabilized types 304 and 316 stainless steels in which the minimum carbon ranges from 0.04% to 0.055% and the maximum upper limit of 0.08% carbon will be specified in order to avoid the interstitial loss in the primary hot-leg piping.

Both carbon and nitrogen levels in types 304 and 316 stainless steel have a strong influence on the ultimate strength, the yield strength, the ductility, and the rupture strength of materials; however, type 316 stainless steel has far superior ultimate strength, ductility, and rupture strength than the type 304 stainless steel for equal amounts of carbon and nitrogen. In the temperature range of 800°F to 1,200°F, the fatigue behavior of type 316 with 0.06% carbon and 0.04% nitrogen had a shorter life than type 304 with 0.053% carbon and 0.052% nitrogen for equivalent heat treatments and test temperatures at the same total strain range. If the comparison is based on stress amplitude, then type 316 is superior.

The tests at the elevated temperatures of 1,110°F to 1,470°F have shown that decreasing the carbon content from 0.05 to 0.004 w% in 18% Cr-12% Ni iron-based alloys lowers the fatigue life and the endurance limit. This happens because of the beneficial effect of carbides in blocking grain boundary sliding and migration, thereby reducing the crack growth. A comprehensive research program is under way to investigate the interstitial transfer on the fatigue of liquid metal fast-breeder reactor (LMFBR) materials. No reduction of fatigue properties because of carbide precipitation is expected either in type 304 or type 316 stainless steel at the operating temperatures of CRBR.

Neutron irradiation above 10^{21} n/cm^2 increases the yield and tensile strengths, decreases ductility, and reduces the creep rupture strength. However, the fluence expected to be seen by the primary heat transport system piping and components will be at least several decades below this level. Therefore, the effect of neutron irradiation on mechanical properties is not a consideration for PHTS materials.

5.11.2.1 Verification and Acceptance Test Programs

The PHTS piping will be hydrostatically tested in accordance with the ASME Code, Section III, NB-6000, and Code Case 1595 (supplemented by RDT E15-2NB-T, October 1975). The lengths of straight pipe will be tested by the manufacturer to satisfy the Code specifications. The thermowell body subassembly will be tested under external pressure, and the IHX ventline flow restrictor will be tested by the manufacturer. Each completed PHTS piping subassembly (spool) will be hydrostatically tested by the spool fabricator at his facility.

5.11.2.2 Insulation

The reactor vessel, pumps, and the IHX will be enclosed by guard vessels. A semi-inert gaseous atmosphere of 0.5%-2% oxygen in nitrogen will be maintained

between the components and the guard vessels. The piping and the upper portions of the components containing sodium external to the guard vessels will also be insulated by alumina silicate blankets, which will be as free from halogens and/or moisture as possible. An annulus will be provided for heaters and leak detection equipment; therefore, there will be no direct contact either with components or the piping. This will minimize any potential contamination of the piping by corrosive elements in the insulation.

5.11.2.3 Intermediate Heat Transport System

The intermediate heat transport system (IHTS) will extract reactor-generated heat from IHX and will deliver this heat to the steam generation system under normal and offnormal operating conditions; for example, decay heat will be removed by pony motor flow or by natural circulation.

The IHTS design, erection, and testing of the components shall be in accordance with the ASME Code, Section III, 1974, with Winter 1975 Addenda, and with applicable Code cases for elevated temperature operation. The basis for high-temperature design and analysis of IHTS components will follow the same rules governing PHTS components as noted earlier, including piping and pump system.

The IHTS will be composed of three independent and physically separated loops, contained within individual cells. Any spill within a cell will be self-contained and the adjacent loops will be unaffected.

The hot-leg piping will be 24-in. (OD) x ½-in. wall type 316H stainless steel and will extend from the IHX outlet nozzle to the superheater inlet nozzle after penetrating the reactor containment wall. From the superheater outlet nozzle, two parallel piping runs of 18-in. (OD) x 9/16-in. wall 2¼% Cr-1% Mo ferritic steel will extend to each of the two evaporator inlet nozzles. The cold-leg piping will start at the evaporator outlet nozzles and will consist of two parallel runs of 18-in. (OD) x ½-in. wall type 304H stainless steel pipes. The lines will be joined by a 24-in. x 24-in. x 24-in. type 304H mixing tee and will continue as a single 24-in. (OD) x ½-in. wall type 304H pipe, which will be connected to a 36-in.-diameter IHTS coolant pump suction through a 7-ft-long diffuser. The cold-leg-A 24-in. (OD) x ½-in. wall type 304H will connect the pump discharge to the IHX inlet nozzle, completing a loop. In case of large or intermediate sodium-water reactions, reaction products might form in the evaporator or superheat modules, thereby subjecting the system to overpressure. The faulted condition of the evaporator or superheater module will be relieved through rupture discs.

The analytical methods and data for the IHST will be developed from the design, construction, and operation of sodium systems developed for the FFTF. Similarly, structural evaluation, stress analysis verification, and compliance with code requirements will be identical to the PHST.

The IHTS piping will be a low-pressure (<225 psia) system and will operate in the temperature range of 650°F to 965°F. The hot-leg piping of IHTS, from containment to superheater inlet, will be highly stressed because of high thermal expansion stresses. The elbows and the 2¼% Cr-1% Mo stainless steel transition piece will also be critical items in the pipe system.

The critical crack sizes for elbows in each leg of IHTS piping were determi based on through-the-wall flaws at the operating and design pressures. The assumed initial flaws were made larger than any flaw allowed by the RDT standards. The final crack lengths were determined by cyclic loading on types 304, 304H, and 316H stainless steel pipings and were found to be much smaller than the critical crack lengths. However, no crack growth calculations have been carried out for 2¼% Cr-1% Mo, ferritic steel and Incoloy materials.

5.11.2.4 Guard Vessels

Guard vessels will be provided for the circulation pump and IHX in the primary sodium loop. The guard vessels will be constructed to the requirements of Section III of the ASME Code, 1974 Edition, including Summer 1974 Addenda, and supplemented by Code Case 1592-4, DOE's RDT E15-2NB-T, November 1974, and DOE's RDT F9-4T, September 1974. The guard vessels will be fabricated from type 304 austenitic stainless steel.

5.11.2.5 Leak Detection

Tests have shown that aerosol-type detectors will reliably detect small sodium leaks (~100 g per hour) when sampling pipe insulation annuli. Aerosol will be generated by the sodium-oxygen reaction in the presence of a small amount of moisture. The plugging filter aerosol detector (PFAD) will embody a differential pressure sensor across a filter, which will get plugged when aerosol collects as the gas passes through the filter. The sodium ionization detector (SID) will have the filament at the proper temperature and will ionize the sodium. Cable and contact detectors will be located in guard vessels and in bellows sealed valves, respectively, to sense the presence of liquid sodium. Radiation monitors will be provided to detect radioactive aerosols released in all cells containing primary sodium.

5.11.2.6 Long-Term Thermal Aging Effects on Welds

The performance of the pressure boundary welds in the primary heat transport piping system for a 30-year life of the plant requires knowledge of long-term effects on welds of thermal aging. However, it is recognized that the data base is limited. The uncertainty associated with the projection of short-term data to long-term performance can be satisfactorily resolved by testing prototypical CRBRP piping and welds. This out-of-plant program will provide several years of lead time on the actual plant materials.

The microstructural stability of the base metal and the weld metal over the plant life was evaluated and reflected microstructural changes on tensile, creep, and fatigue. The fracture toughness properties were also studied.

5.11.3 Evaluation

The CRBR project management provided information on the concept of "leak before break" in the heat transport system. It is the judgment of the staff that a double-ended instantaneous pipe break should not occur and should not be considered as an initiating event for the failure of the heat transport system.

Three avoidance criteria, the excessive plastic strain under worst-case loading, the general fatigue damage under specified cyclic loading and the avoidance of through-wall crack propagation, are very conservative.

Types 316 and 304 austenitic stainless steel pipes used in HTS will be solution annealed to preclude carbide precipitation that lowers corrosion resistance. Test data verify that the yield strength and ductility of the materials will be satisfactory. The chemical composition has been correctly identified. Ultrasonic tests have been employed to check for subsurface defects.

A comprehensive review of the sodium piping experience has shown that no leak in the piping can be attributed to a residual initial defect in the parent metal of the piping system. Out of 29 instances of leaks in the sodium piping, only one was directly attributable to a large initial defect. The 2-lb sodium leak attributable to a large defect was associated with a substandard weld with excessive porosity. This weld was fabricated in 1965 before the publication of ASME Code, Section III. Appendix G of the ASME Code requires a "quarter-thickness" semielliptical surface crack with length equal to six times the depth to ensure toughness of materials under severe conditions. Therefore, the postulated initial defect size for the HTS piping of 0.125-in. depth and 1.5-in. length located with worst-case orientation represents a conservative initial defect for the HTS piping.

The sodium effects such as carbon transfer that the hot leg will encounter, will be monitored by locating type 316 stainless steel specimens on the section of the in-vessel transfer machine port plug below the supressor plate. Type 304 stainless steel cold-leg surveillance samples will be incorporated into the nozzle section of the removable radial shield.

The maximum fluence to which the primary piping loop will be exposed over a 30-year plant life is less than 3×10^{19} n/cm^2 ($E > 0.1$ MeV). The flux spectrum is soft, and less than 1% of the neutrons have energies greater than 0.1 MeV. The degradation of the primary piping is considered to be of no significance.

The primary membrane stresses in HTS are relatively low in comparison to PWR-type piping system, because the pressure is small. Hence, HTS has large margins of safety against the load-controlled mode of failure. The piping has been analyzed using data rules of NB-3650, and primary membranes and primary membrane-plus-bending stresses are below the design limits at all points in the piping. Hence, the piping integrity for the primary 24-in. hot leg and the cold leg is ensured against the abrupt gross failure. The strain limits based on primary-plus-secondary stresses or ratchetting and fatigue damage limits are satisfactory for the cold leg. The hot leg operates in the creep range; inelastic analysis will be required to assess the ratchetting and creep-fatigue damage. The fracture mechanics evaluation has shown that the critical crack lengths for the hot- and cold-leg pipings are in the neighborhood of 20-25 in., and the assumed elliptical flaw (0.125 in. deep and 1.5-in. long) in the wall of the HTS will never reach the critical lengths of 20-24 in. under CRBR loading conditions. Hence, the leak-before-break characteristic is rational and applicable to the HTS.

The design-basis leak rates chosen for the PHTS and IHTS are 8 gpm and 1,000 m, respectively. The 8-gpm leak results from a 4-in. crack in the piping, which is the maximum expected crack size considering the leak-before-break materials behavior and the ability of the sodium leak detection system to detect small leaks. The 1,000-gpm leak results from considering the maximum leak possible in a moderate energy fluid system as defined by BTP MEB 3-1. The selection of a larger leak in the IHTS results from the fact that this piping is not in inerted cells and is not monitored by the safety guide leak detection system, thus presenting the possibility that a small leak could develop undetected into a larger leak.

The staff has reviewed and accepted these design-basis leak rates.

Liquid metal-to-gas leak detection testing has indicated that conductivity type leak detectors (probe and cable) are only useful in detecting either a large leak or leaks in a confined space. Annulus-type aerosol sampling, using either SID or PFAD (plugging filter aerosol detector), provides a sensitive method of detecting leaks in both air and nitrogen down to the range of 2-100 g per hour. When the liquid temperature is above 500°F, large leaks (200 kg per hour) can be detected by aerosol detectors within minutes.

5.11.4 Conclusions

PDC 12 requires that the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The HTS piping system is designed to ensure that stresses, strains, and deformations are within the applicable code criteria and functional limits of the system. DOE's RDT standards imposed inspection limits on the materials and welds; and the leak detection system will ensure that the requirements of PDC 12 are met.

PDC 26 requires that the HTS shall be designed to reliably remove heat from the reactor and transport the heat to the turbine generator or the ultimate heat sinks under all plant conditions--including normal operation, anticipated operational occurrences, and postulated accidents.

Following a normal upset or an emergency condition, HTS will remain sufficiently intact to be capable of performing its decay heat removal function, including maintenance of primary coolant pump pony motor flow. Hence, the intent of the above criterion is satisfied.

The quality of the reactor coolant boundary has been specified according to the ASME Code, Class 1, and DOE's RDT standards, meeting the requirements of PDC 28.

The staff, therefore, concludes that the HTS has been designed according to strict ASME Code and DOE's RDT standards. The cold-leg piping meets all the requirements, but the hot-leg piping does not meet the creep and the fatigue requirements at the highly stressed elbows. However, analysis shows that it does meet the strain limits. Analyses of a postulated 4-in.-long crack in the HTS piping show that the crack will not extend to a critical length under the

cyclic mechanical and thermal loadings. A leak will be detected and the appropriate steps can be taken. Hence, the integrity of the HTS piping can be reasonably ensured by acceptable periodic inservice inspections.

5.11.5 Reactor Vessel Integrity

The staff concludes that structural integrity of the reactor vessel is acceptable and meets the requirements of CRBR PDC 1, 5, 12, 28, 29, and 30. This conclusion is based on the staff's review of the preliminary safety analysis report (PSAR).

The staff has reviewed all factors contributing to the structural integrity of the reactor vessel and concludes there are not special considerations that make it necessary to consider potential reactor vessel failure for this plant. The bases for this conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable NRC regulations, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III. The inherently high toughness of the austenitic materials used in the construction of the reactor vessel ensures the structural integrity and provides assurance that readily detectable leakage would occur before any sizable break could occur.

The integrity of the reactor vessel is ensured because the vessel

(1) will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and any pertinent Code Cases.

(2) will be made from materials of controlled and demonstrated high quality

(3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies

(4) will be operated under conditions and procedures and with protective devices that provide assurance that design conditions of the reactor vessel will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients;

(5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions

Neutron irradiation has the following effect on the mechanical properties of austenitic stainless steels:

(1) Yield and tensile strengths are rapidly increased.
(2) Ductility is decreased.
(3) Creep rupture strength and creep ductility are greatly reduced.
(4) Fatigue life decreases.

The figures indicate that for fluences less than 10^{21} n/cm^2, the effects are not significant. Fluences seen by the PHTS piping and components several decades below this level. For example, the maximum fluences external to the

reactor vessel (at the core midplane) are 3.4 x 10^{18} n/cm^2 and 2 x 10^{20} n/cm for fast (>0.1 MeV) and total fluences, respectively. Therefore the effect of neutron irradiation on mechanical properties is not a consideration for PHTS materials.

5.11.6 Reactor Vessel Materials

(1) The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code and other applicable requirements. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of CRBR PDC 1 and 28.

(2) Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified. Since the applicants have certified that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes used are considered acceptable. Compliance with these Code provisions meets the quality standards requirements of CRBR PDC 1 and 28.

(3) Special methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified and have been found equivalent or superior to the techniques described in ASME Code Section III and RDT standards. Demonstrations have been made using these special techniques and have satisfied all requirements of the Code. The special methods of nondestructive examination are deemed acceptable. This acceptability based on the Code provisions satisfies the quality standards requirements of CRBR PDC 1 and 28.

(4) Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of ASME Code and contract documents. Qualification in accordance with the Code provisions meets the requirements of CRBR PDC 1 and 28 concerning quality standards.

(5) When welding components of austenitic stainless steels, the delta ferrite requirements and determination for filler classification E308L, E308, ER308L, and ER308 shall conform to the requirements of Section III of the 1974 ASME Boiler and Pressure Vessel Code. The delta ferrite range of 5% to 9% shall be used since these limits are consistent with the Code and RDT standards. These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. These controls meet the quality standards requirements of CRBR PDC 1 and 28 and satisfy the requirements of CRBR PDC 12 regarding fabrication to prevent RCPB rapid propagating failure.

Surface finish and cleanliness will also meet all requirements of the ASME Code and the other contract documents. Periodic swab tests of stainless steel surfaces during fabrication in the shop will be performed to ensure that potentially harmful substances such as chlorides do not

contact the components in concentrations greater than specified in applicable codes and standards.

The controls used provide assurance that austenitic stainless steel components will be properly cleaned on site. The controls satisfy Appendix B of 10 CFR 50 regarding controls for onsite cleaning of materials and components.

(6) Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of the ASME Code, and other applicable documents. Compliance with these recommendations satisfies the quality standards requirements of CRBR PDC 1 and 28, the prevention of fracture of the RCPB requirement of CRBR PDC 29, the requirements as detailed in the provisions of the ASME Code, Sections II and III, and other applicable documents.

5.11.7 Reactor Coolant Pressure Boundary Materials

The staff concludes that the plant design is acceptable and meets the requirements of CRBR PDC 1, 4, 5, 12, 28, and 29.

The materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of CRBR PDC 1 and 28.

The materials of construction of the primary heat transport system (PHTS) exposed to the reactor coolant have been identified and all of the materials are compatible with the sodium coolant, which is chemically controlled in accordance with appropriate Technical Specifications. This compatibility has been proven by extensive testing and satisfactory performance. General corrosion of all materials will be negligible. The above evidence of compability with the coolant and compliance with the Code provisions satisfies the requirements of CRBR PDC 4 relative to compatibility of components with environmental conditions.

The materials of construction for the PHTS are compatible with the thermal insulation used in these areas. Conformance with the above recommendations satisfies the requirements of CRBR PDC 12 and 29 relative to prevention of failure of the RCPB. The tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with the provisions of the ASME Code, Section III. Compliance with these Code requirements satisfies the quality standards requirements of CRBR PDC 1 and 28.

Compliance with these provisions and NRC regualations satisfies the requirements of CRBR PDC 29 regarding prevention of fracture of the RCPB.

The applicants have not stated whether electroslag welding has been used in the fabrication of the PHTS.

The applicants also have not committed to special controls for the production of welds made under conditions of limited accessibility.

All materials for the PHTS piping, pumps, valves, heat exchangers, and guard vessels are specified to minimize corrosion and erosion and ensure compatibility with the environment.

In order to avoid the possibility of accelerated erosion, the internal coolant velocity is limited to less than 30 ft per second.

The materials used for PHTS components shall conform to the ASME Code and the supplemental requirements of Department of Energy Standard RDT E15-2NB-T, October 1975. All parts of the PHTS that form part of the reactor coolant pressure-retaining boundaries shall be fabricated from either type 304 or type 316 austenitic stainless steel. Material for service above 800°F will contain 0.55% carbon (minimum).

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprises part of the PHTS such as the loop drains and connecting lines to other systems are also austenitic stainless steel.

The check valve in the PHTS which is in contact with the sodium coolant is constructed primarily of stainless steel. Other materials in contact with the coolant are special materials, such as those for hard surfacing.

All parts of the PHTS coolant pump in contact with the reactor sodium coolant are of austenitic stainless steel except for bearings and special parts.

The exclusive use of austenitic materials for the PHTS provides reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the PHTS.

All parts of the intermediate heat exchanger that form part of the pressure-retaining boundaries or that are exposed to liquid sodium containing environments shall be fabricated from either type 304 or type 316 austenitic stainless steel.

The guard vessels for the intermediate heat exchanger and primary pump components shall be fabricated from type 304 austenitic stainless steel.

Selection of althernate materials shall be based on the mechanical properties, metallurgical stability, sodium compatibility, and response to radiation under the applicable design and environmental conditions. When recommending the use of alternate materials, the supplier shall document the justification which shall include, as a minimum, a summary of available test or experience data and a discussion of the adequacy of the recommonded materials relative to types 304 or 316 stainless steel or other purchaser-specified alternate.

The delta ferrite requirements and determination for filler classifications E308L, E308, ER308L, and ER308 shall conform to the requirements of Section III of the 1974 Edition of the ASME Boiler and Pressure Vessel Code. The delta ferrite range of 5% to 9% shall be used since these limits are consistent with the Code and RDT standards. These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures

during welding and will have high structural integrity. These controls meet the quality standards requirements of CRBR PDC 1 and 28 and satisfy the requirements of CRBR PDC 12 relative to prevention of leakage and failure of the primary heat transport system.

5.11.8 Steam Generator Materials

On the basis of the considerations that follow, the staff concludes that the materials specified for the steam generator are acceptable and meet the requirements of PDC 1, 12, 13, and 29.

The steam generator system will be constructed in accordance with Section III of the ASME Code, Code Case 1592, and DOE's RDT F9-4T and RDT E15-2NB. The steam generator modules are classified as Class 2 components, but will be constructed to Class 1 rules, using the same high-temperature design bases identified for the primary heat transport system, and will comply with RG 1.48.

The maximum design temperatures are 965°F for 2¼% Cr-1% Mo steel and 925°F for type 304 stainless steel. The allowable stress values satisfy Code Case 1592.

The shell and tube material will be 2¼% Cr-1% Mo steel. The compatibility of 2¼% Cr-1% Mo steel with the sodium coolant has been ascertained by experimental data. The initial carbon content will be greater than 0.07% and less than 0.11%. The final carbon content will be adequate for creep rupture properties after decarburization at 965°F. The materials inspection program will be in accordance with the ASME Code, Section XI.

Additional data on 2¼% Cr-1% Mo steel for creep fatigue and time-dependent operational limitations are under investigation at Oak Ridge National Laboratory and General Electric. Adequate margin of safety against fracture for steam generator modules is provided by compliance with Paragraph 3241 of Code Case 1592, which involves Appendix G of Section III of the ASME Code.

5.11.9 Fracture Prevention of Containment Pressure Boundary

This SER assesses the materials in the CRBRP containment system that will constitute the containment pressue boundary to determine if the material fracture toughness is in compliance with the requirements of PDC 42, "Fracture Prevention of Reactor Containment Boundary."

PDC 42 (which parallels GDC 51 of 10 CFR 50, Appendix A) requires that under operating, maintenance, testing, anticipated operational occurrences, and postulated conditions (1) the materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The CRBR containment system includes a ferritic steel containment vessel, which is free standing above the operating floor level and which is embedded in concrete from the foundation up to that elevation.

The materials of the containment pressure boundary which are considered in the assessment are those which will have been applied in the fabrication of the freestanding containment vessel, equipment hatch, personnel lock, penetrations,

and fluid system components including the valves required to isolate the system. Thse components are the parts of the containment system which are not backed by concrete and must sustain loads during the performance of the containment function under the conditions cited by PDC 42.

The staff has determined that the fracture toughness requirements contained in ASME Code editions and addenda typical of those used in the design of the CRBR containment may not ensure compliance with PDC 42 for all areas of the containment pressure boundary. In its licensing reviews of containment pressure boundary materials, the staff has elected to apply the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. Because the fracture criteria that have been applied in construction typically differ in Code classification and Code edition and addenda, the staff has chosen the criteria in the Summer 1977 Addenda of Section III of the Code to provide a uniform review, consistent with the safety function of the containment pressure boundary materials. Therefore, the staff has reviewed the materials of the components of the CRBR containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Considered in the staff review were components of the containment system which will be load bearing and provide a pressure boundary in the performance of the containment function under operating maintenance, testing, anticipated operational occurrences, and postulated accident conditions as addressed in PDC 42. These components are the free-standing containment vessel, equipment hatch, personnel airlock, penetrations, and elements of specific containment penetrating systems.

In some cases, materials had been fracture toughness tested or had been inappropriately tested. Generally, those materials had not been fracture toughness tested because the ASME Code edition and addenda in effect at the time the components were ordered did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested or were inappropriately tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," October 1979, and ASME Code, Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials, with respect to their fracture toughness, was developed from a review of how these materials were fabricated and the thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code, Section III, provided the technical basis for the staff's evaluation of compliance with the Code requirements for materials that were not fracture toughness tested.

5.11.9.1 Conclusions

On the basis of its review of the available fracture toughness data and materials fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, the staff concludes that the

materials of components of the CRBR containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these Code requirements provides reasonable assurance that the materials of the CRBR containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of PDC 42 are satisfied.

5.12 Inservice Inspection Evaluation

5.12.1 Inservice Inspection of the Reactor Coolant Boundary and the Reactor Residual Heat Extraction Systems

5.12.1.1 Examination Requirements

PDC 30, "Inspection of Reactor Coolant Boundary (RCB)," requires, in part, that the components shall be designed to permit periodic inspection and testing of areas and features improtant to safety and to assess their structural and leaktight integrity. PDC 36, "Inspection of Reactor Residual Heat Extraction Systems (RRHES)," requires that the components be designed to permit appropriate periodic inspection of components, such as heat exchangers and piping, to ensure integrity and capability of the system. PDC 37, "Testing of Reactor Residual Heat Extraction System," requires, in part, that the systems be designed to permit appropriate periodic pressure and functional testing to ensure the structural and leaktight integrity of its components.

5.12.1.2 Applicants' Proposed Preservice Inspection and Inservice Inspection Plan for the Reactor Coolant Boundary and Reactor Residual Heat Extraction Systems

The staff reviewed the available information in the PSAR related to preservice and inservice inspections (PSIs and ISIs). The applicants' docketed position on the subject is contained in Appendix G of the PSAR, Amendment 68, dated May 1982. To meet the applicable principal design criteria, the applicants state that the CRBRP PSI and ISI plan will be based on Section XI of the ASME Code, Division 3, "Rules for Inspection and Testing of Components of Liquid Metal Cooled Plants." However, the applicants take exception to or request relief from certain provisions of Section XI of the Code.

The applicants propose to use leakage detection systems and visual examinations as the principal methods for inservice inspection. The bases for this general approach are the ductile characteristics of most of the materials of construction, the relatively low operating pressures, and design features such as guard vessels, guard pipe, and elevated piping system loops that will limit the amount of leakage, if it occurs. Section XI of the ASME Code has requirements for periodic visual examinations and for diverse and redundant leak detection devices to monitor each component. In addition, the first publication of Section XI, Division 3, contains the following requirements:

(A) Volumetric examination of 100% of each of the two dissimilar metal shell welds in the reactor vessel during each 10-year inspection interval.

(B) Volumetric examination of 33% of each of the two dissimilar metal welds in the ex-vessel storage tank (EVST) during each 10-year inspection interval.

(C) Volumetric or surface examination of 33% of all welds on carbon steel guard vessels and guard pipe during each 10-year inspection interval.

(D) Volumetric or surface examination of integrally attachments welded to vessels, pumps, valves and piping.

Several topics are identified in Section XI as "in course of preparation." These topics include examination requirements for the structural welds in the carbon steel reactor vessel closure head and EVST cover, and most of the acceptance standards for nondestructive examination.

The applicants take exception to the requirement of performing a preservice and inservice volumetric examination of the two dissimilar metal shell welds in the reactor vessel and EVST and propose to substitute a visual examination as an alternative. The applicants request relief from the volumetric or surface examination of the only integrally welded attachment, a snubber attachment weld to the check valve. The CRBRP does not currently plan to use carbon steel for the guard vessels and the guard pipe. The applicants propose to reduce the required frequency of inservice visual examinations on several components. The applicants are also evaluating the requirements to perform a visual examination of the reactor vessel internals and may propose alternative provisions.

The preservice examination methods will be limited to those components for which inservice examinations have been proposed. Since the applicants have not committed to performing any inservice volumetric or surface examinations, the initial structural integrity is dependent upon the effectiveness of nondestructive examinations performed during fabrication. Radiography (RT) will be the examination method used during fabrication to accept the full penetration butt welds in the vessels and components.

5.12.1.3 Evaluation of the Applicants' Proposed Preservice Inspection and Inservice Inspection Plan

Staff review of Section XI, Division 3, of the ASME Code, and the applicants' position in Appendix G of the PSAR, have determined that the applicants' proposed PSI and ISI program will not satisfy the requirements of the applicable principal design criteria. Division 3 was first published in the Winter 1981 Addenda and has not been referenced in 10 CFR 50.55a(b). The staff considers Division 3 as an evolutionary document whose requirements are necessary but not sufficient to meet the requirements of the applicable principal design criteria. With trial use and appropriate revisions to increase the extent of volumetric examination, Division 3 of Section XI should become the foundation of an effective ISI plan for CRBRP. Based on its conclusion that additional examinations are required to satisfy the applicable PDC, the staff has established an interim NRC staff position appropriate for the CP review.

The intent of this NRC staff position is to ensure that the following goals are accomplished:

(1) An adequate preservice volumetric examination is performed using techniques that will detect cracks in the weld heat-affected zone (HAZ) and adjacent base material to ensure the original structural integrity of the components.

(2) Adequate designed access for inspection and inspectable weld geometries are provided for future ISI.

(3) Periodic volumetric ISI of a representative sample of weld heat-affected zones and adjacent base metal should be performed in order to detect unanticipated degradation mechanisms.

(4) The specialized inspection tools necessary to perform the required examinations are available, or can be developed without technological breakthroughs, as an integral part of the plant design.

(5) To accommodate the unique operating and accessibility constraints of the CRBR, a concept of ISI coincident with the required maintenance, repair, and modification should be implemented. This concept might result in variable locations and frequency for ISI.

The applicants for CRBRP should be required to develop and demonstrate the following examination technology before an OL is issued:

(1) high-temperature ultrasonic and/or radiographic examination at the normal refueling shutdown temperature or at ambient temperature

(2) the operability and sensitivity of the integrated leakage detection system

(3) surveillance procedures for the reactor internals

(4) surveillance procedures for the IHX tubes to ensure that generic service-induced degradation of the tube bundle has not occurred

5.12.1.4 NRC Staff Position on Preservice Inspection and Inservice Inspection

The NRC should require that the following actions be implemented at CRBRP as a condition of the CP:

Fabrication/Preservice Inspection

(1) An ultrasonic examination should be performed for the full length of all circumferential butt welds in vessels and piping in the PHTS and RRHES with diameters 8 in. and greater. An ultrasonic examination should be performed for the full length of the longitudinal butt welds in reactor vessel, EVST, and piping enclosed by guard vessels and 1 ft on each side of the intersection with a circumferential weld in piping that is not enclosed by guard vessels. An ultrasonic examination should be performed to the extent practical on the accessible weld in the reactor vessel closure head and EVST cover.

(2) Since the leakage detection systems are an integral part of the ISI program, a field test of the installed systems should be performed to verify the assumptions of operability, sensitivity, and redundancy.

Design Access and Inspection Tools

(1) Access and remote equipment suitable for the ultrasonic examination or approved alternative examinations of the two bi-metallic circumferential shell welds in the reactor vessel and the EVST should be designed.

(2) The development of the state of the art mechanized UT system for piping, 8 in. and greater in diameter, should be completed before the issuance of the OL. The system should be capable of examining the piping welds that are not enclosed in guard pipe and outside of the reactor cavity in both the PHTS and the RRHES.

(3) An inspection procedure should be developed to perform surveillance of the reactor vessel internals.

Inservice Inspection

(1) A volumetric examination should be performed on the two bimetallic welds in the reactor vessel. The capability to perform a similar examination should be provided for the bimetallic welds in the ex-vessel storage tank.

(2) A volumetric examination should be performed on selected PHTS and RRHES welds in piping 8 in. and greater in diameter which are made accessible during required maintenance, repair and modification. Examination locations that should be included are the PHTS crossover pipe between the pump and IHX. If defects are detected, additional welds shall be examined at similar locations.

(3) A representative sample of the IHX tubes should be examined volumetrically by surface or approved alternative methods when maintenance or required leakage verification provides direct access to the tubes.

(4) Surveillance of the reactor internals should be performed during the first 10-year inspection interval.

(5) The leakage detection system should be tested according to Technical Specification requirements.

5.12.1.5 Basis for the NRC Staff Position

Fabrication/Preservice Inspection

All components and welds should be examined during construction to provide reasonable assurance that the plant is safe to operate initially and baseline information is obtained for future ISIs. Since conventional radiography (this does not include multi-angle or broad spectrum X-ray) will not ensure that planar flaws, such as cracks, are not present in the welds, the following additional surveillance requirements must be met to maintain the original structural integrity of the component.

(1) Adequate volumetric examination of the PHTS and RRHES welds is required to provide assurance that no significant flaws will escape detection

considering the anticipated high radiation levels, high temperature during the refueling outage, and accessibility constraints. A preservice ultrasonic examination of the piping system welds is within the current state of the art of NDE practice.

(2) Characterization of the through-wall dimension and length of retained acceptable flaws in the PHTS and RRHES should be documented.

(3) Examination methods and techniques must be demonstrated during baseline examinations to support inservice repair, modification, and replacement of components and piping.

(4) Accessibility and identification of nonrelevant ultrasonic indications during the baseline examination are important in the event that periodic ISI is necessary on the specific welds.

Inservice Inspection

(1) Periodic volumetric examination of a representative sample of vessel and piping welds in both the PHTS and RRHES is essential to provide assurance of continuing structural integrity and to reliably identify long-term degradation mechanisms that may result from plant operation. The objective of the volumetric examination is to provide early detection of degradation before leakage occurs. Important locations that should be selected for examination include the two bimetallic welds in the shell of the reactor vessel and EVST and the PHTS crossover pipe between the pump and IHX. A concept of ISI coincident with required maintenance, repair, and modification will determine whether generic degradation is occurring without resulting in excessive radiation exposure to personnel and extended plant outage time.

(2) The benefits of the guard vessels and guard pipe, the ductile material properties, the leakage detection systems, the relatively low stored energy to propagate cracks, and the required augmented fabrication/preservice examination were considered in the concept that designed access, and periodic inservice sampling was not required of welds enclosed in guard vessels and guard pipe.

5.12.2 Conclusions

The conduct of fabrication/preservice examinations, provisions for design access and inspectable weld geometries, development of the specialized examination methods and tools to perform the required examinations, and performance of periodic volumetric inservice examinations in accordance with the requirements of Section XI, Division 3, of the ASME Code and the NRC staff position will provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the requirements of the ASME Code and NRC staff position constitutes an acceptable basis for the CP review for satisfying the inspection requirements of PDC 30, 36 and 37. The staff will continue its evaluation, after the issuance of the CP, until the applicants have established effective programs

to implement the requirements of the ASME Code and the NRC staff position and provides acceptable confirmatory information.

5.12.3 Inservice Inspection of the Intermediate Coolant Boundary, Engineered Safety Features, Steam Generator Tubes, and ASME Code Class 2 and 3 Steam/Water Systems

5.12.3.1 Examination Requirements

PDC 33, "Inspection and Surveillance of Intermediate Coolant Boundary," requires, in part, that the components shall be designed to permit periodic inspection of areas and features important to safety and to assess their structural and leaktight integrity. Means shall be provided for detecting intermediate coolant leakage. PDC 39, "Inspection of Additional Cooling Systems," requires that the components shall be designed to permit appropriate periodic inspections of important components, such as heat exchangers and piping, and to ensure the integrity and capability of the systems. PDC 40, "Testing of Additional Cooling Systems," requires, in part, that the systems shall be designed to permit appropriate periodic pressure and functional testing to ensure the structural and leaktight integrity of their components. The intent of this criterion is to require the design of all cooling systems, not just cooling water, which provide cooling to components and systems important to safety, to allow for periodic testing to ensure that the systems still perform as designed.

5.12.3.2 Applicants' Proposed Preservice Inspection and Inservice Inspection Plan

The staff has reviewed the available information the the PSAR related to preservice and inservice inspections. The applicants' docketed position on the subject is contained in Appendix G of the PSAR, Amendment 68, dated May 1982. To meet the applicable principal design criteria, the applicants state that the CRBRP PSI and ISI plan will be based on Section XI of the ASME Code, Division 3, for systems containing liquid metal and cover gas. Division 1 of Section XI will be used for the steam/water systems. However, the applicants take exception to certain provisions of Section XI.

The applicants propose to use leakage detection systems and visual examinations as the principal method for inservice inspection. The bases for the general approach for the IHTS and engineered safety features are the ductile characteristics of most of the materials of construction, the relatively low operating pressures, and design features such as inerted cells. Section XI has requirements for periodic visual examination and for the installation of leakage-detection devices at selected locations. In addition, the first publication of Section XI, Division 3, contains the following requirements:

(A) Volumetric examination of 33% of the dissimilar metal welds during each 10-year inspection interval

(B) Volumetric examination of 3% of the heating surface in each steam generator during the initial 10-year inspection interval.

Several topics identified in Section XI as "in course of preparation." These topics include most of the acceptance standards for nondestructive examination and the leakage detection requirements for sodium-water leakage in the steam generators. Based on the requirements of Division 1 of Section XI and the ASME Code classification of the CRBRP components, most the steam/water systems will be examined only by a hydrostatic test during each 10-year interval. Although the applicants propose to reduce the required frequency of inservice examination on several components, they have generally committed to follow the extent and method of examinations required by Section XI of the Code.

The preservice examination methods will be limited to those components for which inservice examinations have been proposed. Since the applicants have only committed to performing an inservice volumetric examination of the dissimilar metal pipe welds adjacent to the steam generators, the initial structural integrity is dependent upon the effectiveness of nondestructive examinations performed during fabrication. Radiography will be the examination method used during fabrication to accept the full penetration butt welds in the vessels and components.

5.12.3.3 Evaluation of the Applicants' Proposed Preservice Inspection and Inservice Inspection Plan

Staff review of Section XI, Division 3, of the ASME Code and the applicants' position in Appendix G of the PSAR, have determined that the applicants' proposed PSI and ISI program will not satisfy the requirements of the applicable principal design criteria for the liquid metal and cover gas systems. Staff review has also determined that the use of Division 1 of Section XI for the steam/water components based on safety classification of the components is acceptable. Division 3 of Section XI was first published in the Winter 1981 Addenda and has not been referenced in 10 CFR 50.55a(b). The staff considers Division 3 as an evolutionary document whose requirements are necessary but not sufficient to meet the requirements of the applicable principal design criteria. With trial use and appropriate revisions to increase the extent of volumetric examination, Division 3 of Section XI should become the foundation of an effective ISI plan for CRBRP. On the basis of its conclusion that additional examinations are required to satisfy the applicable PDC, the staff has established an interim NRC staff position appropriate for the CP review.

The intent of the NRC staff position is to ensure that the following goals are accomplished.

(1) An adequate preservice volumetric examination is performed using techniques that will detect cracks in the weld heat-affected zone and adjacent based material to ensure the original structural integrity of the components.

(2) Adequate designed access for inspection and inspectable weld geometries are provided for future ISI.

(3) Periodic volumetric ISI of a representative sample of weld heat-affected zones and adjacent base metal should be performed in order to detect unanticipated degradation mechanisms.

(4) The specialized inspection tools necessary to perform the required examinations are available, or can be developed without technological breakthroughs, as an integral part of the plant design.

(5) To accommodate the unique operating and accessibility constraints of the CRBR, a concept of ISI coincident with the required maintenance, repair, and modification should be implemented. This concept might result in variable locations and frequency for ISI.

The applicants for CRBR should be required to develop and demonstrate the following examination technology before issuance of an OL:

(1) high temperature UT and/or RT at the normal refueling shutdown temperature or at ambient temperature

(2) the operability and sensitivity of the integrated leakage detection system

(3) steam generator tube examination methods (the influence of residual sodium on the interpretation of ultrasonic testing results must be defined.)

5.12.3.4 NRC Staff Position on Preservice Inspection and Inservice Inspection

A "single-loop concept" for the examination sample should be adopted for the systems with a pressure boundary containing liquid metal. This would consist of planning to examine the components and piping that are equivalent to one loop of redundant loops performing the same function once during the life of the plant. The NRC should require that the following actions be implemented at CRBR as a condition of the CP:

Fabrication/Preservice Inspection

(1) An ultrasonic examination should be performed for the full length of all circumferential butt welds in vessels and in piping equivalent to a single loop in the IHTS with diameters 8 in. and greater. An ultrasonic examination should be performed on the longitudinal welds on vessels and piping for a distance of 1 ft on each side of the intersection with a circumferential weld.

(2) A field examination of the steam generator equipment and techniques that will be used during ISI should be performed on all the tubes of all superheaters and evaporators.

This examination should be performed within 1 year of initial sodium fill and should cover the entire length of all tubes. An additional examination of 10% of the tubes should be performed after the initial sodium fill unless a correlation between the dry condition and the residual-sodium wetted condition has been established.

(3) Since the leakage detection systems are an integral part of the ISI program, a field test of the installed systems should be performed to verify the assumptions of operability, sensitivity, and redundancy.

(4) An ultrasonic examination of the steam turbine discs should be performed consistent with light-water-reactor requirements.

Design Access and Inspection Tools

(1) The development of the state-of-the-art mechanized ultrasonic system for coolant boundary piping inspection should be completed before the issuance of an OL. The system should be capable of examining the piping welds in the IHTS and ESF.

(2) Access should be provided to perform inservice volumetric and/or surface examination of all tubes in the superheaters and evaporators.

Inservice Inspection

(1) A volumetric examination should be performed on a sample of the welds in the piping of the IHTS and ESFs. All bi-metallic welds in the IHTS should be examined during each 10-year inspection interval. The majority of welds should be examined when made accessible by required maintenance, repair, and modification. However, selected welds should be examined on a periodic basis even if access is not provided during normal operations.

(2) Volumetric and/or surface examination of an initial sample greater than 3% of the steam generator tubes should be performed during the first 10-year inspection interval. A plan for logically progressive expansion of subsequent sample size should be developed that depends upon the results of the initial sample inspection.

(3) Testing of the leakage detection system based on Technical Specification requirements should be performed.

(4) The turbine discs should be examined at an extent and frequency consistent with similar requirements for light-water reactors.

(5) IHTS sodium boundary tanks and connecting piping that are required to function only under accident conditions should be exempt from periodic inservice inspection. Volumetric examination of the entire system is required if the rupture discs function as a result of a sodium-water reaction.

5.12.3.5 Basis for the NRC Staff Position

Fabrication/Preservice Inspection

Selected components and welds should be examined during construction to provide reasonable assurance that the plant is safe to operate initially and baseline information is obtained for future ISI. Since conventional radiography (this does not include multi-angle or broad spectrum X-ray) will not ensure that planar flaws, such as cracks, are not present in the welds, the following surveillance requirements must be met.

(1) Adequate volumetric examination of selected IHTS and ESF welds is required to provide assurance that no significant flaws will escape detection considering the anticipated high temperature during the refueling outage and accessibility constraints. A preservice ultrasonic examination of the piping system welds is within the current state of the art of NDE practice.

(2) Characterization of the through-wall dimension and length of retained acceptable flaws in the IHTS and ESF should be documented.

(3) Examination methods and techniques must be demonstrated during baseline examinations to support inservice repair, modification, and replacement of components and piping.

(4) Accessibility and identification of nonrelevant ultrasonic indications during the baseline examination are important in the event that periodic ISI is necessary on the specific welds.

Inservice Inspection

(1) Periodic volumetric examination of a representative sample of vessel and piping welds in both the IHTS and ESF is essential to provide assurance of continuing structural integrity and to reliably identify long-term degradation mechanism that may result from plant operation. The objective of the volumetric examination is to provide early detection of degradation before leakage occurs.

(2) Until long-term testing or field experience demonstrates that the heat exchanger tubes will not encounter degradation during the plant life, a conservative ISI program should be required. No technical basis has been provided to the staff to conclude that degradation of tubing in the steam generators will be significantly different from problems experienced in LWR plants.

(3) No technical basis has been provided to the staff to conclude that potential degradation of the turbine discs will be significantly different from problems experienced in LWR and fossil-fired plants.

5.12.4 Conclusions

The conduct of fabrication/preservice examinations, provisions for design access and inspectable weld geometries, development of the specialized examination methods and tools to perform the required examinations and performance of periodic volumetric inservice examinations in accordance with the requirements of Section XI, Divisions 1 and 3, of the ASME Code and the NRC staff position will provide reasonable assurance that evidence of structural degradiation or loss of leaktight integrity during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the requirements of the ASME Code and NRC staff position constitutes an acceptable basis for the CP, review for satisfying the inspection requirements of PDC 33, 39, and 40.

5.13 Plant Startup and Shutdown

5.13.1 Plant Startup

Plant startup basically encompasses the operations that take the reactor plant from either hot standby or refueling conditions to the operating conditions associated with 40% reactor thermal power. The initial plant startup also includes reactor heat transport system (HTS) preheat and sodium fill as well as steam generator system (SGS) fill and heatup. During initial startup, the HTS preheat operation consists of heating the HTS, along with the reactor systems, from a dry, argon-filled, cold condition to 400°F at a nominal rate of 3F° per hour. The SGS will be heated to 400°F by water/steam utilizing the auxiliary steam boiler in addition to recirculation pump heat input. Sodium fill and all subsequent refueling operations will be performed at 400°F in order to preclude the possibility of sodium freezing (208°F).

After completion of all preoperational tests, the reactor systems, HTS and SGS will be heated from refueling to hot standby conditions (600°F) at an average rate not exceeding 50F° per hour. The final phase of startup will consist of taking the reactor plant from hot standby conditions to 40% reactor thermal power. The reactor power will then be increased and the HTS primary and intermediate hot-leg-to-cold leg ΔTs will be formed by releasing steam from the superheater at a rate which will limit the primary hot-leg temperature average heatup rate to 150F° per hour. As steam flow increases, the turbine will be warmed up and loaded while dumping excess steam. At about 15% reactor power, the generator will be placed in parallel with the grid, the initial electrical load will be picked up, and bypass steam will then be secured. At 40% of the reactor power, plant control will be shifted from manual to automatic load follow control.

5.13.2 Plant Shutdown

At the 40% thermal power position, the plant control will be shifted from automatic load follow to manual control. At this point, steam flow to the condenser via the bypass system will be established in order to maintain proper steam pressure. Reactor power will then be decreased at a rate which limits the primary hot leg cooldown to 150F° per hour, while the primary and intermediate sodium flows will be maintained at flows corresponding to 40% power. The turbine will be tripped at approximately 10% power and the generator output breaker opened. The reactor power will then be further decreased to the decay heat level at a rate which limits the primary hot-leg cooldown rate to 15F° per hour. During this phase of the shutdown, the protected air-cooled condenser (PACC) will be placed into operation. When the PACC is removing all the decay heat, the steam bypass flow is secured. If the plant is to be maintained at hot standby conditions, the system temperatures will be controlled by operation of the PACC and electrical resistance heaters. If refueling conditions are desired, the cooldown will be accomplished by heat removal by the PACC so as not to exceed 50F° per hour.

5.13.3 Evaluation

Section 5.7.1 of the PSAR describes the startup and shutdown procedures as suggested in the LMFBR Edition of "Standard Format and Content of Safety

Analysis Reports for Nuclear Power Plants" (USAEC, Feb. 1974). The staff and its consultant, Brookhaven National Laboratory, have reviewed the procedures outlined in this section and agree that they will not result in violation of the safety criteria. However, a more detailed review will have to be completed during the operating license stage of review. The subjects of plant startup and shutdown are covered in greater detail in Section 14 of this SER.

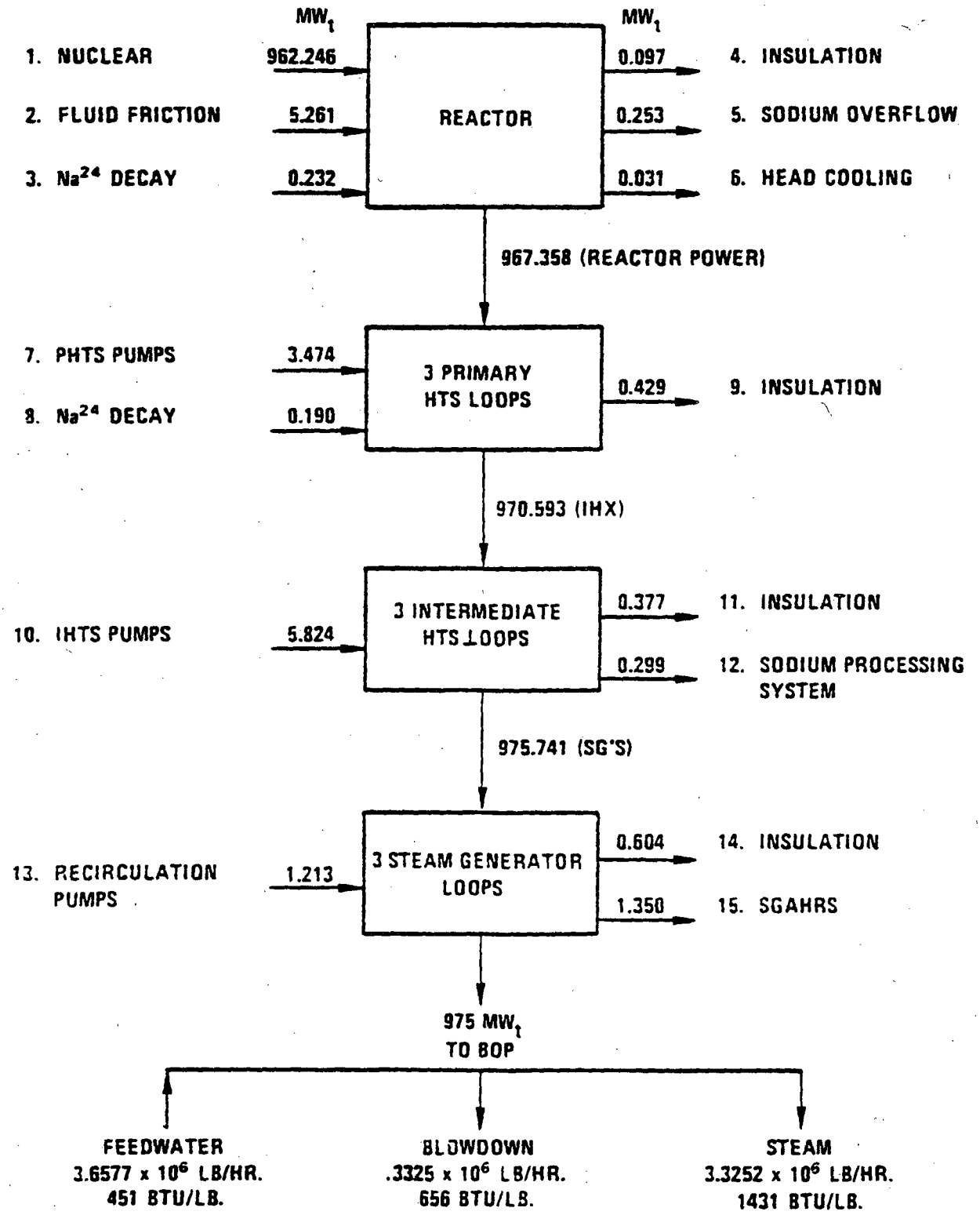


Figure 5.1 Preliminary nuclear steam supply system (NSSS) heat balance 975 MWt at NSSS/balance-of-plant interface

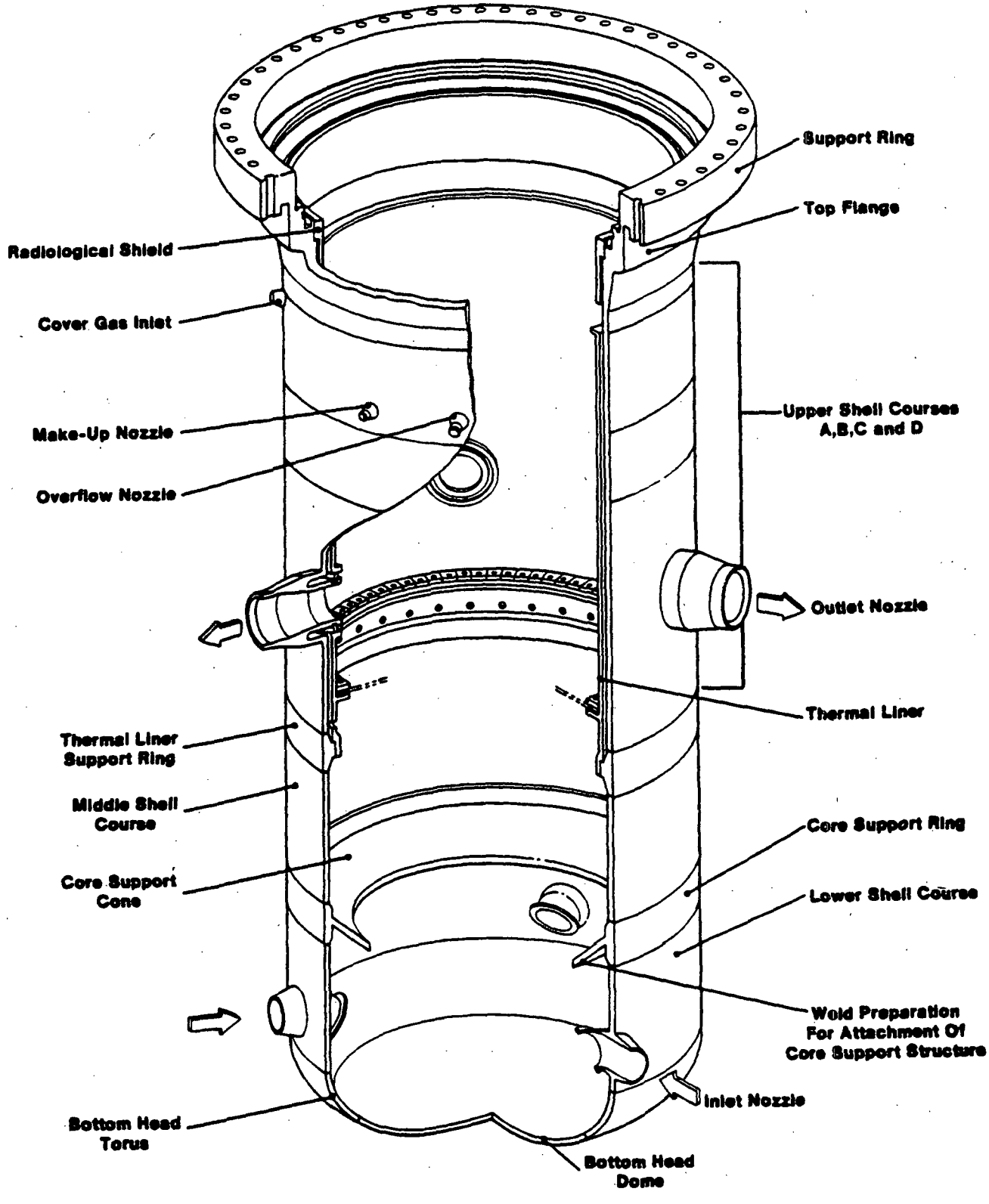


Figure 5.2 Reactor vessel cutaway

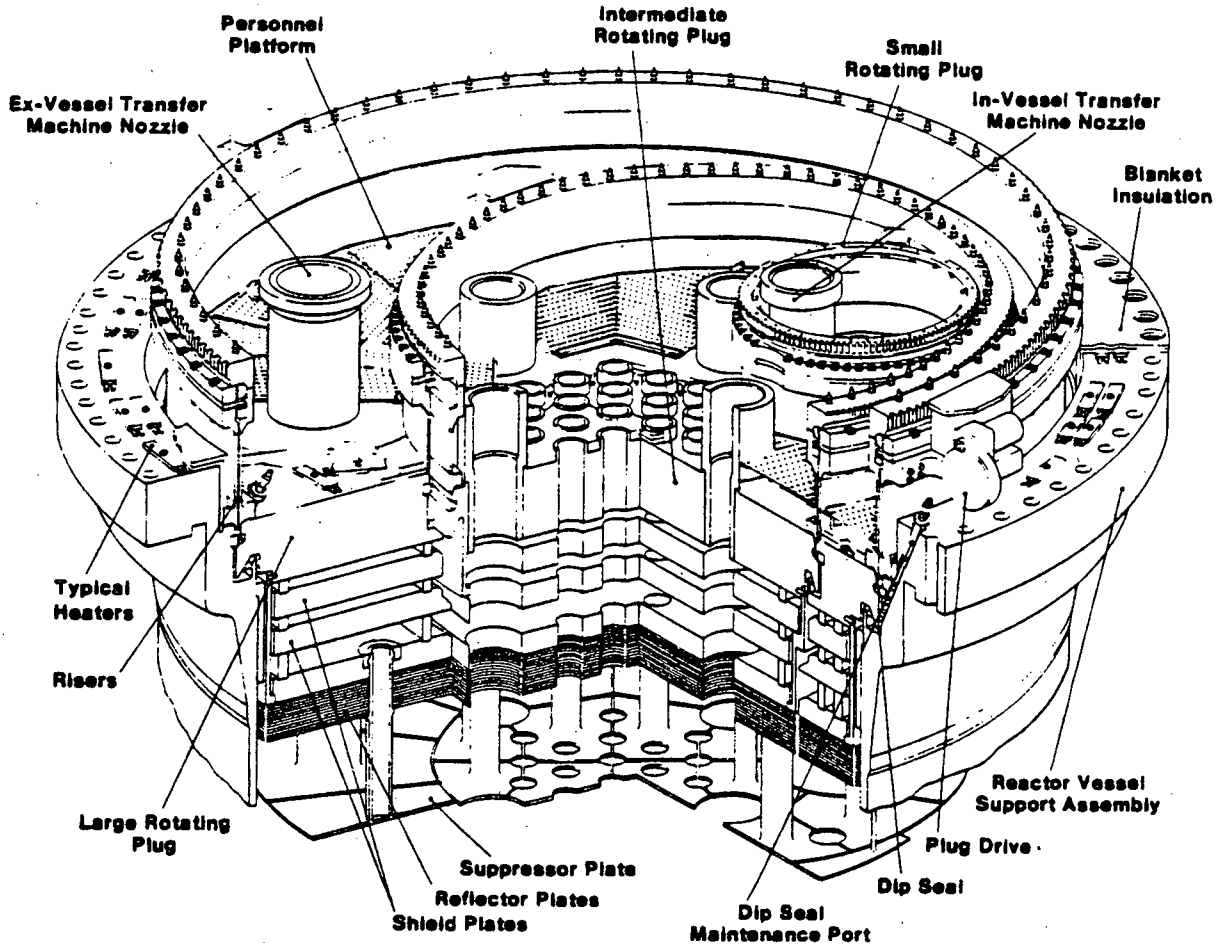


Figure 5.3 Reactor closure heat assembly

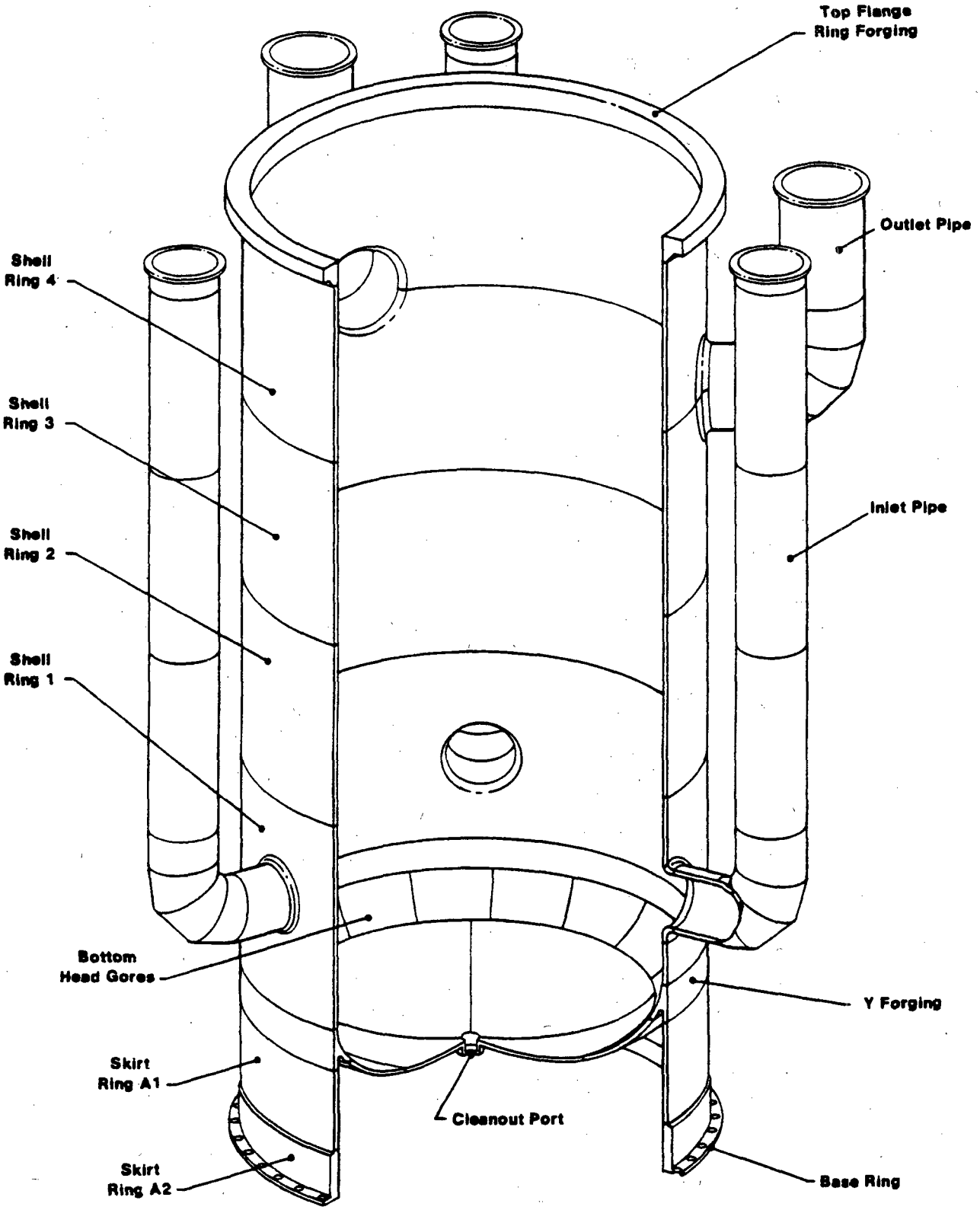


Figure 5.4 Reactor guard vessel cutaway

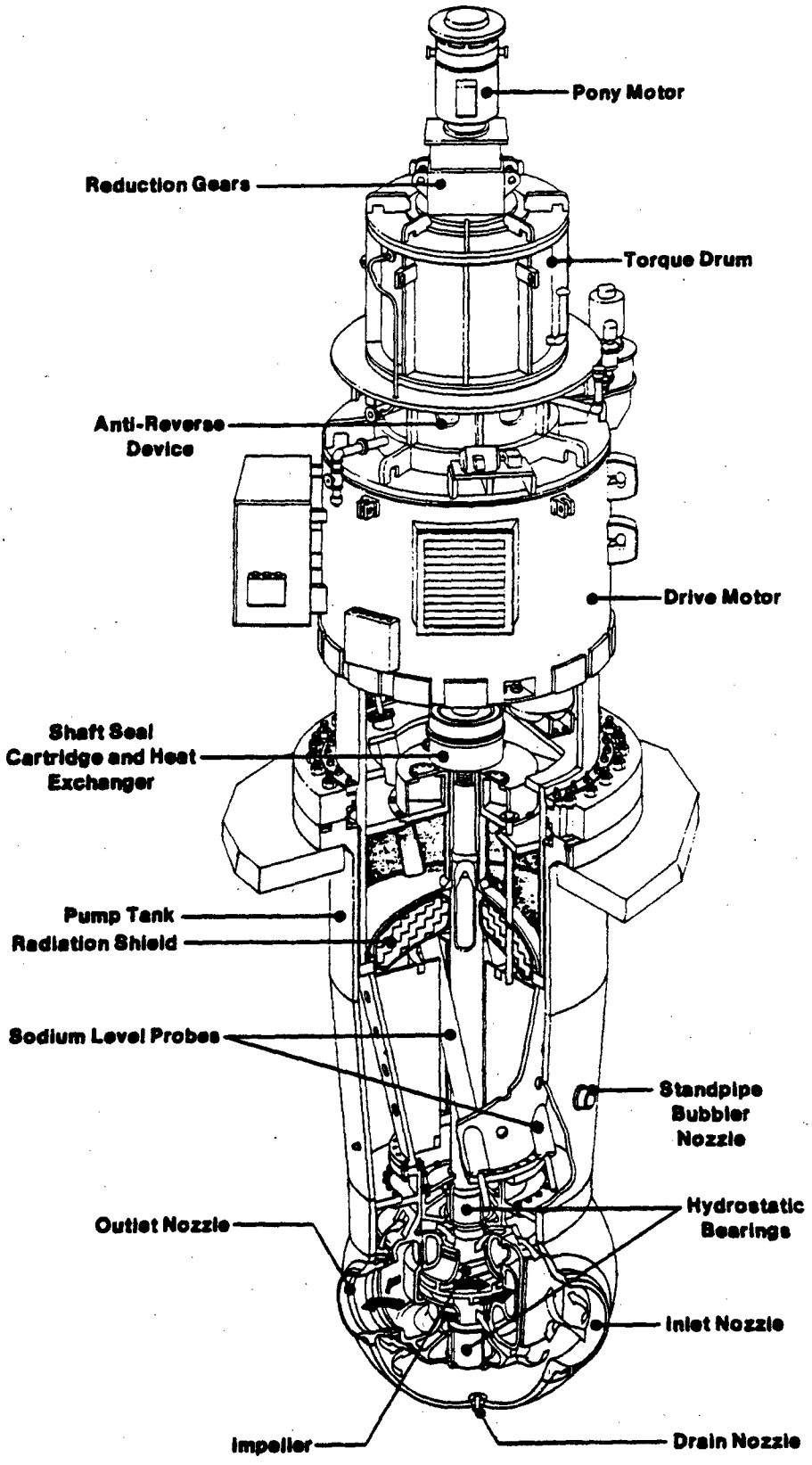


Figure 5.5 Primary system sodium pump and drive motor

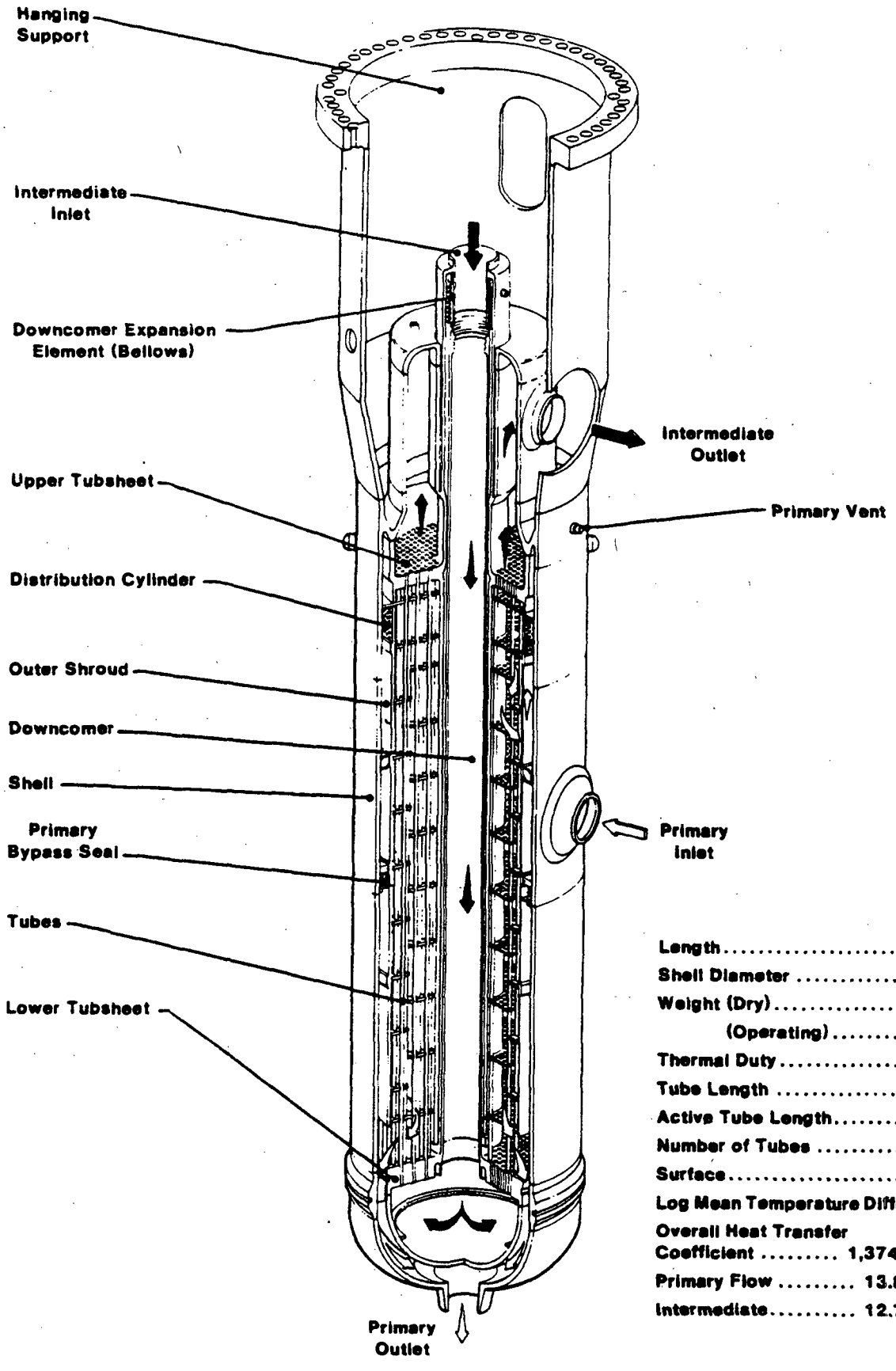


Figure 5.6 Intermediate heat exchanger

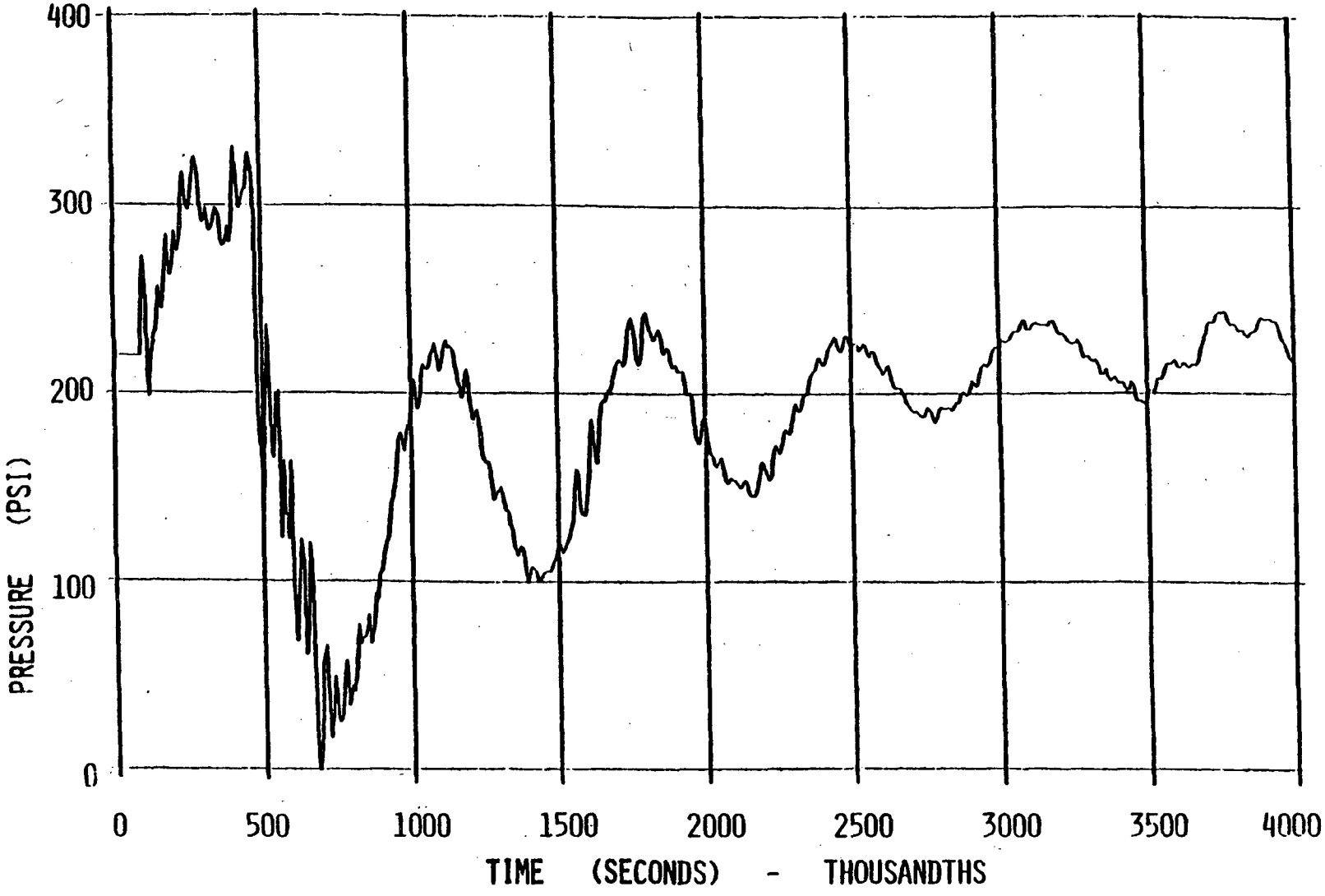


Figure 5.7 Predicted pressure history in IHX for sodium-water reaction design-basis leak in evaporator

Thermal Center
Elevation Differences

CORE TO IHX - 29 ft.
IHX TO STEAM GENERATOR(S/G) - 21 ft.
S/G TO STEAM DRUM(SD) - 13 ft.
SD TO PACC - 39 ft.

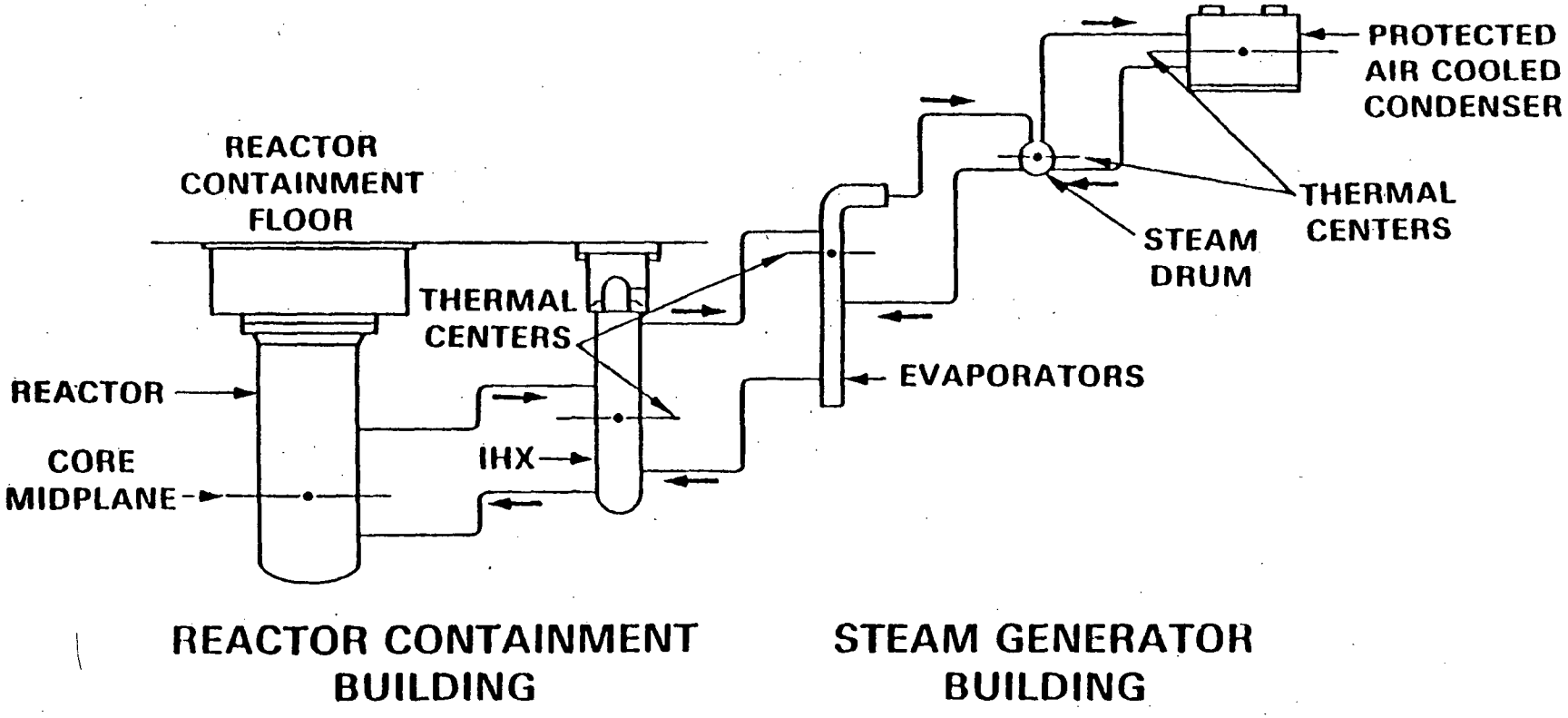


Figure 5.8 Elevation differences in major components

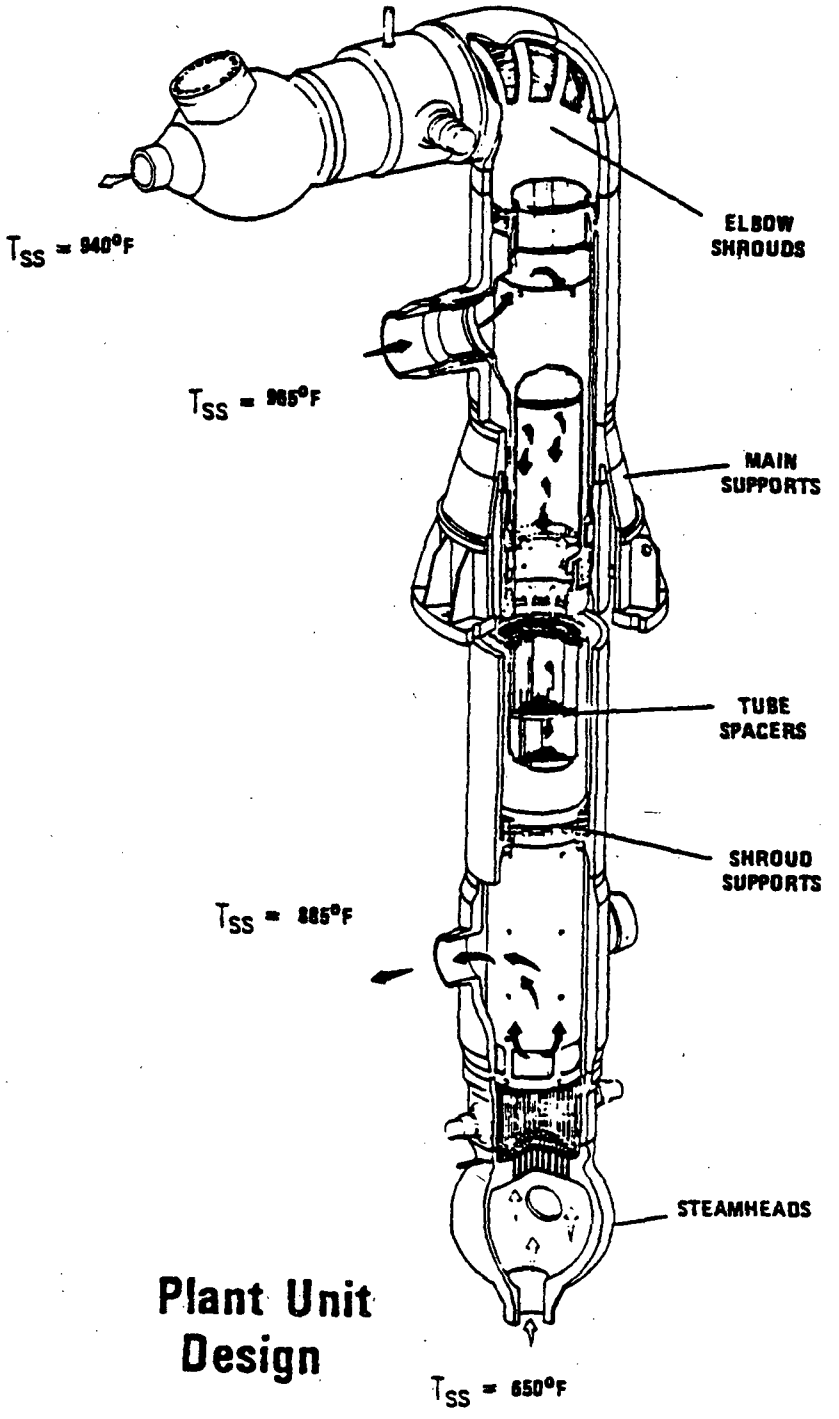
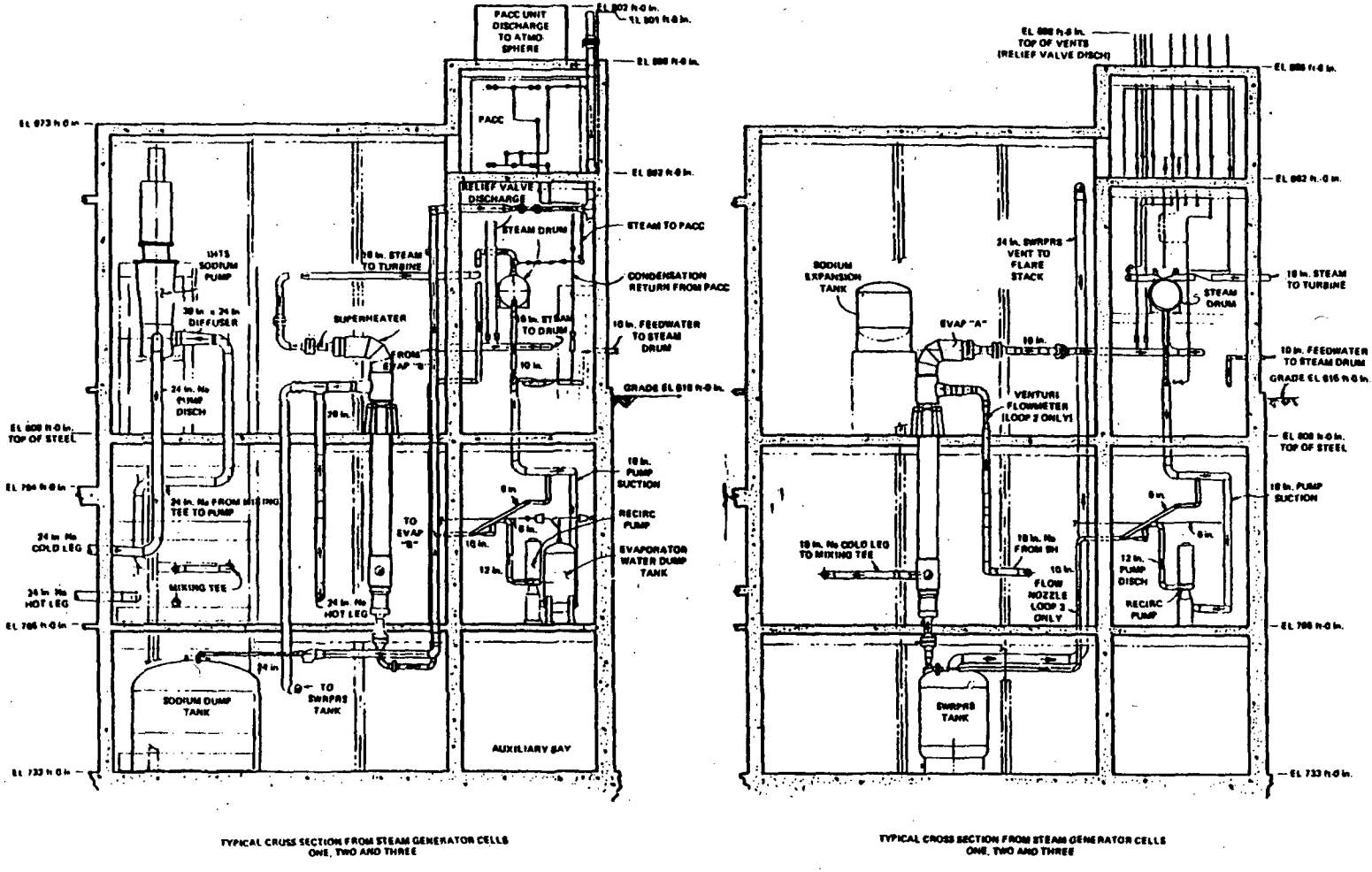


Figure 5.9 CRBRP steam generator module



TYPICAL CRUSS SECTION FROM STEAM GENERATOR CELLS ONE, TWO AND THREE

TYPICAL CROSS SECTION FROM STEAM GENERATOR CELLS ONE, TWO AND THREE

Figure 5.10 Cross-section from steam generator cells one, two, and three

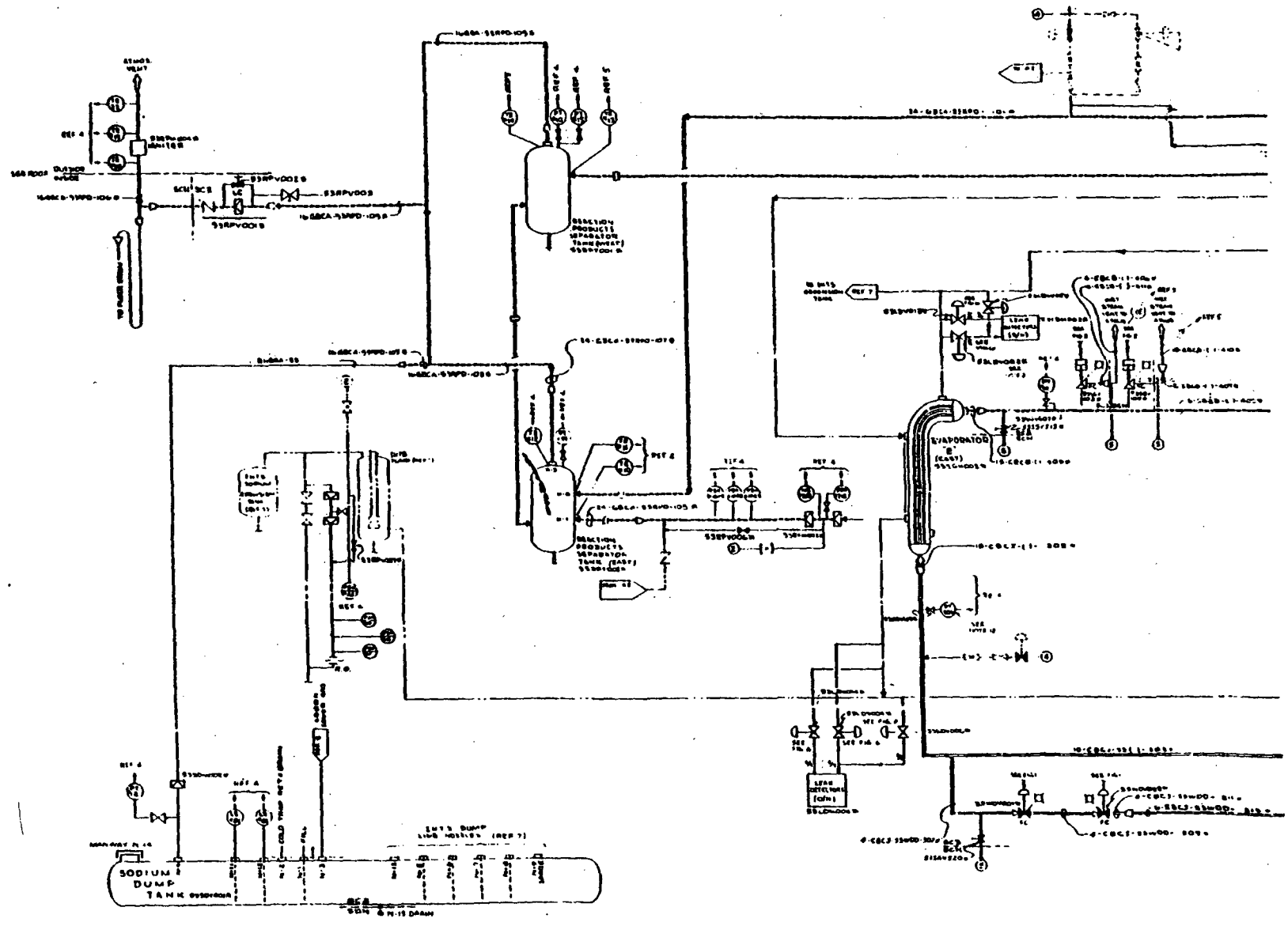


Figure 5.11 Sodium-water reaction pressure relief system

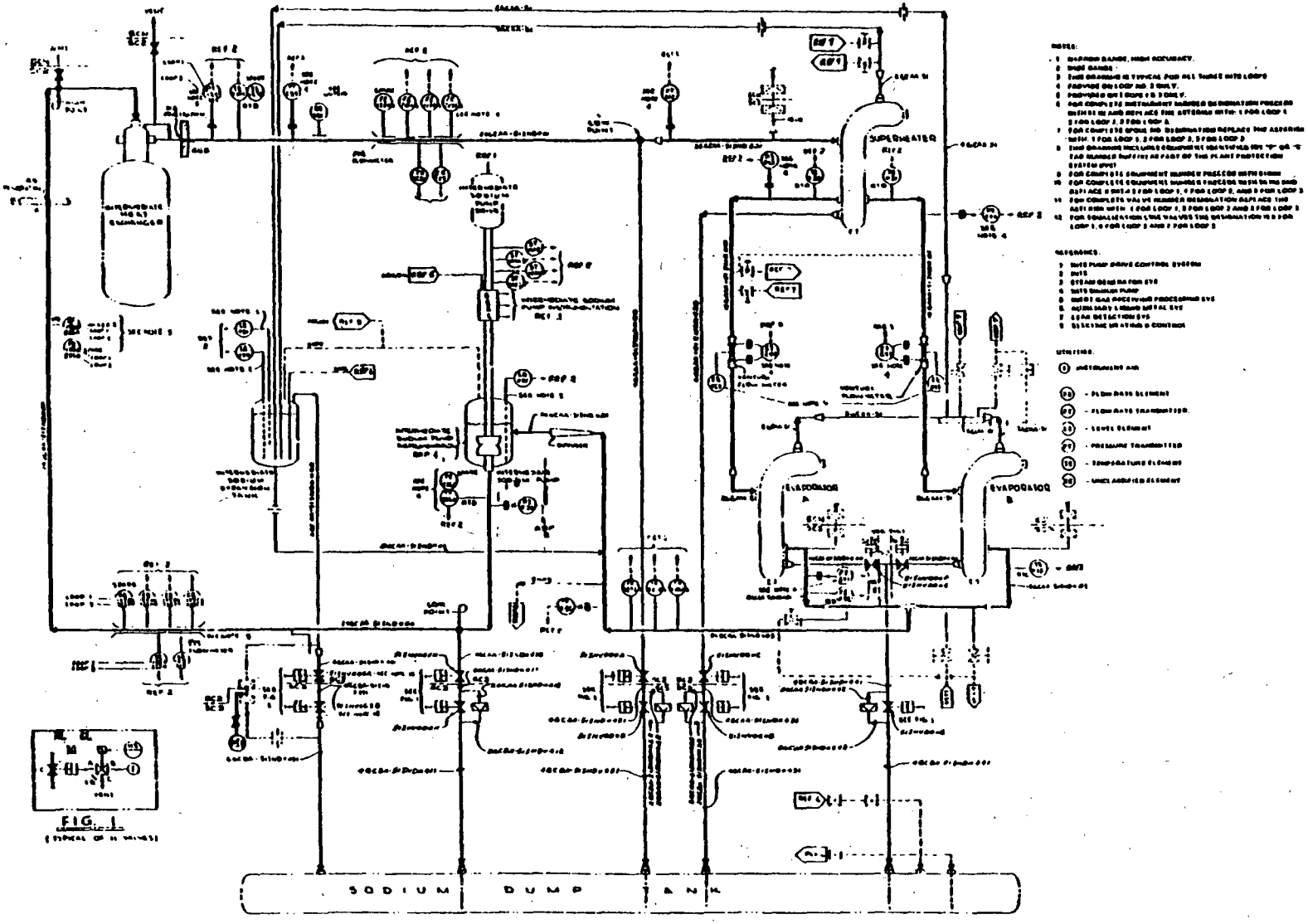


Figure 5.12 Sodium dump subsystem

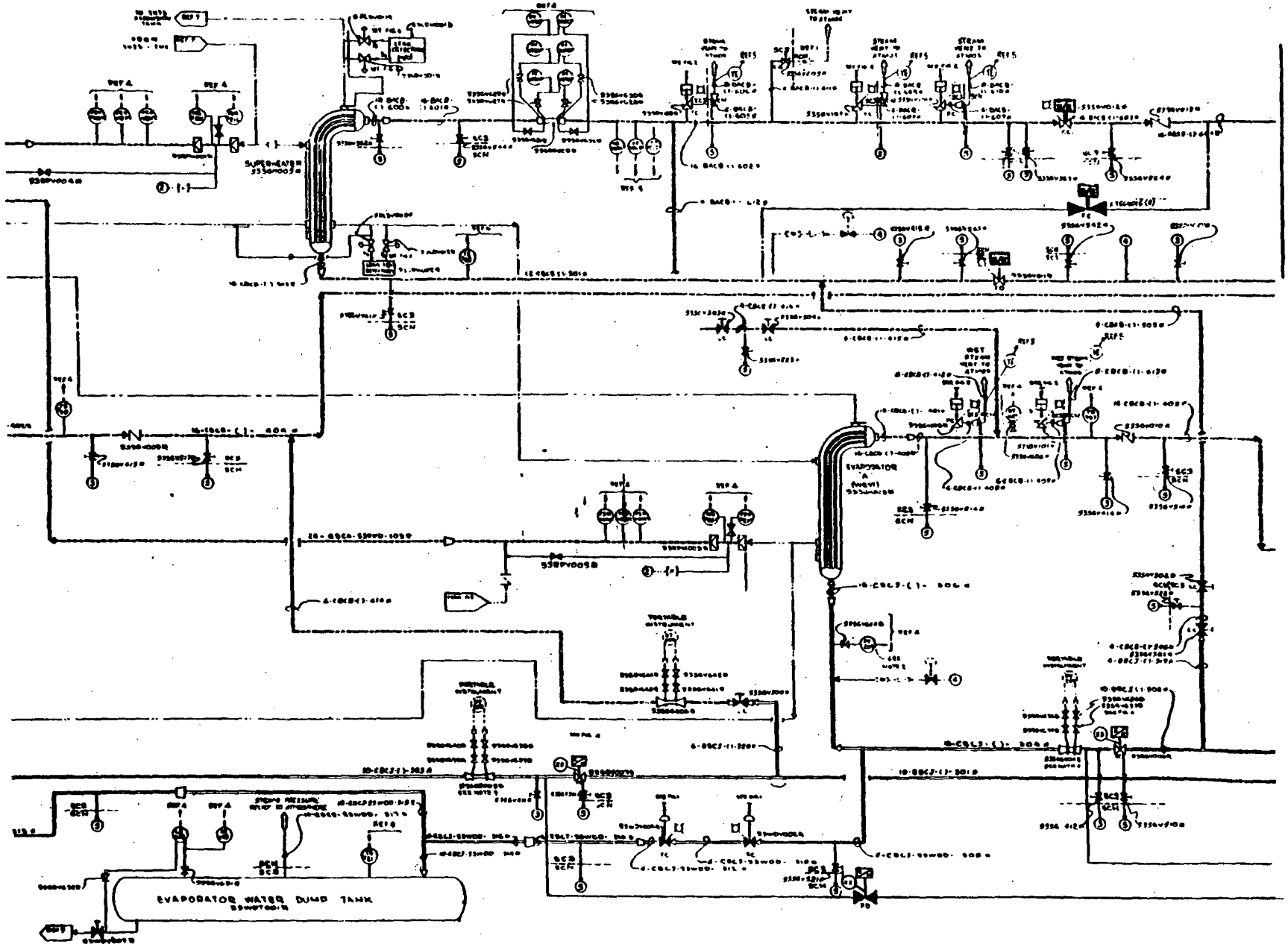


Figure 5.13 Water dump subsystem

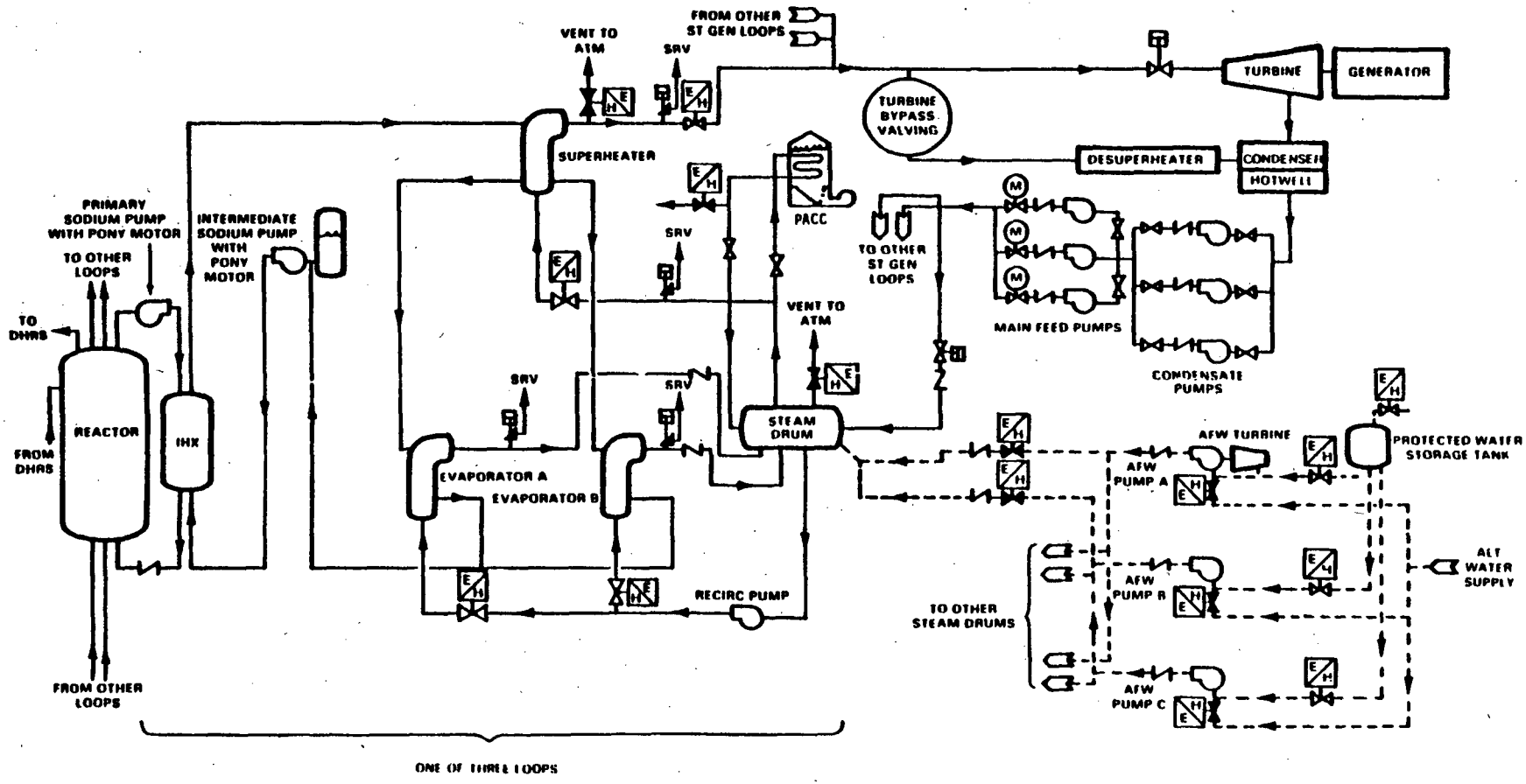


Figure 5.14 Shutdown heat removal system schematic (without direct heat removal service)

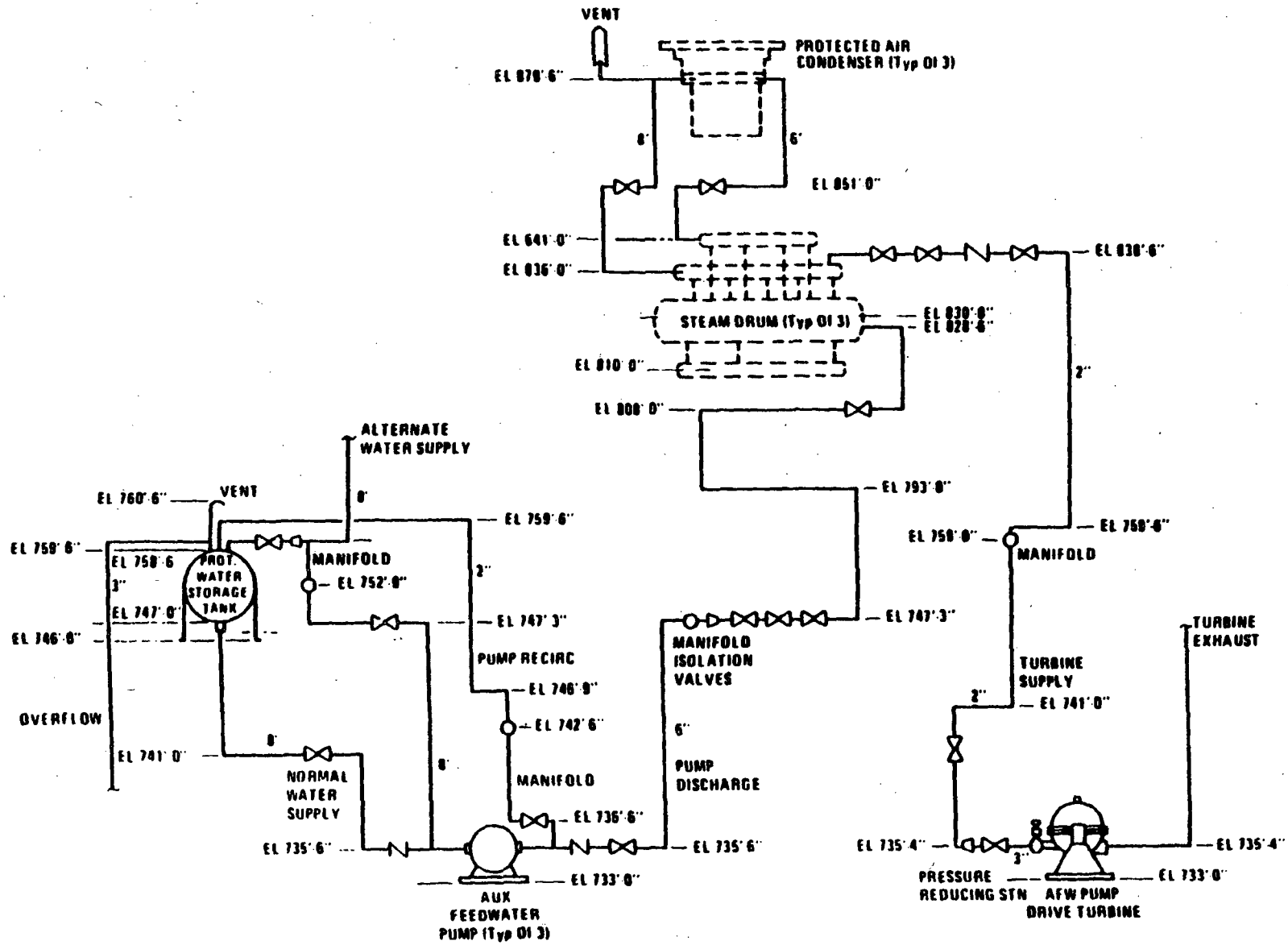


Figure 5.15 Steam generator auxiliary heat removal hydraulic profile

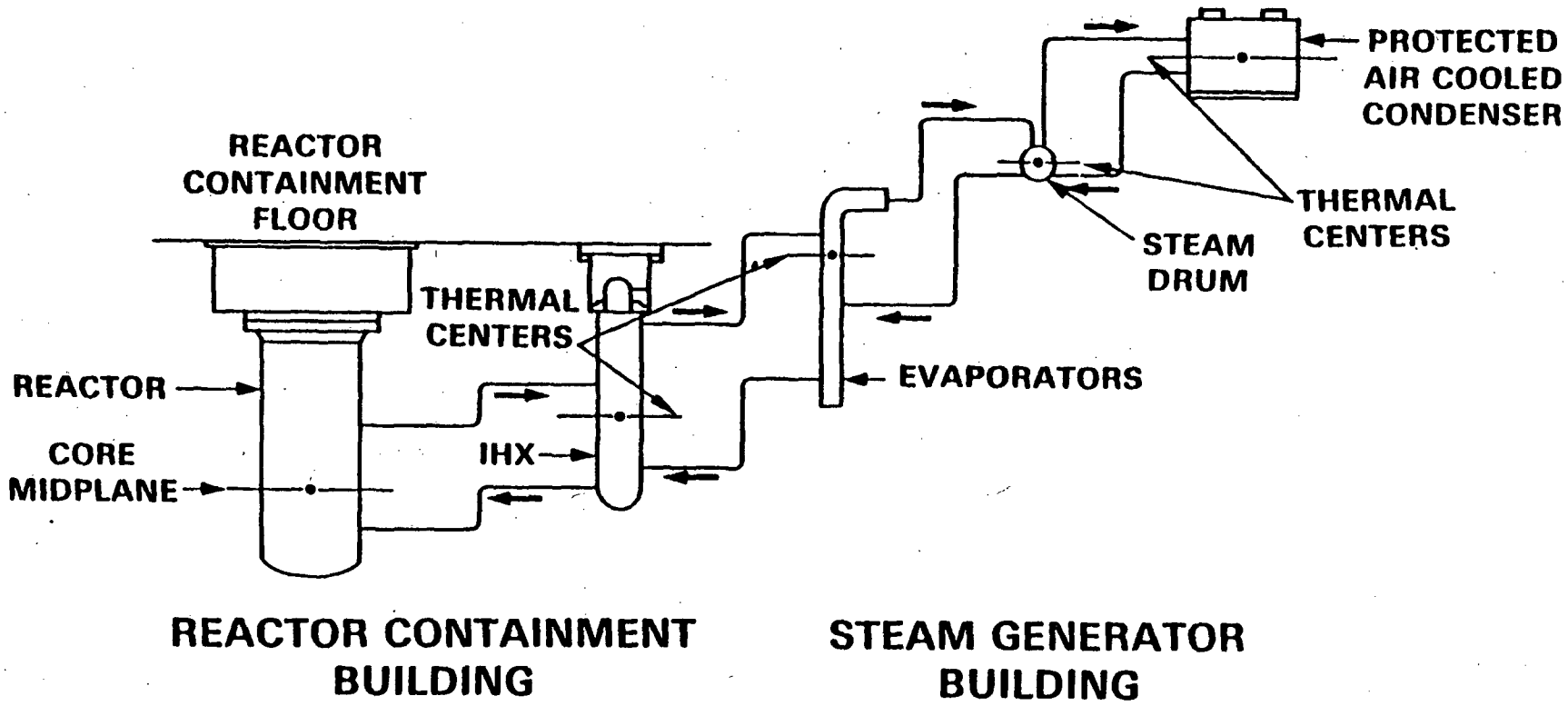


Figure 5.16 Heat transport system hydraulic profile

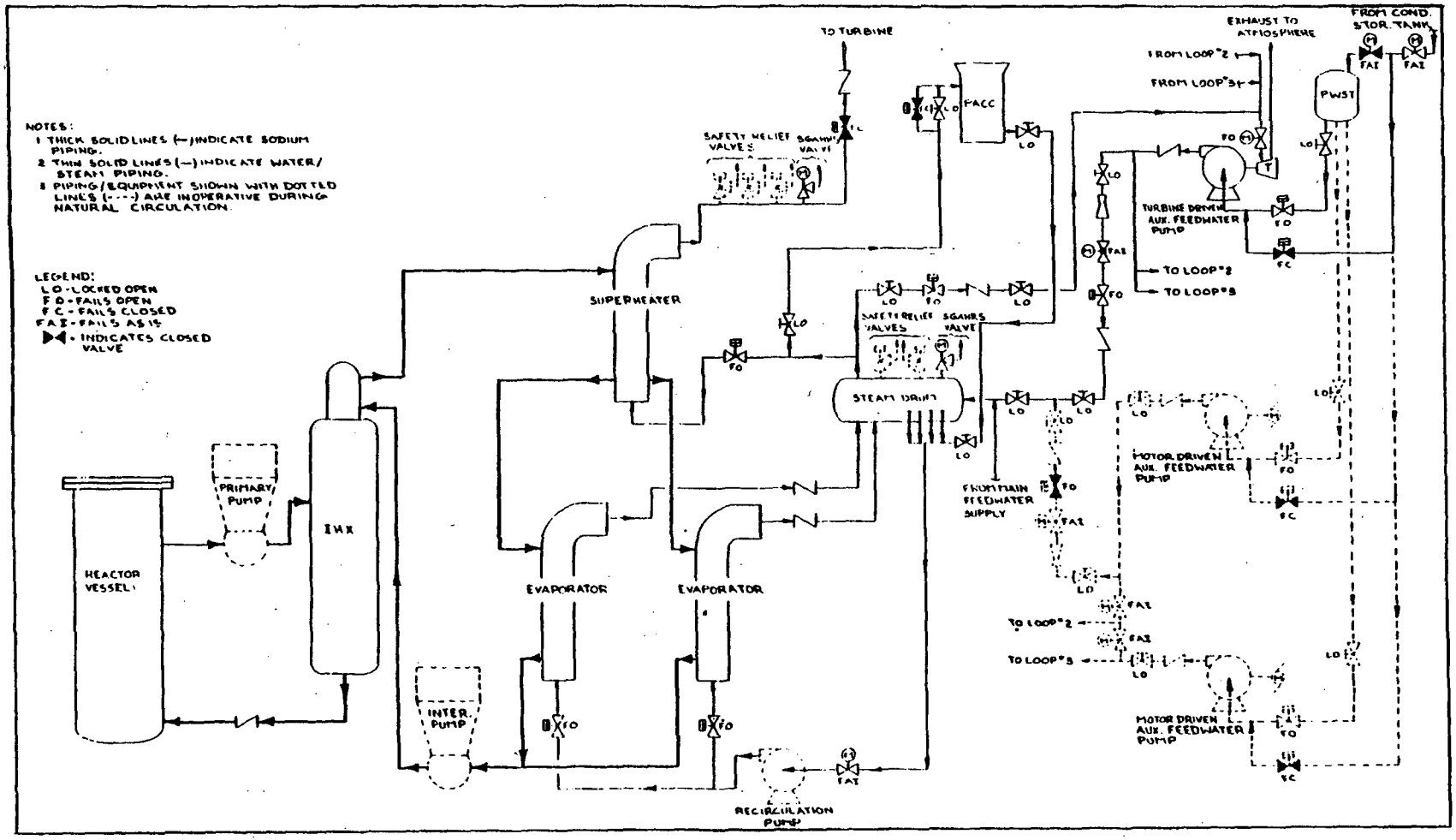


Figure 5.17 Natural circulation decay heat removal train

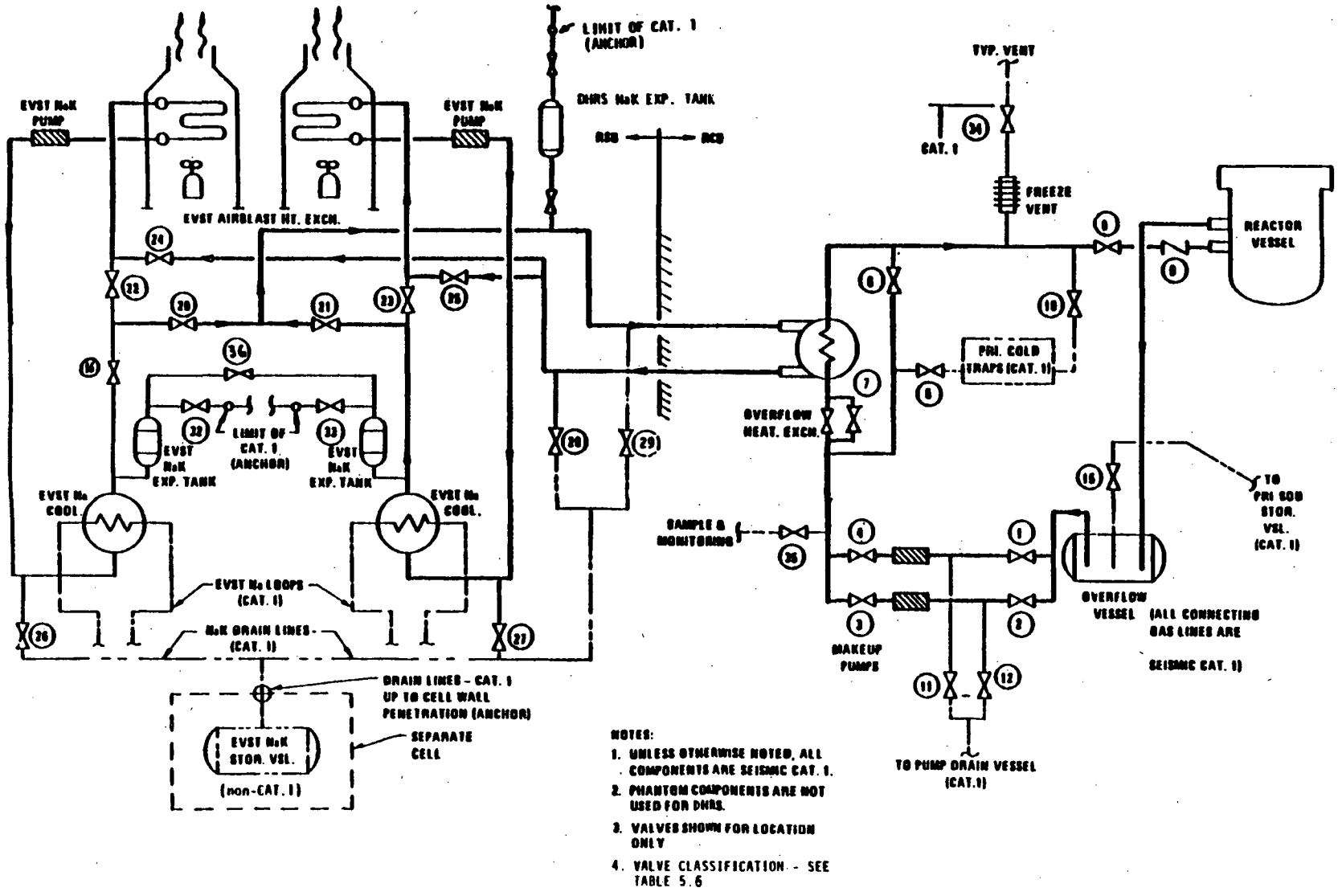


Figure 5.18 Direct heat removal service system and components

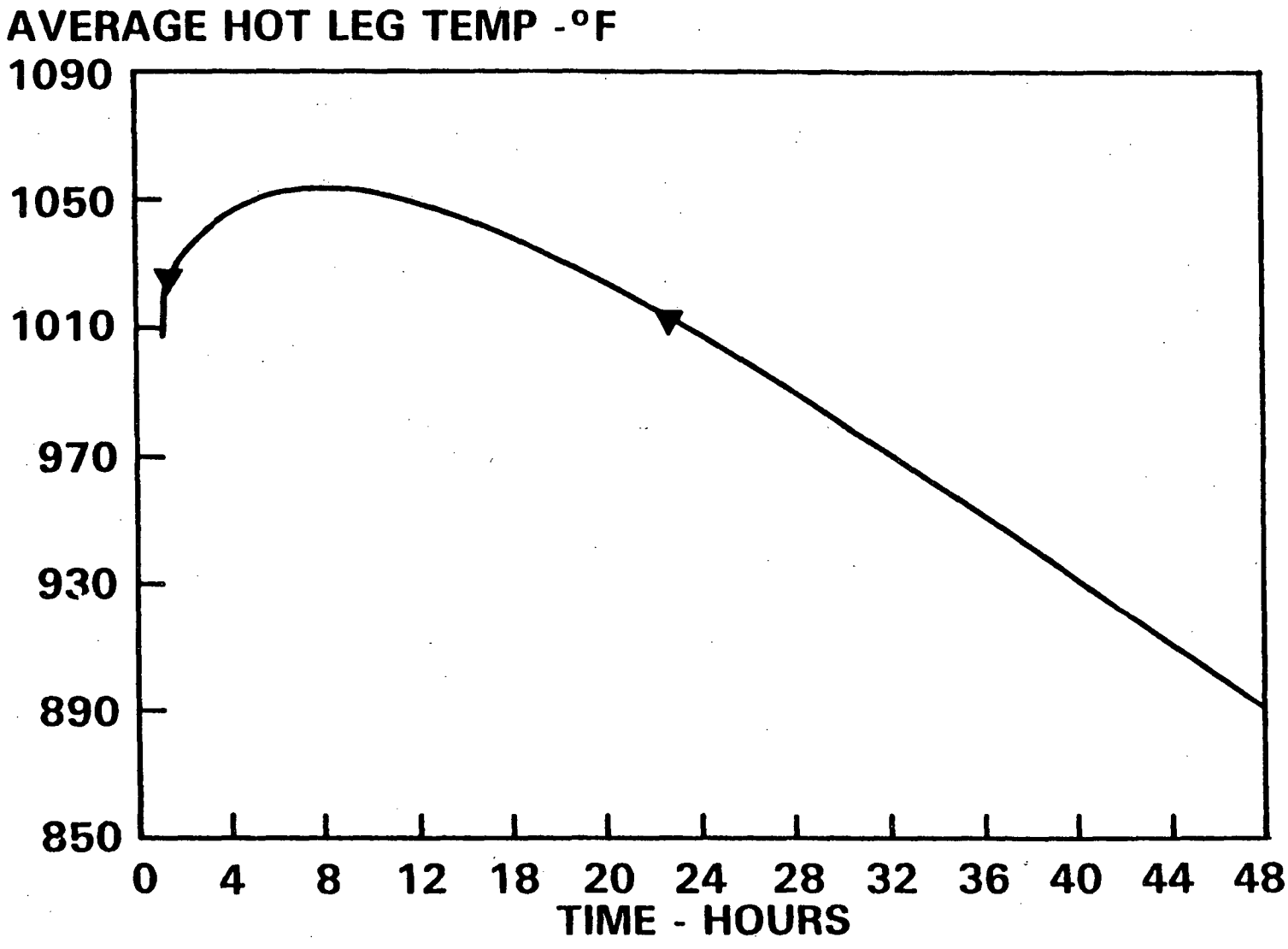


Figure 5.19 Direct heat removal service analysis

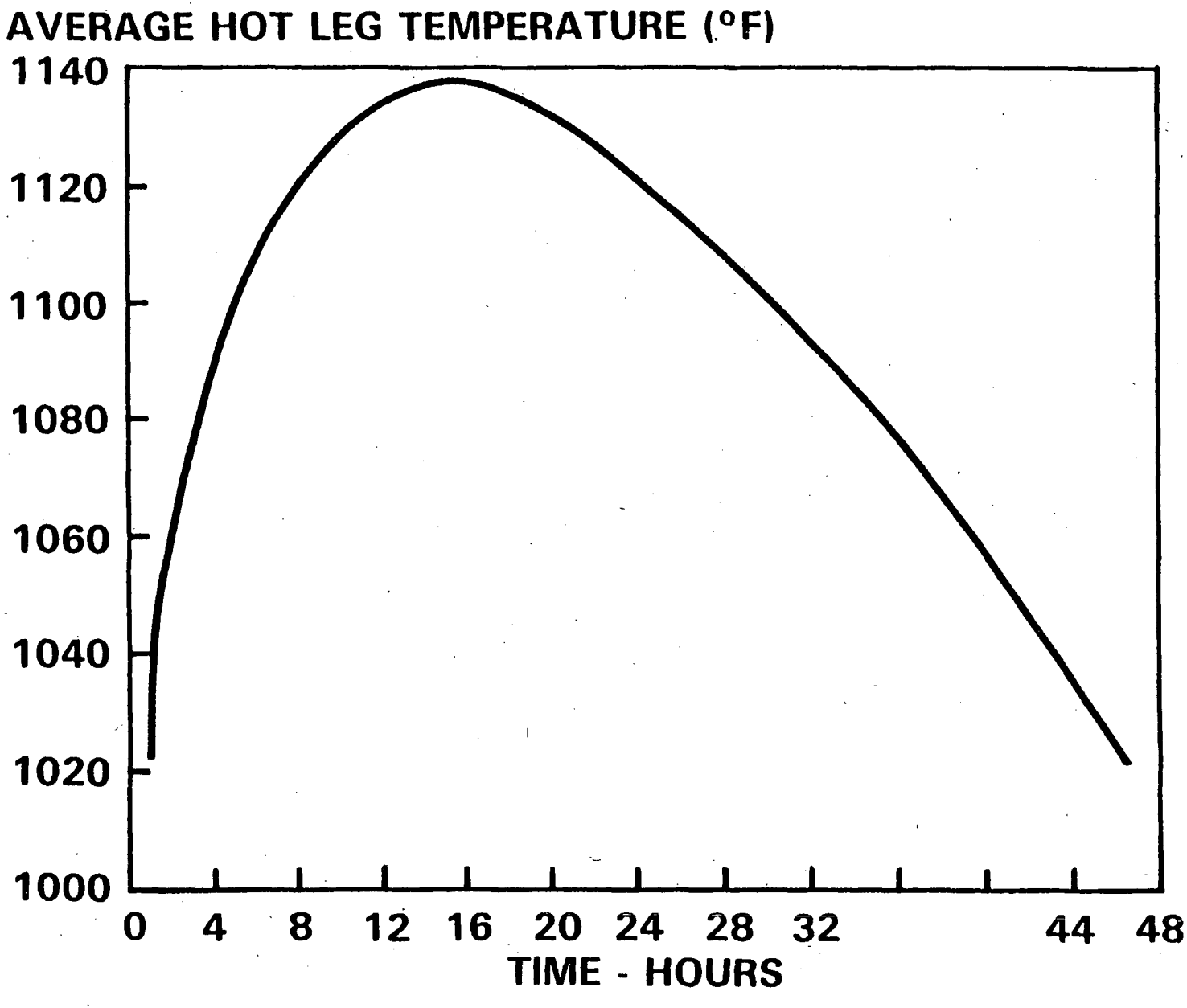


Figure 5.20 Direct heat removal service single-failure analysis

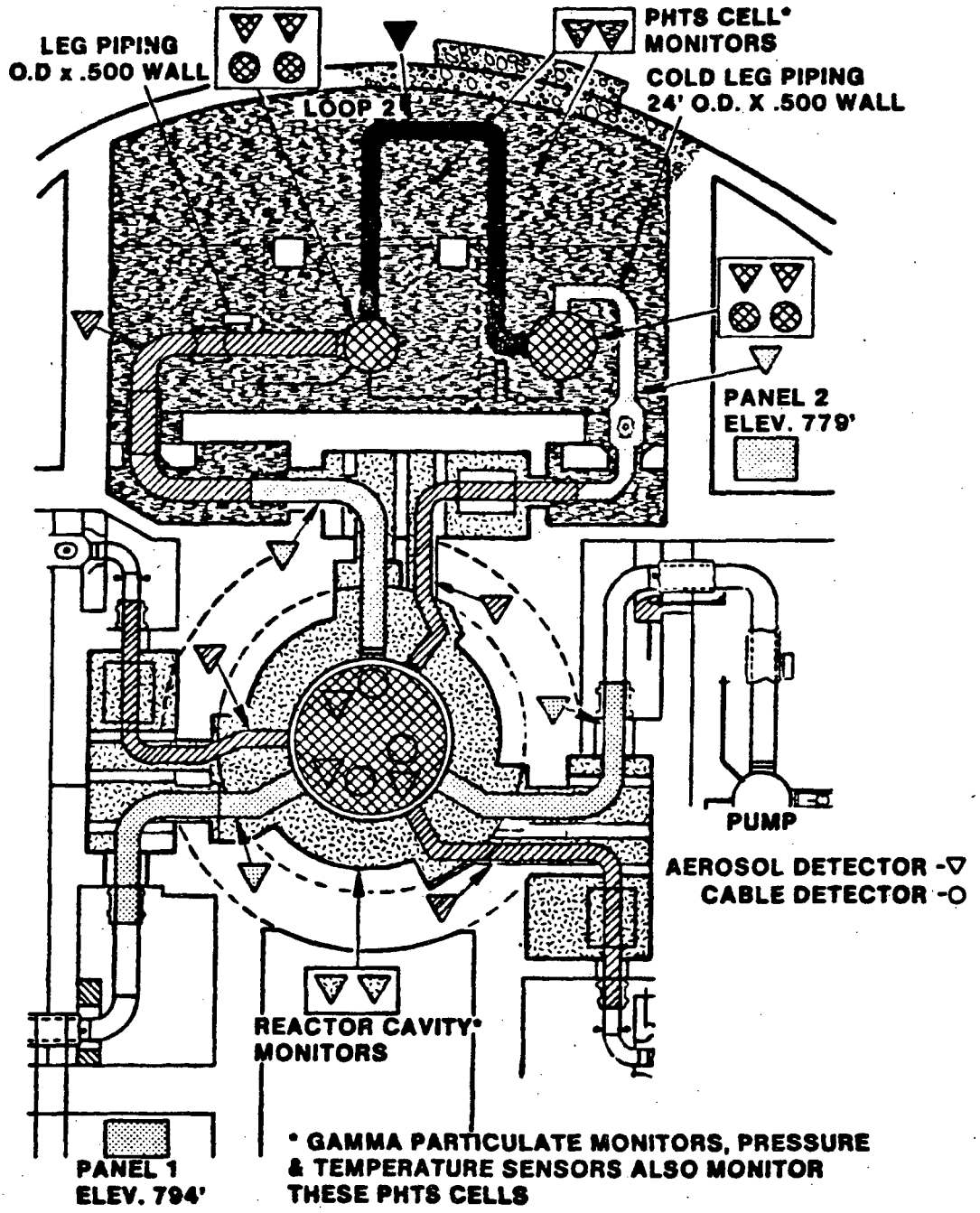


Figure 5.21 General leak detector location and types in the reactor vessel/guard vessel/reactor cavity and PHTS

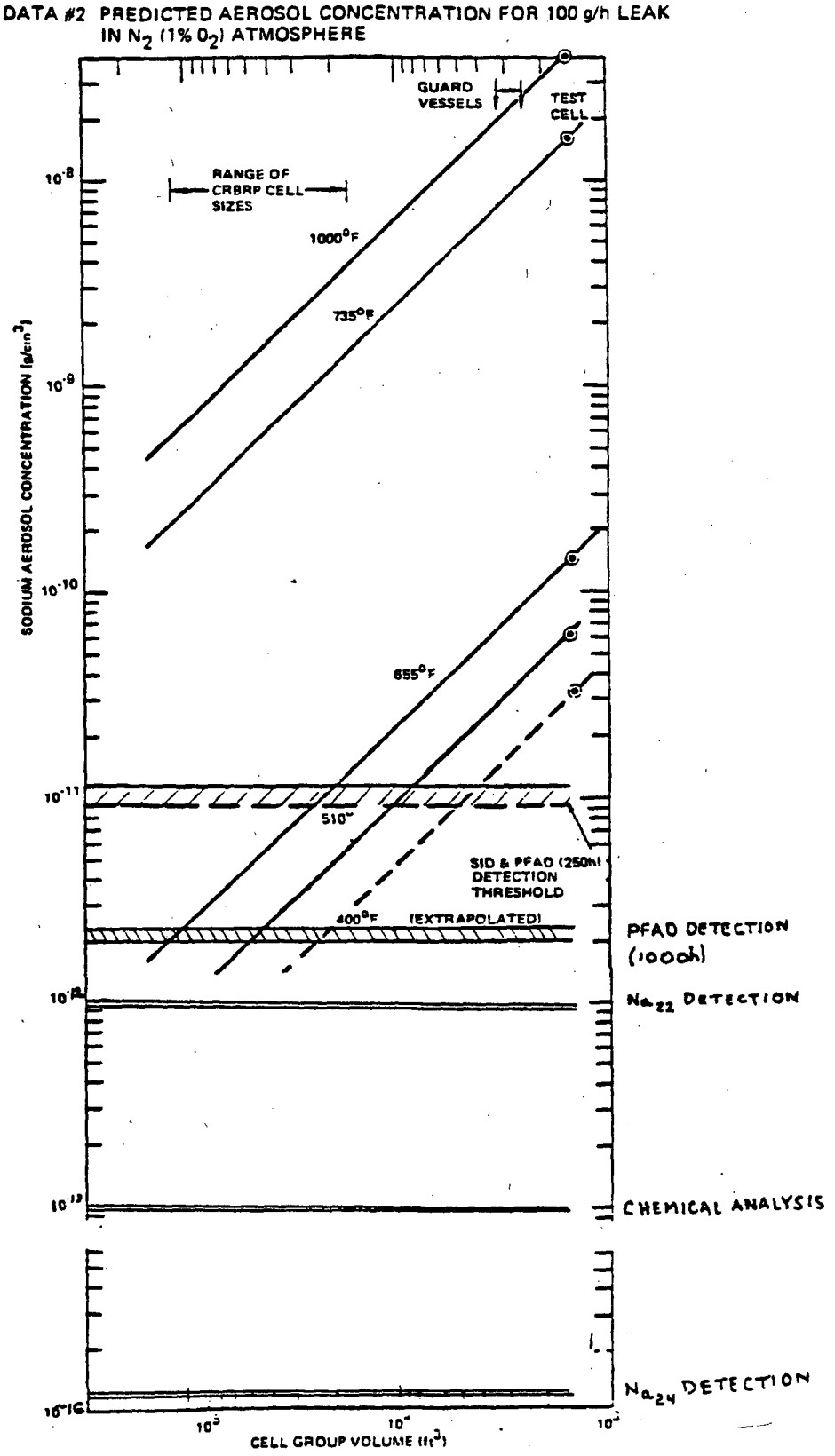


Figure 5.22 Liquid metal-gas detection system

Table 5.1 Partial list of heat transport system hydraulic design conditions

Parameter	Value
Primary system	
Thermal power, MWt	975
Hot-leg temperature, °F	995
Cold-leg temperature, °F	730
Pump flow, gpm at 995°F	33,700
Intermediate system	
Hot-leg temperature, °F	936
Cold-leg temperature, °F	651
Pump flow, gpm at 651 °F	29,500

Table 5.2 Thermal and hydraulic design-basis parameters

Parameter	Value
Power per loop, MWt	325
Flow per loop	
Primary, lb/hr x 10^6	13.82
Intermediate, lb/hr x 10^6	12.78
Primary hot-leg temperature, °F	995
Primary cold-leg temperature, °F	730
Intermediate hot-leg temperature, °F	936
Intermediate cold-leg temperature, °F	651
Maximum primary system pressure drop, psi	160.3
Maximum primary heat transport system loop ΔP (exclusive of reactor), psi	37.3
Required primary pump head (thermal-hydraulic design), ft	458
Maximum primary pump drawdown, ft	8

Table 5.3 Primary heat transport system component design conditions

Component	Parameter	Design conditions	
		Temperature (°F)	Pressure (psi)
Pump	Pump tank	1,015	*
	Hydraulics and discharge assembly	1,015	200
Intermediate heat exchanger (IHX)	Shell side	1,015	200
	Tube side	1,015	325
	Primary to intermediate differential pressure	775	200
	Intermediate to primary differential pressure	1,015	325
	Primary to intermediate hot end ΔT	75	-
	Primary to intermediate cold end ΔT	120	-
Check valve		775	200
Flow meter		775	200
Piping	Reactor to pump	1,015	30
	Pump to IHX	1,015	200
	IHX to check valve	775	200
	Check valve to reactor	775	200

*Design pressure is 325 psig at 775°F in accordance with intermediate pump design for identicality.

Table 5.4 Total volume requirements of the protected water storage tank

Parameter	Case*			
	Pipe break (three-loop operation)	Pipe break (two-loop operation)	Sodium-fire delayed PACC startup	Loss of bulk ac power
Water vented (through direct auxiliary heat removal system (DAHRS)), ft^3	4,650	4,900	4,820	5,400
Pipe-break loss, ft^3**	1,481	1,481	1,481†	---
Leakage, ft^3††	1,564	782	782	2,346
Auxiliary liquid-metal system, ft^3	250	250	250	250
Required usable volume, ft^3	7,945*	7,413	7,333	7,996
Margin (14%), ft^3				1,115
Usable volume, ft^3				9,111
Heel (5%), ft^3				480
Total tank volume, ft^3				9,591

*Cases correspond to those of PSAR Table 5.6-7 (Demo auxiliary heat removal simulation, Appendix A).

**Operator action to isolate break within 10 min is assumed. Break flow sum is based on 110% flow (77 lb/sec) from each of two auxiliary feedwater supply lines for a period of 600 sec.

†In the sodium-fire event, this quantity of water is lost because of the assumed failure of the safety/relief valve, in the open position, at the superheater exit. A 10-min operator action time is assumed to isolate the loop.

††Leakage is a function of the number of loops remaining following the initiating event.

Table 5.5 Design parameters for protected air-cooled condensers

Parameter	Value
Design conditions	
Pressure, psig	2,200
Temperature, °F	650
Thermal-hydraulic performance	
Heat removal, MWt	15 (7.5 MWt per tube bundle)
Steam pressure, psig	1.450
Steam temperature, °F	592
Moisture, %	0
Condensate temperature, °F	592
Air temperature, °F	100
Air pressure, psig	14.3

Table 5.6 Direct heat removal service valve classification

Location	Valve no.*	Active/inactive	Normally** open/closed	Operating mode
Makeup pump suction	1	Inactive	Open	Isolation
	2	Inactive	Open	Isolation
	11, 12	Inactive	Closed	Isolation
Overflow vessel drain/sample return	15	Active	Open	Isolation
Makeup pump discharger	4	Active	Open	Isolation
	3	Active	Open	Isolation
	5	Active	Open†	Isolation
	6	Active	Open†	Flow control
	7	Active	Closed†	Isolation
Overflow heat exchanger sodium outage	8	Inactive	Open	Isolation
	9	Inactive	Open	Check
	10	Inactive	Open	Isolation
Overflow heat exchanger to air blast (hot leg)	24, 25	Active	Closed†	Isolation
	22, 23	Active	Open†	Isolation
Air blast to overflow heat exchanger (cold leg)	20, 21	Active	Closed†	Isolation
NaK drain lines	26, 27, 28 29	Inactive	Closed	Isolation
Ex-vessel storage tank NaK expansion tank	32, 33, 37	Inactive	Open	Isolation
Cover gas	36	Inactive	Closed†	Isolation
Sodium vent (typical)	34	Inactive	Closed	Isolation
Sodium sample	35	Inactive	Open	Isolation

*Valve numbers are those shown in Figure 5.18
**The position indicated is the valve position during normal plant operation.
†Valves that are manually opened or closed before initiation of DHRS operating mode.

NOTE: All active valves are manually operable remotely from the control room.

Table 5.7 Direct heat removal service (DHRS) pump classification

Pump	Active/inactive	Operating mode	
		Normal plant operation	DHRS duty
Makeup pump(s) (operating)	Active	Reduced power flow	100% power flow
Ex-vessel storage tank NaK pump (operating)	Active	Reduced power	100% power
Ex-vessel storage tank NaK pump (standby)	Active	Reduced power	100% power

Table 5.8 Direct heat removal service (DHRS) operating cases and sensitivity evaluations

DHRS operating case*	Peak temperature	Number of PHTS loops	Number of IHTS loops	Number of DHRS trains	Conservative decay heat	EVST heat load	Na flow (gpm)	NaK flow (gpm)
Design case	~1,120°F	3	0	2	Yes	Yes	560	800
Updated design case	~1,052°F	3	0	2	Yes**	Yes	560	800
Single-failure evaluation #1	~1,055°F	2†	2	1	Nominal values used	No	600	600
Single-failure evaluation #2	~1,137	2	0	1	Yes**	No	600	600
Delayed-start design case (~ 45 min to DHRS operation)	~1,078	3	0	2	Yes**	Yes	560	800

*Conservatisms included in all cases: shutdown at end of cycle at full power, no consideration of insulation losses, no consideration of SGS heat capacity.
**Decay heat values revised to reflect current conservative design values.
†The two PHTS loops assumption conservatively envelopes the assumption that a PHTS pump or pony motor fails to operate.

NOTE: PHTS = primary heat transport system; IHTS = intermediate heat transport system; EVST = ex-vessel storage tank.

Table 5.9 Liquid metal-gas leak detection system development testing

Test series	Purpose	Conclusions	Facility	No. of tests
Detector characterization	Survey candidate detector design parameters and performance	Design criteria reasonable Detector performance good	40 ft^3 chamber	24
Mockup/natural convection	Larger scale detection system performance	System performance good: Plugging filter aerosol detector Sodium ionization detector	1,500 ft^3 chamber	21
Verification	Prototypical detection system operation with cell ventilation	Prototypical detection systems meet CRBRP requirements	1,500 ft^3 chamber	8
Long-term performance	Detection system operation in a liquid metal power plant	Continuous, multiyear operation satisfactory	EBR-II	3 detectors

Table 5.10 Liquid metal-gas leak detection system development test parameters

Test	Characterization	Mockup	Natural convection	Small pipe	Verification
Chamber atmosphere oyxgen in nitrogen, %	1, 21	1, 21	1, 21	1, 21	1, 21
Water vapor, ppm	300, 1,000, 3,000, 30,000	1,000, 30,000	1,000, 15,000	1,000, 15,000	300, 15,000
Sodium temperature, °F	400-1,000	350-980	350-980	500-980	500-995
Leak rate, g(Na)/hr	3-30	0.4-100	2-100	2-243,000	100
Gas flow, liter/min	0.6-1.8	0.4, 5.0	0.4, 5.0	0.4, 5.0	0.4, 5.0
Line size, in.	1/4, 1/2	1/4	1/4	1/4	1/4
Line length, ft	50, 100, 200	~100	~100	~100	150

Table 5.11 Liquid metal-gas leak detection system mockup tests--large pipe insulation annulus aerosol detection

[Aerosol general decrease with temperature]

Test	Test number			
	1	2	4	8
Chamber atmosphere				
Oxygen in nitrogen, %	1	1	1	1
Water vapor, ppm	1,000	1,000	1,000	1,000
Sodium temperature, °F*	980	400**	980	500
Sodium leak*				
Rate, g(Na)/hr	100	100	0.4	2
Total, g(Na)	920	830	100	130
Detection time, hr*				
Sodium ionization detectors	0.1	30.7	0.05	NR***
Plugging filter aerosol detectors	0	27	6.5	NR†

*Less aerosol generation at low temperature.

**Temperature of the pipe was increased to 700°F after sodium flow was stopped.

***Detector output did not reach level established to indicate successful detection.

†0.13 mg of sodium aerosol collected on plugging filter aerosol detectors filter membrane in 50 hours, 30 μg required for positive chemical detection.

NOTE: NR = not reported.

Table 5.12 Liquid metal-gas detection system mockup tests--large pipe test chamber atmosphere aerosol detection

Test	Test number 9	10	11	12
Chamber atmosphere				
Oxygen in nitrogen, %	1	1	1	1
Water vapor, ppm	1,000	1,000	1,000	1,000
Sodium temperature, °F*	980	980	980	640
Sodium leak*				
Rate, g(Na)/hr	<100	~100	2	2
Total, g(Na)	350	625	175	200
Atmosphere aerosol concentration, g(Na)/cc	NM	NM	$1x1^{-9}$	$3.5x10^{-10}$
Detection time, hr*				
Sodium ionization detectors	0.25	0.8	0.01	1.2
Plugging filter aerosol detectors	4	4	0.5	12.1

NOTE: NM = not measured.

Table 5.13 Ex-core neutron flux at beginning of Cycle 3

Core midplane location	Total neutron flux (n/cm^2 sec)	Total end-of-life fluence	Fraction of neutron flux (E > 0.1 MeV)	End-of-life fluence (E > 0.1 MeV)
Radial blanket/ removal shield interface	8.5×10^{14}	6.0×10^{23}	0.37	2.2×10^{23}
Fixed radial shield surface*	4.1×10^{13}	2.9×10^{22}	0.24	7.0×10^{21}
Core barrel surface*	6.1×10^{12}	4.3×10^{21}	0.20	8.6×10^{20}
Reactor vessel surfaces*	6.0×10^{11}	4.3×10^{20}	<0.1	4.3×10^{19}

*Design lifetime of nonreplaceable components is 30 calendar years (22.5 equivalent full-power years).

Source: PSAR Table 4.3-35.

6 ENGINEERED SAFETY FEATURES

6.1 General

Engineered safety features (ESFs) are those structures, systems, and components included in a given nuclear plant design to mitigate the consequences of design-basis accidents. For CRBRP features have also been included in the design to mitigate the consequences of accidents of such low probability that the accidents are considered beyond the design basis. The staff's evaluation of these features and low-probability events is provided in Appendix A of this SER.

In general, the ESFs must be seismically qualified and if a pressure boundary designed to ASME Code, Section III, standards. Where instrumentation and controls are involved, they must conform to IEEE 279 guidance. Because the ESF may range from an active system such as the steam generator building aerosol release mitigation system to a passive system such as reactor guard vessel, other special criteria are applied to each specific ESF.

The CRBR ESFs are listed in Table 6.1. Table 6.1 includes the list proposed by the applicants in the PSAR and the list proposed by the staff.

Those ESFs covered in the Standard Review Plan (SRP) such as the reactor confinement/containment system are reviewed in detail in this section. Many of these ESFs are reviewed in depth in at least one other section in this report.

For those ESFs reviewed in other section(s), no review is considered here, but the other sections are referenced to avoid duplication. The sodium leak detection system in the primary heat transport system is included as an ESF because of its importance to the leak-before-break concept and its mitigative potential to detect leaks while they are still small.

6.2 Containment Systems

The basic acceptance criterion used as the basis for the staff's evaluation is that the CRBRP should achieve a level of safety comparable to that of the current generation of light-water-reactor (LWR) plants. Accordingly, the staff has used current licensing criteria set forth in certain sections of the Standard Review Plan (SRP) (NUREG-0800, dated July 1981) taking into account the reactor concept and differences in experience that prevent complete adherence to all of its provisions. Specifically, the CRBRP was reviewed using as guidance the acceptance criteria contained in SRP Sections 6.2.1, 6.2.2, 6.2.3, 6.2.4, 6.2.5, and 6.2.6. These acceptance criteria include the applicable general design criteria (GDC) of Appendix A, 10 CFR 50, as modified for CRBR in the principal design criteria, applicable regulatory guides, branch technical positions, and industry codes and standards as specified in the above-cited sections of the SRP.

6.2.1 Containment Functional Design

The containment systems for the CRBRP (Figure 6.1) will consist of a steel containment enclosed by a secondary leakage control or confinement building and the containment isolation system. The containment will have a design pressure of 10 psig, design temperature of 250°F, and a design leak rate of 0.1 wt% per day. The containment vessel will be a free-standing, welded steel structure with a hemispherical dome, flat bottom, and cylindrical side wall; it will have net free volumes of 3,600,000 ft^3 above the operating floor and 1,400,000 ft^3 below the operating floor. The lower portion of the side wall below the operating floor will be embedded in concrete and be essentially below grade level. Enveloping the containment vessel will be a concrete confinement structure with a vertical wall thickness of 4 ft and a dome thickness of 3 ft. The annular space separating the two structures will be 5 ft wide and will provide a means for heat rejection for those postulated accidents that will be beyond the design basis. The confinement building will include an ex-filtration system to maintain a negative annulus pressure of at least 0.25 in. water gage during normal operation and accident conditions and to ensure that leakage from the containment building will be filtered before it is released to the environment. The applicants have committed to perform preoperational tests to verify that the above condition can be maintained. On the basis of this geometry, heat rejection from the containment/confinement structure will be restricted to that portion above the grade or operating floor level (el 816 ft).

The internal structure will consist of concrete cells that will contain the reactor vessel, the primary heat transport system, and other support systems. The concrete cells will be inerted and partitioned to minimize the consequences of sodium spills. Isolating the space in which the sodium spill occurs will cause oxygen depletion because of the burning of sodium. This technique has been shown to be effective in controlling sodium fires in the absence of sodium-concrete reactions. In applying this concept to CRBRP, it will be necessary to provide the capability of confining the sodium-containing equipment in gas-tight cells and pipeways by using walls, partitions, fire doors, ventilation dampers, and penetration seals. It also is necessary to prevent communication with other inerted cells in the event of sodium fires. The applicant has addressed this issue in the CRBRP design.

The applicants have proposed a design-basis accident involving the rupture of the primary in-containment storage tank located in cell 102A during reactor maintenance. This accident was selected from among others as a worst case. Evaluation of this accident is presented in Section 15.6 of this SER. The accident scenario is based on failure of the storage tank containing 35,000 gal of sodium (drained from one of the primary loops) at 400°F. In addition, it is assumed the cell will be deinerted and the contents of the tanks form a pool on the cell floor. These initial conditions are realistically conservative.

The atmosphere in cell 102A will come into contact with the containment atmosphere through a single 21-ft^2 opening or hatch. The heat transfer mechanism will be natural convection and rely on a countercurrent flow of combustion products in one direction and containment atmosphere in the reverse direction. Although the postulated pathway is appropriately modeled based on experimental studies of the transfer mechanism, this fluid-thermal coupling will have a significant effect on the containment atmosphere temperature and pressure history during the accident. A second pathway between the cell 102A atmosphere and the containment will result in an increased interchange of energy. This permits

unidirectional flow in each pathway. The applicants have addressed this question and believe that proper administrative procedures will preclude there being more than one opening between an inerted cell and the containment. The staff will review these administrative procedures at the OL stage.

The applicants have assumed that sodium combustion will occur as a pool fire, which is a reasonable assumption because of the low system pressure. The GEOSFIRE computer code was used for the analysis. It is similar to the SOFIRE code, which has been previously reviewed and found acceptable. Comparison of this code with small-scale experiments lends credibility to the predictions based on the code although the mechanistic approach used in some areas appears to be questionable. In addition, there is further evidence that sodium conflagrations in small areas (4.5 m^2) lead to both stable combustion rates (40-45 kg/m^3) and stable pool temperatures (730°C). The combustion process at the pool surface appears to be reasonably well understood.

The applicants' analysis of the design-basis sodium spill and the subsequent fire indicates the peak containment gas and wall temperatures to be 138°F and 122°F, respectively; both are well below the containment design temperature of 250°F. The maximum containment pressure was predicted as 0.8 psig (occurs 20 hours after onset of the postulated accident) compared with a design pressure of 10 psig. The differential pressure between the cell and the upper containment was determined by the applicants to be negligible. Therefore, pressure loading on the cell structure is expected to be insignificant. As discussed earlier, these results are believed to be sensitive to the size and number of flow paths between cell 102A and the containment since the cell 102A maximum temperature was calculated by the applicants to be 820°F. It is also worth noting that the heat rejection path from the containment, which was assumed to be restricted to the area above the 816-ft level, will be small compared with the thermal capacity of the containment confinement structure; that is, most of the energy generated during the 550-hour sodium burn will be absorbed by the structure and very little will be rejected to the external environment. This postulated accident poses very little challenge to the containment.

The CRBR reactor containment building (RCB) will be equipped with two independent seismic Category I, Safety Class 2 vacuum relief systems to ensure that the RCB external design pressure, 0.5 psig, will not be exceeded. Each system will consist of two self-actuated valves, one on each side of the RCB, set to open on a negative differential pressure (-3.5-in. water gage) and to close before the pressure in the RCB equalizes with the outside atmosphere (-1.75-in. water gage).

The staff has reviewed the information in the applicants PSAR to ensure conformance with all the applicable acceptance criteria. The staff concludes that the CRBR containment design will conform to CRBR PDC 14, 41, 43, and 44 and is therefore acceptable.

6.2.2 Cortainment Heat Removal System

The applicants do not propose to furnish heat removal systems for the containment to deal with design-basis accidents. The staff agrees with this position since there are no design-basis accidents requiring heat removal.

6.2.3 Annulus Filtration System

The confinement building will be a reinforced concrete structure, totally enclosed, and will form a continuous annular space surrounding the containment vessel above the foundation mat. Two 100% capacity, redundant annulus filtration systems designed to comply with the provisions of RG 1.52 will be provided to ensure that the radioactivity released will not cause offsite doses to exceed the guidelines of 10 CFR 100 and to maintain the containment/confinement annulus space under 1/4-in. water gage negative pressure during both normal plant operation and accident conditions. These systems, which have the capability for operational testing, will be designed to meet seismic Category I and Safety Class 2 criteria, and will be protected against missiles.

The applicants commit to follow the guidelines and testing requirements of SRP Section 6.2.3.

The function of the annulus filtration system will be to collect and process the leakage of radioactive materials from the containment vessel into the atmosphere of the confinement/containment annular space following a design-basis accident. In addition, the annulus filtration system will be capable of filtering the reactor containment building ventilation exhaust during refueling operations. The system will be designed as a partial recirculation system to maintain a slight negative pressure in the annular space following a design-basis accident. The system will be a redundant system. Each train will have a design capacity of 3,000 ft^3 per minute of air and will include the following components: moisture separator, heating coil, high efficiency bag type filter, high efficiency particulate air (HEPA) filter section, adsorber, final HEPA filter section, exhaust fan, and annulus recirculation fan. The equipment and components will be designed to Quality Group B and seismic Category I criteria and will be located in a seismic Category I structure.

From the system description in the PSAR, the staff determined that the annulus filtration system will be designed to be consistent with CRBR PDC 53 and 56. In its evaluation of the system design efficiencies for removal of elemental iodine and organic iodine, the staff concurs with the applicants' system decontamination efficiencies of 95% for elemental and organic iodines for the adsorbers and 99% for particulates for the HEPA filters.

The staff has determined that the annulus filtration system will be designed in accordance with the guidelines of RG 1.52 and will be capable of controlling the release of radioactive materials in gaseous effluents following a postulated design-basis accident.

6.2.4 Containment Isolation System

The containment isolation system will include the containment isolation valves and associated piping and penetrations necessary to isolate the reactor containment building in the event of an accident. The staff's review of this system will include the number of isolation valve locations, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components.

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary to prevent or limit the escape of fission products from a postulated accident. The applicants have proposed the design basis and design criteria as well as the isolation valve arrangements to be used for isolation of containment penetrations.

The containment isolation system will be designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of two isolation valves in series or a closed system and isolation valve, will be provided to ensure that no single active failure will result in the loss of containment integrity. The applicants indicated that the containment isolation system components, including valves, controls, piping, and penetrations, will be protected from internally and externally generated missiles, fluid jets, and pipe whip.

The applicants have proposed certain design criteria for designing the containment isolation system for the CRBRP. The staff's evaluation of these design criteria has been based on the conformance of the containment isolation provisions with the Commission's regulations as set forth in the general design criteria and with applicable regulatory guides, branch technical positions, the Standard Review Plan, and industry codes and standards.

The containmment isolation system will be designed to ASME Code, Section III, Class 2, and will be classified as a seismic Category I system.

The containment isolation provisions for the lines penetrating containment will conform to the applicants' design requirements as described in CRBR PDC 45, 46, 47, or 48, as appropriate.

The staff has evaluated the applicants' proposed design criteria and concludes that they are acceptable because they impose the same design requirements as GDC 54-57 of 10 CFR 50 for light-water reactors.

As provided by CRBR PDC 45, 46, and 47, there will be containment penetrations whose isolation provisions will not have to satisfy the explicit requirements of the design criteria.

Those lines not satisfying these explicit requirements will be the intermediate heat transport system (IHTS) and the chilled water lines.

The IHTS will be maintained intact as an extension to the containment boundary (see Figures 6.2 through 6.4) by indepth protection and integrity monitoring provisions as follows:

(1) A seismic Category I steam generator building will provide protection from tornado, winds, and missiles. It also will provide protection from internally generated missiles by internal structures and impingement shields.

(2) The IHTS piping and components will be designed to maintain their integrity following a safe shutdown earthquake.

(3) IHTS boundary will be monitored through a combination of inservice inspection, sodium-to-air leak detection, ΔP of 10 psi between the IHTS and the primary heat transport system (PHTS) in the intermediate heat exchanger (IHX), and leak detection for the IHX. These mechanisms will provide assurance that leakage between PHTS and IHTS as well as leakage from the IHTS to the atmosphere can be detected early and corrected.

The staff concludes that the proposed indepth protection and integrity monitoring provision provide sufficient assurance that the IHTS will maintain its integrity as an extension to the containment boundary.

For lines of closed systems penetrating containment, CRBR PDC 48 provides the necessary tool for designing the containment isolation provision. The applicants stated that where a closed system will form the second isolation boundary, each of the systems and all components that will form its boundary will be designed to Safety Class 2 and seismic Category I standards.

Since these lines will not contain high-pressure fluids and will be adequately supported, and since the system will be a closed system and will be neither part of nor directly connected to either the containment atmosphere or the primary coolant boundary, the staff concludes that it will represent an acceptable alternative to the requirements of CRBR PDC 45-48.

The applicants stated that the automatic isolation signal (high radiation) will be specified if response to close the valve is necessary within 10 minutes; otherwise, only remote manual initiation will be provided. In addition, certain lines will be provided with back-presure regulated valves that will close automatically if the supply-side pressure drops below a preset limit.

The CRBRP containment isolation system will depend on a single parameter (radiation) for initiation of automatic isolation. This represents a departure from standard licensing practice and the TMI Action Plan requirements, which is to require a diversity of parameters for this purpose.

The applicants indicated that there will be no identifiable accidents within the design bases capable of producing rapid changes in any containment parameters other than radiation level. For example, containment pressure will change by less than 1 psig over a period of more than 20 hours from the onset of a design-basis accident.

In addition, the applicants indicated that the reactor coolant boundary will be contained within five inerted concrete cells with no interconnection between them or with the containment air atmosphere. Although a localized rapid change in temperature, pressure, and/or sodium aerosol concentration might occur, such changes would be translated to containment over a long period of time. It would be overly complex to attempt to connect all of these signals from all cells into the automatic system. However, all these signals will be displayed in the main control room, so the operator will be alerted to isolate the containment. To further justify the appropriateness of the proposed design, the applicants showed by analyses that delay of 1 hour to 2 hours in isolation time would not result in offsite doses above the guideline values for design-basis events. Because of all of the above considerations, the applicants proposed to use the containment radiation as the single parameter for automatic

containment isolation with consideration for diversity provided by the selection of diverse sensors and diverse sensing locations. Specifically, three GM tubes will be located in the head access area, and three scintillation detectors will be located in the containment ventilation exhaust duct. Two-out-of-three logic for each set will initiate containment isolation. The choice of different radiation measurement techniques and instrumentation manufacturers will provide reasonable assurance that common mode failure is not credible.

The staff has completed its evaluation of the applicants' rationale of not adhering to the explicit requirements of SRP Section 6.2.4 regarding diversity of parameters necessary to isolate the containment and concludes that exemption from these requirements is warranted.

6.2.5 Containment Ventilation System

The containment ventilation air supply and exhaust will each be provided with three isolation valves. One valve will be inside containment and two outside containment. This redundancy will ensure proper isolation assuming internal valve failure, resulting from a hostile environment inside the containment, during thermal margin beyond design basis when these systems are called on to function. The staff will require the applicants to address the potential for damage of the outboard isolation valve's internals resulting from the hostile environment inside the containment.

6.2.6 Containment Vacuum Breaker

The CRBRP containment will be equipped with a vacuum breaker assembly consisting of two check valves; one inside and one outside the containment. The applicants agreed to the following staff requirements:

(1) Check each relief valve and associated piping to determine that it is clear of foreign matter at each refueling outage.

(2) Provide redundant position indicators on all vacuum breakers with redundant indication and an alarm in the control room. The vacuum breaker position indicator system should be designed to provide the plant operators with continuous surveillance of the vacuum breaker position.

(3) Operability test all vacuum breakers at monthly intervals to ensure free movement of the valves.

6.2.7 Containment Purge During Plant Operation

The applicants indicated that the CRBRP containment design will incorporate the features of operating with a continuous ventilation purge to allow plant personnel to perform inspections and calibrations during power operation without being encumbered by a fresh air mask. The applicants' rationale to support this position is that unlike light-water reactors where the postulated accident results in a very rapid increase in containment pressure and temperature, the postulated events for CRBRP are very slow developing events. Because of this characteristic, rapid isolation of containment following an accident through provided, will not be essential from the viewpoint of offsite doses.

The staff agrees with the applicants' assertion that there are no identifiable accidents within the design base capable of producing any rapid changes in containment condition (except for radiation) that would necessitate a rapid containment isolation. However, it is the staff's position that containment purging should be minimized; therefore, the staff will require the applicants to:

(1) Address the feasibility of adding an internal filtration system and the effect of such a system on the number of hours needed for purging. The staff proposes that this issue be addressed by the applicants after the construction permit is issued.

(2) Commit to not using the purge system for temperature or humidity control.

(3) Provide the staff with their proposed purging requirements that will include the dose limit at which purging the containment will start and the procedure to minimize the number of entries into containment and consequently the need to purge. This information should be provided during the period following the issuance of the construction permit for implementation at the operating license stage.

6.2.8 Containment Leakage Testing

The staff has reviewed the applicants' containment leak testing program for compliance with the containment leakage testing requirement specified in Appendix J to 10 CFR 50. Such compliance provides adequate assurance that the containment leak-tight integrity can be verified throughout the service lifetime and that leakage rates will be periodically checked during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through potential leak paths will not be in excess of the limits specified for the site.

Specifically, the staff reviewed the containment leak testing program to ensure that the containment penetrations and system isolation valve arrangements will be designed to satisfy the containment integrated leak rate testing requirements and the local leak testing requirements of Appendix J to 10 CFR 50.

The applicants have committed to perform testing of the containment and the containment isolation valves and penetrations in accordance with the guidance of Appendix J with the following qualification:

> The word "pressure" should be omitted from all reference to "reactor coolant pressure boundary."

On the basis of its evaluation of the applicants' commitment, the staff concludes that it constitutes an acceptable fulfillment of the staff's requirements and is, therefore, acceptable.

6.2.9 TMI Action Plan Items

II.B.8 Rulemaking Proceedings on Degraded Core Accident

To reduce the possible consequences of a postulated degraded core accident, the staff requested the applicants to provide a system to control hydrogen that would be generated from a postulated 75% metal-water reaction, which is well beyond the amount of hydrogen specified in 10 CFR 50.44(d). However, because of the differences between CRBR and LWR, different considerations are used to evaluate the hydrogen production following a core disruptive accident. This issue is discussed in Appendix A to this SER.

II.E.4.1 Dedicated Hydrogen Penetrations

The CRBR will not use a hydrogen recombiner for combustible gas control. A backup filter containment purge through a dedicated seismic Category I penetration will be provided. Because there is no recombiner, Item II.E.4.1 is not applicable to CRBR.

II.E.4.2 Containment Isolation Dependability

Because of the unique features of CRBR, complete adherence to the explicit requirements of SRP Section 6.2-4 regarding diversity of the parameters necessary to isolate the containment was not required. The applicants' justification and the staff's evaluation is presented in Section 6.2.4 of this SER.

The applicants have proposed to purge the containment during normal operation. This issue has been identified as a post-CP issue and, therefore, the staff's assessment of Items II.E.4.2 and II.E.4.4 will be addressed in the OL review.

6.3 Control Room Habitability System

The CRBR control room will be a concrete enclosure with sealed doors and an independent ventilation system capable of isolating the control room in the event of an accidental toxic chemical or radiologic release.

The following sections describe the radiological, toxic gas protection, and design aspects of the CRBR control room habitability system.

6.3.1 Radiological Protection

The applicants propose to meet PDC 17 (equivalent to GDC 19 of Appendix A to 10 CFR 50), with respect to radiation, by using concrete shielding and installing two 100% capacity heating, ventilating, and air conditioning (HVAC) pressurization systems (500 ft^3 per minute), which will keep the control room at 1/4-in. positive pressure. This system will contain redundant filter systems consisting of bag filters, two banks of high-efficiency particulate air filters, and a bank of carbon absorbers. The intakes to the filter system will be widely separated; one intake will be located at the southeast corner of the control building roof at el 880 ft, and the other intake will be located at the northeast corner of the steam generator building auxiliary bay at el 858 ft. The system can be placed in the recirculation mode (8,000 ft^3 per minute); thus, control room isolation will result. The control room is classified as type B construction in accordance with RG 1.78 and will be constructed so that the leakage rate

will be about 0.06 air changes per hour. The applicants propose to use quick-acting isolation dampers with a closing time of less than 4 sec. The control room will be maintained at a positive pressure during the isolation period. The applicants propose to install fixed-type airborne radioactivity monitors in the control room to continuously monitor the control room breathing air for airborne radioactivity, including particulate, gaseous, and iodine activities. These systems will be activated automatically either on receipt of a containment isolation signal through the plant protection system or the detection of smoke, toxic chemicals, or high levels of radioactivity in the control room intake ducts. The proposed detector and controls will be designed and installed in accordance with engineered safety feature standards, including redundancy. The staff's independent calculations of the potential radiation doses to control room personnel following an accident similar to the site suitability accident show that the resultant doses will be within the guidelines of PDC 17. The staff concludes that the control room habitability system will provide adequate radiological protection for the control room operators in the event of a design-basis accident.

6.3.2 Toxic Gas Protection

The applicants propose to protect the control room against sodium and sodium combustion products by automatic isolation and initiation of the recirculation mode. The system will be initiated automatically by detectors located in the control room air intakes or by the sodium fire detection system. The selection of the proper signal will be based on criteria in RG 1.78 and will be given in the FSAR. Storage vessels containing toxic materials on site will be provided with instrumentation that will detect a leak, set off an alarm in the control room, automatically isolate the control room HVAC system air intake and discharge ducts, and set the control room in the recirculation mode. The applicants will evaluate the effects of accidental releases of toxic materials stored on site on the control room habitability in accordance with RG 1.78 and will present the analysis in the FSAR. On the basis of the applicants' commitments to conform to RG 1.78 and the inclusion of sodium, sodium combustion products, hydrofluoride acid, ammonia, and chlorine detectors in the control room HVAC intake, which will effect automatic isolation of the control room and initiation of the recirculation mode of the HVAC system, the staff concludes that the proposed system meets the intent of RGs 1.78 and 1.95 and is acceptable.

6.3.3 Control Room Habitability System

The function of the control room habitability system will be to supply non-radioactive air to the control room after a design-basis accident (DBA) and to slightly pressurize the control room to a minimum of 1/8 in. water gage. This system will permit operating personnel to remain in the control room following a DBA. The control room habitability system will be a redundant system, with each system having an intake design capacity of 8,500 ft^3 per minute of air and recirculating design capacity of 8,500 ft^3 per minute of air. Each system will contain the following components: prefilter, a HEPA filter section, carbon adsorber, a final HEPA filter section, and a fan. The equipment and components will be designed to Quality Group C and seismic Category I and will be located in a seismic Category I structure.

On the basis of the above considerations, the applicants have demonstrated that the control room will adequately protect control room operators and remain habitable in accordance with NUREG-0737, Item III.D.3.4.

6.4 Cell Liners

The evaluation of the CRBR cell liners as an engineered safety feature can be found in Sections 3.8.3 and 9.13 of this SER.

6.5 Engineered Safety Feature Filter Systems

6.5.1 Introduction

Section 6.5 of PSAR contains information pertaining to engineered safety feature (ESF) filter systems, their design bases, and applicable acceptance criteria.

6.5.2 Acceptance Criteria

The staff has reviewed the applicants' design, design criteria, and design bases for the ESF filter systems for the CRBRP. The acceptance criteria used as the basis for the staff's evaluation are set forth in Section II of SRP Section 6.5.1 (NUREG-0800), with the exception of the general design criteria of Appendix A to 10 CFR 50. The acceptance criteria include the applicable CRBR PDC; American National Standards Institute (ANSI) Std. N509-1980, "Nuclear Power Plant Air Cleaning Units and Components," and ANSI Std. N510-1980, "Testing of Nuclear Air Cleaning Systems." Guidelines for implementation of the acceptance criteria are provided in the ANSI standards, regulatory guides, and other documents identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the proposed ESF filter systems meet the requirements of 10 CFR 50.

6.5.3 Method of Review

The CRBRP will have three ESF filter systems: the annulus filtration system (Section 6.2.3 of this SER), the reactor service building filtration system (Section 6.5.4 of this SER), and the control room habitability system (Section 6.3 of this SER). Each of these systems was reviewed in accordance with the SRP.

6.5.4 Reactor Service Building Filtration System

The function of the reactor service building filtration system will be to collect and process the bypass leakage that enters the reactor service building atmosphere from the reactor containment building following a design-basis accident. In addition, the system will be designed as a partial recirculation system to maintain a slight negative pressure in the reactor service building following a design-basis accident. The system will provide a redundant train.

Each train will have a design capacity of 18,000 ft^3 per minute of air and will include the following components: a moisture separator, heating coil, prefilter, a HEPA filter section, adsorber, final HEPA filter section, and a fan. The equipment and components will be designed to Safety Class 3 and seismic Category I and will be located in a seismic Category I structure.

On the basis of the system description in the PSAR, the staff has determined that the filtration system of the reactor service building will be designed to

be consistent with CRBR PDC 53 and 56. The staff concurs with the applicants' system decontamination efficiencies of 95% for elemental and organic iodides for the adsorbers and 99% for particulates for the HEPA filters.

The staff has determined that the filtration system of the reactor service building will be designed in accordance with the guidelines of RG 1.52 and will be capable of controlling the release of radioactive materials in gaseous effluents following a postulated design-basis accident.

6.5.5 Evaluation Findings

The staff concludes that the design of the ESF atmosphere cleanup systems, including the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated DBA, will be acceptable. The applicants have met the requirements of CRBR PDCs 17, 49, and 53 by providing ESF atmosphere cleanup systems for the control room, annulus, and reactor service building. The applicants have met the requirements of CRBR PDC 50, 51, and 56 by providing a program for inspecting and testing the ESF atmosphere cleanup systems and monitoring radioactive materials in effluents from these systems. In meeting these regulations, the applicants have demonstrated that the design of the ESF atmosphere cleanup systems will meet the guidelines of RG 1.52 and the ANSI N509 and N510 industry standards. The staff has reviewed the applicants' system descriptions and design criteria for the ESF atmosphere cleanup systems. On the basis of its evaluation, the staff finds that the proposed ESF atmosphere cleanup systems are acceptable and the filter efficiencies presented by the applicants are appropriate for use in accident analyses.

6.6 Catch Pan, and Sodium Fire Suppression Decks

The evaluation of the CRBR catch pans, and fire suppression decks can be found in Section 9.13.2 of this SER.

6.7 Ex-Vessel Storage Tank Cooling System and Siphon Breaks

The evaluation of the CRBR ex-vessel storage tank (EVST) cooling system and siphon breaks can be found in Section 9.1 of this SER.

6.8 Ex-Vessel Storage Tank Guard Vessel

The evaluation of the EVST guard vessel can be found in Section 9.1 of this SER.

6.9 Primary Heat Transport System Guard Vessels

The evaluation of the PHTS guard vessels can be found in Sections 5.2.1.3 and 5.3.2.4 of this SER.

6.10 Steam Generator Cell Aerosol Detectors in the Heating, Ventilating, and Air Conditioning System

The evaluation of the CRBR steam generator cell aerosol detectors in the heating, ventilating, and air conditioning system can be found in Sections 7.3.1 and 9.13.2 of this SER.

6.11 Residual Heat Removal System

The evaluation of the CRBR residual heat removal system can be found in Section 5.6 of this SER.

6.12 Leak Detection System

The evaluation of the CRBR liquid metal leak detection system can be found in Section 5.7 of this SER.

6.13 Sodium-Water Reaction Pressure Relief System

The evaluation of the CRBR sodium-water reaction pressure relief system can be found in Section 5.5.2.2 of this SER.

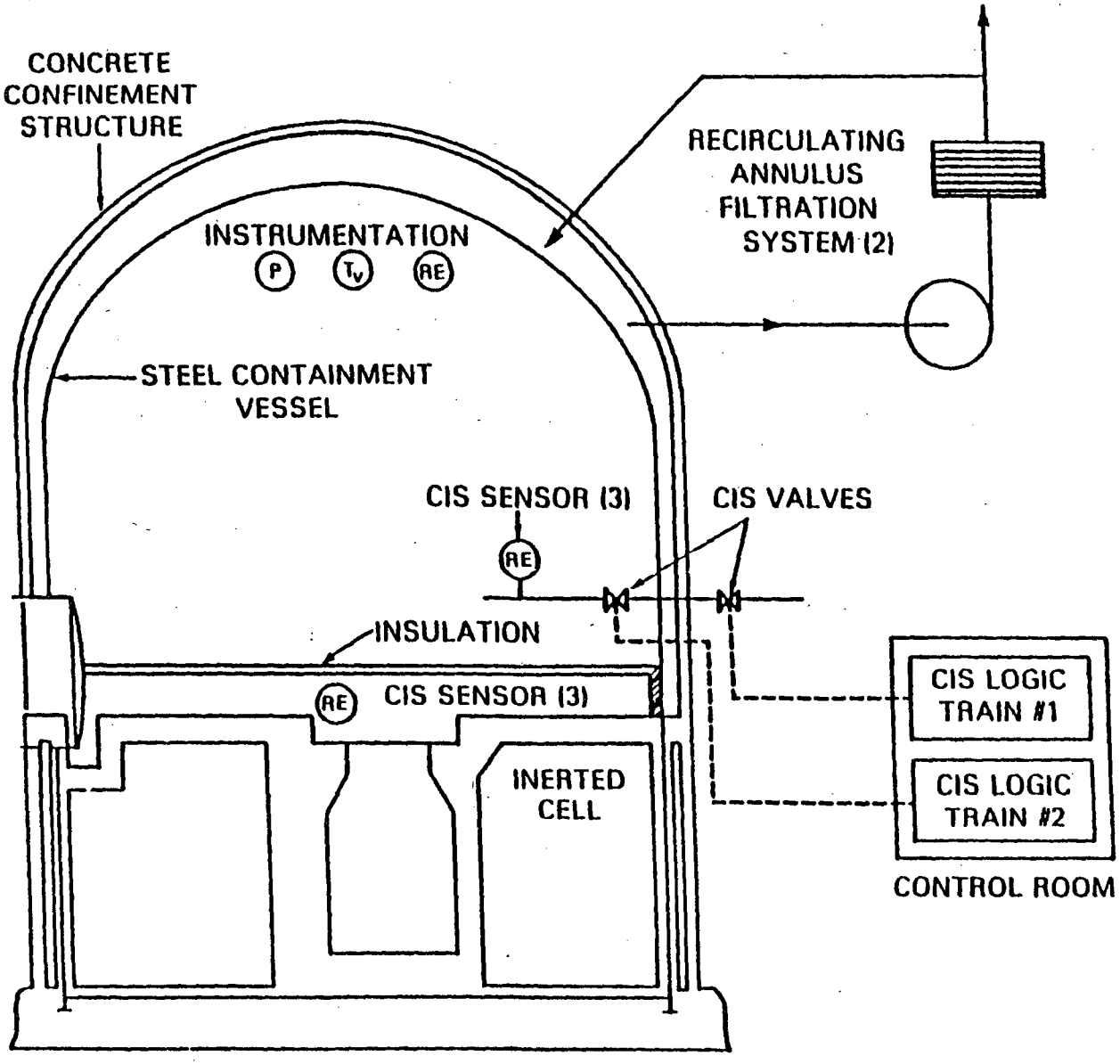


Figure 6.1 Clinch River Breeder Reactor Plant containment system

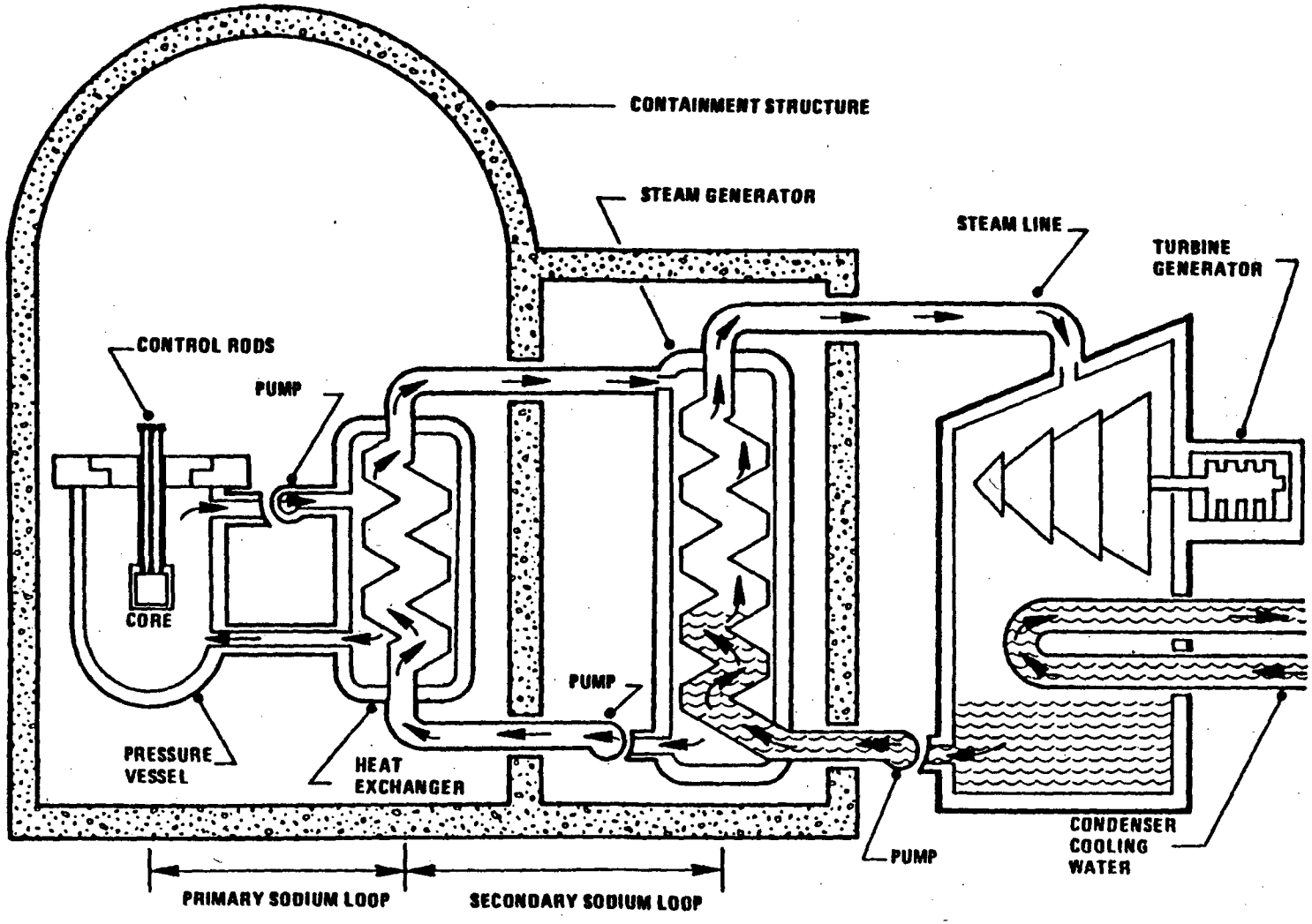


Figure 6.2 Schematic of the heat transport system

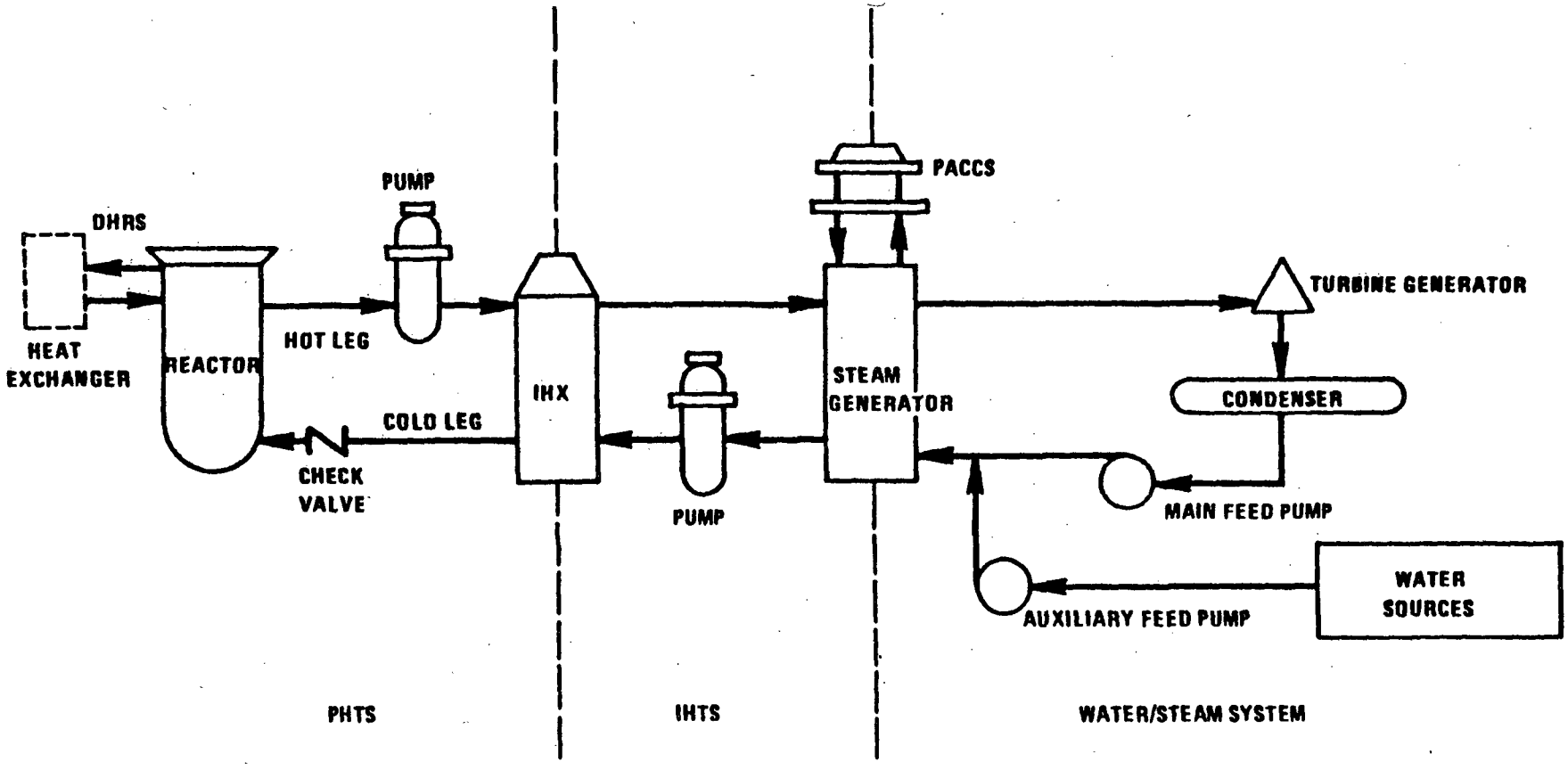


Figure 6.3 Schematic of one of the three heat transport loops plus the direct heat removal service

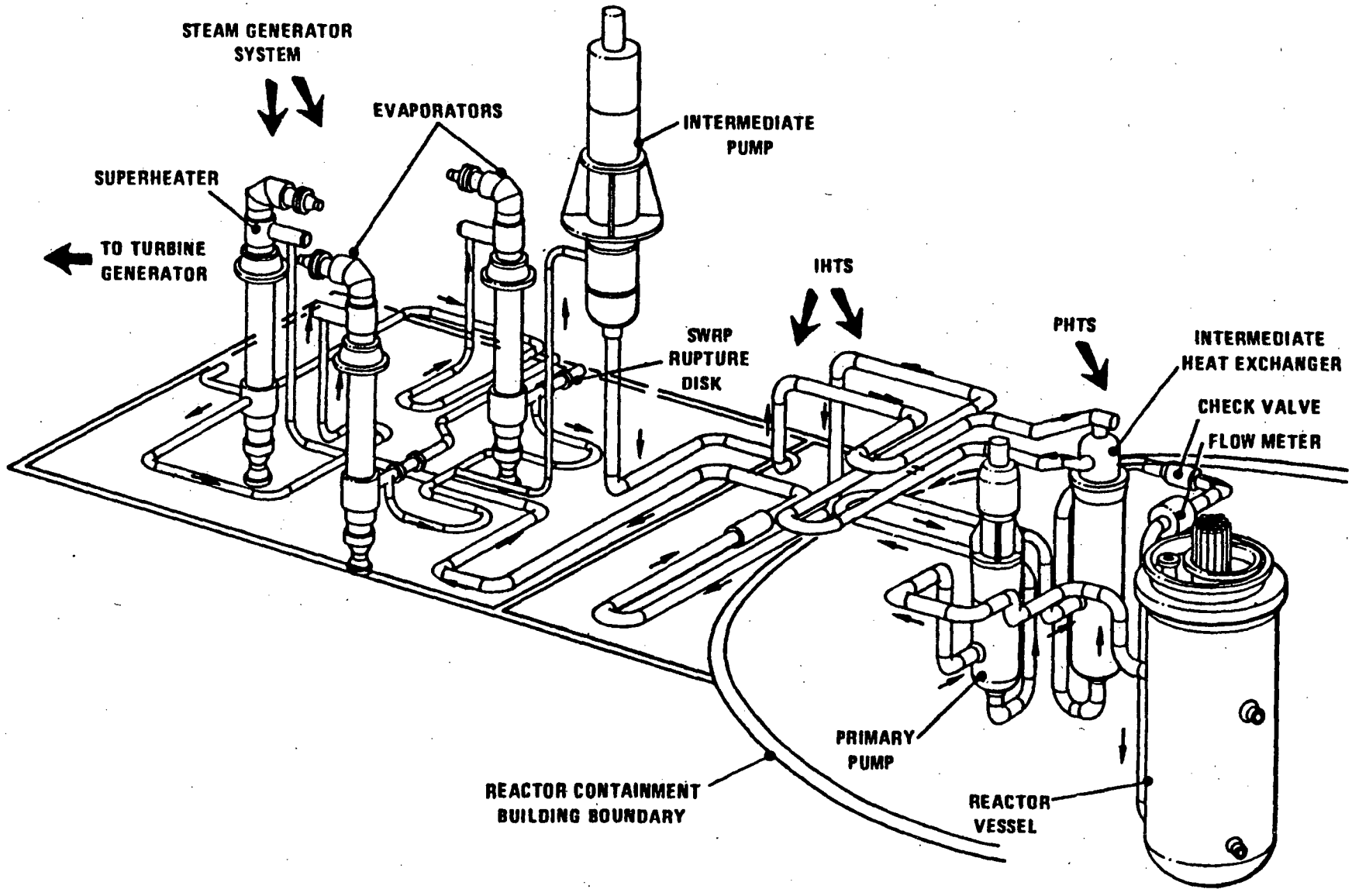


Figure 6.4 Arrangement of one loop of heat transport system

Table 6.1 List of engineered safety features in Clinch River Breeder Reactor Plant proposed by the applicants and the staff

Applicants	Staff	SER section(s)
Reactor confinement/containment	Same	6.2
Reactor containment isolation system	Same	6.2.4, 7.3.2.2
Reactor containment building annulus filtration system	Same	6.5.4.1
Habitability systems	Same	6.5.4.3, 7.3.2.5
Cell liner system	Same	3.8.3, 7.3.2.8, 9.13.2
Reactor service building filtration system	Same	6.5.4, 7.3.2.3
Catch pan and fire suppression deck	Same	3.8A, 7.3.2.8, 9.13.2
_____	Ex-vessel storage tank (EVST) cooling system siphon breakers	9.1
_____	EVST guard vessel	9.1
Primary heat transport system guard vessels	Same	5.2, 5.3, 7.3.2.6
Steam generator building aerosol release mitigation system	Same	7.3.2.4, 9.13.2
Residual heat removal system	Same	5.6, 7.3.2.7
_____	Na leak detection system	5.7
_____	Sodium-water reaction pressure relief system	5.5.2.2

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

Section 7.1 of the Clinch River Breeder Reactor (CRBR) Preliminary Safety Analysis Report (PSAR) contains information pertaining to safety-related and power-generation (nonsafety-related) instrumentation and control systems, their design bases, and applicable acceptance criteria.

7.1.1 Acceptance Criteria

The staff and its consultants (EG&G) have reviewed the design criteria for the instrumentation and control systems for CRBR as identified in the CRBR PSAR Chapter 7 as amended through Amendment No. 72. The acceptance criteria used as a basis for this review are the CRBR Principal Design Criteria and the criteria set forth in Table 7.1, "Acceptance Criteria for Instrumentation and Control Systems Important to Safety," of the Standard Review Plan (SRP), NUREG-0800.

The acceptance criteria listed in Table 7.1 of the SRP include the applicable general design criteria (Appendix A to 10 CFR 50), and IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (10 CFR 50.55a(h)). Guidelines for implementation of the requirements of the acceptance criteria are provided in the IEEE standards, regulatory guides, and branch technical positions (BTPs) identified in Section 7.1 of NUREG-0800.

7.1.2 Method of Review

The review of the CRBR instrumentation and controls important to safety was done in accordance with the SRP Section 7.1 through Appendix 7-A, Revision 2, July 1981 (NUREG-0800). It has been found that although the Standard Review Plan emphasizes criteria for light-water reactors (LWRs), these same criteria are substantially applicable to liquid metal reactors (with the exception of the special consideration that needs to be taken into account with sensing sodium parameters and handling sodium, the principles of design are similar to those for instrumentation and control systems used in LWRs). The staff's basic position was that the CRBR should achieve a level of safety comparable to current-generation LWR plants, according to all current criteria for evaluation, and that the design approaches to accomplish the required level of safety be similar or analogous to current LWR practice.

Since the applicants are applying for a construction permit, detailed reviews of system drawings (logic diagrams, functional control diagrams, electrical schematics, and elementary diagrams) and the tracing of circuits from sensor to the actuated devices was not performed as would have been done for an operating license. However, the staff and its consultant did a drawing audit for several days at the applicants' Waltz Mill, Pennsylvania, engineering offices.

The staff review concentrated on identifying areas of concern found in reviewing the PSAR and areas of concern (found in past reviews of LWR plants) that

were applicable to CRBR. After these areas of concern were identified, the staff transmitted a series of questions to the applicants and held several meetings to discuss the applicants' responses to the questions and other areas of concern regarding the design of the CRBRP instrumentation and control systems. These meetings were held with the applicants and their subcontractors (primarily Westinghouse).

7.1.3 General Findings

The applicants have identified the instrumentation and control systems important to safety. In addition, the applicants have identified the guidelines, consisting of the regulatory guides and industry codes and standards, which are applicable to the systems. The acceptance criteria and guidelines identified by the applicants are provided in Table 7.1-1 of the PSAR.

Based on its review of Section 7.1 of the applicants' PSAR, the staff concludes that the implementation of the identified acceptance criteria and guidelines satisfies the requirements of PDC 1 (GDC 1), "Quality Standards and Records," with respect to the design, fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed. The staff finds that the nuclear steam supply system (NSSS) and the balance-of-plant (BOP) instrumentation and control systems important to safety, addressed in Section 7.1 of the applicants' PSAR, will satisfy the requirements of PDC 1 (GDC 1) and, therefore, are acceptable.

7.1.4 Specific Findings

In review of the applicants' design, design criteria, and design bases for the instrumentation and control systems identified in Chapter 7 of the CRBR PSAR, the following general issues were addressed:

(1) The format of Chapter 7 of the PSAR followed that specified in RG 1.70; however, the applicants had not provided the specific design-basis information in the exact format of RG 1.70. After a meeting with the applicants, it was resolved that although the applicants had not formatted the information according to the recommendations of RG 1.70, the information is contained within the PSAR.

(2) The applicants will review the PSAR and provide consistent definitions of "important to safety" and "safety related" in conformance with the NRC position (letter, Nov. 20, 1981).

(3) The applicants will include in the PSAR wording specifically stating that physical separation between conduits, penetrations, or racks containing redundant instrument channels shall meet the requirements of RG 1.75. This statement would be applicable for "safety-related" systems.

(4) The applicants will include in the PSAR wording specifically stating that physical separation of wiring inside safety-related panels shall meet the requirements of RG 1.75 and the guidance of IEEE Std. 384-1974.

(5) Documentation of the design provisions for conducting response time tests of BOP and NSSS protection systems in accordance with RG 1.118 was provided.

(6) A general description of NSSS and BOP safety-related setpoint methodology was provided. This description verified that environmental allowances will be based on the highest value determined in qualification testing.

(7) The applicants discussed and verified that CRBR has an established seismic qualification program for seismically qualifying the safety-related instrumentation.

(8) The applicants revised the PSAR to indicate the applicability of the updated regulatory guides and IEEE standards to the CRBR design.

7.1.4.1 Open Items

The staff's conclusions noted herein are applicable to the instrumentation and control systems important to safety with the exception of the open items listed below. The staff will review these items and discuss their resolution in a supplement to this SER. The applicable sections of this report which address these items are indicated in parentheses following each open item.

(1) Regulatory Guide 1.75 (7.2.2.6)
(2) Plant Protection System Monitor (7.2.2.7)
(3) Solid-State Programmable Logic System (7.3.2.4)

7.1.4.2 Confirmatory Items

In a number of areas the applicants have committed to make design changes or to provide additional documentation to address concerns raised by the staff during its review. Based upon information provided during meetings and discussions with the applicants, the staff has concluded that the technical issues have been resolved in an acceptable manner. However, the applicants must formally provide final documentation of these items. Confirmatory items will not be discussed in a supplement to this SER unless an unanticipated problem is found. The sections of this report which address items are noted in parentheses.

(1) Control and Protection System Interaction (7.2.2.3)
(2) Sodium Heating Systems (7.3.2.3)
(3) Sodium-Water Reaction Pressure Relief System (7.3.2.4)
(4) Outlet Steam Isolation Instrumentation and Control System (7.4.2.4)
(5) Fuel-Handling and Storage Safety Interlocks System (7.6.2.2)

7.1.4.3 Operating License Review Items

In a number of areas, the staff has identified specific design details, procedures, or tests which are to be implemented or completed before the staff operating license review. The staff has determined that the criteria for these items are adequately defined for a construction permit. However, the acceptability of the applicants' design depends on the successful implementation or completion of these items. The items are listed below with the applicable section of the report which addresses each one indicated in parentheses:

(1) Single-Failure Criterion for Power Supplies and Channel Independence (7.2.2.1)

(2) Primary and Secondary Isolation Devices (7.2.2.2)

(3) Reactor Shutdown System Response Times (7.2.2.9)

(4) Containment Isolation System (7.3.2.1)

(5) Engineered Safety Features Setpoints (7.3.2.2)

(6) Sodium-Water Reactor Pressure Relief System (7.3.2.7)

(7) IE Bulletin 80-06, "Engineered Safety Feature Reset Control" (7.3.2.9)

(8) Protected Air-Cooled Condenser Control (7.4.2.3)

(9) Source Range Subsystem (7.5.2.1)

(10) Delayed Neutron Detection System (7.5.2.5)

(11) TMI Item II.F.3, Postaccident Monitoring Instrumentation (7.5.2.8)

(12) Inoperable Status Monitoring System (7.5.2.9)

(13) Loss of Non-Class 1E Instrumentation and Control Power Bus During Operation (IE Bulletin 79-27) (7.5.2.10)

(14) Radiation Monitoring System (7.6.2.3)

(15) Qualification of Control Systems (7.7.2.2)

7.1.4.4 TMI-2 Action Plan Items

The TMI Action Plan was developed to provide a comprehensive and integrated plan for actions judged necessary to correct or improve the regulation and operation of light-water reactors based on the experience from the accident at TMI-2. NUREG-0718, Revision 1, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," and NUREG-0737, "Clarification of TMI Action Plan Requirements," provide guidance for these requirements. The Action Plan items addressed in this section, appropriately modified for CRBR, are listed below. The specific section of the report addressing each item is indicated in parentheses.

Action Plan Item

(1) TMI Item II.D.3, Relief and Safety Valve Indication (7.5.2.11)

(2) TMI Item II.E.1.2, Auxiliary Feedwater System Automatic Initiation and Flow Indication (7.4.2.2)

(3) TMI Item II.F.3, Instrumentation for Monitoring Accident Conditions (RG 1.97) (7.5.2.8)

7.1.4.5 Fire Protection Review

The review of remote shutdown capability discussed in Section 7.4 of this report covered the compliance with PDC 17 (GDC 19), "Control Room." The aspects of remote shutdown related to fire protection and the review for conformance to 10 CFR 50, Appendix R and BTP CHEB-9.5-1 was performed by the staff and is discussed in Section 9.13 of this SER.

7.2 Primary and Secondary Reactor Shutdown System

7.2.1 Reactor Shutdown System Description

The reactor shutdown system (RSS) consists of the primary and secondary reactor shutdown systems, either of which will be capable of reactor shutdown and heat transport system initiation. The primary and secondary RSSs will each consist of eight subsystems. Each protective subsystem will consist of three separate redundant instrument channels. Each instrument channel comparator will be designed to provide three isolated outputs. For the primary RSS, the three isolated outputs are input to three 2-out-of-3 logic modules, each of which provides an input to the final 1-out-of-24 logic train. This final logic will also be arranged in a 2-out-of-3 coincidence to actuate the scram breakers. The secondary RSS comparator outputs are input to three 1-out-of-16 logic modules. These logic modules then provide input to solenoid-operated pneumatic scram valves in each secondary control rod configured to perform the final 2-out-of-3 actuation logic to release the secondary rods.

Operating bypasses will allow RSS functions to be bypassed during main sodium coolant pump startup and ascent to power. The bypasses cannot be actuated unless certain permissive conditions exist which ensure that adequate protection will be maintained while these protective functions are bypassed. Bypass permissives are part of the RSS, and are to be designed according to RSS requirements. Continuous local and remote indication of bypassed instrument channels will be provided by the applicants in conformance with RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

The design for both reactor shutdown systems will include provisions for on-line testing, with overlapping testing being used for the primary RSS. Insertion of a test signal into a secondary RSS channel will cause the entire train channel (comparator, logic, and scram solenoid valves) to trip.

The instrumentation used by the RSS to detect the occurrence of off-normal plant conditions will include:

(1) Neutron Flux

The primary RSS will use three compensated ionization chamber power-range nuclear sensors evenly spaced around the reactor vessel. The secondary RSS will use three fission chamber wide-range nuclear sensors evenly spaced around the reactor vessel.

(2) Reactor Inlet Plenum Pressure

The primary RSS will use six pressure detectors, two per primary heat transport system (PHTS) loop, located in the elevated primary cold-leg piping. Each set of two detectors will comprise an instrument channel. The outputs of the two detectors in each loop will be auctioneered. The resultant output signal will be provided to the channel comparator.

(3) Sodium Pump Speed

The primary RSS will use three redundant tachometers per primary and intermediate HTS pump to measure pump speed.

(4) Sodium Flow

The secondary RSS will use six permanent magnet flowmeters to measure HTS sodium flows. One flowmeter will be located in each of the primary and intermediate cold legs. Each flowmeter will provide three redundant measurements of loop flow.

(5) Reactor Vessel Sodium Level

The primary RSS will use four sodium-level detectors evenly spaced within the reactor vessel. Three of these detectors will provide redundant active signals to the RSS. The fourth detector will be used as a spare.

(6) Undervoltage Relay

The secondary RSS will use three undervoltage relays, one per coolant loop pump bus. The undervoltage relays will be located on the HTS pump buses.

(7) Steam Flow

The primary RSS will use three redundant steam mass flow signals per loop. Each steam supply loop will have one Venturi flowmeter, three differential pressure sensors, three temperature sensors, and three pressure sensors located between the superheater exit and the main steam header. Three redundant steam mass flow signals will be generated by pressure and temperature compensation of the Venturi flowmeter analog signal.

(8) Feedwater Flow

One Venturi flowmeter, three differential pressure sensors, and three temperature sensors per steam generator loop will supply the primary RSS with three redundant temperature-compensated feedwater mass flow signals.

(9) IHX Primary Outlet Sodium Temperature

The primary RSS will use three redundant thermocouples, mounted in three thermowells per loop to measure the sodium temperature in the primary cold leg.

(10) Steam Drum Level

The secondary RSS will use three redundant reference column level sensors to determine the water level in each steam drum. The level sensor will be density compensated.

(11) Evaporator Outlet Sodium Temperature

The secondary RSS will use three redundant thermocouples per loop, mounted in three thermowells. These thermocouples will be provided to monitor the sodium temperatures in each intermediate cold leg.

(12) Underfrequency Relay

The primary RSS will use three underfrequency relays, one per coolant loop pump bus. The underfrequency relays will be located on the HTS pump buses.

(13) Sodium-Water Reaction

The secondary RSS will use three redundant pressure sensors located in the reaction products vent line immediately downstream from each rupture disc to detect if the rupture discs have blown.

The primary RSS will consist of the high flux subsystem, the flux-delayed flux subsystem, the flux-pressure subsystem, the pump speed mismatch subsystem, the HTS pump frequency subsystem, the pump electronics, the reactor vessel level subsystem, the steam-feed flow mismatch subsystem, and the IHX primary outlet temperature subsystem. The secondary RSS will consist of the modified nuclear rate subsystem, the flux-total flow subsystem, the startup flux subsystem, the flow mismatch subsystem, the steam drum level subsystem, the evaporator outlet sodium temperature subsystem, the sodium-water reaction subsystem, and the HTS pump voltage subsystem.

The reactor shutdown systems are designed so that, once initiated, a protective action at the system level must go to completion. Return to normal operation will require manual reset by the operator because the primary RSS scram breakers or secondary scram latch circuitry must be manually closed following trip.

The reactor shutdown system includes means for manual initiation of each protective action at the system level with no single failure preventing initiation of the protective action. Manual initiation depends on the operation of a minimum of equipment, because the manual trip directly operates the scram breakers of the solenoid scram valve power supply.

7.2.2 Specific Findings

The staff reviewed the CRBR primary and secondary reactor shutdown systems (RSSs) for conformance to the requirements of the applicable principal design criteria and the guidelines of the applicable regulatory guides and industrial codes and standards outlined in SRP Section 7.2, Parts II and III. From an evaluation of the design criteria invoked for the primary and secondary RSSs by the applicants, the staff concluded that the applicants have adequately identified the criteria applicable to the primary and secondary RSSs.

The proposed design of the primary and secondary RSSs has been reviewed based on the information in the CRBR PSAR. The applicants have included in the design of the primary and secondary RSS subsystems provisions for isolation between safety-related* and nonsafety-related instrumentation, bypass capabilities for subsystem channels which are allowed to be bypassed for startup or operation, channel redundancy, and on-line instrument channel testing, calibration, and maintenance. The applicants have also established an environmental qualification program for all safety-related instrumentation and control systems.** The design of subsystem channels includes means for indicating channel status, that is, whether it is in the tripped state, nontripped state, bypassed mode, or inoperable. In meeting the single-failure criteria, it has been common practice to utilize common sensors between a single channel of different subsystems and to utilize a common bypass permissive signal. This type of design practice meets the single-failure criteria because failure only affects a single channel of the subsystems leaving two unaffected channels which can provide protective action. A common sensor has been used in the design of the flux-delayed flux subsystem, the flux-pressure subsystem, and the high flux subsystem, and a common permissive bypass signal has been used in the design of the pump speed mismatch subsystem and the HTS pump frequency subsystem.

7.2.2.1 Single-Failure Criterion for Power Supplies and Channel Independence

The primary and secondary RSS power supply distribution has been designed so that each of the three subsystem instrument channels will receive power from a separate power division; that is, one channel of a subsystem will be powered by division 1 power, another channel by division 2 power, and the remaining channel by division 3 power. The concern of the staff is that the division 1 power for the primary RSS instrument channels will come from the same vital bus as the division 1 power for the secondary RSS instrument channels, since there will be only three Class 1E sources for CRBR. The applicants have, in a meeting, provided one-line diagrams of the electrical distribution depicting the distribution from the vital buses to the primary and secondary instrument channels and the primary and secondary logics. Discussion was provided on the isolation and overvoltage/overcurrent protection used within the division power supplies at the channel level. This design is carried through on all safety systems powered by the three vital buses. Because it is unique to have two reactor shutdown systems with the isolation and protection being provided in the shared power supplies, the staff has accepted a single division to power single channels of primary and secondary instrumentation.

However, the staff remains concerned that a fault would propagate from the primary RSS through the common portion of the shared power supply to the secondary RSS, from the secondary RSS through the common portion of the shared power supply to the primary RSS, or from the shared power supply to both the primary and secondary RSS, thus disabling both the primary and secondary RSS.

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*Throughout this report the term safety-related instrumentation and control system(s) is used synonymously with the term Class 1E equipment.

**The environmental qualification will be accomplished through a disciplined program discussed in Reference 13 of PSAR Section 1.6, "CRBRP Requirements for Environmental Qualification of Class 1E Equipment."

To alleviate this concern, the applicants have committed to perform a test to verify that the isolation capability will be adequate between the primary and secondary RSSs for all credible fault and transient conditions. The above commitment resolves this issue for the CP review. However, the applicants are required to submit the following for staff review and approval within 1 year following issuance of the CP.

(1) description of each planned test phase--this should include maximum fault qualification--both common mode and transverse mode

(2) sketches showing the test mockup and test points for application of test voltages

(3) values of test voltages

(4) justification of maximum credible voltages for the application on CRBR

(5) test acceptance criteria

7.2.2.2 Primary and Secondary Reactor Shutdown System Isolation Devices

Various instrumentation and control system circuits in the plant (including the reactor shutdown system (RSS), engineered safety features actuation system, and instrument power supply distribution system) will rely on devices to provide electrical isolation capability in order to maintain the independence between redundant safety circuits, between redundant safety circuits and other safety circuits, and between safety circuits and nonsafety circuits.

The staff questioned the applicants as to (1) the types of isolation devices that will form the Class 1E boundary for interfaces between the safety circuits and nonsafety circuits, (2) the acceptance criteria used for the above-identified isolation devices, and (3) the type of testing that will be conducted on these isolation devices to ensure adequate protection against electromagnetic interference (EMI), short-circuit failures, voltage faults, and/or surges.

The applicants identified five types of isolation devices that will be used in the nuclear steam supply shutoff system. These are light-emitting diode (LED)-phototransistor pairs, transformer/LED-phototransistor pairs, relays, buffers, and isolation transformers. For the balance-of-plant systems, either optical (phototransistor pairs) isolators or relays will be used. All of the above isolation devices will be qualified to Class 1E qualification criteria.

From discussions with the applicants and a review of the criteria used for the isolation devices, the staff has concluded that its concerns as to the isolation methods employed at CRBR are resolved. The applicants have committed to provide a description, before the OL review, of the test program and acceptance criteria for demonstrating that faults on the nonsafety side of the above isolators will not propagate in such a way as to compromise the RSS, and that faults cannot propagate across the isolators in such a way as to compromise the independence of safety circuits. The staff will verify the results of the test program during the OL review. This test program should include line-to-line faults and surges on the terminals of the isolators (e.g., faults and surges across output terminals).

7.2.2.3 Control and Protection System Interaction

Section 4.7 of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and CRBR PDC 22 (GDC 24), "Separation of Protection and Control Systems," discuss requirements regarding control and protection system interactions.

The CRBR design uses plant protection system sensors for control functions. The applicants have stated that there are certain design advantages to be gained from the practice of "sharing" sensors. For example, redundant control signals increase the availability of information for plant control; the number of sensor penetrations is reduced; the shared channels are subject to protection system maintenance and test schedules; and the use of shared channels provides for greater and more frequent operator surveillance of the behavior of protection system sensors. The disadvantage to the practice of "sharing" sensors is that the instances for adverse control and protection system interactions are increased.

The CRBR design prevents undesired control and protection system interactions by buffering all signals from the protection system with Class 1E qualified isolators (Section 7.2.2.2 of this SER), locating these isolation devices within the protection system boundaries, providing median selectors in the control system that prevent incorrect control action on a single channel failure, monitoring redundant plant protection system channels to alert the operator to signal mismatches, and by not feeding the control system signals into the protection system.

The applicants recognize that in a limited number of situations, during testing or maintenance of a channel which supplies signals to both the protection and control systems, a single sensor failure could initiate plant control actions requiring protective action and simultaneously prevent proper protective channel response. In these situations, the redundant functions in the alternate RSS will provide the protection as required by Section 4.7.3 of IEEE Std. 279-1971. However, for normal operation (non-test), the primary and secondary RSSs individually meet the criteria of IEEE Std. 279-1971.

The likelihood of potentially undesirable situations is limited by the use in the control system of a median select arrangement which precludes a response to an abnormally high or low signal. The staff questioned the applicants regarding the qualification criteria used for the median selector. The applicants stated that the median selector is being designed and fabricated under an ANSI N45.2 quality assurance program and, thus, meets the 10 CFR 50, Appendix B, requirements. Also, the median selector is being designed to be functional after an operating basis earthquake (OBE). In addition, most failures of the median selector are self-annunciating (i.e., abnormal plant response is detected by the operator). The staff will require that appropriate Technical Specifications be applied to the median selector including a requirement for periodic testing of the median selector. The details of these Technical Specifications will be established during the OL review.

The applicants provided a list of the shared sensors (primary RSS in all cases), the protection system functions, and the control system functions. An analysis was then performed using the failure of a channel having a sensor common to

control and protection as the initiator, while another channel is in test. The resulting control system response and the resulting protection system response were then discussed for each situation in which a sensor is shared. For all cases, the CRBR control and protection system design leaves intact redundant protection channels which will be capable of providing the protective action even when degraded by a second random failure.

The sharing of components does not lead to a situation in which a single event both initiates an incident through plant control system malfunctions and prevents reactor shutdown system action. When a malfunction in a shared component occurs and action is required by the shutdown systems, the results will be enveloped by the safety analysis of a transient associated with normal operation of the shutdown systems. In addition, this performance will not be affected by the removal from service of a shutdown system channel for testing and the simultaneous presence of an additional random single failure in the shutdown system.

The staff has reviewed the CRBR design with respect to control and protection system interactions and concludes that it meets IEEE Std. 279-1971, Section 4.7.3, and is acceptable. The concern regarding a single failure causing multiple control system failures is discussed in Section 7.7.2.1 of this SER.

The applicants are required to revise the information in the control and protection system interface portion of PSAR Section 7.2.2 to document that when testing a protection channel, and a part of the protection system shared with the control system fails, the damage levels are enveloped by those of the anticipated operational occurrences even when the most limiting additional random failure within the reactor shutdown system is assumed. The staff will confirm that final acceptable documentation is provided by the applicants.

7.2.2.4 Reactor Shutdown System Bypass Capability

The applicants have modified the PSAR in reference to two-loop operation. Bypass features included within the primary and secondary RSS hardware for two-loop operation will be deactivated during all three-loop operating modes so that the three-loop operating configuration cannot be affected by the bypass feature either by operator action or by two-loop hardware failure. The staff finds this modification acceptable.

7.2.2.5 Primary and Secondary Reactor Shutdown System Diversity

It is the staff's position that because of the inherent differences in nuclear characteristics between LWRs and CRBR, two independent, diverse, redundant rapid shutdown systems should be provided for CRBR. This will reduce the likelihood of a failure to scram. Within the design of the primary and secondary RSSs, it was a concern of the staff that the primary RSS was not totally diverse from the secondary RSS.

As a result of discussions and meetings that were held, the applicants expanded the description in the PSAR by providing a discussion of the diverse design features of the primary and secondary RSSs. The diversity extends from the sensors used as input to the two systems, through the logic utilized to the actuation devices required to trip the two different control rod designs.

Table 7.1 presents the principal diverse design features between the primary and secondary RSSs. The diverse methods employed in the CRBR RSS design satisfy the concerns identified by the staff relating to the diversity of the primary and secondary RSSs. The staff concludes that this part of the RSS design meets PDC 24 and, therefore, is acceptable.

7.2.2.6 Regulatory Guide 1.75

The staff has concluded that the redundant instrument channels (1, 2, and 3) in the primary RSS will be physically separated from one another in accordance with the provisions of RG 1.75, "Physical Independence of Electric Systems." Also, redundant instrument channels (1, 2, and 3) in the secondary RSS will be physically separated from one another in accordance with RG 1.75. CRBR requirements for separation are delineated in the PSAR Sections 7.1.2.2 and 8.3.1.2.14.

However, the staff is concerned that physical separation within the same divisions of the primary and secondary (i.e., division 1 primary and division 1 secondary) RSSs will not meet RG 1.75. It is the CRBR Program Office's position that because of the inherent nuclear characteristics between LWRs and CRBR (i.e., positive sodium void coefficient and potential for recriticality), two independent, diverse, redundant shutdown systems should be provided for CRBR to reduce the likelihood of a failure to scram.

PDC 24 states in part that two independent reactivity control systems of different design principles shall be provided. This criterion pertains to both the sensing portion of the system (including its associated electronics and logic) and the mechanical portion of the system. This is a more conservative criterion than applied to LWRs (GDC 26).

The criteria for physical separation within the same divisions of the primary and secondary RSSs were clarified after reviewing the applicants' response to staff questions and during discussions with the applicants.

The applicants indicated that the physical separation between the same division of the primary and secondary RSSs will not always meet the provisions of RG 1.75. The separation criteria meet the provisions of RG 1.75 in nonhazard areas (control room and cable spreading room). For wiring internal to safety-related cabinets, the provisions of IEEE Std. 384-1974 and RG 1.75 will be met.

However, in other areas of the plant (e.g., reactor containment building), the primary and secondary RSS equipment and cables will be located in the same hazard area and, therefore, will not meet the provisions of RG 1.75. In addition, the applicants indicated three other exceptions where RG 1.75 separation is not maintained.

First, both the primary and secondary RSSs will use the same instrument ground (plant will have one instrument ground). The applicants stated that physically separated cables will be run to this ground from each instrument channel in the primary and secondary RSSs to eliminate the common mode failure of the ground cable.

Second, the primary and secondary RSSs will obtain power from the same three power divisions. However, each power division will utilize separate Class 1E

uninterruptible power supplies, including batteries, rectifiers, and inverters, which are designed to prevent transient electrical power fluctuations. The commonality of the power supplies between the primary and secondary RSSs is discussed in Section 7.2.2.1 of this SER.

Third, both the primary and secondary RSSs trip each heat transport loop (HTS) pump breaker through separate trip coils. The separate trip coils and relay logic used to drive them provide sufficient independence to prevent failure propagation from the pump trip circuits to the control rod trip circuits of either shutdown system.

The staff position is that the separation between the same division (channel) of the primary and secondary RSSs should meet the provisions of RG 1.75 (especially in hazard areas) unless the applicants justify the design criteria used in those areas where the separation criteria do not meet the provisions of RG 1.75. A similar situation exists for the direct heat removal service (DHRS) and steam generator auxiliary heat removal systems (SGAHRSs). Section 7.6.2.1 of this report describes the separation criteria utilized for DHRS, SGAHRS, and between the same divisions of DHRS and SGAHRS. Section 7.6.2.1 states the staff position for the separation between DHRS and SGAHRS and asks the applicants to justify the design criteria used for DHRS and SGAHRS.

The applicants are required to provide justification(s) for the design criteria used when the physical separation does not meet RG 1.75 for the same divisions of the primary and secondary RSSs and for the same division of DHRS and SGAHRS (separation between DHRS and SGAHRS is discussed in more detail in Section 7.6.2.1 of this report).

7.2.2.7 Plant Protection System Monitor

The plant protection system (PPS) monitor will be designed to monitor the operational status of the 24 primary RSS logic trip functions, the three primary steam generator auxiliary heat removal systems logic functions, and the three SGAHRS secondary logic functions. The applicants have stated that the monitor is not the same as, but is equivalent to, the logic tester provided by some LWR designs. In comparison, the PPS monitor will have the equivalent test capability of the LWR design with the exceptions that it will not inhibit all trips from the logic train during testing, will not do testing on individual logic trains (the PPS monitor checks multiple logic trains to determine 2-out-of-3 logic capability), and will have broader testing capability (the PPS monitor checks comparator logic drivers, logic input isolators, and interconnecting field cables).

The PPS monitor is a non-Class 1E on-line test device that has test signals coupled to and from the RSS/SGAHRS initiation logic through Class 1E electro-optical isolation devices located within the RSS/SGAHRS Class 1E equipment boundary. The testing will consist of supplying an eight-step sequence of short-duration pulses to the appropriate on-line test inputs of the primary RSS and SGAHRS trip comparator or at the secondary SGAHRS interface card. These pulses will simulate all possible combinations of trip/reset conditions for that logic subsystem. The test pulse is of such short duration that it cannot trip the scram breakers or initiate SGAHRS operation. Abnormalities detected by the monitor will be displayed by 96 error indicators (showing either correct

or abnormal primary RSS/SGAHRS operation with regard to subsystem and associated logic trains) and will initiate upon detection of a logic error a remote alarm.

The PPS monitor provides for automatic and manual test sequences. When initiated, the automatic mode of operation continually tests each subsystem in numerical sequence until the automatic mode switch is actuated again or the selector switch is placed in the manual test position. The manual mode provides for two test sequences--"normal" or "repeat." When the manual mode is initiated, a complete on-line test of one logic subsystem is initiated. Upon completion of a subsystem test, the monitor does not step to the next subsystem until the manual switch is actuated again.

When in the manual mode, and placed in the "repeat" position, the monitor will allow the initiation of as many test sequences on the same subsystem as desired. To step to another subsystem while in the "repeat" mode, an "index" switch must be depressed. The monitor is so designed that the actuation of any manual mode switches will not have any effect while the monitor is in the automatic mode (reverse is also true). The monitor also provides two functions for self-checking of logic error detection circuits to ensure that the PPS monitor has not malfunctioned. The interfacing of the monitor is done with Class 1E isolation devices, and the monitor interfaces with the primary RSS only and not the secondary RSS.

The staff was concerned that the PPS monitor would decrease the independence, separation, testability and, therefore, the reliability of the RSS. After reviewing the PPS monitor design, the staff remains concerned about certain provisions of the monitor. The common tie-in point for the three primary RSS subsystem channels, the use of a non-Class 1E test device, the lack of provisions for an optional or redundant test method, and the lack of a defined method for periodically verifying correct operation of the single test device are among the staff's concerns. The staff is continuing its review of the PPS monitor design and will report its findings in a supplement to this SER.

7.2.2.8 Common Instrument Lines/Taps

The applicants were asked to identify where instrument sensors or transmitters supplying information to more than one protection channel, to both a protection channel and control channel, or to more than one control channel, will be located in a common instrument line or connected to a common instrument tap. The intent of this question was to verify that a single failure in a common instrument line or tap (such as a break or blockage) cannot defeat required protection system redundancy. Also, the staff required the applicants to identify any sensors or circuits that will be used to provide input signals to the protection system which will be located or routed through nonseismically qualified structures.

The applicants responded by stating that no redundant protection channel, protection and control channel, or control channel sensors or transmitters will be located in common instrument lines or taps. In addition, all sensors and circuits that provide input signals to the protection system will be located and routed through seismically qualified structures, will be designed to IEEE Std. 279-1971 criteria, and will be seismically and environmentally qualified. The staff has concluded that these issues are resolved.

7.2.2.9 Reactor Shutdown System Response Times

The applicants have clarified Table 7.2-3 and other sections of the PSAR where RSS response times have been called out. This has eliminated confusion in determining overall system response times. The staff has found these modifications acceptable. The detailed response times will be reviewed during the Technical Specification implementation phase (OL review).

7.2.3 Findings and Conclusions

The staff has conducted an audit review of the reactor shutdown system for conformance to the principal design criteria and the guidelines of applicable regulatory guides and industry codes and standards. In Section 7.1 of this SER, the staff concluded that the applicants had adequately identified the guidelines applicable to these systems. Based upon its audit review of the design for conformance to the guidelines, the staff finds that, upon satisfactory resolution of the open items identified in Sections 7.2.2.6 and 7.2.2.7 of this SER, there is reasonable assurance that the systems will conform to the guidelines applicable to them.

Staff review has included the identification of those systems and components for the RSS which will be designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicants have identified the systems and components consistent with the design bases for the RSS. Sections 3.10 and 3.11 of this SER address the qualification program to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of the systems and components satisfies this aspect of PDC 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," and PDC 5 (GDC 4), "Environmental and Missile Design Bases."

On the basis of its review, the staff concludes that the RSS will conform to the design-basis requirements of IEEE Std. 279-1971. The RSS will include the provision to sense accident conditions and anticipated operational occurrences and initiate reactor shutdown consistent with the analyses presented in Chapter 15 of the PSAR. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 18 (GDC 20), "Protection System Functions."

The RSS will adequately conform to the guidance for periodic testing in RG 1.22 and IEEE Std. 338 as supplemented by RG 1.118. The bypassed and inoperable status indication will adequately conform to the guidance of RG 1.47. The RSS will adequately conform to the guidance on the application of the single-failure criterion in IEEE Std. 379, as supplemented by RG 1.53. On the basis of its review, and the resolution of the open items described in Sections 7.2.2.6 and 7.2.2.7, the staff concludes that the RSS will satisfy the requirements of IEEE Std. 279 with regard to system reliability and testability. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 19 (GDC 21), "Protection System Reliability and Testability."

The staff believes with reasonable assurance that the RSS will adequately conform to the guidance in IEEE Std. 384 as supplemented by RG 1.75 for the protection system independence. On the basis of its review and the resolution of the open items in Sections 7.2.2.6 and 7.2.2.7, the staff concludes that the

RSS will satisfy the requirements of IEEE Std. 279 with regard to the independence of systems. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 20 (GDC 22), "Protection System Independence."

On the basis of its review of the failure modes and effects analysis for the RSS, the staff concludes that the system is designed to fail into a safe mode if conditions such as disconnection of the system, loss of energy, or postulated adverse environment are experienced. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 21 (GDC 23), "Protection System Failure Modes."

On the basis of its review of the interfaces between the RSS and plant operating control systems, the staff concludes that the system will satisfy the requirements of IEEE Std. 279 with regard to control and protection system interaction. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 22 (GDC 24), "Separation of Protection and Control Systems."

On the basis of its review of the reactor shutdown system, the staff concludes that the system will satisfy the protection system requirements for malfunctions of the reactivity control system, such as accidental withdrawal of control rods. Section 15 of the PSAR addresses the capability of the system to ensure that fuel design limits are not exceeded for such events. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 23 (GDC 25), "Protection System Requirements for Reactivity Control Malfunctions."

On the basis of its review, the staff finds that, upon satisfactory resolution of the open items in Sections 7.2.2.6 and 7.2.2.7, the RSS will consist of two independent reactivity control systems of different design principles. Each system will have the capability to shut the reactor down with the assumption of an active single random failure of any component. Therefore, the staff finds that the RSS will satisfy the requirements of PDC 24 (GDC 26), "Reactivity Control System Redundancy and Capability."

The staff's conclusions, noted above, are based on the requirements of IEEE Std. 279 with respect to the design of the RSS. Therefore, the staff finds that the RSS will satisfy the requirement of 10 CFR 50.55a(h) with regard to IEEE Std. 279.

The staff's review of the RSS has examined the dependence of this system on the availability of essential auxiliary support (EAS) systems. On the basis of its review, the staff concludes that the design of the RSS is compatible with the functional performance requirements of EAS systems. Therefore, the staff finds the interfaces between the RSS design and the design of the EAS system acceptable.

In summary, the staff concludes that the design of the reactor shutdown system and the design of the essential auxiliary support systems are acceptable and will meet the relevant requirements of PDC 2, 5, 18, 19, 20, 21, 22, 23, and 24 (GDC 2, 4, 20, 21, 22, 23, 24, 25, and 26) and 10 CFR 50.55a(h) with satisfactory resolution of the open items identified in Sections 7.2.2.6 and 7.2.2.7 of this SER.

7.3 Engineered Safety Features Systems

7.3.1 System Description

This section describes the review for that portion of the protection system used to initiate and control operation of the engineered safety features (ESF) and essential auxiliary support systems to mitigate the consequences of postulated accidents and transients. The ESF systems are identified in Table 6.1-1 of the PSAR and are as follows:

(1) reactor confinement/containment isolation
(2) containment isolation
(3) the annulus filtration
(4) reactor service building filtration
(5) steam generator building aerosol release mitigation
(6) habitability
(7) residual heat removal

Other ESF systems are identified in Table 6.1-1 of the PSAR. However, these systems are passive in nature having no instrumentation and control features, and therefore are not reviewed.

Although the residual heat removal system is an ESF system, the review and evaluation is handled in Sections 7.4 and 7.6 of this SER. The safety-related HVAC system portions associated with the annulus filtration system, the reactor service building filtration system, and the habitability systems are reviewed and evaluated in Section 7.6 of this SER.

7.3.1.1 Reactor Confinement/Containment Isolation System

The reactor confinement/containment isolation system will be designed by the applicants to ensure that an acceptable upper limit of leakage of radioactive material will not be exceeded under design-basis accident conditions. Primary containment is the reactor containment vessel. The containment functions will be dependent on or enhanced by other associated features and systems. These include (1) containment vessel and inner cell structures, (2) annulus filtration system (AFS), (3) vacuum relief system (VRS), and (4) the containment isolation system (CIS).

The containment vessel and inner cell structures are passive structures. Entrance into reactor containment will be via airlocks. These airlocks for personnel and equipment will be power operated and subject to interlocks which will prevent either of the airlock doors from opening at the same time as the railroad door in the reactor service building (RSB) is opened. The testing system for the personnel and equipment airlock doors will detect failure or degradation. Airlock leakage is manually vented to the annulus filtration system.

The VRS was developed to ensure that the RCB external design pressure (0.5 psig) will not be exceeded. The containment vessel will have two independent VRS valves. Each system will have two, series-connected, self-actuated valves set to open on a negative differential pressure allowing outside atmosphere to enter the containment vessel. The valves must close before the pressure in the vessel

equalizes with the outside atmosphere in order to preclude the possibility of leakage out of containment. These valves have no interconnection to the CIS.

7.3.1.2 Containment Isolation System

The containment isolation system (CIS) will provide the means to close valves in the lines penetrating the containment to establish the required barriers to prevent release of radioactive gas or particulate matter. Closure of valves may be manually or automatically initiated, depending on the type of line penetrating the containment.

All lines penetrating the containment which connect directly to the containment atmosphere will have two isolation valves, one inside and one outside the containment. Those that normally would not be closed during plant operation will automatically be closed by an actuation signal from the CIS.

The argon and nitrogen lines that penetrate the containment and connect to the coolant boundary primary cover gas space or inerted cell atmosphere will have two automatic isolation valves, each of which may be actuated closed by the CIS, or manually, depending on a need for a response time of less than 10 min.

The argon and nitrogen supply lines will each have two automatically initiated containment isolation valves. The valves will not be controlled by the CIS. They will instead be backpressure regulated to close automatically when the supply side pressure drops below a preset limit.

Closed systems (except for the intermediate heat transfer systems) whose lines penetrate the containment will have at least one isolation valve which will either be automatic, locked closed, or manually operated remotely. The valves will be located outside the containment except for those on the NaK lines serving the direct heat removal service (DHRS).

The intermediate heat transfer systems (IHTSs) will utilize closed loops that penetrate the containment without isolation valves. The IHTS boundary will be maintained intact by in-depth protection of piping components. The steam generator building (SGB) will be designed to seismic Category I and will provide a barrier for protection from extreme environmental conditions.

The necessary response characteristics of CIS-actuated isolation valves will be system and incident dependent. Meeting the guidelines will be a function of air transport time, detection time, valve-closure time, and the point at which the incident is sensed. The design will be based on 10-sec air transport time (from the detector to the valve), 3-sec detection time, and 4-sec closure time. All automatically actuated valves will be designed to close on loss of power or pneumatic energy supply.

The CIS will be actuated either manually or by a high radiation signal. Other instrument channels will monitor the environment of the reactor containment building (RCB). These include instrument channels that will monitor containment air pressure, temperature and hydrogen concentration, containment vessel temperature, seismic activity, and smoke and leak detection.

Radiation sensing will be provided in two areas: the exhaust duct of the containment ventilation and the head access area. Three independent, redundant measurements will be provided at each location. If the measurement signal is greater than the setpoint, a comparator will initiate a trip signal to a 2-out-of-3 logic which in turn will initiate the CIS. The logic for the automatic containment isolation will functionally be identical to that used for the primary reactor shutdown system. The comparator output will be optically coupled to the logic. Within the logic, the two comparator outputs (one from the head access area, the other from the exhaust duct of the containment ventilation) will be combined to feed 2-out-of-3 coincidence logic modules. One logic train will drive the in-containment automatic valves. Another logic train will drive the ex-containment automatic valves. No permissives or bypasses will be provided for the radiation monitoring system.

Provisions for on-line testing of electrical and mechanical equipment will be included in the design. A test source will be used to test the instrument channel through the comparator. Signals will be inserted before the 2-out-of-3 module to test the remainder of the logic and the closure of the automatic valves. Since closure of the valves will not require reactor shutdown, this test can be performed during power operation.

Radiation channel output meters will be included to provide the operators with early indication of anomalous instrumentation performance. This equipment will not be safety related.

There are no interlocks included in the design, nor is there any sequencing of the closure of the valves.

Manual initiation capability will be provided locally at the CIS breaker cabinets and within the control room on the main control board to close all CIS valves. Capability to close the CIS valves will be provided if loss of access to control room is assumed. Position-indicating lights will be provided in the control room for all containment isolation valves.

There will be three categories of CIS cabling: cables between the radiation monitoring sensors and logic panels; cabling between the logic panels and the power breakers; and cabling from the breakers to the valve actuators. Wiring for the three CIS instrument channels will be routed exclusively with the three secondary protection system instrument channels.

CIS logic train actuation wiring will be routed through two separated and independent conduits. A conduit will contain only wiring from a single CIS logic train.

7.3.1.3 Annulus Filtration System

The annulus filtration system will be designed to maintain the containment/confinement annulus space under 1/4-in. water gage negative pressure during normal plant operation and accident conditions, to filter the containment/confinement annulus exhaust during normal operation, to filter the RCB ventilation exhaust air during refueling operations when the RCB/RSB refueling hatch is open, to filter and recirculate the annulus air during accident conditions, and to accomplish proper mixing in the annulus using the recirculating duct system.

7.3.1.4 Reactor Service Building Filtration System

The reactor service building (RSB) filtration system will be designed to filter the RSB exhaust air to limit the possible spread of radioactive contaminants during accident conditions. The RSB will be maintained at a minimum of 1/4-in. water gage negative pressure under all conditions, except when the railroad door is open. The system will be designed to exhaust air from potentially radioactive areas and to prevent the spread of airborne radioactivity from contaminated to clean areas within the building.

7.3.1.5 Steam Generator Building Aerosol Release Mitigation System

The steam generator building aerosol release mitigation system will be designed to limit the quantity of sodium aerosols that may be released from the steam generator building in the event of a design-basis leak in one loop of the intermediate heat transport system. Upon detection of aerosol concentration greater than 10^{-7} g per cm^3 in any of the SGB loop exhaust ducts, the SGB, RCB, and RSB HVAC systems will be closed to the outside atmosphere.

Other aerosol detectors will be located in the air inlet to each protected air-cooled condenser (PACC) unit. Detection of aerosol concentration exceeding 10^{-7} g per cm^3 precludes a PACC unit from starting for a period of 5,000 sec. Following the 5,000-sec shutdown period, an affected PACC unit will automatically restart.

The aerosol detectors will be employed redundantly in a 2-out-of-3 logic design and will be powered from three separate Class 1E uninterruptible power sources. A total of 18 aerosol detectors will be used for monitoring the SGB loop HVAC exhaust ducts and another 18 will be used for monitoring the air inlet to the PACC units.

7.3.1.6 Habitability Systems

The habitability systems will be required to maintain an environment within the control room at acceptable working conditions following a major radioactive release. This will be done by the shielded design of the control room along with a low leakage enclosure and special design of the heating and ventilation system.

There will be two independent full-capacity air conditioners and exhaust fans serving the control room HVAC systems. Two widely separated air intakes will be incorporated into the design. The air intake will be filtered and controlled to maintain a slightly positive pressure in the control room with respect to its surroundings at all times.

The system will have two modes of operation. The normal operating mode will be used during all normal plant operations and most minor accidents. The accident mode of operation will be initiated by the receipt of a containment isolation signal through the plant protection system, or by the detection of smoke, toxic chemicals, or high levels of radioactivity in the control room air intake ducts.

7.3.1.7 Residual Heat Removal System

The residual heat removal (RHR) system is an engineered safety feature. The instrument and control aspects of the RHR are evaluated as a part of the safe shutdown systems in Sections 7.4 and 7.6 of this SER.

7.3.2 Specific Findings

7.3.2.1 Containment Isolation System

The CIS is actuated by radiation sensors. These radiation sensors are provided in two areas: the exhaust duct of the containment ventilation and the head access area (HAA).

The staff questioned the lack of diverse actuation signals for the CIS. In response, the applicants stated that there is no diversity in the actuation signals because a diverse variable does not exist. The containment design-basis event (DBE), which is the most severe event that challenges the CIS, is the primary sodium in-containment storage tank fire during maintenance (PSAR Table 15.1.3-2). This event does not produce significant containment atmosphere pressures or temperatures in a time short enough to isolate the containment on either signal. This DBE envelopes the other events listed in PSAR Table 15.1.3-2 regarding pressure and temperature differentials. The only variable common to all of the listed events is radiation. The radiation is sensed in two different locations (HAA and containment exhaust) by three detectors in each location. The HAA detectors are of the Geiger Mueller type and the exhaust detectors are of the scintillation type. The detection schemes offer diversity in measurement and in manufacturers.

The HAA radiation monitors are listed as having a range of 0.1 to 10^4 mR per hour and an accuracy of ±40%. The staff questioned the adequacy of the accuracy parameter for these monitors.

The applicants stated this was the most accurate detector available with the required response time. In addition, this accuracy was used in the development of the PSAR Chapter 15 safety analyses. The final numerical values for these detectors will be evaluated during the OL review.

The staff has reviewed the CIS design criteria and concludes that they are acceptable (Section 6.2.4 of this SER).

7.3.2.2 Engineered Safety Feature Setpoints

The staff asked the applicants to provide and describe information for safety-related setpoints and provide verification that environmental error allowances will be based on the highest value determined in qualification testing. The applicants responded by stating that performance requirements are part of the input in calculating safety-related setpoints. All safety-related instrumentation and control equipment shall meet the performance requirements indicated in the ordering data under any combination of environmental conditions as defined in the applicable equipment specifications. These environmental conditions include normal and accident environments.

In addition, the applicants stated that if equipment errors obtained during equipment qualification testing will be greater than the specified performance requirements, the instrument setpoints will be suitably adjusted to ensure conservative margins. This response and commitment is acceptable to the staff for the CP review. The staff will verify the final setpoint methodology used during the OL review.

7.3.2.3 Sodium Heating Systems

CRBR PDC 7, "Sodium Heating Systems," requires that heating systems be provided for systems and components important to safety which contain, or may be required to contain, sodium or sodium aerosol. It also requires the heating systems and their controls to be designed with suitable redundancy to ensure that changes in temperature are maintained within design limits assuming a single failure.

Staff requirements that have been transmitted to LWRs were used along with PDC 7 to form the design criteria for the sodium heating systems. These additional staff requirements are as follows.

(1) An environmental control system (heating and ventilation or heat tracing) should be provided so that sodium or sodium aerosol instrumentation sensing lines for systems important to safety are protected from freezing (sodium freezes at 208°F).

(2) The environment associated with these instrumentation sensing lines should be monitored and alarmed so that appropriate corrective action can be taken to prevent loss of or damage to systems important to safety upon loss of the environmental control system.

(3) The environmental control systems (Position 1 above) should be electrically independent of the monitoring and alarm systems (Position 2 above) so that a single failure in either system does not affect the capability of the other system.

(4) The environmental control and monitoring systems should be designed to standards commensurate with their importance to safety and with consideration of administrative controls that will be implemented to address events or conditions which could render the systems inoperable. Plant procedures should address appropriate actions to be taken for such considerations.

(5) The Technical Specifications should include surveillance requirements for testing the environmental control and monitoring systems.

PSAR Sections 9.4, 5.2.1.5, and 5.2.1.6 were reviewed along with the applicants' responses to questions. Heating systems will be provided, including temperature-measuring and controlling instrumentation, for the following sodium- and sodium-aerosol-containing process systems and components:

(1) reactor enclosure (guard vessel and head)
(2) primary and intermediate reactor heat transport systems
(3) steam generator dump tanks and sodium-water reaction product tanks
(4) auxiliary liquid metal systems
(5) sodium impurity monitoring system

(6) inert gas processing system
(7) ex-vessel storage tank and its associated sodium piping

PSAR Table 9.4-1 provides a more detailed list of the safety-related components requiring heating. A staff evaluation regarding the acceptability for only these selected areas to be provided with sodium heating systems is presented in Section 9.4 of this SER.

The CRBR sodium heating system has been reviewed. The design criteria for this system meet the above-stated staff position and are, therefore, acceptable with the following provisions.

(1) A positive statement needs to be made concerning the electrical independence of the environmental control systems and the monitoring and alarm systems for each heating system.

(2) For those systems where heating is required for the system to function in a postaccident condition (i.e., decay heat removal), a commitment should be made to provide seismic Category I qualified heating systems, unless it can be demonstrated heating is not required. Other heating systems should be seismic Category II.

(3) The staff requirements resulting from the review presented in Section 9.4 of this SER should be met.

The above are considered to be confirmatory items requiring a formal response from the applicants.

7.3.2.4 Solid-State Programmable Logic System

The applicants were asked to identify and document where microprocessors, multiplexers, or computer systems may be used in or may interact with safety-related systems. The applicants, in a meeting, stated that most of the above systems are used in non-Class 1E applications and where a microprocessor, multiplexer, or computer interfaces with a Class 1E signal, that signal will be isolated by a qualified Class 1E isolator before being utilized by a non-Class 1E system. The two systems which use microprocessors, multiplexers, or computers for Class 1E application are the solid-state programmable logic system (SSPLS) and the radiation monitoring system (Section 7.6 of this SER).

The solid-state programmable logic system controls and actuates safety-related Class 1E equipment. It contains the control logic, signal conditioners, isolation devices, and auxiliary circuitry. The applicants have stated that the SSPLS may use microprocessor-based circuitry. The SSPLS will be qualified to IEEE Stds. 323, 344, and 383 as required for all Class 1E devices in order to minimize the possibility of common mode failures. In addition, the SSPLS will consist of three separate and redundant safety-related divisions so that a failure in one division will not affect any component or device in the other divisions.

The staff questioned the applicants with regard to the manual initiation capability of the SSPLS. The applicants stated that all motor-operated or pneumatically actuated valves controlled by the SSPLS can be operated or actuated manually; however, pumps, fans, and dampers require the SSPLS in order to operate.

It was not apparent from the PSAR which CRBR systems will use the SSPLS and which of the equipment controlled by the SSPLS will be safety related. The staff has concluded in past reviews (WNP-2 Docket No. 50-397) that safety-related equipment controlled by multiplexers, microprocessors, or minicomputers should be made available even if the multiplexing, microprocessing, or minicomputing system malfunctioned.

The staff requires that the applicants clearly define which systems will use the SSPLS and which equipment controlled by the SSPLS will be safety related. The applicants must verify that the above equipment will be manually operable following failures of any multiplexers, microprocessors, or minicomputers that may be used. The resolution of this item will be discussed in a supplement to this SER.

7.3.2.5 Freeze-Protection/Water-Filled Instrument and Sampling Lines and Cabinet Temperature Control

The applicants were asked to describe features of the CRBR environmental control system which ensure that instrument sensing and sampling lines (water) for systems important to safety will be protected from freezing during extremely cold weather.

The applicants' response was that all safety-related process, instrument, and sampling lines (water) will be contained entirely within environmentally controlled buildings and, therefore, will not be exposed to cold weather. The nuclear island (NI) heating, ventilating, and air conditioning (HVAC) system will maintain a minimum temperature of 55°F in all areas of the nuclear island buildings that will contain safety-related equipment.

Exceptions to the above are the protected air-cooled condensers (PACCs) which will be located outside the environmentally controlled buildings. However, secondary steam from the turbine will continuously heat the PACCs to protect them from extremely cold weather.

Another staff concern on LWR plants has been the use of unqualified heaters to control temperature and humidity within cabinets housing electrical transmitters that provide signals to the reactor protection system. In responding to this concern, the applicants stated that the only CRBR Class 1E equipment which will use cabinet heaters is the sodium pump drive system reactor shutdown system (RSS) breakers. These heaters are Class 1E and are qualified to environments of 125°F and 90% relative humidity.

The staff concludes that these concerns are resolved.

7.3.2.6 CRBRP Liquid Metal-to-Gas Leak Detection System

The staff has reviewed the applicants' liquid metal-to-gas leak detection system for the purpose of determining its adequacy to perform the detection and monitoring function in accordance with applicable regulatory guides. The applicants' performance design and test specifications for this system were also reviewed.

PDC 28, "Quality of Reactor Coolant Boundary," states in part that means shall be provided for detecting and, to the extent practical, identifying the location

of the source of reactor coolant leakage. RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System," deals with acceptable methods for implementing PDC 28 with regard to the selection of leak detection systems for the reactor coolant boundary.

The CRBR design provides for several types of methods for the detection of liquid metal-to-gas leaks. Contact detectors are designated for use between the bellows and the stem packing collar on bellows sealed valves. These detectors will consist of two open contact wires that are gold plated, sheathed in stainless steel, insulated with mineral oxide, with the sheath and insulation closed off by a glass seal. Cable detectors are to be placed in the bottom of guard vessels and below large tanks. These detectors will consist of a loop of wire contained in a stainless steel sheath insulated with magnesium oxide. Contact with the wire loop by leaking sodium would cause a low resistance path to ground. Plugging filter aerosol detectors (PFADs) will be used to monitor aerosol by measuring the pressure drop across a membrane filter with a constant flow of gas from certain areas. The applicants plan to install these detectors in the space within guard vessels and in cells containing liquid metal systems. Sodium ionization detectors (SIDs) will be used for monitoring cell atmosphere. The SID will consist of a detector housing through which sample gas will be channeled. Sodium ions will be picked up on a collector plate that will be located opposite an ionizing filament. Sodium ion current at the plate will be read out and monitored.

The applicants have described other detection methods as part of the total liquid metal-to-gas leak detection system. Among these methods are pressure and temperature monitoring in the inerted cells, particulate radiation monitoring in seven cell groups, and liquid sodium level sensors for indication of large leaks.

Indicators and alarms will be provided in the control room to indicate leakage. The various detectors will have different thresholds; some will be proportional detectors and others will be integrating detectors. Provisions will be made for periodically testing the leak detectors for operability and proper calibration during plant operation. Technical Specifications will be imposed to limit plant operation in the event of a leak or in the event that the liquid metal-to-gas leak detection system capability is degraded.

The applicants' design bases consist of design criteria and quantitative and qualitative performance requirements. Pertinent design criteria are:

(1) The leak detection system must provide diverse means (in inert cells containing Class 1 piping) for detecting and locating liquid metal-to-gas leaks throughout the plant at normal operating conditions.

(2) The leak detection system must operate at lower temperature conditions (700°F), though its sensitivity may be reduced. The liquid metal-to-gas leak detection system shall detect and locate liquid metal-to-gas leaks throughout the plant between the temperatures of 375°F and 1,000°F as required to fulfill continuous monitoring requirements of Appendix G to the PSAR, "CRBR Plan for Inservice and Preservice Inspections."

(3) The leak detection system must perform its function during and after an OBE, even in the event of loss of offsite power.

The applicants were asked to provide a description of the CRBR design compliance with RG 1.45. The applicants provided the response to this question in the Westinghouse Advanced Reactors Division Piping Integrity Report (WARD D-0185). The response for compliance with RG 1.45 was evaluated by the staff and, in Section 5.7.2 of this SER, it was concluded to be satisfactory.

The liquid metal-to-gas leak detection system is the principal backup for the liquid metal systems designed and stressed to the leak-before-break concept. The staff considers the liquid metal-to-gas leak detection system to be a system which prevents or mitigates accidents. This fulfills the requirements and satisfies the definition of an engineered safety feature and, therefore, is required to be qualified to seismic Category I, be placed on Class 1E power, and be redundant and diverse. In response to these requirements and to additional requirements that are discussed in Section 5.3.2.3 of this SER, the applicants have committed to

(1) update the PSAR to reflect the 20-hour leak detection required for a 100-gal-per-hour leak in the IHTS air-filled cells.

(2) prepare Technical Specifications for the IHTS leak detection operability.

(3) upgrade the leak detection capability in the PHTS, DHRS, and EVST systems to be safety related, i.e., seismic Category I, Class 1E power, environmental qualification, and redundance or diversity, but not both.

The staff has reviewed the design criteria and performance capability for the liquid metal-to-gas leak detection system and concludes that it is acceptable for the CP review.

7.3.2.7 Sodium-Water Reaction Pressure Relief System

The sodium-water reaction pressure relief system (SWRPRS) is a three-channel secondary RSS subsystem and will be designed by the applicants to meet the safety-related channel performance and reliability requirements of the CRBRP principal design criteria, IEEE Std. 279, applicable NRC regulatory guides, and other appropriate criteria and standards by complying with the applicable design requirements delineated in Section 7.1.2 of the PSAR.

PSAR Section 7.5.6.1.3 provided preliminary information on SWRPRS interlocks and the applicants stated that more information on the interlocks will be provided in the FSAR along with justification as to whether or not the interlocks need to be Class 1E. The staff required that these interlocks be safety related if their failure will prevent an engineered safety features actuation system (ESFAS) from occurring when needed.

The applicants were required to submit design information for the SWRPRS including the safety classification for the various portions of the system.

In a letter dated February 15, 1983, from John R. Longenecker to J. Nelson Grace, the applicants provided revised sections of PSAR Chapters 3 and 7. Revised Table 3.2-3 indicates that the SWRPRS instrumentation and controls will be classified as Class 1E. In addition, a revised Table 7.1-1, "Safety-Related Instrumentation and Control Systems," shows that the SWRPRS instrumentation, initiation, and control logic which achieve adequate isolation and blowdown of the water side of a superheater or evaporator in the event of a sodium-water reaction will be Class 1E. The instrumentation used to initiate the isolation and blowdown valves will be the rupture disc pressure detectors located downstream of the rupture disc. The other pressure and temperature instrumentation distributed throughout the sodium-water reaction pressure relief system will be used for status indication only and will not be Class 1E.

Revised PSAR Section 7.5.6.1.1 reiterates the above by stating that the SWRPRS instrumentation and controls which will be used to initiate and control the SWRPRS function will be Class 1E. Furthermore, revised PSAR Section 7.5.6.1.3 states that bypasses and interlock functions which are required for operation of the initiation and control logic will be Class 1E.

The staff's interpretation of the revisions noted above indicates that the SWRPRS instrumentation and controls required to initiate and control the SWRPRS functions listed in a through g of Section 7.5.6.1.2 of the PSAR will be Class 1E and that the bypasses and interlocks required for the above functions will be Class 1E.

The staff has reviewed the applicable sections of the PSAR and the applicants' responses to its questions regarding SWRPRS. The applicants have committed to provide Class 1E instrumentation and controls for SWRPRS as discussed above. This is acceptable to the staff.

However, the applicants are required to formally confirm that the instrumentation and controls required to initiate and control the SWRPRS functions listed in a through g of Section 7.5.6.1.2 of the PSAR will be safety related and, thus, Class 1E, or provide acceptable justification.

In addition, for the OL review the applicants are required to provide final system design information including schematics that detail the functions and safety classifications of the instrumentation and controls for SWRPRS.

7.3.2.8 Level Measurement Errors That Result From Environmental Temperature Effects on Level-Measuring Instrument Reference Legs

The applicants were asked to describe how the effects of high temperatures on reference legs of steam drum level-measuring instruments subsequent to high energy line breaks were evaluated and included in the CRBR design. This issue was addressed for operating LWRs in IE Bulletin 79-21. The applicants responded by stating that the steam drum water-level instrument will use a reference leg which operates at ambient temperature and has a length of 38 in. Each loop (three loops total) will be located in a separate cell and the environment from a high energy pipe break will only affect one loop. The maximum temperature of the cell following a high energy pipe break would be 215°F.

This maximum cell temperature (215°F) and the steam drum's short reference leg (38 in.) will limit the indicated water level to approximately 1.4 in. above the actual water level. This increase of 1.4 in. in indicated level above the actual level is within the setpoint margin.

Level-measuring instruments in the reactor primary heat transport system and intermediate heat transport system do not use reference legs.

The staff has reviewed the CRBR design regarding the effects of high temperatures on reference legs of level-measuring instruments and concludes that this issue is resolved.

7.3.2.9 IE Bulletin 80-06, "Engineered Safety Feature Reset Control"

IE Bulletin 80-06 addressed concerns related to safety equipment not remaining in its emergency mode upon reset. The staff asked the applicants to review the CRBR design for all systems serving safety-related functions to determine whether or not, upon reset of an ESFAS, the safety-related equipment will remain in its emergency mode and to provide justification for the safety systems that will not remain in the emergency mode.

The applicants provided information on two design features for the electrical system and examples of system reset designs. The two design features used are:

(1) Circuit breakers will close on the presence of an emergency signal where driven equipment is powered through medium- or low-voltage switchgear. The breakers will remain closed even after the actuating signal has been reset. Opening of the breakers will be achieved through any of the following: manual operation, electrical fault, or absence of process interlocks which otherwise are necessary for continuous operation of the equipment.

(2) Where operated equipment is powered through motor-control centers or power-distribution panels, seal-in circuitry will be provided for the momentary contacts. The circuit will remain energized even when the actuating signal resets, and can be deenergized only by manual operation, or electrical fault, or by the absence of process interlocks which are otherwise necessary for continuous operation of the equipment.

It was not apparent to the staff that the CRBR design satisfied the concerns that were addressed in IE Bulletin 80-06. In particular, the applicants' response to this concern does not provide clarification that the term "reset" includes both manual and automatic means. In fact, it is apparent that the operator can manually reset ESFAS that will cause safety-related equipment to not remain in its emergency mode.

The applicants were required to readdress the concerns of IE Bulletin 80-06 and to justify any places in the design of CRBR safety system logic where safety equipment will not remain in its emergency mode upon reset (manual or automatic).

In a letter dated January 26, 1983, from John R. Longenecker to Paul S. Check, the applicants provided a clarification to the previous response regarding IE

Bulletin 80-06. The applicants stated that there are no places in the design of the CRBRP safety program logic where, once actuated, safety equipment will not remain in its emergency mode upon automatic or manual reset of an engineered safeguards actuation signal. Equipment can only be returned to its normal condition by deliberate manual action.

The only exception to the above is the containment isolation system (CIS). Operation of the CIS automatic back pressure valves is an exception because manual action is not required for these valves to return to their normal position (open). However, these valves will automatically return to their emergency position (closed) when an actuation condition (low pressure) is present.

After its review of the applicable sections of the PSAR and the applicants' responses to the concerns expressed in IE Bulletin 80-06, the staff has concluded that the CRBRP ESFAS designs will meet the staff's guidelines regarding ESF reset controls, and, therefore, this concern is resolved. However, for the operating license, the staff will require the applicants to perform preoperational tests to demonstrate that all ESF equipment (except as noted above) will remain in its emergency mode upon removal of the actuation signal and/or resetting of the various isolation or actuation signals.

7.3.3 Findings and Conclusions

The audit review of the instrumentation and control aspects of the engineered safety feature systems included the engineered safety features actuation systems (ESFASs) and the engineered safety features (ESF) control systems. The ESFAS will detect plant conditions requiring the operation of an ESF system and/or essential auxiliary support (EAS) system and will initiate operation of these systems. The ESF control system will regulate the operation of the ESF systems following automatic initiation by the protection system or manual initiation by the plant operator.

The staff has conducted an audit review of these systems for conformance to guidelines of the regulatory guides and industry codes and standards applicable to these systems. In Section 7.1 of this SER the staff concluded that the applicants have adequately identified the guidelines applicable to these systems. The staff finds that, subject to resolution of the previously stated open item in Section 7.3.2.4, there is reasonable assurance that the design of the ESF actuating and control systems will satisfy the applicable guidelines for these systems. The staff's review has included the identification of those systems and components for the ESFAS and ESF control systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicants have identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of the systems and components satisfies this aspect of PDC 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," and PDC 5 (GDC 4), "Environmental and Missile Design Bases."

On the basis of its review, the staff concludes that the ESFAS conforms to the design-bases requirements of IEEE Std. 279. The system includes the provisions to sense accident conditions and anticipated operational occurrences to initiate the operation of ESF and EAS systems consistent with the analyses presented in Chapter 15 of the PSAR. Therefore, the staff finds that the ESFAS satisfies the requirements of PDC 18 (GDC 20), "Protection System Functions."

The ESFAS will conform to the guidelines for periodic testing in RG 1.22 and IEEE Std. 338 as supplemented by RG 1.118. The bypassed and inoperable status indication will conform to the guidelines of RG 1.47. The ESFAS will conform to the guidelines on the application of the single-failure criterion in IEEE Std. 379 as supplemented by RG 1.53. On the basis of its review, the staff concludes that the ESFAS will meet the criteria of IEEE Std. 279 with regard to the reliability and testability of the system. Therefore, with the exception of the open item listed above, the staff finds that the ESFAS will satisfy the requirement of PDC 19 (GDC 21), "Protection System Reliability and Testability."

The staff concludes that the ESFAS adequately conforms to the guidance in IEEE Std. 384 as supplemented by RG 1.75 for the protection system independence. On the basis of its review, the staff concludes that the ESFAS satisfies the requirements of IEEE Std. 279 with regard to the system's independence and satisfies the requirement of PDC 20 (GDC 22), "Protection System Independence."

On the basis of its review of the analysis for the ESFAS, the staff concludes that the system will be designed with due consideration of safe failure modes if conditions such as disconnection of the system, loss of energy, or postulated adverse environment are experienced. Therefore, the staff finds that the ESFAS will satisfy the requirements of PDC 21 (GDC 23), "Protection System Failure Modes."

On the basis of its review of the interfaces between the ESFAS and plant operating control systems, the staff concludes that the system satisfies the requirements of IEEE Std. 279 with regard to control and protection system interactions. Therefore, the staff finds that the ESFAS satisfies the requirements of PDC 22 (GDC 24), "Separation of Protection and Control Systems."

The staff's conclusions noted above are based upon the requirements of IEEE Std. 279 with respect to the design of the ESFAS. Therefore, the staff finds that the ESFAS satisfies the the requirements of 10 CFR 50.55a(h) with regard to IEEE Std. 279.

The staff's review of the ESFASs and ESF control systems has included the dependence of these systems on the availability of essential auxiliary supporting (EAS) systems. On the basis of its review and coordination with those having primary review responsibility of the EAS systems, the staff concludes that the design of the ESFASs and ESF control systems is compatible with the functional performance requirements of EAS systems. Therefore, the staff finds that the interfaces between the ESFASs and ESF control systems and the EAS systems are acceptable.

The staff's review of the ESF control systems included conformance to the requirements for testability, operability with onsite and offsite electrical

power, and single failures consistent with the principal design criteria applicable to these ESF systems. The staff concludes that the ESF control systems will be testable and will be operable on either onsite or offsite power (assuming only one source is available), and that the controls associated with redundant ESF systems will be independent and satisfy the single-failure criterion and, therefore will meet the relevant requirements of PDC 35 and 49 subject to the resolution of the open item in Section 7.3.2.4 of this SER.

In summary, the staff concludes that upon satisfactory resolution of the open item in Section 7.3.2.4 the design of the ESFASs and ESF control systems will be able to meet the relevant requirements of PDC 2, 5, 11, 17 through 23, 58, 35, and 49 and 10 CFR 50.55a(h). The counterpart general design criteria are 2, 4, 13, 19 through 25, 29, 34, and 41, respectively.

7.4 Systems Required for Safe Shutdown

7.4.1 System Description

This section describes the equipment and associated controls and instrumentation of systems required for safe shutdown. It also describes controls and instrumentation outside the main control room that enable safe shutdown of the plant in case the main control room needs to be evacuated. The following systems have been identified as being necessary for safe shutdown.

(1) steam generator auxiliary heat removal system (SGAHRS) which includes:

   (a) protected air-cooled condensers (PACCs)
   (b) auxiliary feedwater subsystem (AFWS)
   (c) steam venting subsystem
   (d) protected water storage tank (PWST)

(2) outlet steam isolation system (OSIS)

(3) direct heat removal service (DHRS)

(4) pony motors/primary heat transfer system (PHTS) and intermediate heat transfer system (IHTS)

(5) remote shutdown system

Also required for safe shutdown are the auxiliary support systems necessary for the operation of those systems identified above.

One or more of the three loops in the PHTS will be used for all modes of reactor shutdown and residual heat removal. The primary pumps will utilize pony motors to drive the primary pumps at reduced speed corresponding to the requirements for shutdown. The IHTS will be able to carry the heat from the primary system via the intermediate heat exchanger to either the steam generator system (normal operation) and the SGAHRS. The main condenser/feedwater train will be the normal path for heat removal from the SGS. However, it will consist of nonsafety-related equipment, and its operation will not be required for safe shutdown of the plant.

The SGAHRS will be a safety-related system design to provide cooling upon loss of normal feedwater or heat sink. The SGAHRS will include the AFWS, the PACCs, and the steam venting system. There will be three PACCs and three auxiliary feedwater pumps. One of the feedwater pumps will be steam driven. The other two will be electrically driven from Class 1E power supplies.

The PHTS, IHTS, steam generator system (SGS), and SGAHRS will be designed to provide decay and sensible heat removal from the reactor via natural circulation in combination with steam venting utilizing the steam-turbine-driven auxiliary feedwater pump. Pony motor operation, although not required for the natural circulation mode of operation, is the preferred mode of operation. However, the pony motors are required if the backup DHRS is to be placed in operation at full capacity.

The PHTS and IHTS will provide sufficient heat removal to prevent loss of coolable geometry following a normal pump coastdown from full-power operation with three loops at natural circulation flow. The SGAHRS protected water storage tank capacity and the PACC heat removal capability will be such that the entire long-term decay heat load can be carried by a combination of extended steam venting from one SGS loop and operation of one PACC with water-side natural circulation and air-side forced circulation.

The direct heat removal service (DHRS) will be a backup shutdown heat removal system that does not depend on the IHTS, SGS, or SGAHRS. The DHRS will consist of one heat removal path capable of removing approximately 11 MWt from the primary coolant. The capability will be adequate to prevent loss of coolable geometry assuming the SGS heat sink is lost immediately upon shutdown from rated power and active operation of DHRS is initiated ½ hour after shutdown. The system will use an NaK loop and air heat exchanger to dissipate heat and will be dependent on part of the PHTS for operation. The instrumentation and controls for the DHRS are described in Section 7.6 of the PSAR.

The instrumentation and control systems necessary for safe shutdown will include those necessary to actuate, monitor, and control those systems previously identified. Also included will be the instruments and controls associated with monitoring core criticality, outlet steam isolation, and control room habitability.

Core criticality will be monitored by the flux monitoring system which is described in Section 7.5 of the PSAR. The instrumentation and controls necessary to maintain the control room habitability are described as part of the engineered safety features in Section 7.3 of the PSAR and safety-related HVAC systems in Section 7.6 of the PSAR.

7.4.1.1 Steam Generator Auxiliary Heat Removal Instrumentation and Control System

The steam generator auxiliary heat removal system (SGAHRS), in conjunction with the plant protection system, will be designed to detect the need for, initiate, and control the alternate heat removal path when the normal heat sink is unavailable.

When actuated, the SGAHRS will draw water from a protected water storage tank and pump it to each steam drum. Two supply lines will be provided for each

steam drum. One line will be supplied by two half-sized, motor-driven feedwater pumps; the other will be supplied by a full-sized, turbine-driven pump. A flow control valve and an isolation valve will be provided in each supply line at the inlet to each steam drum. Isolation valves will be provided to isolate the auxiliary feedwater system from the steam generator system during power operation and to provide leak isolation during SGAHRS operation.

There will be one protected air-cooled condenser (PACC) for each steam drum. The condenser portion of the PACC will reject heat to the atmosphere via convection. Saturated steam will be supplied to the condenser from the steam drum and saturated water will be returned. The PACC steam and water loop will be driven by natural circulation. Each PACC unit will consist of two tube bundles, two sets of louvers, and two fans. The fans will be used to force air across the condenser tubes. Control of the air flow will be accomplished by a combination of fan blade pitch control and air inlet louver position control.

7.4.1.1.1 Auxiliary Feedwater Pump Control

Upon receipt of the SGAHRS initiation signal, the two motor-driven AFW pumps will be started, resulting in both pumps coming on line and operating at constant speed. In addition, the isolation valves in the steam supply lines from the steam drums to the turbine-driven pump will be opened. At the turbine inlet, a pressure-regulating valve will reduce the steam supply pressure to the 1,000 psig required by the turbine drive. Each AFW pump can also be actuated manually at the operator's discretion.

Each AFW pump control will include a "normal/long-term cooldown (LTC)" mode selector. In the "normal" mode, the pumps will start on SGAHRS initiation. In the LTC mode, the operator may shut down any or all AFW pumps provided the steam drum water level is above the trip point setting. When in the LTC mode, the AFW pumps come on line automatically when the water level in the steam drum drops to a low level trip point.

7.4.1.1.2 Auxiliary Feedwater Flow Control

The flow to the steam drum will be controlled with a control valve that is positioned by a single controller. Manual control of the auxiliary feedwater control valves will be provided at the main control panel and at the local SGAHRS panel.

7.4.1.1.3 Protected Air-Cooled Condenser Control

The protected air-cooled condenser louvers will be opened and the fans started upon receipt of either a scram or the SGAHRS initiation signal. Airflow through the condenser will be controlled by a combination of fan blade pitch and inlet louver position. The fan blades and inlet louvers will be positioned by automatic controllers. Manual control of the inlet louver position and fan blade pitch will also be provided. Manual controls will also be provided for the blower motors. The outlet louver will be interlocked with the inlet louver and will open automatically when the inlet louver actuator is energized. If a high concentration of sodium aerosol in a PACC cell is detected, redundant trip logic will shut down the affected PACC system for approximately 1½ hours.

7.4.1.1.4 Pressure-Controlled Bypass Valve

To prevent the auxiliary feedwater pumps from overheating at reduced flow, each pump will be provided with a bypass line from the discharge back to the protected water storage tank. The valve in the bypass line will normally be open during pump startup. After startup, the valve will close. When pump discharge pressure rises to 1,970 psig, the valve will open. When the pressure drops below 1,820 psig, the valve will go to the closed position.

7.4.1.1.5 Auxiliary Feedwater Isolation Valves and Pump Inlet Isolation Valves

Isolation valves in each of the supply lines to the steam drums (AFW isolation valves) will be provided to ensure that an uninterrupted supply of auxiliary feedwater will be available to unaffected loops following failures in a loop. Isolation valves at the pump inlets will be provided to prevent loss of water from the protected water storage tank (PWST) in the event of a failure between these valves and the AFW isolation valves, and to allow the switching of suction from the PWST to the condensate storage tank. The auxiliary feedwater isolation valves will be opened upon receipt of the SGAHRS initiation signal.

During SGAHRS operation, these valves will close automatically upon detection of a sodium-water reaction, a high steam drum level, a steam drum pressure less than 200 psig, or AFW flow greater than 150% of full flow for 5 sec. This automatic closure will occur only in the affected loop. If the valves are closed by a high drum level signal, they will reopen automatically when the drum level falls to the low drum level trip point.

7.4.1.1.6 Superheat and Steam Drum Vent Control Valves

The superheat and steam drum vent control valves will be opened upon SGAHRS initiation and will depressurize the steam drums to the valves' respective setpoint levels. The superheater vent control valve setpoint will be 1,475 psig, and the steam drum vent control valve setpoint will be 1,550 psi. The valves will function to provide steam release during the venting period until the PACC units can remove the heat load in a closed-loop manner.

7.4.1.1.7 Initiating Circuits

The reactor shutdown system will provide redundant primary and secondary initiation signals to SGAHRS to sequentially start the three auxiliary feedwater pumps and the three PACCs when either a low steam drum level or high steam-to-feedwater flow mismatch occurs in any one of the three steam generator system (SGS) loop subsystems. The PACCs will also be initiated for all reactor scrams. However, initiation of the PACCs upon the occurrence of a reactor scram will not be a safety-related function, but will be provided in order to increase the rate of the steam cooling of the superheater outlet tubesheet. In each of the two plant protective subsystems, the three trip signals for low steam drum level and the three trip signals for high steam-to-feedwater flow mismatch will each be isolated and input to redundant 2-out-of-3 logic networks. The outputs from the redundant logic networks will each be isolated within the SGAHRS divisional control system and combined in a 1-out-of-4 logic which will initiate SGAHRS. If two of three trip signals occur in any subsystem, the SGAHRS will be initiated.

The sequence of decay heat removal events is shown in Table 7.4-1 of the PSAR. The scheme used for initiating the SGAHRS is shown in Figure 7.4-1 of the PSAR.

Sensors provided at the PACC airside inlet will be used to detect sodium aerosols in the event of an IHTS loop sodium leak. When an aerosol concentration limit of 10^{-7}g per cm^3 is detected at any PACC inlet, the affected PACC will automatically shut down for a 5,000-sec period and then will automatically restart. A manual override will be provided for the operator to restart the PACC before the 5,000-sec period has ended if the operator has determined this can be done without adverse effect on PACC operation.

Since the automatic activation and control of auxiliary feedwater flow will be necessary to ensure decay heat removal, provisions will be included in the design to ensure that the automatic initiation takes precedence. A startup signal to the feedwater pumps will override a manual control signal. Similarly, a signal to open the isolation valves will override a manual closure signal.

7.4.1.1.8 Bypasses and Interlocks

Bypasses are required on the steam-to-feedwater flow mismatch and steam drum level subsystems to allow system reset and reactor startup without initiating SGAHRS. These bypasses will be implemented as described in the section on the reactor shutdown system (Section 7.2 of the PSAR).

The following are interlocks provided in the SGAHRS components control circuits.

(1) Each auxiliary feedwater pump (preferred) inlet valve may be closed only after the associated alternate inlet valve has been fully opened. The preferred inlet valve will open automatically anytime the alternate inlet valve starts to close.

(2) A switch will be provided on the back panel to permit the operator to bypass the sodium aerosol protection circuit of the PACC. The bypassed position will be annunciated on the main control room panel.

(3) The PACC outlet louver will open automatically whenever the inlet louver is not fully closed. When the outlet louver is fully open, the PACC blower may be started either automatically or manually.

7.4.1.2 Outlet Steam Isolation Subsystem

The outlet steam isolation subsystem (OSIS) will be designed to provide isolation of steam system breaks. Isolation will be provided on a per-loop basis on a low superheater outlet pressure signal or a trip of the sodium-water reaction pressure relief subsystem (SWRPS). For all other automatic actuations, isolation will be provided on a three-loop basis. A high steam-to-feedwater flow mismatch is indicative of a main steam supply leak downstream from the flow meter or insufficient feedwater flow. The OSIS will be initiated by the SGAHRS initiation signal. An initiation signal will close the superheater outlet isolation valves in all three loops when a high steam-to-feedwater flow mismatch or a low steam drum level occurs in any loop. In each steam generator system loop, the three trip signals for high steam-to-feedwater flow mismatch and the low steam drum level will input to a 2-out-of-3 logic. If two of three trip

signals occur in any of the three loops, the OSIS will be initiated, and all three loops will be isolated. The OSIS instrumentation and control will include design features allowing for manual bypass, redundancy, isolation and separation between redundant channels, and closure of isolation valves upon loss of electrical or control air supply.

7.4.1.3 Direct Heat Removal Service

The direct heat removal service (DHRS) is not listed in the PSAR as part of the safe shutdown systems. The staff has listed it here because it is a backup shutdown heat removal system. The description of the DHRS instrumentation and controls is found in Section 7.6 of this SER.

7.4.1.4 Pony Motors--Primary Heat Transport System and Intermediate Heat Transport System

Six pony motors, one in each primary and intermediate heat transport loop, will provide sodium flow for decay heat removal. Each pony motor will run continuously during all mode of plant operation except during reactor shutdown for maintenance, which involves incapacitating an HTS loop. During maintenance only one loop will be permitted to be out of service.

Normal pony motor start will be through a non-Class 1E permissive sequence circuit which will start the vertical drive motor's external circuit which in turn will start the vertical drive motor's external lubricating oil cooling system and high-pressure lube oil pump. This method of starting is not classified as safety related and will be used for starting the pony motor during reactor shutdown.

The Class 1E controls will start the pony motors without the use of the external lubricating oil cooling system or high-pressure lube oil pump.

7.4.1.5 Remote Shutdown System

A remote shutdown system will be provided. It will consist of the following provisions.

(1) remote local control of

  (a) steam generator auxiliary heat removal system (SGAHRS)
  (b) diesel generator system
  (c) ex-vessel storage tank (EVST) cooling system

(2) remote shutdown panel instrumentation:

  (a) primary and intermediate hot-leg temperatures

  (b) primary and intermediate cold-leg temperatures

  (c) primary and intermediate pump speeds

  (d) steam drum pressure, temperature, and level

(e) steam flow

(f) reactor sodium level

(g) diesel generator power output, current, voltage, frequency, revolutions per minute, fuel oil level

The remote shutdown system will also include the following features.

(1) Plant instrumentation and control systems required for remote shutdown operations will have transfer switches located at the local panels to permit the plant operating personnel to select to operate from the local panels while isolating the remote controls or, conversely, to operate from the control room while isolating the local controls. The transfer of control of a plant system from the remote to the local mode is annunciated in the control room.

(2) If offsite power is lost coincident with having to achieve a safe shutdown condition in the reactor plant from outside the control room, the diesel generators will start and function in accordance with the design provided by the building electrical power system. Any operator actions required in conjunction with operating and loading the diesel generators will be done in the local operating mode at the diesel generator local panels.

(3) An independent sound-powered communications network will be provided between the local safety shutdown control stations.

(4) All plant protection system signals necessary for remote shutdown control will be isolated from the control room.

The need for remote shutdown (i.e., control room evacuation) is assumed not to occur simultaneously with the recovery from any other conditions except the loss of offsite power.

7.4.2 Specific Findings

7.4.2.1 Auxiliary Feedwater Isolation Valves

The applicants submitted a diagram of the auxiliary feedwater (AFW) system showing the division of power assignments for all valves and safety-related instrumentation and controls. A discussion was given that indicated the normal position and the position upon a loss of power to each valve. In this discussion, and in PSAR Section 7.4.1.1.6, credit is taken for the feedwater isolation valves to fail safe in the open position upon loss of electrical power. The staff required that the applicants verify that the fail-safe analysis covers all power-supply failures including overvoltage transients, shorts, and grounds. The staff was concerned that valves in the same division that must change position following a line break may be subject to a common failure, causing a malfunction that may prevent feedwater flow to any of the operable steam drums.

There will be six isolation valves, two in parallel to each steam drum. One valve will supply water from the electric-driven pumps, the other from the turbine-driven pump, and each will be capable of supplying 100% water flow to the steam drum. These two valves will be supplied with power from separate Class 1E power divisions. The failure of one valve to open would not result in the loss of water to a steam drum, as the required water would be supplied by the parallel line. These isolation valves will be closed for normal plant operation. In addition there will be flow control valves in series with these isolation valves. Each of the flow-control valves will be supplied with power from a Class 1E power division different from the Class 1E power division supplied to the isolation valve in the same line. The flow-control valves will be open during normal plant operation and will modulate to control the auxiliary feedwater flow to maintain steam drum level.

The applicants stated that the design will provide for failure of the control power (loss of power) to result in a "fail-safe" position (open) for the isolation valve, and an "as-is" position for the flow control valve; therefore, the capability to maintain the AFW supply is ensured.

For the hot short condition, since the wiring and controls for the isolation or flow control valves will be in separate Class 1E power divisions and separation will be provided in accordance with IEEE Std. 383-1974, hot shorts between the valves will not occur. The only hot short that could occur would be within the same power division between the valve and control circuits. This could result in two of the six isolation valves not opening when called upon. This would not result in a loss of water to any of the three steam drums.

The staff asked the applicants to discuss the resulting alignment that would occur if there were a break in steam or feedwater line associated with one steam drum and a failure in the division of power that does not supply the affected loop. The applicants stated that the loop would isolate the break and that the other loops would continue to supply feedwater to the steam drums since the isolation valve affected by the power failure would go to its failure position (open) and the affected flow control valve would remain in its normal position (open). If the power failure is of such a nature that it prevents an isolation valve from obtaining its failure position (a hot short), the affected flow-control valve on the parallel line would remain in its normal position (open), and flow would be controlled by the opening and closing of the isolation valve on the other division of power.

The AFW valves are safety related and are qualified to perform their function in the worst accident environment condition they are expected to experience.

The staff questioned the applicants about common mode failures for these valves. The applicants stated that inherent design features precluded common mode failures to these valves.

The control and isolation valves are manufactured by different valve suppliers and are provided with different types of actuators.

The staff has reviewed PSAR Figures 5.1-5 and 5.1-5a, Section 7.4, and the applicants' response to its questions and has concluded that the design criteria

for this system are acceptable and this issue is resolved. However, the acceptability of the design is based upon the applicants' statement that the control valves will be left in their open position during normal operation.

7.4.2.2 TMI Item II.E.1.2, Auxiliary Feedwater Automatic Initiation and Flow Indication

TMI Item II.E.1.2 basically requires that plants with auxiliary feedwater (AFW) systems provide automatic and manual AFW system initiation (Part 1) and AFW system flow indication in the control room (Part 2).

The applicants have stated (PSAR Appendix H) that the CRBR design will incorporate the automatic and manual initiation design for the AFW system in accordance with the requirements of TMI Item II.E.1.2. The discussion of the automatic initiation design of the AFW system as a part of the SGAHRS is presented in Section 7.4.1.1.7 of this SER. In addition, the applicants have satisfactorily described the manual initiation capability for the AFW system. The staff review has led to the conclusion that the AFW initiation capability (automatic and manual) meets the requirements of TMI Item II.E.1.2, Part 1 of NUREG-0737 and is, therefore, acceptable.

The PSAR Appendix H indicates that the CRBR design will include a Venturi flowmeter in each AFW system line to a steam drum and the provision of an indication in the control room of AFW flow through each Venturi flowmeter. The AFW flow indication will be safety grade and will be powered from the emergency power supplies. The staff has concluded that the flow rate indication meets all the requirements of TMI Item II.E.1.2, Part 2 of NUREG-0737 and is, therefore, acceptable.

7.4.2.3 Protected Air-Cooled Condenser Control

It was not clear from the review of the PSAR which instrumentation and control functions of the protected air-cooled condenser (PACC) were considered safety related. The staff asked the applicants to discuss the PACC and how the air flow will be controlled and to provide the safety classification with this discussion. The applicants stated that the outlet louvers will have discrete open and closed position sensors and that these will provide indication at both the local control panel and the main control panel in the control room. The inlet louvers will have both discrete open and closed position sensors and a continuous position sensor. The continuous position sensor will provide feedback to the louver control.

The fan blade pitch is sensed by continuous position sensors for both control and indication. This indication is provided locally and in the main control room. Both the discrete and continuous sensors are integral to the actuator. The applicants stated that all of the above instrumentation will be Class 1E with the exception of the indicating lights. Furthermore, the statement was made that all instrumentation and controls necessary for the PACC to perform its intended safety function will be safety related.

The applicants were asked if the operator would have any way of knowing the water level in the PACC loops. The applicants stated that there would be no level detectors on the PACC loops. However, the PACC condensate return line

will contain a Venturi flowmeter which will indicate condensate return flow locally and in the main control room. During normal plant operation, the water level in the PACC condensate return line will be essentially the same as the water level in the steam drum. During PACC operation, water level in the PACC condensate return lines will be higher because of the flow-induced pressure drop in the lines. The height of the water level will depend on the heat being removed by the PACC and, therefore, will be proportional to the condensate flow rate.

After a staff review of the PSAR and the applicants' responses to its questions, the staff concludes that the design criteria for the PACC instrumentation and controls are acceptable and that the applicants have satisfactorily identified the safety-related instrumentation and controls for PACC operation. However, the applicants are required to provide a detailed listing of the instrumentation and controls that are safety related for the PACC. The staff will verify this detailed listing during the operating license review.

7.4.2.4 Outlet Steam Isolation Instrumentation and Control System

The applicants have stated that the instrumentation and controls for the outlet steam isolation system (OSIS) have been designed to IEEE standards listed in PSAR Table 7.4-3 which includes IEEE Std. 279. Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the OSIS instrumentation and control system to perform its intended safety function. The staff concludes that the design criteria for the OSIS is acceptable; however, the applicants are required to complete their description of the OSIS bypasses and interlocks using design criteria acceptable for a safe shutdown system.

The applicants are required to complete the PSAR description of the OSIS bypasses and interlocks to demonstrate that they meet acceptable design criteria.

7.4.2.5 Pony Motor Control Circuits

The PSAR provided an inadequate description of the pony motor instrumentation and control systems for staff review. The applicants, in a letter dated October 4, 1982, provided an amended (Amendment 72) PSAR Section (7.4.3) which gave a design description for the pony motors and controls.

Six pony motors (75 hp, 480-V ac, 3 phase, 60 Hz), one in each primary and intermediate heat transport loop, will provide sodium flow for decay heat removal. An overrunning clutch will allow the pony motor to run continuously during all modes of plant operation and will automatically drive the pump when the vertical motor speed decreases below the output speed of the reducing gear.

The pony motor will be controlled using both a non-Class 1E and a Class 1E circuit. The non-Class 1E circuit will be isolated from the Class 1E circuit and will be overridden by the Class 1E circuit. Once started by either the Class 1E or the non-Class 1E control, the pony motor will automatically restart following the loss of offsite power and will then be powered by the Class 1E diesels. There will be no bypasses in the Class 1E control circuit. However, a sodium-to-water leak detection in the steam generators will result in an automatic trip of the affected intermediate heat transport loop pony motor.

The staff concludes that the design criteria for the pony motor control circuit are acceptable.

7.4.2.6 Safe Shutdown System Testing Procedures

The staff asked the applicants to describe the sequence for periodic testing of the outlet steam isolation valves, main feedwater control valves, main feedwater isolation valves, auxiliary feedwater system, and the pressure relief valves at the superheater. This description was to include features to ensure the availability of the safety function during test and measures taken to ensure that equipment cannot be left in a bypassed condition after test completion.

The applicants submitted the procedures for testing the above safe shutdown systems and they were discussed during a meeting with the applicants. The staff reviewed the procedures for testing these systems and concludes that they are acceptable for the construction permit review.

7.4.2.7 Remote Shutdown System

To comply with PDC 17 (GDC 19), the CRBR plant design should provide redundant safety-grade capability external to the control room to maintain the unit in a safe shutdown condition assuming no fire damage to any required systems and equipment and assuming no accident has occurred. Credit may be taken for manual actuation (exclusive of continuous control) of systems from locations that are reasonably accessible from the remote shutdown panel. Credit may not be taken for manual actuations involving jumpering, rewiring, or disconnecting circuits. The design should provide redundant safety-grade capability for subsequent control of the reactor at any coolant temperature lower than the hot shutdown condition through the use of suitable procedures.

The staff questioned the applicants with regard to meeting this criterion. The applicants stated that remote shutdown is accomplished with the use of the local controls for SGAHRS. The SGAHRS instrumentation and controls will be qualified to Class 1E requirements for each of the SGAHRS loops, A, B, and C. These redundant sets of controls will be located in separate cells on the 836-ft level of the steam generator building (SGB). In addition, the SGAHRS local panels will be seismically qualified. However, the remote shutdown monitoring panel (RSMP) originally was not to be seismically qualified. The staff asked the applicants to justify providing a nonseismically qualified RSMP. The applicants deferred providing this justification and, instead, stated in a letter dated December 6, 1982, that the RSMP will be seismically qualified.

Redundancy will be provided at CRBR by separating the A, B, and C loops of the SGAHRS electrically and physically so that failures in the equipment for one loop will not affect the other loops or the capability to remove heat from the plant. Similarly, the RSMP will provide separate indications for the A, B, and C loops for the PHTS, IHTS, and steam parameters. Separate indications will also be provided on the RSMP for the electrical parameters of the division 1, 2, and 3 diesel generators.

Remote shutdown operations at CRBR will not rely on jumpering, rewiring, or disconnecting the wiring of circuits. All of the instrumentation and controls used for remote shutdown operation can be switched between the normal mode and

the local mode of operation by built-in transfer switching capability. The manual transfer to the remote shutdown operation mode will not disable any automatic actuation of engineered safety features (ESF) equipment, nor change the operating status of this equipment.

The local operation of SGAHRS will also be used to remotely provide the capability for subsequent control of the reactor at any coolant temperature lower than the hot shutdown condition (i.e., refueling condition).

The CRBR design for remote shutdown will permit achieving and maintaining plant shutdown conditions if a loss of offsite power occurs coincident with remote shutdown operations. This will be accomplished by only using normal plant safety-related equipment which is designed to be applicable following a loss of offsite power.

The staff has reviewed the CRBR remote shutdown system design criteria and concludes that the system is acceptable and meets the requirements of PDC 17 (GDC 19).

7.4.3 Findings and Conclusions

The review of systems required for safe shutdown included the sensors, circuitry, redundancy features, and actuated devices that provide the instrumentation and control functions that prevent the reactor from returning to criticality and provide means for adequate residual heat removal.

The staff concludes that the systems required for safe shutdown can be designed to meet the relevant principal design criteria subject to resolution of the confirmatory item in Section 7.4.2.4 of this SER. Relevant criteria are found in PDC 2, 5, 11, 17, and 35, which correspond with GDC 2, 4, 13, 19, and 34, respectively. GDC 35 and 38 listed in the Standard Review Plan and 10 CFR 50, Appendix A (normally a requirement for safe shutdown of water reactors), are not considered applicable to the CRBR. However, several special CRBR principal design criteria for which no comparable general design criteria in 10 CFR 50, Appendix A, exist, are listed in Chapter 3 of the PSAR. These are

(1) CRBR PDC 4, "Protection Against Sodium and NaK Reactions"
(2) CRBR PDC 26, "Heat Transport System Design"
(3) CRBR PDC 36, "Inspection of Reactor Residual Heat Extraction System"
(4) CRBR PDC 37, "Testing of Reactor Residual Heat Extraction System"

The staff has conducted an audit review of these systems to determine conformance to the guidelines of the regulatory guides and industry codes and standards applicable to these systems. In Section 7.1 of this SER, the staff concluded that the applicants had adequately identified the guidelines applicable to these systems. Based upon its audit review of the current system design concepts for conformance to the guidelines, the staff finds that there is reasonable assurance that systems can be made to conform to the guidelines applicable to these systems.

The staff's review has included the identification of those systems required for safe shutdown which are to be designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of

its review, the staff concludes that the applicants have identified those systems and components consistent with the design bases for those systems.

Section 3.1.3.1 of the PSAR states a commitment to meet the requirements of PDC 2 and 5 (GDC 2 and 4) regarding environmental and natural phenomena for safety-related systems. The safe shutdown systems are safety related. The natural phenomena considered applicable are given in Sections 2.3, 2.4, and 2.5 of the PSAR. The environmental conditions are discussed in Chapters 2, 3, 6, 9, and 15 of the PSAR. Therefore, the staff finds that the identification of those systems and components satisfies this aspect of PDC 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," and PDC 5 (GDC 4), "Environmental and Missile Design Bases."

On the basis of its review, the staff concludes that instrumentation and controls will be provided to monitor and control variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant boundary, and the containment and its associated systems. Therefore, the staff finds that the systems required for safe shutdown will be able to satisfy the requirements of PDC 11 (GDC 13) "Instrumentation and Control."

Instrumentation and controls have been provided within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including a shutdown following an accident. Equipment and instrumentation will be located outside the control room for the purpose of maintaining safe shutdown. Therefore, the staff concludes that the systems required for safe shutdown satisfy the requirements of PDC 17 (GDC 19), "Control Room."

The staff's review of the instrumentation and controls required for safe shutdown has included the dependence of these systems on the availability of essential auxiliary support (EAS) systems. These supporting systems are reviewed in Section 7.6 of this SER. On the basis of its review, the staff concludes that the design of the EAS systems will be compatible with the functional performance requirements of these systems. Therefore, the staff finds the interfaces between the design of safe shutdown systems and the design of EAS systems to be acceptable.

The staff's review of the instrumentation and control systems required for safe shutdown included conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures consistent with the principal design criteria applicable to safe shutdown systems. The information contained in the PSAR gives reasonable assurance that these systems will be testable and operable on either onsite or offsite power. The applicants have not provided reasonable assurance that the backup DHRS instrumentation and control is completely independent and separate from the SGAHRS instrumentation and control (see Section 7.6 of this SER). Therefore, the staff finds that the systems required for safe shutdown will satisfy the relevant requirements of PDC 35 (GDC 34) with the resolution of the open item stated in Section 7.2.2.6 of this SER and described in Section 7.6.2.1 of this SER.

In summary, the staff concludes that the systems required for safe shutdown are acceptable and meet the relevant requirements of PDC 2, 5, 11, 17, 35, 36, and 37, subject to satisfactory resolution of the confirmatory item identified in

Section 7.4.2.4 of this SER and the open item identified in Section 7.2.2.6 of this SER.

7.5 Information Systems Important to Safety

7.5.1 Description

The information systems important to safety are composed of display instruments that provide information for manually initiated and manually controlled safety functions, to indicate that plant safety functions are being accomplished, and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. Information that the operator needs to maintain the plant in a hot shutdown condition or to proceed to a coolant temperature lower than the hot shutdown condition (i.e., refueling) within the proposed limits of the proposed Technical Specifications is also displayed. The operator uses these information systems to monitor the reactor, the heat transport systems, and the containment and process systems during normal operation of the plant, including anticipated operational occurrences and for postaccident monitoring. The display systems also include bypassed and inoperable status information. Various displays associated with safe shutdown are duplicated at the remote shutdown panel as described in PSAR Section 7.4.

The protective system provides the operator with information pertinent to systems status and safety. All transmitted signals (flow, levels, temperature, etc.) which can cause a reactor trip are indicated for each channel, including all neutron flux signals (wide-range log mean square voltage power level, wide-range linear power level, and primary power-range power level). Parameters associated with automatic actuation as well as those required to enable the operator to manually initiate engineered safety features systems are displayed. Any reactor trip will actuate an alarm and an annunciator. Such protective actions are indicated and identified down to the channel level.

The instrumentation and monitoring systems identified in Section 7.5 of the PSAR are the flux monitoring system, the heat transport instrumentation system, the reactor and vessel instrumentation system, the fuel failure monitoring system, the leak detection instrumentation systems, the sodium-water reaction pressure relief system, the containment hydrogen monitoring system, the containment vessel temperature monitoring system, the containment atmosphere temperature system, and the containment pressure monitoring system.

7.5.1.1 Flux Monitoring System

The flux monitoring system will be composed of the source-range subsystem, the wide-range subsystem, and the power-range subsystem. Each subsystem will provide three separate, independent channels of instrumentation. The primary reactor shutdown system (RSS) will use the power-range and the secondary RSS will use the wide-range instrumentation. The flux monitoring system will measure neutron flux proportional to reactor power over a span of more than 10 decades from shutdown to above full power and will provide indication and electrical outputs for plant protection, plant control, data handling and display, recording, and annunciation.

7.5.1.2 Heat Transport Instrumentation System

The heat transport instrumentation system will be composed of those subsystems which monitor the primary and intermediate sodium loops, the sodium pumps, and the steam generator. The primary and intermediate sodium loops will be monitored by the reactor inlet pressure subsystem (a three-channel primary RSS subsystem), the primary and intermediate loop flow subsystem (a three-channel secondary RSS subsystem), the intermediate heat exchanger (IHX) primary outlet temperature subsystem (a three-channel primary RSS subsystem), the primary and intermediate hot- and cold-leg temperature subsystem, the primary and intermediate pump discharge pressure subsystem, the IHX outlet pressure subsystem, the IHX differential pressure subsystem, the intermediate pump inlet pressure subsystem, the intermediate expansion tank level subsystem, and the evaporator sodium outlet temperature subsystem (a three-channel primary RSS subsystem). The sodium pumps will be monitored by the pump tank sodium level subsystem, the primary and intermediate speed subsystem (three channels will be used in the primary RSS), and the pony motor running subsystem.

The steam generator will be monitored by the sodium flow subsystem, the sodium temperature subsystem (a three-channel secondary RSS subsystem), the sodium pressure subsystem, the steam and water flow subsystem (three channels will be used in the primary RSS), the steam and water temperature subsystem, the steam and water pressure subsystem, the evaporator and super heater outlet pressure subsystem, the steam drum level subsystem (a three-channel secondary RSS subsystem), and the water chemistry subsystem.

7.5.1.3 Reactor and Vessel Instrumentation System

The reactor and vessel instrumentation system will include the monitoring of all in-vessel temperature, sodium level, and vibration sensors required for the reactor shutdown system, for the process control system, for required surveillance, and for design verification.

7.5.1.3.1 Sodium Level Subsystem

The sodium level subsystem will be composed of six channels of instrumentation which monitor the reactor vessel sodium level. Three of the channels will be used for the primary RSS. An installed spare level channel will be provided to maintain the three RSS operating channels without a shutdown in the event of a level-sensor failure. The two remaining instrument channels will be used to provide indication in the reactor control room for accident-monitoring purposes.

7.5.1.3.2 In-Vessel Temperature Monitoring Subsystem

The in-vessel temperature monitoring subsystem will be a multitemperature monitoring subsystem which inputs the temperature sensor signals to the plant data handling and display system (PDH&DS) and to the plant control system for surveillance and design verification. Three hundred thirty eight sensors will be utilized for accident-monitoring instruments.

7.5.1.3.3 In-Vessel Vibration Monitoring Subsystem

This subsystem will consist of four nonreplaceable biaxial accelerometers which sense the vibration within the reactor vessel. The subsystem will function for

design verification and need not function beyond the first 6 months of reactor operation; however, it will be required to physically withstand the sodium environment for the life of the reactor.

7.5.1.4 Fuel Failure Monitoring System

The fuel failure monitoring system will be composed of those subsystems whose function is to detect and locate fuel or blanket failure in the presence of up to 4 existing failures, detect and locate fuel or blanket failure in the presence of 59 previously failed pins, characterize failed pins as to burnup and other information to permit correlation with core and blanket history, and detect less than 1.5 cm^2 of fuel area exposed to sodium. The subsystems which will make up the fuel failure monitoring system are the cover gas monitoring subsystem, the reactor delayed neutron monitoring subsystem, and the failed fuel location subsystem.

7.5.1.4.1 Cover Gas Monitoring Subsystem

The cover gas monitoring subsystem will continuously sample the cover gas and determine through gamma analysis (1) the concentration of selected radioactive fission gases to detect each instance of fuel or radial blanket pin cladding failure, and (2) the concentration of radioactive fission gases to characterize failed pins as to burnup and other information.

7.5.1.4.2 Reactor Delayed Neutron Monitoring Subsystem

The reactor delayed neutron monitoring subsystem will continuously monitor the sodium coolant for the presence of fission products which decay with the emissions of neutrons. A predetermined increase in the neutron signal from the sodium in the primary heat transport system above the normal background level will be taken as an indication of fuel contact with sodium.

7.5.1.4.3 Failed Fuel Location Subsystem

The failed fuel location subsystem will consist of a gas tag sampling cryogenic trap and a mass spectrometer. Each fuel and radial blanket assembly pin will contain a tag gas and fission gases. All individual pins of a given assembly will contain the same unique mixture of tag gases. If a pin failure occurs, a fraction of the tag gas and fission gases in the pin escapes to the cover gas and will be detected by the cover gas monitoring subsystem which will initiate operation of the gas tag sampling cryogenic trap and mass spectrometer. The failed assembly will be identified by matching the results of the mass spectrometer analysis with previously determined analyses of all gas tags in the reactor, suitably corrected for burnout, production, and background. The subsystem will locate a fuel or blanket failure in the presence of up to four existing failures.

7.5.1.5 Leak Detection Instrumentation Systems

The leak detection instrumentation systems for the CRBR will consist of the liquid metal-to-gas leak detection system, the intermediate to primary heat transport system leak detection system, and the steam generator leak detection system.

7.5.1.5.1 Liquid Metal-to-Gas Leak Detection System

The liquid metal-to-gas leak detection system will provide a continuous surveillance of the liquid metal systems boundaries and will detect and locate a liquid metal break. The system will monitor (1) the reactor enclosure, (2) the primary heat transport system, (3) the intermediate heat transport and steam generator system, and (4) the impurity monitoring and analysis, refueling and auxiliary liquid metal system, for any liquid metal leaks.

7.5.1.5.2 Intermediate to Primary Heat Transport System Leak Detection System

The intermediate heat transport system (IHTS) will be maintained at $\geq$10 psi higher than the primary heat transport system (PHTS) at the IHX to prevent radioactive primary sodium from entering the IHTS in the event of a tube leak. Maintaining a positive pressure differential across the IHX will ensure that the leakage across the IHX tube barrier will result in an inflow of sodium into the primary system, causing a loss of sodium level in the IHTS. The sodium inventory in the IHTS will be monitored by tracking sodium levels and correcting for loop temperature effects. The reactor operator will be alerted upon detection of a large loss of IHTS sodium inventory.

7.5.1.5.3 Steam Generator Leak Detection System

The steam generator leak detection system will be designed to detect small water-to-sodium and steam-to-sodium leaks in the steam generator modules, to identify the module in which the leak has occurred, and to alert the control room operator. The leak detection will be based on the measurement of the hydrogen and oxygen concentration in the sodium. Hydrogen- and oxygen-detection instrumentation will monitor the sodium exiting from the superheater and the evaporators. The sodium-filled vent lines from the evaporator vents and the superheater vent will also be monitored. An increase above the background concentration of hydrogen or oxygen will indicate the presence of a leak.

7.5.1.6 Sodium-Water Reaction Pressure Relief System Instrumentation and Controls

The sodium-water reaction pressure relief system (SWRPRS) will be designed to detect the inception of a large or intermediate water-to-sodium leak in any of the steam generator modules.

For a large leak, three Class 1E pressure sensors (nine per loop) will be provided immediately downstream from each pair of rupture discs in the superheater and evaporator (two) reaction products vent lines. The signals will be transmitted to the secondary RSS which initiates a reactor trip and PHTS and IHTS sodium pump trip. Buffered signals will be transmitted to the SWRPRS trip logic which isolates the affected loop. A group alarm will be transmitted to the plant annunciation system (PAS). For intermediate leaks, three pressure sensors will be provided in the IHTS sodium expansion tank equalization line to the sodium dump tank, downstream of the rupture discs. These signals will be transmitted directly to the SWRPRS trip logic via a 2-out-of-3 coincidence logic which isolates the affected loop. The SWRPS is discussed in more detail in Section 7.3.2.7 of this SER.

7.5.1.7 Containment Hydrogen Monitoring System

The containment hydrogen monitoring system will be designed to monitor the hydrogen concentration in the upper levels of containment during events that lie beyond the design basis for the plant.

The hydrogen instrumentation will consist of two fully redundant analyzer channels. The containment atmosphere will be sampled through an entry filter located near the top of the reactor control building (RCB). The air samples will be pumped to the analyzer which will be located in the SGB and will operate on the principle of thermal conductivity. From there, signals will go to the control room where the hydrogen concentration readout will be provided.

7.5.1.8 Containment Vessel Temperature Monitoring System

The containment vessel temperature monitoring system will be designed to monitor the containment vessel temperature.

The temperature instrumentation will consist of two fully redundant channels. Each channel will consist of eight thermocouples mounted at various locations on the inside of the containment wall, with each thermocouple providing a signal to conditioning instrumentation in the steam generator building (SGB). The instrumentation will send signals to the control room where individual readouts will be provided. This system will also be required to perform functions for events which lie beyond the design basis for the plant.

7.5.1.9 Containment Pressure Monitoring System

The containment pressure monitoring system will provide indication in the control room of the pressure inside the containment above the operating floor.

The pressure instrumentation will consist of a pressure detector inside the containment vessel. Signals will be provided to the display and alarm panel in the control room so that continuous readout will be provided to the plant operator. This system will also be required to perform its functions for events which lie beyond the design basis for the plant.

7.5.1.10 Containment Atmosphere Temperature Monitoring System

The containment atmosphere temperature monitoring system will provide indication in the control room of the atmosphere temperature inside the containment building.

The temperature instrumentation will consist of two fully redundant and independent channels. Each channel will provide signals to conditioning instrumentation which in turn will provide signals to the control room where individual readouts will be provided. This instrumentation system will be required to perform functions for events which are beyond the design basis for the plant.

7.5.1.11 Accident Monitoring Instrumentation

The PSAR discussion of accident monitoring was limited to identifying criteria for the instrumentation and those functions which the instrumentation is to

perform. At the present time, Table 7.5-1 of the PSAR identifies that accident monitoring will be associated with the flux monitoring system, core sodium exit temperature instrumentation, and sodium level above core instrumentation.

However, additional accident monitoring instrumentation has been identified in other areas of the PSAR as a result of the requirements of RG 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." The requirements of RG 1.97, Revision 2, are applicable in principle to CRBRP. The discussion regarding accident monitoring instrumentation and how it will meet the implementation requirements of RG 1.97, Revision 2, is presented in Section 7.5.2.8 of this SER.

7.5.2 Specific Findings

The information systems important to safety have been reviewed on the basis of the information presented in the CRBR PSAR. The review included determining conformance of the information systems important to safety to the applicable principal design criteria, general design criteria, regulatory guides, and industrial codes and standards outlined in SRP Section 7.5, Parts II and III. A comparison of these criteria with those that will be invoked by the applicants in the design of the information systems important to safety was also made. For those informations systems which have been categorized by the applicants as safety-related instrumentation and control systems (refer to Table 7.1-1 of the PSAR), the applicants have invoked compliance of the same criteria to the design of these information systems as those invoked for the primary and secondary RSS (refer to Section 7.2.2 of this SER).

7.5.2.1 Source-Range Subsystem

The staff questioned whether the source-range subsystem should be safety related and comparable to that in a PWR plant (i.e., with overlapping system ranges). The applicants provided the following response. In order to make CRBRP critical, the secondary control rods must be withdrawn first. Thus, at startup, the major investment of "shutdown" reactivity will be under the control of the secondary reactor shutdown system. The startup flux protection functions in the secondary reactor shutdown system will be based on the use of three circumferential ex-vessel fission chambers, which will provide a continuous measurement of reactor power extending from shutdown to full power, by means of a Class 1E, wide-range system. The wide-range system will consist of three channels. These channels can be compared and tested before startup to ensure that continuous and consistent measuring capability in conformance with previous calibration will exist over the entire range.

Trip and bypass functions are therefore not required to ensure that successive detectors will be operational during startup (as is usual with a conventional three-range system). An analysis has shown that the startup trips related to the secondary system will be able to meet the requirements of PDC 8 and 24 (GDC 10 and 26) for rod withdrawal accidents initiated from any startup condition.

The source-range flux monitoring system will be supplied as a highly sensitive refueling monitor. It will provide a verification signal for the wide-range channels, but will not otherwise have any safety-related functions.

The primary reactor shutdown system also has flux trip functions initiated from three Class 1E linear power channels which monitor three ion chambers located around the reactor core. These power range channels will cover the top two to three decades of power. The startup trips related to these channels will safely shut down the reactor to meet the requirements of PDC 8 and 24 in the event of the failure of the secondary shutdown system following a startup reactivity excursion.

In addition, diversity will be provided in the primary and secondary shutdown systems (Section 7.2 of this SER) and the CRBR design will have the capability of monitoring the core during refueling operations after a seismic event with the wide-range seismically qualified system.

The staff has reviewed the information submitted and has concluded that the present nonsafety-grade source range subsystem is acceptable. However, the applicants are required to provide Technical Specification requirements for testing the operability of the secondary system immediately before startup.

7.5.2.2 Sodium-Level Probes

The staff asked the applicants to provide a discussion regarding the CRBR sodium-level measurement instrumentation with emphasis on the adequacy of the reactor vessel level gages.

The discussion provided indicated that mutual inductance-type sodium-level instruments will be used for all continuous sodium-level measurements in the reactor vessel, sodium expansion tank, sodium dump tank, and the sodium pump tank. Other types of level probes (balanced bridge inductive-level probes, displacement-float-type-level transducers, delta-type-level transducers, and time domain reflectometry transducers) were evaluated by the applicants and were considered not nearly as reliable as the mutual-inductance-type probes.

The mutual-inductance-level probe will have a primary and secondary inductance coil. Excitation will be applied to the primary coil which will develop a signal in the secondary coil. The signal magnitude in the secondary coil is dependent upon the height of the sodium.

To compensate for sodium temperature changes, a temperature compensation circuit is integral with the signal condition equipment and works on the concept of resistance changing with temperature. The compensation circuit measures the voltage and current in the primary coil and evaluates changes to determine the resistance change and automatically adjusts the output of the signal conditioner based on the resistance change.

The reactor vessel will contain four narrow-range probes, three of which will be used by the primary reactor shutdown system (RSS), and two wide-range probes which are designated to be part of the accident monitoring (AM) system. The range for the narrow-range probes (30 in.) will be wide enough to cover the normal operating range in the vessel but will be narrow enough that the accuracy of the measurement is maximized. The measurement range for the wide-range probes (189 in.) is based on the ability to monitor the level down to the reactor vessel outlet nozzles.

Each primary pump will contain two redundant wide-range probes (80.5 in.) to monitor sodium level over the full elevation of the pump tank. Sodium level will be measured in the intermediate system via the sodium pump and expansion tank. The intermediate sodium pumps will have a single wide-range probe (86.9 in.) installed in the pump tank which will monitor the full range of the sodium level in the pump tank. Two level probes will be installed in the sodium expansion tank: a wide-range probe to measure the full range of anticipated steady-state and transient sodium levels in the tank and a narrow-range probe for accuracy during fill of the system. The wide-range-level probe in the expansion tank also will provide a signal for a high- and low-level alarm. The pump tank-level probe will provide a signal for a high- and low-level alarm, and isolation of IHTS argon cover gas system.

Two wide-range-level probes will be installed in each sodium dump tank. These probes will be arranged with overlap to provide for monitoring sodium levels during sodium-fill and drain operations of the intermediate heat transport system.

The staff has reviewed the information provided by the applicants in the PSAR and in the responses to questions and concludes that the design criteria for the sodium-level probes are acceptable.

7.5.2.3 Design Criteria for the Fuel Failure Monitoring System

The fuel failure monitoring system is not defined in Chapter 7 of the PSAR as safety related.

The applicants have stated that the radiation detectors in the radioactive argon processing system (RAPS) will detect failed fuel in the required time. This time has been established in the PSAR Chapter 15 safety analyses. The CRBR Program Office has concurred that these detectors will be adequate.

In addition, the fuel failure monitoring system is neither intended nor required to perform after an accident and will not be needed during or after a shutdown.

Therefore, the staff has concluded that the nonsafety-related classification for this system is acceptable.

7.5.2.4 Loose Parts Monitoring

The CRBR Program Office has conducted the review of the loose parts monitoring system. This review is discussed in Section 4.4.7 of this SER.

7.5.2.5 Delayed Neutron Detection System

The delayed neutron detection (DND) system will include a delayed neutron monitor consisting of three gas-filled proportional neutron detectors that will continuously monitor the coolant sodium for delayed neutrons emitted by decay of radioactive precursors in the sodium. The delayed neutron detection will occur immediately upon fuel exposure to sodium, thus having the best capability for rapidly detecting failures that have been caused by or can lead to an uncoolable geometry.

The applicants have stated that the reactor will be designed to accommodate operation with failed fuel. Thus, the fuel failure monitoring system will not be required to prevent accidents associated with the release of radioactivity. In fact, the radiation monitoring system will provide the means for detecting, measuring, and warning of excessive radiation levels within the plant, and of excessive radionuclide concentrations within and external to the plant.

As discussed in Section 15.4, "Local Failure Events," of this SER, the staff considers that the DND system should be viewed as an instrumentation system important to safety, should at a minimum be designed for seismic Category II conditions, and that plant operation should be contingent upon at least two of the three PHTS loops having their delayed neutron detectors in operation.

Whether the DND system needs to provide a signal for automatic shutdown of the reactor (i.e., be made part of the PPS) or should provide a signal for operator action only will depend on the results of the run beyond cladding breach testing and fast flux test facility experience and will be determined in the FSAR review.

As part of developing the FSAR, appropriate limits on the DND signal should be established which would result in a plant shutdown or scram upon indication of a situation where coolable geometry can no longer be ascertained. In addition, it is suggested that the system be designed in accordance with IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," so as to provide flexibility for incorporating this system into the plant protection system (PPS) if found necessary.

On the basis of available limited evidence, it appears highly unlikely that rapid failure propagation could result from any of the identifiable events for CRBR. However, in view of the lack of final test data, the applicants should maintain a degree of flexibility in the design criteria for the DND system. The staff will determine during the operating license (OL) review if the DND system needs to provide a signal for automatic shutdown of the reactor.

7.5.2.6 Leak Detection Systems

The staff reviewed the liquid metal-to-gas leak detection system and discussed it in Section 7.3.2.6 of this SER. In this section the steam generator leak detection system is discussed.

The steam generator leak detection system is provided to give early indication of small water/steam-to-sodium leaks, thereby allowing the operator to take corrective action to reduce the system pressure before significant steam generator damage occurs and before discs rupture. This leak detection system continuously monitors the hydrogen and oxygen concentrations in the sodium stream. This is discussed in the PSAR Section 7.5.5.3.2.

The applicants have stated that this leak detection system serves only to decrease component damage and plant down time in the event of a small sodium-water reaction. Therefore, no corrective action is initiated by signals from the oxygen and hydrogen meters. The leak detection system is not a part of the CRBR plant protection system and is not subject to the requirements of IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

The staff has reviewed this system and has concluded that the applicants are not required to provide automatic actuation signals and that the principal design criteria for the steam generator leak detection system are acceptable.

Overpressure protection for the intermediate heat transport system (IHTS) is provided by the sodium-water reaction pressure relief system (SWRPRS). SWRPRS is discussed in Section 7.3.2 of this SER.

7.5.2.7 Containment Monitoring System

The containment hydrogen monitoring system, the containment vessel temperature monitoring system, the containment atmosphere temperature monitoring system, and the containment pressure monitoring system are only briefly discussed in the PSAR as informational systems important to safety. The applicants, however, stated that the design criteria and design requirements for these systems are enveloped by the requirements of the thermal margin beyond design base (TMBDB) which are described in Westinghouse's document CRBRP-3, Volume 2, Revision 1, pages 2-11 to 2-16 and 2-29 to 2-34d.

In this document, the statement is made that these systems shall be designed, manufactured, and qualified to all standards applied to Class 1E instrumentation. The channels for these systems will be completely independent and physically separated in accordance with RG 1.75 and powered from the Class 1E power system. However, a review of this document indicates that these systems will normally be connected to a non-Class 1E power source and that remote manual connection to the Class 1E power source will be made from the control room after other loads which are nonessential are removed.

These systems (designated TMBDB instrumentation) are required for continuous readout and for events beyond the design basis. They are used to aid in the proper operation of the engineered safety features (ESF) which are required to mitigate design-basis accidents. Because of their importance to safety, the staff required further information on the procedure for the manual switchover to the Class 1E power source. The applicants were required to justify the normal power mode (non-Class 1E) and the remote manual switchover to a Class 1E power source (i.e., location of switches and time required).

In a letter dated February 15, 1983, from J. R. Longenecker to J. N. Grace, the applicants stated that this concern arises from a misinterpretation of wording in CRBRP-3. Section 2.2.13 of CRBRP-3 has been modified to state that the TMBDB instrumentation will be connected to Class 1E electrical power and will be energized during both normal and emergency plant operation. Other electrical loads for TMBDB features will be connected to Class 1E electrical power supplies; however, they will remain deenergized. When the equipment is required to operate during the TMBDB event, it will be remote manually energized from the control room. This is acceptable because manual initiation (operator action required) of the TMBDB features will not be required for the first 24 to 36 hours of the event. The staff has reviewed this additional information and has concluded that this concern is resolved.

7.5.2.8 TMI Action Plan Item II.F.3, Postaccident Monitoring Instrumentation

The requirements of RG 1.97, Revision 2, are applicable in principle to CRBR. Proper provisions for instrumentation to monitor plant variables and systems

during and following an accident are as necessary for CRBR as they are for an LWR plant. However, the requirements delineated in RG 1.97, Revision 2, are specifically for LWR plants and a different set is required for the CRBR design because of the technological differences.

The applicants were required to discuss the implementation of the RG 1.97, Revision 2, requirements. This discussion was provided in Appendix H, Item II.F.3 ("Instrumentation for Monitoring Accident Conditions," (RG 1.97)).

In Appendix H, the applicants committed to provide accident monitoring instrumentation in accordance with RG 1.97, Revision 2, as it applied to CRBR and provided a discussion of the design basis for the accident-monitoring functions. In addition, preliminary information describing the specifics of the accident-monitoring design were included in a response to a staff question (760.6, PSAR Amendment 71).

Commission policy requires that conformance to the provisions of RG 1.97, Revision 2, be addressed in the broader context of the requirements for emergency response capability. The staff will reserve the review of RG 1.97, Revision 2, implementation until information describing the specifics of the accident-monitoring design is provided. The staff will verify the resolution of this concern during the OL review. The applicants have made sufficient commitments with respect to accident monitoring for the CP stage.

7.5.2.9 Inoperable Status Monitoring System

Upon staff request, the applicants provided information pertaining to the CRBR bypassed and inoperable status indication system. This information described the means to be used for compliance with the recommendations of RG 1.47, the design philosophy to be used in the selection of equipment/systems to be monitored, and how the design of the bypassed and inoperable status indication systems will comply with Positions B1 through B6 of BTP-ICSB 21.

There are two active safety systems which are not included in the inoperable status monitoring system (ISMS): the reactor shutdown system, which is provided with separate indication, and the containment isolation system, for which no bypasses or deliberately induced inoperable states have been identified. The safety functions and systems which are monitored by ISMS are shown in Table 7.5-4 of the PSAR.

The PSAR states that the ISMS data-acquisition system will obtain the sensor-status information, process the sensor data, and provide the status indication to the unit operator.

Further review and discussions revealed that the ISMS will be supported by the plant annunciation system (PAS) and the plant data handling and display system (PDH&DS). The activation of the inoperable status indication will be provided by a computer program which will not be accessible to the plant operator. The applicants have stated that all safety inputs (connections) to the ISMS and the PDH&DS will be electrically isolated from the safety-related signal sources.

The applicants have committed to meeting the guidelines of RG 1.47 and BTP ICSB-21. This is acceptable to the staff. However, it is not clear to the

staff the exact role that the PDH&DS has in the ISMS and whether or not the ISMS uses its own dedicated data-acquisition system (i.e., microprocessor, minicomputer). For the OL review, the applicants are required to provide additional information detailing the ISMS design, the role of the PDH&DS in the ISMS, the capability for manual actuation of the safety system inoperable status indicators, and the primary location for the inoperable status indication (the PDH&DS cathode-ray tube or the ISMS panel).

7.5.2.10 Loss of Non-Class 1E Instrumentation and Control Power Bus During Operation (IE Bulletin 79-27)

If reactor controls and vital instruments derive power from common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. IE Bulletin 79-27 addresses several concerns related to the above subject.

The applicants stated in a meeting that the concerns of the staff enumerated in IE Bulletin 79-27 have been listed in the Summary Report on the Conduct of the Clinch River Breeder Reactor Plant (CRBRP) Key Systems Reviews, dated February 1982.

PSAR Reference 7.9.1 describes the CRBR Key Systems Reviews and includes the Class 1E and non-Class 1E electrical power distribution and instrumentation and control review. Section 4.12 of the Summary Report on the Conduct of the Key Systems Reviews provides a discussion of the CRBR review effort which will be performed regarding IE Bulletin 79-27.

The following is the scope of effort for the review which will be performed.

(1) Identify the instrument and control system loads connected to the buses and evaluate the effects of loss of power to these loads including the ability to achieve shutdown and safely remove decay heat.

    The loads will consist of the individual sensors, signal conditioners, indicators, or other elements that make up an instrument or control loop.

(2) Identify the cross-connection of power supplies, particularly in the Class 1E area, that is, sensor and indicator on one loop supplied from different power divisions.

(3) Identify alternate indications and/or control circuits which may be powered from other Class 1E or non-Class 1E instrumentation and control buses.

(4) Generate operating base procedures to be used by the operator upon loss of power to each Class 1E and non-Class 1E bus supplying power to safety and nonsafety-related instrument and control systems.

The staff has concluded that the applicants' commitment and documented scope of effort are acceptable for the CP review, and the staff will verify resolution of this issue during the OL review.

7.5.2.11 TMI Action Plan Item II.D.3, Direct Indication of Relief and Safety Valve Position

TMI Action Plan Item II.D.3 states that the reactor coolant system relief valves on an LWR shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe. This item was provided to resolve a concern for LWRs involving the failure of relief and safety valves to close on the reactor coolant pressure boundary resulting in small loss-of-coolant events. An unambiguous indication of the position of the valves was to aid the operator in detecting a failure and taking the proper corrective action.

In a letter dated February 15, 1983, from J. R. Longenecker to J. N. Grace, the applicants stated that the potential for loss of coolant to the reactor through safety relief valves such as that which occurred at Three Mile Island does not exist for the CRBRP. TMI Action Plan Item II.D.3 deals with the inability to cool the reactor as a result of loss of coolant to the reactor through safety relief valves. Therefore, the applicants stated that neither the requirement nor the principle of the requirement has any application for CRBRP.

Valves will exist in the steam generator auxiliary heat removal system (SGAHRS) and in the steam generator system (SGS) that will be directly in the heat transport path. The valves involved are the SGAHRS steam drum and superheater vent control valves and the SGS safety valves located on the superheater outlet, the evaporator outlets, and the steam drum.

The CRBR design for valve position indication for the above valves is as follows.

(1) The SGAHRS steam drum and superheater vent control valves will be provided with position indicators in the main control room. Acoustic sensors located on the vent piping downstream of the valves will detect any steam leakage past the seat of a closed valve and the leakage will be alarmed and annunciated in the main control room.

(2) The SGS safety valves will be pilot-operated valves which will open when system pressure reaches their setpoints. In addition, the evaporator outlet and superheater outlet safety valves will open when actuated by an air-operated actuator. Main control room indication of pilot stem position, not main valve stem position, will be provided for these valves.

(3) To provide a backup to the pilot stem position indicators, an acoustic sensor will be added to the vent piping downstream of each SGS safety valve. These sensors will detect either a stuck-open valve or any steam leakage past the seat of a closed valve. These conditions will be alarmed and annunciated in the main control room.

The staff has reviewed the information provided by the applicants, the applicable parts of the PSAR, and the requirements of TMI Action Plan Item II.D.3 as stated in NUREG-0737 and has concluded that the requirements of this TMI item are not applicable to the CRBRP.

In addition, the staff has concluded that there will be adequate indication of valve position for the SGAHRS and SGS valves discussed above. Loss of water

inventory at a rate sufficient to be of a safety concern can be detected and the affected loop isolated by Class 1E instrumentation. This instrumentation will consist of steam drum pressure, steam drum level, and main feedwater flow or SGAHRS auxiliary feedwater flow instrumentation.

In addition, the applicants will be required to verify during the operating license review that the acoustic monitors will be capable of distinguishing between valve "leakage" and significant flows which could impact plant safety.

7.5.3 Findings and Conclusions

The information systems important to safety are to provide the operator with information on the status of the plant and to allow manual safety action to be performed when necessary. The review has included the applicable acceptance criteria and guidelines and design bases, including those for indication of bypassed or inoperable safety systems. The review has also included the applicants' analyses of the manner in which the design of information systems conforms to the acceptance criteria of the principal design criteria and guidelines which are applicable to these systems as noted in the staff's Standard Review Plan.

The staff has conducted an audit review of these systems for conformance to guidelines of the applicable regulatory guides and industry codes and standards. In Section 7.1 of this SER, the staff concluded that the applicants had adequately identified the guidelines applicable to these systems. On the basis of its audit review of the systems designs for conformance to the guidelines, the staff finds that there is reasonable assurance that the systems will conform to the guidelines applicable to them.

The staff's review has included the identification of those systems and components of the information systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicants have identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of these systems and components satisfies this aspect of PDC 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," and PDC 5 (GDC 4), "Environmental and Missile Design Bases."

The staff concludes that the redundant safety-grade information systems adequately conform to the guidance for the physical independence of electrical systems provided in RG 1.75.

The staff concludes that the information systems important to safety will include appropriate variables and that their range and accuracy will be consistent with the plant safety analysis. Therefore, the staff finds that the information systems will satisfy the requirements of PDC 11 (GDC 13), "Instrumentation and Control," for monitoring variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions. This conclusion is based on the review of the PSAR data concerning the

appropriate variables, their ranges and their accuracies. The available data were considered to be appropriate for the PSAR review and the staff expects that the ensuing FSAR data will be consistent with the plant-safety analysis.

The staff finds that conformance to PDC 11 (GDC 13) and the applicable guidelines satisfies the requirements of PDC 17 (GDC 19), "Control Room," with respect to information systems provided in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions. It should be noted that the CRBR design is being upgraded to comply with the provisions of RG 1.97, Revision 2, as discussed in Section 7.5.2.8 of this SER.

In summary, the staff concludes that the information systems important to safety will be acceptable and will meet the requirements of PDC 2, 5, 11, and 17 (GDC 2, 4, 13, and 19).

7.6 Other Instrumentation and Control Systems Required for Safety

The instrumentation and control systems included in this section are those required for safety, but not previously discussed in Sections 7.2 through 7.5. These systems are the instrumentation and control systems for the following:

(1) emergency plant service water systems
(2) emergency chilled water system
(3) direct heat removal service
(4) heating, ventilation, and air conditioning systems
(5) fuel-handling and storage safety interlock systems
(6) radiation monitoring systems
(7) recirculation gas cooling system
(8) steam generator building flooding protection

7.6.1 Description

7.6.1.1 Emergency Plant Service Water System

The service water systems are described in Section 9.9 of the PSAR. The normal plant service water system is listed as a nonsafety-related system. The emergency plant service water (EPSW) system is listed as a seismic Category I, safety-related system. It will be designed to provide sufficient cooling water to permit safe shutdown and the maintenance of the safe shutdown condition of the plant in the event of an accident resulting in the loss of the normal plant service water system or the loss of the plant's ac power supply and all offsite ac power supplies (motors will be switched automatically to the standby diesel generator). The emergency plant service water system will not be used during normal plant operation. The system will provide the emergency chilled water system chiller condensers and the standby diesel generators with cooling water. Additionally, this system will provide fire-fighting water for the seismically qualified fire pumps of the nonsodium fire-protection system. The emergency plant service water system will include the emergency cooling towers and emergency cooling tower basin.

The EPSW system will be fully automatic, normally controlled from the main control panel in the control room. Redundant controls will be provided that

will allow full operation of the system from a control panel in the diesel generator building.

Instrumentation for the EPSW system will be provided for local and/or remote (control room) indication. A flow switch, located in the return line from each diesel generator and emergency chiller, will detect an abnormal low flow condition and energize an annunciator in the control room.

Level, temperature, and air flow instrumentation will be provided at the emergency cooling tower structure with signals transmitted to the control room.

7.6.1.2 Emergency Chilled Water System

The chilled water systems are described in Section 9.7 of the PSAR. The normal chilled water system is a nonsafety-related system not required to cool any safety-related equipment during accident conditions. The emergency chilled water system (ECWS) is safety related and will be designed to seismic Category I and connected to the Class 1E ac power supply.

The ECWS will provide a backup source of chilled water for the chilled water coils of the air handlers and unit coolers located in the control building, reactor service building, reactor containment building, and steam generator building. It will be designed to operate during accident conditions. The system will be in standby during normal plant operation.

Failure of any component in the operating chilled water loop will activate an alarm in the control room. A single-failure analysis of the emergency chilled water system is given in Table 9.7-5 of the PSAR. If, during normal operation, the normal chilled water supply is interrupted, flow switches in the emergency chilled water supply header will close the isolation valves between the two sytsems and automatically start the emergency plant service water system and then the emergency chilled water system.

The PSAR states that the ECWS instrumentation and controls will be periodically tested according to RG 1.22.

7.6.1.3 Direct Heat Removal Service

The direct heat removal service (DHRS) will be a backup shutdown heat removal system that will not depend on the IHTS, SGS, or SGAHRS. The DHRS will consist of one heat removal path capable of removing approximately 11 MWt from the primary coolant. The capability will be adequate to prevent loss of coolable geometry assuming that the SGS heat sink is lost immediately upon shutdown from rated power and the active operation of DHRS is initiated ½ hour after shutdown. The system will use two NaK loops and air-heat exchangers to dissipate heat.

The DHRS instrumentation and control system will be designed to permit the monitoring of system conditions and to provide alarm indication of off-normal conditions. These will be the same instrumentation and controls that will be provided for ex-vessel storage tank (EVST) cooling (PSAR Section 9.1.3.1.5) and the reactor primary sodium overflow circuits (PSAR Section 9.3.2.5) with the addition of few temperature-monitoring instruments located on the NaK lines connecting the overflow heat exchanger with the EVST NaK cooling loops.

The DHRS will be remote manually activated and will be controlled from the control room. Instrumentation required to monitor the condition of the DHRS will consist of thermocouples on the EVST sodium outlet lines (three loops) and level indicators in the EVST and the reactor vessel (RV). These instruments will confirm that the sodium level in the RV remains above the loop outlet nozzles and that temperatures remain below design limits. Other DHRS diagnostic instrumentation will not be essential for DHRS operation as the pumps and airblast heat exchanger will be operated at the maximum design rating. When the reactor decay heat load has dropped sufficiently, the cooling capacity of the system may manually be reduced by lowering the flowrates, the fan speed, or by shutting down one of the EVST cooling trains.

Instrumentation in the primary sodium storage and processing system that is a part of the DHRS is described in PSAR Section 9.3.2.5 ("Overflow and Makeup Circuit"). Instrumentation in the ex-vessel storage sodium-processing system that is a part of the DHRS is described in PSAR Sections 9.3.3.5 and 9.1.3.1.5.

When the DHRS will be activated, all automatic controls of the EVST airblast heat exchangers will be bypassed, the pumps and valves, which will be remotely set, will provide maximum flow through the DHRS loops, and the airblast heat exchangers, which will be remotely set, will provide maximum cooling capability. Control of the pumps and the airblast heat exchangers will be provided from three separate locations: (1) a field panel adjacent to or in a cell adjacent to the equipment, (2) a local panel in the same building as the equipment, and (3) the control panel in the main control room. The capability to provide power directly to the pumps, by bypassing all panel voltage and interlock control functions, will also be provided so that no control function failure can keep the DHRS electrical equipment from operating. The DHRS will be interconnected with the primary sodium processing and the EVS sodium processing systems. The DHRS functions will be controlled from separate, redundant control panels, one local and the other in the main control room.

Functional testing of all portions of DHRS that will not be used during the course of normal operations will be performed on an annual basis during reactor refueling.

7.6.1.4 Heating, Ventilating, and Air Conditioning Systems

The safety-related HVAC systems will be provided with local controls and remote controls from the control room. The remote controls located in the control room will override the local controls, unless the "key lock" transfer switch located on the local panel commands local control preference over the control room operation, during control room unavailability. These systems will also be provided with local monitoring and alarms and remote monitoring and alarms in the control room.

The HVAC systems identified in Chapter 9 of the PSAR as safety related are

(1) control and diesel generator building emergency HVAC system
(2) control room HVAC system
(3) portion of the reactor containment building HVAC system
(4) annulus cooling and filtration system
(5) portions of the reactor service building HVAC system including:

(a) fuel-handling area HVAC systems

(b) radioactive argon processing system (RAPS) and cell atmosphere processing system (CAPS) exhaust and unit cooler system

(c) cell cooling systems

7.6.1.5 Fuel-Handling and Storage Safety Interlock System

The PSAR states in Section 7.6 that the fuel-handling and storage safety interlock system is an instrumentation and control system required for safety. Discussion of the interlocks is found in the PSAR in sections relative to the fuel-handling and storage system.

7.6.1.6 Radiation Monitoring System

There are several CRBR plant systems that will make use of radiation monitors. These are described in Section 11.4 of the PSAR. These include the following

(1) postaccident containment atmosphere monitors
(2) reactor containment isolation monitors
(3) building ventilation exhaust monitors
(4) control room inlet air monitors
(5) inerted cell atmosphere monitors
(6) RAPS and CAPS monitors
(7) radiation disposal system liquid effluent monitors

Some of the monitors listed above are used to initiate protective action. These include those used in the containment isolation system and the control room HVAC system, which were discussed in Section 7.3 of this SER.

The PSAR states that the monitors will be provided in accordance with PDC 56 (GDC 64). Process systems will be monitored to give early warning of equipment malfunctions, indicate potential radiological hazards, and prevent release of radioactivity to the environment in excess of 10 CFR 20 limits.

The features of the monitors will include loss-of-signal instrument failure alarms and redundant readout in the control room and the health physics area of the plant service building.

7.6.1.7 Recirculation Gas Cooling System

The recirculation gas cooling system will be a safety-grade system providing heat removal and temperature control in the inerted cells. The sodium makeup pumps, vessel, pipeways, and ex-vessel storage loops will be located in these cells.

7.6.1.8 Steam Generator Building Flooding Protection System

The steam generator building (SGB) flooding protection system will be provided to detect flooding of SGAHRS equipment resulting from postulated SGS water-steamline ruptures, thereby ensuring the availability of SGAHRS for reactor decay heat removal following water-steamline rupture events.

7.6.2 Specific Findings

The PSAR states that the safety-related instrumentation and control systems will be designed to meet the applicable requirements of the regulatory guides, the IEEE standards, and the CRBRP Principal Design Criteria. All of the systems discussed in this section have been identified as being safety related and are, therefore, subject to the above criteria.

7.6.2.1 Direct Heat Removal Service Instrumentation and Control System

The direct heat removal service (DHRS) is intended to provide a supplementary means of removing long-term decay heat in the unlikely case that none of the steam generator decay heat removal paths are available. The staff questioned the applicants about the degree of independence provided between the DHRS instrumentation and controls and the SGAHRS instrumentation and controls and the design criteria used for DHRS. The applicants stated that the DHRS will be separate in function and in equipment location from the SGAHRS, and that there will be no sharing of instrumentation and controls between them. However, the DHRS will be powered from Class 1E, divisions 1 and 2, which will also provide power to SGAHRS.

The applicants stated that, in general, the cables, raceways, and other equipment of the SGAHRS and DHRS will be located in different places in the plant, except in the control building and the steam generator building where the cables of the same safety division will be located in the same area but will not prevent the other divisions from performing their safety functions. The staff questioned the rationale for not providing separation between the DHRS and SGAHRS according to the guidelines of RG 1.75. The applicants have committed to expand the discussion in the PSAR by identifying the particular areas where DHRS and SGAHRS of the same division will be routed together and to provide justification(s) for this routing.

The instrumentation required by the DHRS to monitor the functional performance of the decay heat removal process loops will be redundant from the sensor out to and including the readout panel, so that a single failure of an instrument, interconnecting cable, or panel will not prevent the process loop from being monitored. In some cases a redundant sensor will not be provided, but separate indicators on separate panels will be provided. Where redundant sensors will not be provided, loss of the sensor will not prevent the acquisition of the equivalent variable from other sensors on the process loop. A list of DHRS variables monitored and annunciated is given in PSAR Sections 7.6.3.1.2 and 7.6.3.1.3 and in a letter dated November 3, 1982.

In addition, the applicants stated that remotely operated valves for the DHRS will incorporate either "fail safe" or "fail-in-place" features and will be provided with direct manual (reach rods on sodium valves) override capability in event of instrumentation and control failures. Equipment required to provide power to the DHRS pumps, air blast heat exchangers, and the monitoring instrumentation will be Class 1E power designed and tested to seismic Category I requirements.

The staff questioned the applicants concerning the existence of interlocks for the DHRS. The applicants stated that the only interlocks that will be present

when the DHRS is activated are those that are associated with protecting the NaK and sodium pumps against high pump stator temperatures.

A manual override of these interlocks can be performed with the knowledge that pump damage and early failure could result. In addition, the applicants provided assurance that there will be no process measurement dependent interlocks that will limit operation of the DHRS.

The PSAR states that the DHRS must be "adjusted manually." The staff asked for clarification of this statement. The applicants stated that the manual adjustment referred to will consist of turning six switches in the control room on the DHRS panel from the normal to the DHRS position to manually initiate the system. This initiation is on-off control rather than an "adjustment." These control switches will be provided to position DHRS valves and control the pumps and air blast heat exchangers.

The staff questioned the total time involved for this procedure and the minimum amount of time available to initiate DHRS. The applicants stated that a conservative estimate of the time required for operation of these switches and the automatic sequencing to operate is 12 min. In the worst-case transient analysis performed for the DHRS (extremely unlikely event of DHRS initiation following reactor shutdown from 100% power with loss of all heat transfers through the primary heat exchangers at the time of the reactor trip), it is assumed that no heat will be transferred to the DHRS for ½ hour after shutdown. The heat capacity of the primary system will be used as the principal heat sink during this period with no operator action required to ensure the heat sink. The NRC CRBR Program Office is reviewing the acceptability of the above analysis.

The applicants have provided additional design information for the DHRS and, in addition, have committed to design the DHRS to be single-failure tolerant. The staff has reviewed the question of the sharing of power supplies (divisions 1 and 2) between the DHRS and the SGAHRS. This review is presented in Section 8 of this SER. The question of system diversity between DHRS and SGAHRS and the final acceptability of manual initiation of DHRS was reviewed by the CRBRP Program Office.

The staff has concluded that the instrumentation and controls design criteria for the DHRS meet the regulatory requirements for a system that is important to safety. However, the applicants have not provided total physical separation between DHRS and SGAHRS. The staff's position is that the separation between DHRS and SGAHRS instrumentation and controls should meet the provisions of RG 1.75 unless the applicants justify the design criteria used in those areas where the separation does not meet RG 1.75. This issue is considered to be part of the open item discussed in Section 7.2.2.6, "Regulatory Guide 1.75," of this SER.

7.6.2.2 Fuel-Handling and Storage Safety Interlock System

Section 7.6 of the PSAR lists the fuel-handling and storage safety interlock system as a safety-related system. The PSAR, however, does not include sufficient information on the system to evaluate compliance with the applicable criteria for systems important to safety. The staff asked the applicants to clarify the design criteria for this system. The applicants stated that the

fuel-handling and storage safety interlock system was not a safety-related system and that the PSAR was in error.

Postulated reactor refueling system (RRS) accidents with potentially severe consequences were analyzed in detail to determine requirements for the refueling interlocks. The techniques employed included safety assurance diagrams, fault trees, mechanical and thermal analyses, and radiological release calculations. The applicants stated that none of the analysis results showed offsite doses exceeding those presented in Sections 15.5 or 15.7 of the PSAR. The offsite doses in Sections 15.5 and 15.7 of the PSAR resulting from postulated RRS accidents are all well below the 10 CFR 100 guideline exposures without taking credit for interlocks. The staff believes that the fuel-handling and storage safety interlock system is correctly identified as a system not required for safety and concludes that its design criteria are acceptable. To ensure that an anticipated event cannot lead to fuel damage and subsequent radiation release, the staff will require that a single failure not be capable of removing the function performed by an interlock or interlocks.

The applicants are required to correct the PSAR to indicate that the fuel-handling and storage safety interlock system is not a safety-related system and to provide within the PSAR the justification for this interlock system not being safety related. This justification should indicate that a single failure will not prevent an interlock from performing its function as discussed above.

7.6.2.3 Radiation Monitoring System

The applicants were asked to identify where microprocessors, multiplexers, or computer systems will be used in or will share a common boundary with safety-related systems. The applicants stated that many microprocessors, multiplexers, and computers will be used in CRBR systems; however, all but two will be used in non-Class 1E applications. The two Class 1E system applications are the solid-state programmable logic system (SSPLS), which is described in Section 7.3.2.4 of this SER, and the radiation monitoring system remote processor stations.

The applicants have provided the following information describing the radiation monitoring system remote processor stations and how they function. The radiation monitoring system remote processor stations are microprocessor based, radiation monitoring electronic and communication assemblies. The microprocessor receives raw count rate and process system data and manipulates the data into the desired form. Data exchange and monitor control is by channel dedicated multiplexed signal paths. The overall design concept and use of the microprocessor in the radiation monitoring system for data management is acceptable to the staff. However, the applicants are required to have a software verification and validation program for this system. The staff will review the software verification and validation program at the operating license stage.

7.6.2.4 Recirculating Gas Cooling System

The recirculating gas cooling system (RGCS) is described in PSAR Section 7.6.6 as a system required for safety. However, Section 7.7.1.10 of the PSAR lists the RGCS as a nonsafety-related system. The applicants were asked to provide a summary of the instrumentation and control (I&C) system functional design, re-

dundancy, and a rationale for the classification (safety related or nonsafety related) of the I&C for the RGCS.

The applicants responded to this request by providing revised PSAR Sections 7.6.6 and 7.7.1.10. Revised Section 7.6.6 discusses the safety-related subsystems of the RGCS and states that the RGCS will maintain the required cell temperature in the inerted cells where the following components will be located: sodium makeup pump and vessel (subsystem MA), sodium makeup pump and pipeways (subsystem MB), ex-vessel storage loop 1 (subsystem EA), and ex-vessel storage loop 2 (subsystem EB).

Revised PSAR Section 7.7.1.10 references PSAR Section 9.16 for the nonsafety-related portions of the RGCS. PSAR Section 9.16 describes the RGCS and its 13 subsystems and provides their safety classification (four will be safety related (listed above) and 9 will not be safety related). Furthermore, in a letter dated February 15, 1983, from J. R. Longernecker to J. N. Grace, the applicants stated that the 9 nonsafety-related subsystems will not be required for safe shutdown of the plant and that the loss of cooling to the areas serviced by these subsystems will not result in the loss of capability to bring the plant to a safe shutdown condition.

The staff has reviewed the safety classification for the 13 RGCS subsystems and has concluded (Section 9.16 of this SER) that the systems are correctly classified.

7.6.2.5 Steam Generator Building Flooding Protection System

It was not clear from the description in the PSAR if the steam generator building flooding protection (SGBFP) system will meet the criteria applicable to equipment performing a safety function. The staff questioned the applicants with regard to this system and the applicants provided a more detailed description. The applicants stated that the SGBFP system will have both safety and nonsafety functions. The safety function will be the detection of a major pipe rupture and the isolation of the feedwater supply system and the affected loop.

Upon detection of a major pipe rupture, the startup and main feedwater control valves and the feedwater isolation valves will be closed by two independent and separate Class 1E logic trains. One logic train will close the startup and main feedwater control valves, the other the feedwater isolation valve. Actuation of each logic train will require concurrent 2-out-of-3 signals from both temperature and moisture from the same cell of any one of the four cells in each heat transport loop. The temperature and moisture instrument channels are Class 1E.

The staff has reviewed the design criteria for this system and concludes that they are acceptable.

7.6.2.6 Auxiliary Liquid Metal System

The staff asked the applicants to provide information describing the design criteria used for the auxiliary liquid metal system instrumentation and controls. The applicants stated that PSAR Section 7.7.1.10 identifies those instrumentation and control systems that will not perform safety-related functions. Included in this PSAR section are portions of the auxiliary liquid metal system.

However, other portions of this system will be used by the DHRS and the spent fuel storage system which will perform safety-related functions. Therefore, the instrumentation and controls for these portions are classified as safety-related and will be accordingly designed to the requirements for safety-related systems. The applicants provided an amended PSAR Section 7.6.3 that described the safety-related portions for the auxiliary liquid metal system. The staff has reviewed the design criteria for this system and concludes that they are acceptable.

7.6.3 Findings and Conclusions

From its audit review, using the acceptance criteria and procedures outlined in SRP Section 7.6 (Parts II and III) (NUREG-0800), the staff concludes that the design of the systems discussed in Section 7.6 of the PSAR is acceptable and meets the relevant requirements of PDC 2 and 5 (GDC 2 and 4). This conclusion is based on the following.

On the basis of the staff review of the systems in Section 7.6 of the PSAR, it is concluded that their design bases are consistent with the plant safety analyses and the systems importance to safety. The staff further concludes that the design of these systems, with respect to single failures, redundancy, independence, qualification, and testability, is adequate to ensure that the functional performance requirements will be met.

The staff's review included the identification of those systems and components that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of this review, the staff concludes that the applicants have identified the systems and components consistent with their design bases. Section 3.10 of this SER addresses the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of the systems and components satisfies this aspect of PDC 2 and 5.

On the basis of its review of Section 7.6 of the PSAR, the staff concludes that the applicants' design criteria as described in this section are sufficient to provide reasonable assurance that the final design will conform to the design bases and applicable criteria with an adequate margin for safety.

7.7 Control Systems

Section 7.7 of the PSAR is entitled "Instrumentation and Control System Not Required for Safety." The systems described in this section fit the descriptions and areas of review found in SRP Section 7.7. The SRP defines these systems as "control systems used for normal operation that are not relied upon to perform safety functions following operational occurrences or accidents but which control plant processes having significant impact on safety." The systems fitting this description, identified in Section 7.7 of the PSAR, are

(1) supervisory control system
(2) reactor control system
(3) primary and secondary control rod drive mechanism controller systems
(4) rod position indication system

(5) sodium flow control system
(6) rod misalignment rod block system
(7) steam generator feedwater flow control system
(8) feedwater pump speed control system
(9) recirculation flow control system
(10) sodium dump tank pressure control system
(11) steam dump and bypass control system
(12) fuel-handling and storage control system
(13) nuclear island auxiliary instrumentation and control systems
(14) balance-of-plant instrumentation and control systems

7.7.1 Description of Systems

7.7.1.1 Supervisory Control System

The supervisory control system will operate on steam temperature, steam pressure, steam flow, and load demand signals and will control the primary flow, intermediate flow, steam temperature, and pressure. The supervisory control system will also produce a feed forward reactor temperature demand signal and an intermediate flow demand signal, which will be used by the primary flow control system and the reactor control system.

7.7.1.2 Reactor Control System

The reactor control system will use the weighted core outlet temperature and neutron flux to control the reactor outlet temperature by maneuvering the control rods. Multiple sensor inputs will be used to ensure control reliability. Rod withdrawal blocks based on high flux to flow ratio and high flux will be provided. The temperature setpoint will be generated either manually or by the supervisory controller.

The reactor control system will position the control rods to the desired reactor thermal power and outlet temperature. The inner loop flux control, as part of the reactor control system, will use inputs from three filtered and buffered power range flux signals. The feedback signal will then be sent to a difference amplifier that, along with the power demand, will generate the error signal for the flux control compensation network.

The outer loop temperature control will use the signals from 30 core exit thermocouples to generate the feedback signal. The thermocouple outputs will first be individually filtered and then processed through an averaging and reject circuit. The feedback signal will be compared with the temperature demand by means of a difference amplifier that will generate the error signal for the temperature dead zone circuit.

7.7.1.3 Primary and Secondary Control Rod Drive Mechanism Controller Systems

The primary control rod drive mechanism (CRDM) control system will transform three-phase power to pulsed dc necessary to operate the control rod drive mechanism which will respond to input commands from the reactor control system. The drive mechanism will be actuated by a four-pole, six-winding reluctance stepping motor. Driving power will be supplied from the site power systems through redundant motor-generator sets, through the reactor shutdown system

scram breakers, then to a three-phase-to-six-phase transformer that will provide power to silicon controlled rectifiers in the individual controllers supplying power to the stator windings of the CRDM.

There will be two groups of primary control rods. One group of three rods will be manually controlled and will normally be fully withdrawn during reactor operation. The second set of six rods will be responsive to either manual control or to an analog signal from the reactor control system. Rod speed limits will be included in the reactor controller as well as in the individual CRDM controller.

Reactor scram will be initiated by interrupting the power to the primary CRDM stator coils. With the collapse of the rotor field, the rollers will disengage from the lead screw and the primary rod will fall under force of gravity and spring assist.

7.7.1.4 Rod Position Indication System

There will be two independent rod position indication systems provided for each control rod. One will be the absolute position indication system that will not require rezeroing after a scram or loss of power. For the primary control rod system, it will be a solid-state system using ultrasonics and magnetics to provide a dc signal indication of rod position.

The second system for the primary control rod system will be the relative rod position indication (RRPI) system. It will use pulse counters triggered by pickup coils located near magnetic poles mounted on the rotor mechanisms. This system must be reset following a scram or power failure.

7.7.1.5 Sodium Flow Control System

The sodium flow control system will consist of six controllers used to drive the three primary and three intermediate sodium pumps. Each controller will consist of a cascade system with an inner loop using speed as the feedback signal and an outer loop using a flow feedback signal. The flow control range will be 30 to 100% of rated flow. The flow setpoints will be generated either manually or by the supervisory control system.

7.7.1.6 Rod Misalignment Rod Block System

The rod misalignment rod block system will be an interlock system designed to prevent a row 7 control rod from being withdrawn more than a set distance above the average position of the six control rods in row 7.

7.7.1.7 Steam Generator Feedwater Flow Control System

The feedwater control system will automatically control the flow of water into the steam drums to maintain the water level in the drum within specified limits. Primary control will be effected by modulating a feedwater flow control valve at the inlet to the steam drum. In addition, feedwater pump speed will be varied to maintain the pressure drop across the feedwater flow control valves at a value that will not change excessively with load.

The feedwater flow control valve will be positioned by a conventional three-element cascade control system, which will consist of drum level and feedwater flow controllers.

The drum level controller will compare measured drum level with a predetermined setpoint value. The resulting error signal will be summed with a signal representing drum steam flow and will constitute the input (i.e., flow setpoint) to the feedwater flow controller. The output of the feedwater flow controller will be the position demand signal to the feedwater flow control valve. The net effect is that feedwater will be controlled by both the feedwater flow and steam flow signals through the two cascaded controllers.

Instrumentation required by this control system will include (1) steam drum level measured by a differential pressure transmitter, (2) steam flow measured in the outlet line of the superheater by a differential pressure transmitter, and (3) feedwater flow measured in the inlet line of the steam drum by a differential pressure transmitter.

7.7.1.8 Feedwater Pump Speed Control System

The feedwater pump speed control system will control the feedwater pump speed by means of a fluid drive coupling located between the pumps and their drive motors. Control of the fluid coupling will be by an electric control that will take its signal from a differential pressure controller, which will compare the difference between steam drum and feedwater header pressure signals and a setpoint value. Instrumentation required for this system will include (1) feedwater pressure measured by a bourdon tube pressure transmitter with sensor located upstream of the feedwater control valve and (2) steam drum pressure measured by a bourdon tube pressure transmitter.

7.7.1.9 Recirculation Flow Control System

The recirculation pump will be constant speed and no flow control instrumentation will be used for this system.

7.7.1.10 Sodium Dump Tank Pressure Control System

The sodium dump tank pressure control system will function to maintain the argon cover gas pressure over the sodium dump tank within prescribed limits. Pressure control of the sodium dump tank will be maintained by the supplying or venting of argon gas to or from the cover gas region through a 2-in. line. Two pressure transmitters will be located on this line, one for indication and off-normal alarm, the other for control of the argon supply and vent valves.

The argon supply and vent valves will be controlled by an "on-off" type pressure controller, which will cycle the supply and vent valves to maintain the cover gas pressure between the lower and upper limits. Manual overrides for the supply and vent valves will be provided in the main control room and may be used at the plant operator's discretion. High- and low-pressure alarms will alert the operator to off-normal conditions that may result from a malfunction of the pressure control system.

7.7.1.11 Steam Dump and Bypass Control System

The steam dump and bypass control system will provide the control and instrumentation necessary to operate the turbine bypass system.

Redundant interlocks will be provided to prevent bypass operation if the condenser will be unable to accept steam flow (e.g., high condenser back pressure or loss of circulating water flow).

Independent instrumentation and control channels will be provided for the detection of load rejection (e.g., reactor power exceeding net generation by 20%). These two channels will be arranged for a 2-out-of-2 coincidence and failure of one channel in any mode will not cause the operation of the steam dump and bypass control system. Time response will be such that the proper bypass valve(s) lift will be obtained within 3 sec of system actuation to maintain bypass steam flow approximately proportional to reactor power.

A pressure control channel will be provided for the regulation of main steam pressure following reactor trip, during decay heat removal operation, and during turbine standby, loading, and unloading operations. The pressure control mode will be automatically selected for reactor power levels below 40% and will operate two of the four bypass valves.

At reactor power levels above 40%, the steam dump and bypass control system will automatically position bypass valves to regulate total steam flow approximately proportional to reactor power; however, the pressure control mode may be manually selected by operating personnel at any power level.

7.7.1.12 Fuel-Handling and Storage Control System

The fuel-handling and storage control system will include several subsystems that will be used in loading and unloading core components. The system will make use of a computer for controlling linear and rotary motions of the refueling equipment. It will also provide (1) data acquisition information, (2) inventory control, (3) system status information, and (4) monitoring for out-of-limit conditions of refueling equipment. The system will include interlocks to prevent damage to fuel or system components. Equipment controlled by the system will include (1) the in-vessel transfer machine (IVTM), (2) the ex-vessel transfer machine (EVTM), (3) the reactor rotating plugs (RRPs), (4) the ex-vessel storage tank (EVST), and (5) the auxiliary handling machine (AHM).

Two events are identified that are to be prevented by the interlocks. These are (1) rotation of one or more rotating plugs with the assembly partially inserted into core or storage location and (2) rotation of one or more rotating plugs while the EVTM is located over the reactor.

7.7.1.13 Nuclear Island Auxiliary Instrumentation and Control Systems

The PSAR identifies several auxiliary supporting instrumentation and control systems that will not perform safety-related functions and the failure of which will not prevent the functioning of safety-related systems. These instrumentation systems are

(1) recirculation gas systems
(2) auxiliary cooling fluid systems
(3) heating and ventilating systems
(4) sodium-fire protection system
(5) inert gas receiving and processing systems
(6) impurity monitoring and analysis system
(7) auxiliary liquid metal systems

These systems are further discussed in Chapter 9 of the PSAR.

7.7.1.14 Balance-of-Plant Instrumentation and Control Systems

The PSAR does not discuss the nonsafety-related instrumentation and control systems that will support balance-of-plant systems. It does, however, state that these systems will be designed so that their failure will not prevent the function of safety-related systems.

7.7.2 Specific Findings

Section 7.7 of the PSAR evaluates the consequence of failure of the key systems considered in this section. Those evaluated are (1) the supervisory control system, (2) the reactor control system, (3) the sodium flow control system, (4) the steam generator feedwater flow control system, and (5) portions of the balance-of-plant instrumentation and control system.

There are numerous identifiable single element malfunctions that could direct the plant away from desirable operating limits. The supervisory control system, the steam generator feedwater control system, and the sodium flow control system could have malfunctions that could result in mismatch of plant variables. The PSAR states that the supervisory control system will use multiple sensors and average/reject, auctioneer, or median select circuitry to minimize the possibility that single sensor failure will result in inappropriate control system action. There is also an inherent limit to the speed at which an inappropriate control action can be taken.

The ultimate protection for failure of these control systems will be the plant protection system (PPS). It is stated in the PSAR that the supervisory control system will be inherently incapable of initiating a transient that is more severe than the PPS design basis.

The reactor control system will depend on flux and core exit temperature to effect control rod movement. Failure in this system could result in erroneous movement of the control rods. This system will include rod withdrawal block circuitry, which will be independent and designed to prevent control electronic failures from causing reactor trip. Withdrawal blocks will be initiated for both high power and power-to-flow ratio. The control rod drive controller and rod sequencer will include overspeed detector and block circuitry to limit rod withdrawal speed if failure occurs in other areas of reactor control. The system will include alarms to indicate malfunction and manual override. The final protection for reactor control malfunction will be the PPS, which is to be independent from the control system.

7.7.2.1 Effects of Control System Failures

A number of concerns have been expressed about the adequacy of safety systems in mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in PSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions insofar as failures of single control systems are concerned, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are, in the staff's judgment, those resulting from failure or malfunction of power supplies or sensors common to two or more control systems. To provide assurance that the design-basis-event analyses adequately bound multiple control system failures, the staff asked the applicants to provide the following information.

(1) Identification of those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indication of which, if any, of the control systems identified will receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses resulting from the failure of higher level distribution panels and load centers.

(3) Indication of which, if any, of the control systems identified will receive input signals from common sensors, common hydraulic headers, or common impulse lines.

(4) Verification that failure of a power supply, sensor, hydraulic header, or impulse line supplying two or more control systems will not result in consequences more severe than those acceptable for an anticipated operational occurrence.

A response to this concern was received (PSAR Amendment 69 dated July 1982) from the applicants but was determined to be incomplete. The applicants agreed to amend this response and to use the format submitted in the response to this issue on the Callaway Unit 1 docket (Docket STN 05-0483). In addition, the applicants agreed to document that when a protection channel is in test, and a common part of the protection system fails, multiple control system failures will not cause failure consequences beyond the Chapter 15 analyses.

The applicants transmitted the amended response in a letter dated December 14, 1982. This amended response listed seven CRBR control systems which may affect functions important to safety and provided an analysis for each event (loss of a single instrument, break of a single instrument line, and loss of a common power source) for these control systems.

The amended response documented that the break of an instrument line common to more than one control system is not applicable to CRBR and also documented that when a protection channel is in test, and a loss of any single control system sensor or a loss of power to a protection system group occurs, the resulting

multiple control system failures will not cause failure consequences beyond the PSAR Chapter 15 analyses.

The applicants' amended response with regard to multiple control system failures caused by a failure or malfunction of a common sensor line was acceptable and the staff concluded that this portion of the concern was resolved.

However, the staff's review of the amended response for the remaining parts of the multiple control system failures concern led to the conclusion that additional information was required from the applicants to clarify the criteria used in the applicants' evaluation of this issue.

The staff required the applicants to provide a definitive statement that the single event that causes multiple control system failures (failure or malfunction of shared power sources, or common sensors), concurrent with a protection channel in test, and any additional single random failure within the protection system will not result in consequences more severe than those acceptable for CRBRP for anticipated operational occurrences.

In a letter dated February 15, 1983, from John R. Longenecker to J. Nelson Grace, the applicants revised their previous response by stating that control system failures (including failures caused by malfunctions of shared power sources or common sensors), which will cause plant transients requiring reactor shutdown system action, will be terminated by the shutdown system action, will be terminated by the shutdown system within the CRBRP limits for anticipated operational occurrences. This includes the condition of a protection channel in test and any additional single random failure within the reactor shutdown system.

The staff has reviewed the information provided by the applicants and concludes that it is acceptable.

7.7.2.2 Qualification of Control Systems

IE Information Notice 79-22, "Qualification of Control Systems," addressed consequential control systems failure following a high-energy-line break. The staff was concerned that these consequential failures may result in event sequences more severe than those events analyzed in Chapter 15 of the PSAR.

The applicants have committed to perform a review before submission of the FSAR to ensure that consequential control failures following a high-energy-line break do not lead to events more severe than those analyzed in Chapter 15. This response is acceptable to the staff for a CP review. In their review, the applicants should assume the high-energy-line break, all consequences (including consequential control failures) of the high-energy-line break, and a concurrent single random failure within the safety systems. The review must also address the effects of the high-energy-line break occurring at a time when a protection system channel is in test. The staff will verify the resolution of this concern during the OL review.

7.7.2.3 Steam Generator/Steam Drum Level Control

The staff asked the applicants to discuss the consequences of a steam drum water level control channel failure. The applicants submitted a response to

this item and also provided an updated PSAR Section 7.7.1.5. Failure of the level control circuit (including median select circuit) which could result in flooding of the steam drum will be mitigated by two independent Class 1E high steam drum water level trips which will be set at 8 in. and 12 in. above normal water level. The 8-in. logic train will close the steam drum isolation valve and the main and startup bypass feedwater control valves. The 12-in. logic train will close the feedwater isolation valve.

Failure of the control valves which results in an increased steam drum water level will result in the same trips as discussed above for a failure in the level control circuit. The Class 1E trip circuits also will isolate the steam generator auxiliary heat removal system, auxiliary feedwater (AFW). Valves in the motor-driven AFW pump loops will close at the 8-in. trip; the valves in the turbine-driven AFW pump loop will close at the 12-in. trip. In addition, the steam drum outlet nozzle which will provide steam to the superheater will be located 35-in. above normal water level and the steam dryers will also be located well above the 12-in. trip setting.

The staff finds the design criteria for the steam drum level control system satisfactory.

7.7.2.4 Rod Position Indication System

The applicants were asked to discuss the design criteria for the primary rod position indication system (RPIS). In the response to this question, the applicants stated that the basic criteria for the RPIS are to provide redundant indication of primary control rod position over the full range of possible rod movement and to provide position information necessary to ensure that maximum control rod misalignments are limited to a value less than ±1.5 in.

To meet the first criterion, the applicants have provided two diverse and independent measuring systems. Each system will be capable of measuring the positions of the primary rods throughout their range of motion. The absolute rod position indication (ARPI) system will determine the position of the control rod absorber through the position of the mechanism lead screw. The ARPI will provide a direct measurement of rod position and will not lose its reference after a scram or a temporary loss of power.

The second system, the relative rod position indication (RRPI) system, will determine the position of the control rod absorber by monitoring the rotation of the roller nut which operates the lead screw. This system can lose its reference and must be recalibrated to zero after a scram.

To ensure that the second criterion is met, the applicants have committed to meeting a system (both RRPI and ARPI) design accuracy of better than ±0.3 in. The staff finds the applicants' design acceptable.

7.7.3 Findings and Conclusions

An audit review of the control systems used for normal operation that are not relied on to perform safety functions but that control processes having a significant impact on plant safety has been performed following the acceptance criteria and procedures outlined in SRP Section 7.7 (Parts II and III) (NUREG-0800).

The staff concludes that the proposed control systems are acceptable and meet the relevant requirements of PDC 11 and 17 (GDC 13 and 19).

BTP ICSB-14, "Spurious Withdrawal of Single Control Rods in PWRs," contained in NUREG-0800 is directed toward the interpretation of PDC 18 (GDC 20), "Protection System Functions," and PDC 23 (GDC 25), "Protection System Requirements for Reactivity Control Malfunctions." Although not identified in Chapter 7 of the PSAR, these criteria are addressed in the PSAR Section 3.1.3.1. The staff concludes from the statements made in Section 3.1.3.1 of the PSAR that BTP ICSB-14 will be met as will PDC 18 and 23.

The conclusions of the analyses of anticipated operational occurrences and accidents as presented in Chapter 15 of the PSAR have been used to confirm that plant safety is not dependent upon the response of the control systems. The staff concludes that failure of the systems themselves or as a consequence of supporting systems failures (such as power sources) will not result in plant conditions more severe than those bounded by the analyses of anticipated operational occurrences.

Finally, the staff concludes that with consequential effects of anticipated operational occurrences and accidents will not result in control system failures that would cause plant conditions more severe than those bounded by the analyses of these events.

Table 7.1 Principal diversities in design features of the primary and secondary reactor shutdown systems (RSSs)

Feature	Primary RSS	Secondary RSS
Logic	Local coincidence	General coincidence
Sensors (parameter)	Inlet plenum pressure Primary pump speed Intermediate pump speed Heat transport system bus frequency Steam flow Feedwater flow Intermediate heat exchanger primary outlet temperature	Primary loop flow Intermediate loop flow HTS bus voltage Steam drum level Reaction products flow Evaporator outlet sodium temperature
Logic isolation	Light-emitting diodes	Direct dc coupled--no isolation required
Equipment		
· Circuitry	Integrated circuits	Discrete components
· Power supplies	Manufacturer A	Manufacturer B
· Potentiometers	Manufacturer C	Manufacturer D
· Buffers	Light coupling	Magnetic coupling
· Control rod release	Circuit breakers in 2/3 arrangement	2/3 solenoid-operated pneumatic valve
· Control rods	Gravity with spring assist Unlatching at roller nut Partially inserted	Gravity with hydraulic assist Unlatching at top of core Fully withdrawn
· Sensors (equipment)	Compensated ion chambers	Fission chambers

8 ELECTRIC POWER SYSTEMS

8.1 General

The bases for acceptance used by the staff in the evaluation of the applicants' designs, design criteria, and design bases for the CRBR electric power systems are set forth in SRP (NUREG-0800) Section 8.1, Table 8-1, "Acceptance Criteria and Guidelines for Electric Power Systems." These acceptance criteria and guidelines include the general design criteria of Appendix A to 10 CFR 50 and the guidelines of branch technical positions, regulatory guides, and NRC reports that are applicable to electric power systems used at LWR plants.

The objective of the following staff evaluation is to determine that there is reasonable assurance that the final design will satisfy the above-cited acceptance criteria and guidelines and will perform its design functions during normal operation, anticipated operational occurrences, and accident conditions. In addition, the staff will determine that the above-cited acceptance criteria and guidelines provide a sufficient basis for acceptance of the electric power systems used at a liquid metal fast breeder reactor plant.

8.2 Offsite Electric Power System

The safety function of the offsite power system (assuming the onsite power system is not functioning) is to provide sufficient capacity and capability to ensure that structures, systems, and components important to safety perform as intended. The objective of the staff review is to determine that there is reasonable assurance that the final design of the offsite power system will satisfy the requirements of CRBR Principal Design Criteria (PDC) 6, 15, and 16 (General Design Criteria (GDC) 5, 17, and 18 of 10 CFR 50, Appendix A) and that the system will perform its design function during all plant operating and accident conditions.

8.2.1 Compliance With PDC 6

The final plant design will satisfy the requirements of PDC 6, "Sharing of Structures, Systems, and Components," with respect to sharing of circuits of the preferred power system. PDC 6 is the same as GDC 5 of Appendix A to 10 CFR 50. The CRBRP will be a single-unit plant and thus will have no shared systems.

8.2.2 Compliance With PDC 15

The final plant design will satisfy the requirements of PDC 15, "Electric Power Systems," with respect to the offsite power system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety, (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies, (3) physical independence of circuits, and (4) availability of circuits.

PDC 15 with respect to offsite circuits is the same as GDC 17 of Appendix A to 10 CFR 50. The following items address the problem areas revealed during the staff review and their resolution.

8.2.2.1 Physical Independence of Offsite Circuits

The CRBR PSAR did not adequately describe or analyze physical independence of offsite circuits. The applicants by letter dated June 1, 1982 have stated that the two offsite circuits that will pass through the reserve switchyard will provide the two physically independent offsite power sources. The PSAR, in contradiction, implied that the offsite circuits that will pass through the generating switchyard will be immediate access circuits and, thus, the preferred offsite circuit as defined by IEEE Std. 308-1974. It is the staff's position, in accordance with the requirements of PDC 15, that electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

By letters dated November 2, 1982 and December 6, 1982, the applicants provided additional descriptive information and analysis regarding physical separation of the offsite circuits. On the basis of this information and analysis, the staff concludes that the final plant design will satisfy PDC 15 and is acceptable.

8.2.2.2 Availability of Offsite Power Circuits

PDC 15 requires, in part, that each of the offsite circuits be designed to be available in sufficient time following a loss of all onsite ac power supplies and other offsite electric power circuits to ensure that specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded. On the basis of information in the PSAR, the staff was unable to conclude that the period of time that the plant would remain in a safe condition assuming no immediate ac power would be available is greater than the time that would be required to reestablish ac power from the offsite grid to the onsite Class 1E distribution buses (see SRP Section 8.2, Part III, Item 1d).

In response to this item, the applicants, by letter dated November 2, 1982, indicated that offsite power to the onsite distribution system would be supplied through two physically and electrically independent switchyards. Given failure of one switchyard, offsite power would be automatically transferred to the other switchyard. The staff concludes that this design meets the requirements of PDC 15 and is acceptable.

8.2.2.3 Sequencing of Loads on the Offsite Power System

It is the staff's position that the offsite power system should have sufficient capacity to supply all required loads without sequencing of loads on the offsite power system. By letter dated November 2, 1982, the applicants indicated that each one of the offsite ac power supplies would have the capacity for feeding all the required plant auxiliary loads without a load sequencer. The staff concludes that this capacity meets the provisions of PDC 15 for minimizing the probability of losing electric power from any of the remaining supplies as

a result of, or coincident with, the loss of power from the onsite electrc power supplies and is, therefore, acceptable.

8.2.2.4 Loss of the Normal Offsite Power Source Causing Loss of the Reserve Offsite Power Source

The applicants have documented in Section 3.1 of the PSAR that automatic transfer from the normal power source to the reserve power source would be initiated by fault sensing relays in the normal power supply. Given loss or failure of the normal power supply, it was not clear how fault-sensing relays will ensure the availability of the reserve offsite power source.

It is the staff's position, in accordance with PDC 15, that provision be included to minimize the probability of losing electric power from reserve offsite power source as a result of, or coincident with, loss of power from the normal offsite power source.

By Amendment 74 to the PSAR, the applicants provided additional descriptive information as to the design for transferring power between the normal and reserve offsite power sources. On the basis of this additional information, the staff concludes that the design meets PDC 15 and is acceptable.

8.2.2.5 Surveillance of Offsite Circuits

Section 5.2.3(5) of IEEE Std. 308-1974 requires that offsite power supplies be monitored to the extent that they are shown to be ready to perform their intended function. Description of the monitoring that is to be provided for CRBR offsite power circuits was not described in the PSAR. Subsequently, by Amendment 74 to the PSAR, the applicants provided the subject description. The staff concludes that the monitoring of offsite power supplies described meets the guidelines of IEEE Std. 308-1974 and is, therefore, acceptable.

8.2.3 Compliance With PDC 16

The final plant design will satisfy the requirements of PDC 16, "Inspection and Testing of Electric Power Systems," with respect to the capability to test systems and associated components during normal plant operation and the capability to test the transfer of power from the nuclear power unit, the offsite preferred power system, and the onsite power system.

PDC 16 is the same as GDC 18 of Appendix A to 10 CFR 50. The following item addresses the problem area revealed during the staff review and its resolution or status.

8.2.3.1 Capability To Test Transfer of Power Among the Various Power Supplies

The CRBR design will include provisions for transferring power between the plant power supply, the normal ac power supply through the generating switchyard, the reserve ac power supply from the reserve switchyard, and the onsite standby diesel generator power supply. By letter dated June 1, 1982, the applicants implied that the design capability to test the transfer among the above-listed listed power supplies during power operation was not provided in accordance with PDC 16.

By letter dated November 2, 1982, the applicants documented that the CRBR design includes the capability to test the transfer of power supplies among the plant power supply, the normal ac supply through the generating switchyard, the reserve ac supply through the reserve switchyard, and the onsite standby diesel generator power supplies. In addition, the applicants documented that the sensors that detect loss of power will have the capability to be tested during plant operation. The staff concludes that this design capability meets the requirements of PDC 16 and is acceptable.

8.2.4 Evaluation Findings

The review of the offsite power system for the CRBRP covered descriptive information, preliminary analyses, electrical single-line diagrams, and preliminary physical arrangement and layout drawings. The basis for acceptance of the offsite power system in the staff's review was conformance of the CRBR design criteria and bases to the Commission's regulations as set forth in the general design criteria of Appendix A to 10 CFR 50. The staff concludes (1) that the acceptance criteria and guidelines set forth in SRP Table 8-1 (NUREG-0800) provide a sufficient basis for acceptance of offsite electric power systems used at a liquid metal fast breeder reactor plant and (2) that there is reasonable assurance that the final plant design will meet the requirements of PDC 6, 15, and 16 (GDC 5, 17, and 18) and will conform to the applicable guidelines of regulatory guides and branch technical positions and is acceptable.

8.3 Onsite Power System

The safety function of the onsite power system (assuming the offsite power system is not functioning) is to provide sufficient capacity and capability to ensure that structures, systems, and components important to safety perform as intended. The objective of the review is to determine that there is reasonable assurance that the final design of the onsite power system will satisfy the requirements of PDC 2, 5, 6, 15, 16, and 41 (GDC 2, 4, 5, 17, 18, and 50 of 10 CFR 50, Appendix A) and will perform its intended function during all plant operating and accident conditions.

The onsite power system consists of an ac power system and a dc power system. Compliance with PDC 2, 5, 6, 16, and 41 as they relate to both ac and dc systems is evaluated in Section 8.3.3 of this report. Compliance with PDC 15, as it relates to ac systems, is evaluated in Section 8.3.1 and, as it relates to dc systems, is evaluated in Section 8.3.2.

8.3.1 Onsite AC Power System's Compliance With PDC 15

The final plant design will satisfy the requirements of PDC 15, "Electric Power Systems," with respect to the onsite ac system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety, (2) independence, redundancy, and testability to perform its safety function assuming a single failure, and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

PDC 15 is the same as GDC 17 of Appendix A to 10 CFR 50, except as noted in Section 3.1 of this report. The following items address the problem areas revealed during the staff review and their resolution.

8.3.1.1 Interconnection Between Redundant Divisions

In Section 8.3.1.2.1 of the PSAR, the applicants stated: "The standby onsite power supply network has provisions to manually cross-connect the 4.16 kV buses of the Division 1 and 2 power supplies in case of extreme emergency." By letter dated June 1, 1982, the applicants defined extreme emergency conditions to be loss of offsite power, loss of one diesel generator, and failure of a critical system load on the remaining operative diesel generator. It appeared that the failed load, since it has been characterized as critical, is needed to mitigate the consequences of some design-basis event. Thus, it appeared that the interconnection between divisions 1 and 2 would be used to meet the redundancy requirement of PDC 15 at the expense of the independence requirement of PDC 15.

Subsequently, by letter dated November 2, 1982, the applicants documented that the electrical distribution system will consist of three functionally redundant divisions so that any one of the three will have the capability to safely shut down the plant. The applicants further documented that the failed load is not needed to mitigate the consequences of any design-basis event and that the system design will meet all the redundancy requirements of PDC 15 without the subject cross-connection. On the basis of the above, the staff concludes that the design will meet PDC 15 and is acceptable.

8.3.1.2 Suitable Electric Interconnections

The terminology "suitable electric interconnections" contained in Criterion 26 of Section 3.1 of Amendment 32 to the PSAR is inconsistent with the terminology in PDC 15. PDC 15 requires that the onsite electric distribution system have sufficient independence and redundancy to perform its safety function assuming a single failure. The applicants' terminology "suitable electric interconnection" in Criterion 26 allows electric interconnection between redundant distribution systems in violation of the independence requirement of PDC 15.

The applicants, by letter dated November 2, 1982, changed the terminology "suitable electric interconnection" in Criterion 26 of the PSAR to "suitable interconnection." The staff, therefore, considers this item resolved.

8.3.1.3 Redundancy of the Three Independent Load Groups

Section 8.3.1.1 of the PSAR indicates that three independent load groups will be provided with load group 1 redundant to load group 2. No description as to the redundancy of load group 3 was provided in Chapter 8 of the PSAR. By letter dated June 1, 1982, the applicants indicated that each of the three load groups will have the capability to shut down the plant safely and that since not all the loads in load group 3 will be identical or similar to those in load groups 1 and 2, load group 3 was not identified as redundant to load groups 1 and 2 in the PSAR.

Even though the loads in load group 3 will not be identical or similar to those in load groups 1 and 2, it appears that the three load groups should be functionally redundant since it has been stated that each will have the capability to shut down the plant safely. Subsequently, the applicants, by letter dated November 12, 1982 and Amendment 74 to the PSAR, provided clarification by documenting that the three divisions will be functionally redundant. The staff, therefore, considers this item resolved.

In addition, Section 8.3.1.2.1 of the PSAR stated that the Class 1E safety-related loads will be separated into three load groups so that loss of any one load group will not prevent safe shutdown of the plant. This statement implied that as a minimum the remaining two load groups would be needed to shut down the plant. This statement contradicted other statements in the PSAR that imply that only one load group would be needed to shut down the plant. By Amendment 74 to the PSAR, the applicants corrected the PSAR to indicate that only one load group would be needed to shut down the plant. The staff, therefore, considers this item resolved.

8.3.1.4 Non-Class 1E Loads Powered From Class 1E System

Section 8.3.1.2.14 of the PSAR indicates that non-Class 1E loads will be connected to one division of the Class 1E system through an isolation device. The isolation system design is evaluated as follows:

(1) The proposed design for the isolation device primarily addressed protection of the Class 1E system from worst-case faults in the non-Class 1E system. By letter dated June 1, 1982, the applicants stated:

> Faults and failure modes other than the worst-case fault (three-phase) have also been addressed in the design of the isolation system. Protective devices have been provided in the design to clear any fault on the non-Class 1E system such as phase-to-ground, phase-to-phase, and three-phase faults, within a reasonable time such that there will be no degradation to the Class 1E system. A phase-to-ground fault (which is the most likely mode of failure) on a non-Class 1E circuit will have no effect on the Class 1E system since the isolation system includes a 4.16-kV/480-V delta-wye connected transformer with the high resistance grounded neutral. The neutral is grounded through a 55.4-ohm resistor which will limit the phase-to-ground fault current to approximately 5 amperes. The Class 1E 480-V and 4.16-kV circuit breakers will be tripped to clear a ground fault in the case that the affected non-Class 1E breaker fails to trip.
>
> Any phase-to-phase or three-phase fault on the non-Class 1E circuits will be isolated by instantaneous operation of the affected branch feeder circuit breaker. Backup protection is provided by fast operation of the 480-V supply circuit breaker (0.2 to 0.3 sec clearing time) or by the 4.16-kV unit substation transformer feeder circuit breaker (0.6-0.7 sec clearing time). In addition undervoltage sensors are provided at the input terminals of the 480-V supply circuit breaker. These undervoltage sensors will initiate tripping of the 480-V and 4.16-kV circuit breakers within

5 sec upon sensing the undervoltage caused by loss of power or failure of the circuit breakers to clear a fault.

The applicants did not address, in their June 1, 1982 letter, failure modes such as the failure of the Class 1E breakers to operate and the imposition of high voltage on the output terminals of the transformer or isolation system, nor did they address proposed testing. By letter dated November 2, 1982, the applicants addressed each of these items as follows:

(a) Failure of both Class 1E breakers, given a fault in the non-Class 1E system, may cause loss of Class 1E division 3. However, both Class 1E divisions 1 and 2 will be independent of division 3, and each will have the capability to shut down the plant safely.

(b) The imposition of high voltage on the output terminals of the transformer will be detected and cleared by the 4.16-kV Class 1E circuit breaker.

(c) The component parts of the isolation sytsem will be subject to manufacturer, preoperational, and periodic testing to ensure system performance. In addition, the protective relays and devices of the system will be designed so that they will be testable during plant power operation.

(2) The isolation device was to be designed, as indicated in the PSAR, so that voltage on the Class 1E system buses would not drop below 70 or 80% of nominal power given a worst-case fault in the non-Class 1E system. Since most Class 1E equipment will be designed to operate at not less than 90% of nominal power, the staff was concerned that Class 1E systems not operate at the lower voltage. By letter dated June 1, 1982 the applicants stated:

> The isolation system is designed so that impedance of the system is high enough that the worst possible fault (three-phase bolted fault) on the 480-V non-Class 1E bus will not degrade the voltage at 4.16-kV Class 1E bus below the following levels:
>
> (1) When the 4.16-kV Class 1E bus is being supplied from offsite power supply, the voltage at the bus will not drop below 80% of nominal.
>
> (2) When the 4.16-kV Class 1E bus is being supplied from onsite (standby) power supply, the voltage at the bus will not drop below 75%.

The minimum voltage levels of 80 and 75% of nominal are chosen to be the same as the allowable minimum voltage levels during the sequential loading of the 4.16-kV Class 1E bus or during starting of the largest motor after the bus has been fully loaded.

As discussed in Item (1) at the beginning of Section 8.3.1.4, any fault on 480-V non-Class 1E system will be cleared within 5 sec. After the fault has been cleared, the voltage at the 4.16-kV bus will be restored

to a minimum of 90% of nominal power within 2 sec, which will allow all connected loads to operate continuously.

The staff concludes that the proposed isolation system design described above will provide an acceptable level of isolation so that the Class 1E systems will be able to accomplish their safety functions regardless of any failure or combination of failures that may occur in non-Class 1E system components.

8.3.2 Onsite DC Power System's Compliance With PDC 15

The final plant design will satisfy the requirements of PDC 15, "Electric Power Systems," with respect to the onsite dc system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety, (2) independence, redundancy, and testability to perform its safety function assuming a single failure, and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

8.3.3 Common Electrical Features and Requirements

This section presents common electrical features and requirements of the onsite ac and dc power system that deal with distinct aspects of the onsite ac and dc power systems. The common electrical features and requirements addressed in this section are as follows.

8.3.3.1 Compliance With PDC 2 and 5

The final plant design will satisfy the requirements of PDC 2, "Design Basis for Protection Against Natural Phenomena," and PDC 5, "Environmental and Missile Design Bases," with respect to structures, systems, and components of the onsite ac and dc power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, missiles, and environmental conditions associated with normal operation and postulated accidents. The onsite power system and components (1) will be located in seismic Category I structures, which provide protection from the effects of tornadoes, tornado missiles, turbine missiles, and external floods, (2) will be given the quality assurance designation "Class 1E," (3) will be designated as being seismically and environmentally qualified, and (4) will be designed to accommodate or will be protected from the effects of missiles and environmental conditions associated with normal operation and postulated accidents.

The following items address the problem areas revealed during the staff review and their resolution.

8.3.3.1.1 Environmental Qualification of Cables and Terminations

Section 8.3.1.4 of the PSAR states that environmental tests will be performed on cables and terminations that are required to function in a hostile environment. This statement implies that cables or terminations that are not required to function in a hostile environment will not be environmentally qualified and may not be in compliance with IEEE Std. 323-1974. By letter dated June 1, 1982, the applicants provided the following corrected statement:

Environmental type test will be performed on all cables and terminations for their expected environment.

On the basis of this corrected statement, the staff concludes that there is reasonable assurance that cables and terminations will be designated as being environmentally qualified and are, therefore, acceptable.

8.3.3.1.2 Fire Protection

See Section 8.3.3.3.2 of this report.

8.3.3.2 Compliance With PDC 6

The final plant design will satisfy the requirements of PDC 6, "Sharing of Structures, Systems, and Components," with respect to structures, systems, and components of the ac and dc onsite power systems. PDC 6 is the same as GDC 5 of Appendix A to 10 CFR 50. The CRBRP will be a single-unit plant and thus will have no shared systems.

8.3.3.3 Independence (Compliance With PDC 15)

8.3.3.3.1 Physical Independence of Cables

Section 8.3.1.2.14 of the PSAR indicates that physical separation of circuits and equipment comprising or associated with the Class 1E power system, Class 1E protection systems, and Class 1E equipment will be in accordance with the criteria set forth in Paragraph 8.3.1.4 of the PSAR. Separation criteria stated in Sections 8.3.1.2.14 and 8.3.1.4 of the PSAR were not clear and did not meet the guidelines of IEEE Std. 384-1974 and RG 1.75. For example, the PSAR indicated that non-Class 1E cables in panels will be separated from Class 1E cables so that they will not provide a combustion path between different divisions. Section 5.6.5 of IEEE Std. 384-1974 states that non-Class 1E cables shall be separated by 6 in. or a barrier. In general, no criteria had been stated for separation of Class 1E and non-Class 1E cables. Other examples include:

(1) No criteria had been stated for separation between cable trays and conduits of another division.

(2) Criteria stated for the separation of the third division were confusing. (The design indicated there are three divisions but only two redundant divisions. Separation criteria refer to the two redundant divisions in many cases versus the three divisions.)

(3) The definition of associated cables was confusing.

(4) No criteria had been stated for the separation between associated cables and non-Class 1E cables

(5) No criteria had been stated for the separation before and after an isolation device.

By letter dated June 1, 1982, the applicants stated that the separation design criteria for CRBR are fully consistent with the guidelines in IEEE Std. 384-1974

and RG 1.75 (Rev. 2). In addition, the applicants provided a revised PSAR Section 8.3.1.4 (Item F) to clarify each of the above-described examples of noncompliance. On the basis of a review of the information contained in the revised PSAR, the following items remained as concerns: (1) adequacy of metal conduit or steel wire ducts as barriers inside panels and (2) the RG 1.75 requirement that non-Class 1E circuits meet all the requirements placed on Class 1E circuits.

In a letter dated November 2, 1982, the applicants addressed the concern regarding the adequacy of metal conduit or steel wire ducts as barriers inside panels by stating that metal conduit or steel wire ducts might be installed in addition to the 6-in. special separation specified in RG 1.75. Where used, the design will exceed the guidelines of RG 1.75 and is acceptable.

The remaining concern is addressed in Section 8.3.3.3.5 of this report.

8.3.3.3.2 Fire Protection

The applicants stated in Section 8.3.1.4 of the PSAR that only one safety division will be routed in a fire hazard zone and that this one division will be suitably protected so that a fire in the zone will not affect the safety functions of the other safety groups.

Protection of safety-related electrical equipment and circuits from the effects of a design-basis-event fire within the fire zone is addressed in Section 9.5.1 of this report.

8.3.3.3.3 Pipe Break Protection

Separation of Class 1E raceways from high-energy pipelines, as defined in the PSAR, will be more than 15 ft, or less than 15 ft if the pipe will be suitably restrained so as not to whip and strike the raceway. Current regulatory guidelines require that the Class 1E raceway be protected so that pipe whip missiles, jet impingement, or environmental effects of the pipe break will not cause failure of the Class 1E raceway. The staff believes that 15 ft of space is not adequate protection.

By letter dated June 1, 1982, the applicants stated that any damage to cable trays caused by pipe whip, missiles, jet impingement, or environmental effects will be limited to the same safety division to which the pipe belongs. The two other divisions, each capable of shutting down the plant, will remain unaffected by the pipe break. Thus, the staff concludes that there is reasonable assurance that the final design will meet the guidelines of Sections 4.1 and 4.2 of IEEE Std. 308-1974 and Sections 4.1 and 4.4 of IEEE Std. 384-1974 and the independence requirement of GDC 17, and is, therefore, acceptable with one exception. Section 8.3.1.4 of the PSAR stated in contradiction that "Under no circumstances do safety-related raceways run less than fifteen feet from high-energy pipelines of the opposite safety system." The staff's original concern remained; 15 ft of space is not adequate.

Subsequently, by letter dated November 2, 1981, the applicants revised Section 8.3.1.4 of the PSAR to state that additional protection will be provided to prevent cable tray damage caused by high-energy pipe lines in the vicinity

of Class 1E raceways. The design of restraints and/or barriers will be determined by analysis to meet BTP ASB 3-1. This revision resolves the remaining concern. The staff's evaluation of the acceptability of restraints and/or barriers is given in Section 3.6.1 of this report.

8.3.3.3.4 Separation Between Associated and Non-Class 1E Circuits

Section 8.3.1.2.14 of the PSAR indicates that analyses and testing of associated circuits will be performed in accordance with Sections 4.5(3), 4.6.2, and 5.1.1.2 of IEEE Std. 384-1974.

By letter dated June 1, 1982 the applicants indicated that all associated circuits will be treated as Class 1E circuits and routed with circuits of the same Class 1E division with which they are associated. Associated circuits will be separated from non-Class 1E circuits in the same way as Class 1E circuits. The staff, thus, concludes that there is reasonable assurance that the final design will meet the guidelines of IEEE Std. 384-1974 and the independence requirements of GDC 17 and is acceptable.

8.3.3.3.5 Associated Circuit Compliance With All Requirements Placed on Class 1E Circuits

Section 8.3.1.2.14 of the PSAR states that associated circuits will be installed in accordance with the requirements placed on Class 1E circuits, such as separation from non-Class 1E cables, cable derating, environmental qualification, flame retardance, splicing restrictions, and raceway fill limitations. On the basis of this PSAR statement, it appeared that associated circuits may meet only the requirements listed. Items not listed, such as protection from design-basis events, quality assurance, and seismic qualification of the associated circuit from and including its supply breaker and actuated equipment, are of concern. In accordance with Position C.4 of RG 1.75, it is the staff's position that associated circuits meet all requirements placed on Class 1E circuits or the adequacy of the proposed design be justified. By Amendment 47 to the PSAR, the applicants documented that (1) the electric distribution system is designed to keep the number of associated circuits to a bare minimum and (2) the associated circuits will comply with one of the following paragraphs of IEEE Std 384-1974; 4.5(1), 4.5(2), or 4.5(3), and Positions C.4 and C.6 of RG 1.75, Revision 2. The staff concludes that this design and commitment of compliance for associated circuits is acceptable.

8.3.3.3.6 Electrical Isolation Between Control Room and Remote Control Locations

Section 8.3.1.2.22 of the PSAR indicates that indicators and controls will be provided outside the control room in compliance with Section 4.4 of IEEE Std. 308-1974. In addition, it is a current staff position in regard to fire protection that these indicators and controls provided outside the control room be designed so that no single failure in the control room can cause failure at the remote location. By letter dated June 1, 1982, the applicants indicated that controls and indicators located in the control room will be electrically separated from controls and indicators at remote locations. Separation will be provided by independent overcurrent protection for each source so that overcurrent in the power source for control and indication at the control room will not affect operation of the power source for remote control and indication.

Both control circuits (control room and remote) will interface in the common control logic in the solid-state programmable logic system cabinet where they will be electrically isolated. The staff concludes that there is reasonable assurance that the final design will meet the position stated above and is, therefore, acceptable.

8.3.3.3.7 Physical Separation of Class 1E Equipment

It is the staff's position that Class 1E equipment that supplies power to different load groups should be located in separate rooms of a seismic Category I building. On the basis of information in the PSAR, the staff concludes that each diesel generator will be located in a separate room and that two of the three 4.16-kV Class 1E switchgear buses will be located in separate rooms. The separation of the remaining Class 1E equipment was not clear.

Subsequently, by Amendment 74 to the PSAR, the applicants documented that each division of Class 1E equipment of divisions 1, 2, and 3 will be located in separate rooms, which will be separated by a barrier with a fire rating of at least 3 hours. On the basis of this clarification, the staff considers this item resolved.

8.3.3.3.8 Conduits as Fire Barriers

Section 8.3.1.1.2.1 of the PSAR states: "Cables of the division 3 power supply are separated from cables of the remaining Class 1E plant ac power supplies by routing them in conduits or cable trays in separate fire hazard areas." On the basis of this statement, it appears that conduits may be used as a fire barrier for protecting division 3 cables.

By letter dated November 2, 1982, the applicants stated that conduits will not be used as a fire barrier for protecting division 3 cables. This statement resolves the staff concern regarding conduits; however, routing of division 3 cables in separate fire hazard areas remains a concern. Section 8.3.1.4 of the PSAR in contradiction indicates that division 3 cables will be separated from other division cables by a 1-ft-horizontal and 3-ft-vertical barrier in the cable spreading room. If the cable spreading room is defined as a fire hazard zone (which appears to be the case), redundant cables should be located in a separate fire zone. By letter dated January 26, 1983, the applicants documented that division 3 cables within the control room and upper cable spreading room areas are routed in raceways embedded in concrete floors and walls up to the point of entry to the division 3 panels. There are no division 3 cables or raceways in either the upper or lower cable spreading rooms. On the basis of this clarification, the staff considers this item resolved.

8.3.3.3.9 Protection of Class 1E Cables From Rotating or Dropped Equipment

Separation between redundant raceways as defined in the PSAR takes into consideration the presence of rotating equipment, monorails, and equipment removal paths and the possibility that heavy equipment could be lifted and dropped and possibly cause failure of two raceway channels. The PSAR states that the minimum separation between two raceway channels at CRBRP will be such as to preclude failure of both channels. Current regulatory guidelines, however, require protection of each raceway as well as separation so that the dropped

equipment will not cause failure of either raceway. An alternative to protection may be a design that provides an additional two independent systems, each capable of shutting down the reactor and separated so that neither will be affected by the dropped equipment or failure of rotating equipment.

By letter dated June 1, 1982, the applicants indicated that routing of safety-related raceways will be such that any dropped equipment will not result in a failure of any safety-related raceway. The staff concludes that safety-related raceways will be protected from dropped equipment, will meet review guidelines, and are acceptable.

The applicants also stated that the safety system design includes three divisions, each of which will be capable of shutting down the reactor and located so that failure of rotating equipment will not cause failure of more than one safety division. This statement was found unclear as to the rotating equipment that will have failed. Subsequently, the applicants, by letter dated November 12, 1982, indicated that failure of Class 1E rotating equipment will only affect its associated Class 1E division cables. In regard to non-Class 1E rotating equipment, the applicants indicated that the equipment will be designed to maintain structural integrity under a seismic event and will not become missiles. This design is acceptable for the seismic event. However, for other events, such as turbine overspeed, it was not clear that the non-Class 1E rotating equipment will not become missiles. By letter dated January 26, 1983, the applicants provided clarification by documenting that the likelihood of rotating equipment missile damage to Class 1E equipment is minimized or eliminated by one or a combination of the following:

(1) qualification of non-Class 1E and Class 1E rotating equipment to prevent missiles during the worst case seismic event (which envelopes normal operating conditions) for that rotating equipment

(2) segregate rotating equipment from Class 1E equipment

(3) provide missile protection by walls or barriers for the Class 1E equipment

(4) provide redundant equipment necessary to meet the single failure criterion

On the basis of this clarification the staff concludes that Class 1E circuits will be protected, meet review guidelines, and are acceptable. Missile protection is further addressed in Section 3.5 of this report.

8.3.3.4 Sufficient Redundancy (Compliance With PDC 35)

The acceptability of the onsite ac and dc power system, with regard to redundancy, is based on conformance of the system to the same degree of redundancy that is required by the criterion of the safety system it supplies (SRP Section 8.3.1, Part III, Item 1). CRBR PDC 35, "Reactor Residual Heat Extraction System," requires that the safety function of the residual heat extraction system be accomplished, assuming a single failure, with at least two flow paths remaining available for residual heat removal. To meet PDC 35, four flow paths or an apparent redundancy of four in the electric power system would be required. The electric power system does not meet this redundancy requirement. On the basis of discussions with the applicants and the review of electric power

supplies needed to maintain residual heat removal, the staff concludes that two flow paths will remain for the following event: loss of Class 1E division 1 power supply and associated distribution system with concurrent reactor trip, loss of offsite power, and failure of Class 1E division 2 power supply and associated distribution system as the single failure. The staff, therefore, concludes that the redundancy of three in the Class 1E power system is sufficient, meets the redundancy requirement of PDC 35, and is acceptable.

8.3.3.5 Compliance With the Guidelines of NUREG-0737

Two TMI items relating to GDC 17 are identified in NUREG-0737. These items are II.E.3.1, "Emergency Power Supply for Pressurizer Heaters," and II.G.1, "Emergency Power for Pressurizer Equipment." The background, the NRC positions, and clarification of the positions are included in NUREG-0737.

The specific requirements of these positions are not directly applicable to the liquid metal fast breeder reactor plants. This item is thus considered resolved.

8.3.3.6 Compliance With PDC 16

The final plant design will satisfy the requirements of PDC 16, "Inspection and Testing of Electric Power Systems," with respect to the onsite ac and dc power system. The onsite power system will be designed to be testable during operation of the nuclear power generating station as well as during those intervals when the station is shut down.

PDC 16 is the same as GDC 18 of Appendix A to 10 CFR 50. The following items address the problem areas revealed during the staff review and their resolution.

8.3.3.6.1 Testing of Circuit Breakers

Section 8.3.1.1.2 of the PSAR, under the subheading "Testing and Inspection," implies that when 4.16-kV and 480-V switchgear circuit breakers are being tested, their capability to respond to a bonafide accident signal during plant operation has not been maintained in accordance with the guidelines of Section 5 of IEEE Std. 338-1977. By letter dated June 1, 1982, the applicants have further implied that the subject breakers will not be designed, in accordance with the guidelines of Section 5 of IEEE Std. 338-1977, to be functionally tested during plant operation.

The applicants, by letter dated November 2, 1982, indicated that the Class 1E power system will be divided into three functionally redundant load groups so that each load group will have the capability to safely shut down the plant. Thus, there will be sufficient redundancy within the Class 1E power system to ensure safe shutdown during testing, even when degraded by a single random failure. The staff concludes that the design will meet the requirements of Section 5 of IEEE Std. 338-1977 and is, therefore, acceptable.

8.3.3.7 Compliance With PDC 41

The final plant design will satisfy the requirements of PDC 41, "Containment Design Basis," with respect to electrical penetrations containing circuits of

the safety and nonsafety onsite power systems. PDC 41 requires, in part, that the reactor containment structures, including penetrations, be designed so that the containment structures and internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences, or any of the postulated accidents.

PDC 41 is the same as GDC 50 of Appendix A to 10 CFR 50. The following items address the problem areas revealed during the staff review and their resolution.

8.3.3.7.1 Design of Electric Penetration

Section 8.3.1.2.11 of the PSAR indicates that conductors of the penetration will be designed to withstand the maximum short-circuit currents on the basis of the interrupting capability of the protection device associated with the penetration assembly conductors. Position C.1 of RG 1.63, on the other hand, states that the electrical penetration assembly (versus the conductor) should be designed to withstand the maximum short-circuit condition. By Amendment 74 to the PSAR, the applicants corrected the PSAR to state that both the conductor and penetration will be designed to withstand the maximum short-circuit currents. The staff concludes that the design will meet Position C.1 of RG 1.63 and is, therefore, acceptable.

8.3.4 Evaluation Findings

The review of the onsite ac and dc power system for the CRBRP covered single-line diagrams, preliminary station layout drawings, and descriptive information. The basis for acceptance of the onsite power system in the staff's review was conformance of the design criteria and basis to the Commission's regulations as set forth in the GDC of Appendix A to 10 CFR 50. The staff concludes (1) that the acceptance criteria and guidelines set forth in SRP Table 8-1 (NUREG-0800) provide a sufficient basis for acceptance of onsite electric power systems used at a liquid metal fast breeder reactor plant and (2) that there is reasonable assurance that the final plant design will meet the requirements of PDC 2, 5, 6, 15, 16, and 41 and will conform to the applicable guidelines of regulatory guides, branch technical positions, and NRC reports and is acceptable.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

Introduction

The CRBR fuel-handling system will consist of those components and systems necessary for the receipt, storage, handling, preparation, and insertion into the reactor of new fuel and for the removal from the reactor, handling, storage, examination, and preparation for offsite shipment of irradiated fuel. The major components and systems that will be in the fuel-handling area and their functions are:

(1) *New fuel shipping containers (NFSCs)--used to transport unirradiated fuel assemblies to the CRBR site and provide temporary storage in the reactor service building (RSB).

(2) New fuel unloading stations (NFUSs)--consist of two pits in the RSB used to support the NFSCs in an upright position for fuel unloading.

(3) Ex-vessel storage tank (EVST)--a sodium-filled vessel in the RSB used to store new or irradiated core assemblies.

(4) Auxiliary handling machine (AHM)--used to set up equipment needed for fuel handling (such as the in-vessel transfer machine) and serve as a maintenance cask for some components.

(5) Ex-vessel transfer machine (EVTM)--used to transfer irradiated and non-irradiated components to or from the EVST, NFUS, fuel-handling cell, or the in-vessel transfer locations.

(6) In-vessel transfer machine (IVTM)--used to transfer irradiated and non-irradiated components to or from the in-vessel transfer locations to the reactor core.

(7) Rotating guide tube (RGT)--used in conjunction with the EVTM to access the in-vessel transfer locations.

(8) Fuel-handling cell (FHC)--used to examine irradiated core assemblies and to load them into the spent fuel shipping cask for offsite shipment.

(9) Floor valves--used to seal openings in the reactor head, EVST head, and FHC when their normal seal plugs are removed for refueling. They will interface with the fuel-handling machines and can be opened and closed from the operating deck to allow insertions and removals by the fuel-handling machines.

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*It should be noted that the NFSC and the SFSC are to be licensed separately and are included in this section only to show their interface with the fuel-handling system and the safety aspects of their use.

(10) Fuel transfer port adapter and cooling inserts--used to provide cooling for removal of decay heat if an irradiated fuel assembly becomes immobilized in a reactor, EVST, or FHC fuel transfer port during transfer.

(11) *Spent fuel shipping cask (SFSC)--used to transport irradiated assemblies off site.

The core assemblies that will be handled by the fuel-handling system are:

(1) fuel assemblies
(2) inner blanket assemblies
(3) radial blanket assemblies
(4) control assemblies
(5) removable radial shield assemblies
(6) special assemblies (used before and during initial fuel load to position the core, filter the sodium and hydraulically simulate the fueled assemblies)

The following is a summary description of the sequence of events that will be used in the fuel-handling operation, from receipt of new fuel in the reactor service building to loading the irradiated assemblies into a cask for offsite shipment. Refer to Figures 9.1 and 9.2 for locations of equipment and visual representation of the flowpath. A discussion on the individual major components of the system follows this summary.

New fuel assemblies will be brought into the reactor service building in a new fuel shipping container. The shipping containers will be placed upright in the new fuel unloading station, opened, and inerted with argon. The EVTM then will pick up the assembly and insert it into an ex-vessel storage tank preheat position to preheat the assembly to avoid thermal shock when it is inserted into the 400°F ex-vessel storage sodium pool. After preheat the EVTM will insert the assembly into a sodium-filled core component pot (CCP) in the ex-vessel storage tank to await the start of refueling operations and insertion into the reactor core.

Refueling operations will begin with reactor shutdown and with the sodium at refueling temperature (~400°F). The allowable reactor cover gas activity for refueling is specified in the Technical Specifications and will have to be met before refueling is started. The equipment hatch between the reactor containment building (RCB) and the RSB will be opened. The control rod drive lines (CRDLs) will be disconnected from the control assemblies, which will be fully inserted in the core. The drive lines and the upper internals then will be raised to clear the tops of the core assemblies to allow the reactor head plugs to rotate.

Adapters and floor valves will be installed on the reactor head over the IVTM and fuel transfer ports. The plugs then will be removed from these ports by the auxiliary handling machine and the EVTM, respectively.

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*It should be noted that the NFSC and the SFSC are to be licensed separately and are included in this section only to show their interface with the fuel-handling system and the safety aspects of their use.

The auxiliary handling machine will be used to install the in-vessel section of the IVTM in the small rotating plug of the reactor head. Then the drive section of the IVTM will be installed, using the overhead crane in the reactor containment building. The reactor will now be ready for refueling.

The IVTM will be located in the small rotating plug that will be located in an intermediate rotating plug, which in turn will be located in the large rotating plug in the reactor head. The design will be such that the IVTM can be positioned over any core assembly by appropriately rotating these three plugs, as well as over the four transfer positions that will be outside the core but inside the reactor vessel.

The EVTM and IVTM will work in conjunction during refueling to exchange spent core assemblies in the reactor with new core assemblies in the ex-vessel storage tank. Two operations will occur simultaneously--the EVTM will remove a new core assembly in a sodium-filled core containment pot (CCP) from the ex-vessel storage tank, and the IVTM will remove a spent core assembly from the core and deposit it in a CCP located in a transfer position. The EVTM then will place the CCP containing the new core assembly in an adjacent transfer position inside the vessel, the rotating guide tube will rotate to the first transfer position, and the EVTM will remove the CCP containing the spent core assembly from the first transfer position. During this time, while the EVTM is at the reactor, the IVTM and rotating plugs must remain stationary. The EVTM then will move away from the reactor to deposit the CCP with the spent core assembly in the ex-vessel storage tank and pick up a new assembly in a CCP from the ex-vessel storage tank while the IVTM, operating in conjunction with the rotating plugs, will install the new core assembly in the open core lattice position. These operations will be repeated until the refueling is completed. During refueling in the core, the ex-vessel neutron count rate will be monitored. This monitoring will have the capability of detecting gross fuel-loading errors.

Spent blanket and control assemblies will be exchanged first to increase core negative reactivity. Then, spent fuel assemblies will be exchanged for fresh fuel assemblies in inverse order of decay power. As a consequence of this refueling sequence, time is permitted for decay, thus reducing the decay heat that must be removed by the EVTM during fuel handling. This sequence will reduce the maximum spent fuel assembly decay power handled by the fuel-handling equipment to $\sim$15 kW as compared with the 20-kW capability of the fuel-handling machines.

After refueling is completed, the IVTM will be removed from the reactor and stored in the reactor containment building. The fuel transfer and IVTM port plugs will be reinstalled in the reactor head, and both floor valves and adapters will be removed to their respective storage areas in the reactor containment building. The reactor upper internals will be lowered, and the control rod drive lines will be reconnected to the control assemblies. Finally, the hatch between the reactor containment building and reactor service building is closed, and the reactor will be ready for operation.

After spent fuel has decayed for $\sim$100 days in the ex-vessel storage tank, it can be loaded into the spent fuel shipping cask (SFSC). Spent assemblies to be loaded into the SFSC will be transferred in sodium-filled core component pots

from the ex-vessel storage tank to the fuel-handling cell by the ex-vessel transfer machine. In the fuel-handling cell, they will be stored temporarily in a three-position spent-fuel transfer station. The assemblies will be removed one at a time from the core component pots in the transfer station, the exterior will be dimensionally and visually examined, if desired, and residual sodium will be drained before loading.

The SFSC will be brought on site by railroad car. The cask will be removed from the railroad car and lowered down a shaft onto a transport dolly by the reactor service building crane, and the outer shipping cover will be removed. The dolly will move the cask under the fuel-handling cell floor where it will be sealed to the bottom of the cell. An access plug in the floor of the cell will be removed by an in-cell crane, and spent fuel assemblies will be loaded into the cask. The cask will then be detached from the fuel-handling cell (FHC), the shipping cover will be reinstalled, and the cask will be removed from the FHC shaft and loaded onto the rail car. The cask will then be ready for shipment off site.

Scope of Review

PSAR Sections 9.1, "Fuel Storage and Handling," and 15.5, "Fuel Handling and Storage Events," were reviewed along with SRP Sections 9.1.1, "New Fuel Storage," 9.1.2, "Spent Fuel Storage," 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," 9.1.4, "Light Load Handling System," and 9.1.5, "Overhead Heavy Load Handling System." In addition, the following documents were reviewed and considered applicable:

(1) American Nuclear Society (ANS) Std. 57.1, "Design Requirements for LWR Fuel Handling Systems"

(2) ANS Std. 57.2, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"

(3) ANS Std. 57.3, "Design Requirements for New LWR Fuel Storage Facilities" (not formally issued)

(4) Regulatory Guide (RG) 1.13, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"

(5) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

(6) NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants"

CRBR Principal Design Criteria (PDC) 53, "Fuel Storage and Handling and Radioactivity Control," 54, "Prevention of Criticality in Fuel Storage and Handling," and 55, "Monitoring Fuel and Waste Storage," were also reviewed since they specifically apply to the design.

Evaluation Criteria

The fuel storage and handling systems were reviewed

(1) For compliance with PDC 53, 54, and 55

(2) For conformance with the applicable portions of SRP Sections 9.1.1 through 9.1.5, ANS Stds. 57.1 and 57.2, RG 1.13, NUREG-0612, and NUREG-0554. Although ANS Std. 57.3 has not yet been formally published, the requirements of this standard were also evaluated for applicabilty, and those considered applicable were used in the evaluation.

Although these ANS standards, SRP Sections, and RG 1.13 are written for light-water reactors, many of the basic principles and requirements they address are considered applicable to a liquid metal fast breeder reactor. The staff reviewed each of these documents and made a judgment regarding the applicability of the requirements. In general, all requirements that were not unique to a water system were considered to apply.

(3) In general, for

(a) acceptability of proposed design requirements and preliminary design
(b) potential for causing inadvertent criticality
(c) potential for causing pin or assembly damage/failure
(d) ability to minimize the spread of radioactive material
(e) ability to withstand postulated failures/accident conditions
(f) provisions for instrumentation to monitor system operability
(g) provisions for adequate shielding
(h) ability to minimize fuel misloading errors in the core

Evaluation

The fuel storage and handling system is to be designed to limit the cladding temperature of fuel assemblies to the limits specified in Table 9.1. These limits were chosen to maintain cladding integrity for normal and anticipated events and limited pin failures (fission gas release only) for unlikely and extremely unlikely events. For assemblies to be reinserted into the reactor or examined, the limits were chosen to preclude any additional significant cladding damage.

The following paragraphs describe in more detail the systems, components, and safety aspects of the fuel-handling system and are organized to parallel the PSAR sections.

9.1.1 New Fuel Storage

New fuel will be stored on site in the new fuel shipping containers (NFSCs) and in the ex-vessel storage tank. Each NFSC can only hold one core assembly and is about 20 in. in diameter. The containers will be removed from the truck one at a time by the reactor service building (RSB) crane and lowered into a new fuel unloading station. The shipping cover will not be removed from the container until it is installed in the unloading station.

Two new fuel unloading stations will be located between the ex-vessel transfer machine (EVTM) gantry-trolley rails in the RSB. A new fuel shipping container will be supported and placed in a vertical position at each station to permit loading or unloading of a new core assembly by the EVTM.

Each station will consist of a pit in the operating floor with a structural steel support for the shipping container. An adaptor and valve assembly will

be mounted on the top of the unloading station to provide an interface between the unloading station and the EVTM. The adaptor will mate with and seal to the EVTM closure valve. Argon gas to inert and purge the shipping containers will be provided from a floor service station.

Fuel in the NFSCs is to be protected from criticality by the physical dimensions of the NFSCs so that even if the containers are stacked in a large array with new fuel of 33% Pu enrichment and flooded with water, criticality will not result. Decay heat is not a concern with new fuel.

New fuel in the ex-vessel storage tank (EVST) can be stored in a sodium-filled core component pot (CCP) or in one of the 24 EVST argon-filled preheat stations. The design of the EVST is to be such that the physical separation between storage locations and permanently installed B_4C absorbers will maintain the K_{eff} of the EVST below 0.95 (with or without sodium) even if fully loaded with fresh fuel of 33% Pu enrichment.

9.1.2 Spent Fuel Storage

Reactor spent fuel will be stored in the EVST (Figure 9.3). The EVST is a cylindrical vessel made of type 304 stainless steel and designed and fabricated to ASME Code, Section III, Class 2 requirements. It is to be designed for a steady-state temperature and pressure of 650°F and 15 psig, respectively. The EVST will be located inside a guard vessel (an ASME Code, Section III, Class 3 vessel) that will be contained in a steel-lined, nitrogen-inerted concrete cell beneath the floor of the RSB. The EVST guard vessel is to prevent the loss of Na inventory from the EVST in the event of a leak. This is an analogous function to the reactor guard vessel. Accordingly, it is designated as ESF (see Section 6). The EVST will be filled with sodium and supplied with an argon cover gas. It will contain storage tubes within which will fit the CCPs with the assemblies inside. A center-to-center distance of 9 in. or more will be provided between storage tubes; the EVST will be capable of storing 650 core components in two tiers. The storage tubes will be mounted on a turntable, which can be rotated beneath a fixed head; nine fuel transfer ports in this head will allow core components to be installed or removed from any storage position.

The EVST fixed-head assembly will consist of a 6.5-in. plate that will be part of the RSB floor, a 12-in.-thick closure head 10 in. below the floor plate, and a multilayer steel thermal insulation assembly between the closure head and the surface of the sodium. The nine fuel transfer ports penetrating the head assembly will each be fitted with a cooling sleeve that will extend from the closure head to a point between the insulation and the sodium surface. Double-sealed stepped plugs will be installed in the fuel transfer ports except during core component transfer operations.

Nine storage positions in the upper tier and nine corresponding positions in the lower tier will contain B_4C-filled neutron absorber assemblies and will be inaccessible for fuel storage or for removal by the fuel-handling equipment. These poison assemblies along with the storage position geometry will limit the value of K_{eff} for the EVST to <0.95. In addition, the capability to install a neutron count rate monitor at the EVST location to confirm subcriticality will be provided.

The EVST will be designed with sensors and interlocks to prevent any unscheduled movement of the turntable while the EVTM is being mounted on the EVST. The interlocks will allow the turntable to rotate only when the EVTM grapple is in the full up position.

Temperature instrumentation and sodium-level sensing probes will monitor cooling capability. High EVST sodium outlet temperature and high or low sodium levels will sound an alarm. Sodium-leak detectors will monitor the space between the storage vessel and the guard tank. An argon cover-gas activity monitor will be provided. An area monitor of the gamma scintillation type will measure the gamma radiation on the RSB operating floor above the EVST.

An insulating blanket and heaters outside the guard vessel will help maintain the desired sodium temperature in the EVST.

The EVST is designed to be a seismic Category I structure.

The grid structure securing both ends of the EVST storage tubes will be designed to preserve the fuel element spacing during a seismic event and prevent criticality with the maximum loading of 650 fuel elements.

The EVST head will be designed to absorb the impact of the heaviest load that can pass over it without damage or relocation of the stored fuel. The heaviest loads are defined as (a) the heaviest load carried by the RSB overhead crane main hook lowered at maximum crane speed and (b) the dropping of the maximum load from the RSB overhead crane auxiliary hook from maximum height.

Sufficient shielding will be provided by the EVST structure and its vault to maintain personnel exposures below the requirements of 10 CFR 20 during normal operation.

The EVST will be designed to contain the entire fission gas inventory of one spent fuel assembly by limiting leakage through penetration seals, that causes radioactive releases, to values below the guidelines of 10 CFR 100. In addition, the EVST will have a cover gas cleanup system associated with it. Section 15.5 summarizes releases for unlikely and extremely unlikely events for the fuel storage and handling system.

9.1.3 Spent Fuel Cooling and Cleanup System

Removal of decay heat from the spent fuel stored in the EVST and purification of the EVST sodium will be accomplished by the ex-vessel storage (EVS) processing system. The details of this system are described in Section 9.3, "Auxiliary Liquid Metal Systems." The design requirements and description associated with the decay heat removal function of this system and of the EVST are discussed in this section.

There will be three independent, redundant, and separate means of providing heat removal from the EVST, each of which will be capable of providing adequate cooling. The three heat removal paths will consist of two identical forced-circulation EVS sodium processing loops and one natural convection loop of a design different from that of the forced circulation loops. Each of these

loops will transfer its heat to a NaK loop, which in turn will dump the heat to the atmosphere by means of a heat exchanger.

Figure 9.4 provides a diagram of these three loops. Each of these loops will be physically separated, with the sodium portion of the loop housed in separate nitrogen-inerted lined cells.

During normal operation the EVST sodium temperature will be maintained by one of the EVS forced-circulation loops at 400°F inlet and ~510°F outlet (corresponds to a fuel clad temperature of ~660°F), with 1,800 kW heat removal. 1,800 kW is the design value heat removal capacity for each of the three heat removal loops for normal operation. It corresponds to the decay heat that would be in the EVST assuming a complete core unloading occurring along with two annual loads of spent fuel assemblies already in the EVST. If the natural convection loop is used for cooling with 1,800 kW of decay heat, the EVST sodium temperatures will be 400°F inlet and ~600°F outlet with the peak fuel cladding temperature being ~750°F.

The NaK portion of the two forced-circulation loops also will be part of the reactor direct heat removal system (DHRS), described in Section 5. When the DHRS will be in use for removing reactor decay heat, EVST heat will be removed by the EVST natural circulation loop.

All components of the normal sodium and NaK loops that require electrical power will be on the Class 1E power system to ensure continuous EVST cooling and reactor decay heat removal (i.e., in the event of complete loss of external power to the plant, power to both of the normal cooling circuits will be provided by the plant diesels). Separate diesels will power each forced-circulation loop. The only "active components in the backup loop will be two redundant dampers on the natural draft heat exchanger. Each will be operated manually and, therefore, will not require connection to the emergency power system. Each will be capable of providing full air flow for adequate cooling.

Immediate activation of the diesel-power supply will not be necessary for the EVST sodium cooling, since the sodium volume within the EVST will provide a heat sink to attenuate the sodium temperature rise during loss of forced circulation; in addition, the natural circulation heat removal path alone will be adequate to remove EVST decay heat. Activation of the emergency power supply to the NaK pumps and air-blast fans will be provided within ½ hour, however, to support the availability of DHRS for reactor decay heat removal.

Even if all forced and natural circulation heat removal paths are lost, the heat removal by conduction through the EVST insulation into the inerted cell will be sufficient to maintain the EVST sodium temperature below 1,500°F for up to 20 hours under this condition with 1,800 kW of decay heat. For lower decay heat levels this time would be longer. This means that immediate action on loss of all EVST cooling will not be required (since the integrity of the EVST vessel and coolable geometry of the fuel will be maintained at this temperature) and that considerable time will be available to restore a cooling path either by repairing the inoperable equipment or by establishing an alternate heat removal path through cross-connection between EVST cooling loops or the DHRS. It should be noted that the design provides significant flexibility for the establishment of alternate flow paths.

During normal operation the sodium level will be maintained at a high enough elevation so that normal fluctuations resulting from changes in temperature or number of stored components will not uncover the top of the CCPs in which the spent fuel is stored. During offnormal conditions, such as a leak or rupture in either the vessel or a cooling system, the vessel sodium outside the CCPs cannot fall below a level that would affect the operation of the remaining cooling loops. This will be accomplished by the guard vessel around the EVST vessel, which limits the loss of the EVST sodium inventory, by the sodium nozzle locations in the EVST which will be in the upper elevations of the vessel wall, and by the EVST sodium inlet lines which will contain antisiphon devices to prevent a cooling system leak from lowering the vessel sodium to a level that will affect the operation of the remaining cooling loops.

The EVST cleanup system will provide the capability to maintain the sodium purity and will be independent of the reactor and primary loop sodium purification system.

The EVST sodium purity will be maintained by cold traps. The limits for sodium purity are 5 ppm oxygen and 0.4 ppm hydrogen. The basis for establishing these oxygen and hydrogen limits is corrosion, which is temperature dependent. The corrosion effects from these impurity levels for the EVST operating at 500°F will be comparable to corrosion effects from 2 ppm oxygen and 0.2 ppm hydrogen for the reactor sodium operating at 1,000°F.

The radiochemical limits currently imposed for the EVST sodium are 10 ppb plutonium and uranium. These limits are established to minimize potential maintenance problems resulting from contamination of components owing to potential sodium spills.

EVST sodium samples will be taken by the multipurpose sampler (MPS) for laboratory chemical analyses. Chemical analysis will be performed monthly with the following exceptions:

(1) During periods of anticipated changes (initial fill, refueling, and maintenance activities) in impurity levels, the frequency will be increased.

(2) When it has been established that an element impurity level is no longer changing after several years of plant operations, the frequency of analysis may be reduced.

Indication of oxygen and hydrogen concentrations by the plugging meter technique will be obtained routinely once a day, or if there is no apparent change over a long period of operation, at a decreased frequency.

The criterion for forced-circulation cold trap replacement is an increase in cold trap differential pressure, ΔP, at design flow. The limiting ΔP is currently estimated to be 22 psid. When this limiting ΔP is reached, cold trapping may be continued at decreased flows provided impurity limits are not exceeded.

The following instrumentation and controls will be provided to ensure safe operation of the EVS sodium processing system and adequate EVST decay heat removal.

Temperature monitoring of the inlet and outlet of all heat source and sink components, in conjunction with loop flow measurements, will be provided for all systems to monitor their status. Critical temperatures and flows will be alarmed in the control room. All electromagnetic pumps will be provided with winding temperature measurements and winding coolant low-flow indication. These measurements will be alarmed for offnormal conditions and interlocked to automatically shut down the pump to prevent damaging it.

The EVST and the NaK expansion and storage tanks will be provided with level measurements, which will be alarmed for abnormal low and/or high level. This information, in conjunction with leak detection data, will be used to diagnose external liquid metals leaks. The operator will be alerted to NaK-to-sodium leakage by NaK expansion tank high-low level alarms. All the bellows seal valves will be provided with leak detectors. All valves will have position indicators. Accessories (solenoids, pressure gages, I/P converters, etc.) for remotely operated valves installed in inerted cells will be installed off the valves in adjacent accessible air cells. To provide for continued operation and prevent possible system damage resulting from control system failures, manual control capability will be provided on all controllers. The design requirements for the safety-related instrumentation were evaluated in more detail in Section 7 of this SER.

9.1.4 Fuel-Handling System

The CRBRP fuel-handling system will consist of the following subsystems and components:

(1) ex-vessel transfer machine (EVTM)
(2) in-vessel transfer machine (IVTM)
(3) auxiliary handling machine (AHM)
(4) floor valves
(5) fuel transfer port adapter and inserts
(6) rotating guide tube (RGT)
(7) fuel-handling cell (FHC)

Each of these components is discussed in more detail in the following sections.

9.1.4.1 Ex-Vessel Transfer Machine

The primary function of the ex-vessel transfer machine (EVTM) will be to transfer core assemblies between the reactor, ex-vessel storage tank, new fuel unloading station, and the fuel-handling cell. The EVTM will be designed to handle both new and irradiated core assemblies in sodium-filled core containment pots and bare new core assemblies.

The EVTM is a shielded, inerted, single-barrel fuel-handling machine (see Figure 9.5). The pressure boundary will be designed in accordance with ASME Code, Section III, Class 3 requirements. It will be mounted on a trolley that, in turn will be positioned on rails on top of the gantry. The gantry will move on crane rails between the reactor containment building and the reactor service building. The trolley rails will be perpendicular to the gantry rails, allowing complete indexing of the ex-vessel storage tank, new fuel unloading station, fuel-handling cells, and reactor vessel.

The major subassemblies of the EVTM will be the grapple drive system, several cask body modules that will provide structural support and shielding, service platforms, a view port assembly to permit visual inspection of the cell containment pots, port plugs, drip pans, a drip-pan assembly to collect sodium drippage, an extender that will raise and lower the closure valve, a closure valve that will mate with the floor valve and provide the gas-tight seal, the cold-wall assembly, electrical instrumentation, and control equipment.

Irradiated core assemblies will be handled by the EVTM only in sodium-filled core containment pots (CCPs). New core assemblies will be handled both in CCPs (between the reactor and the ex-vessel storage tank (EVST)) and bare (between the new fuel unloading stations and the preheat stations in the EVST and between the preheat stations and the storage stations in the EVST)). Different grapples will be required for the bare assembly and the CCP. Changing of grapples will be done in the fuel-handling cell.

All seals on the fission gas boundary will be double seals with a pressurized buffer gas between them. This will allow seal leak testing. Each inflatable seal will be provided with a backup independent pressurization supply in the event of a loss of offsite power or main pressure supply.

The EVTM will be connected at each fuel transfer port by lowering the closure valve and mating it with the floor valve. This interface will be sealed with double seals on the top of the floor valve. The space between the seals will then be pressurized with argon buffer gas.

The grapple will be lowered into the handling socket of the CCP or core assembly by the chain drive system. The grapple fingers will be actuated and will engage the lip on the handling socket. The assembly or CCP containing the new or irradiated core assembly will then be raised into the EVTM cold-wall assembly. After the assembly or CCP is fully raised, the drip pan will be closed to prevent sodium drippage on the closure valve or floor valve. The closure valve and floor valve will be shut, and radioactive gas trapped between the valves will be purged.

Spent fuel assembly decay heat will be removed by a cold-wall system consisting of a sealed cold wall, approximately 8 in. in inside diameter, with an array of axial fins attached to the outside (see Figure 9.6). Heat from the 3-ft-high fueled region of the spent fuel assembly will be distributed over the 15-ft length of the CCP by natural convection of the sodium in the CCP. The heat will be transferred from the surface of the CCP to the cold wall primarily by thermal radiation and secondarily by conduction across a stagnant argon-filled gap. The cold wall will be cooled by forced convection of ambient air circulated past the axial cold-wall fins. Forced-air convection will be provided by a blower with the capacity to circulate sufficient air to maintain fuel cladding temperature to less than that seen at full power. Total heat removal capability will be 20 kW. In the event forced air cooling is lost, natural convection air flow will limit cladding temperature to less than the limits for unlikely events in Table 9.1. Instrumentation is to be provided so that the operator will know the status of the EVTM cooling. Adequate shielding for radiation protection will be provided in the design of the EVTM to meet the radiation protection requirements of 10 CFR 20.

The activity released from a damaged or leaking spent fuel assembly while in the EVTM will be contained in the EVTM by proper sealing or welding of penetrations and openings. No gas cleanup system will be associated with the EVTM. The elimination of radioactive gases, if necessary, will be by purging when the EVTM is mated to a component with gas cleanup capability (reactor vessel, ex-vessel storage tank, or fuel-handling cell). Radioactive leakage and diffusion through seals will be well below the limits specified in 10 CFR 100. Section 15.5 describes the limiting event of this type and the resultant consequences.

Mechanical damage to fuel assemblies could be caused by the dropping of a grappled CCP or new fuel assembly or by tipping over of the EVTM. Dropping of new fuel assemblies or of CCPs containing new or spent fuel assemblies will be prevented by the design of the grapple fingers and CCP grappling lip and by suitable interlocks. In addition, brakes are to be provided to stop assembly motion in the event of a loss of power. If, however, a CCP and assembly should drop, the bottom closure and drip pan would remain intact and retain the EVTM pressure boundary. In addition, the CCP would remain vertical, thus retaining its sodium, and cooling would still take place. The EVTM gantry and trolley will be designed with anti-liftoff restraints and rail stops to prevent derailing of the gantry and trolley under combined normal operating and safe shutdown earthquake loads. Mechanical collision between the EVTM on its gantry and other equipment, especially the control rod drive lines, in-vessel transfer machine, and equipment hatch between the reactor containment building and reactor service building will be prevented by a combination of stops, interlocks, and procedures. Movement of the EVTM while mated to the reactor vessel, ex-vessel storage tank, or fuel-handling cell is to be prevented by two independent interlocks. In addition, rotating plug motion while the EVTM is mated to the reactor head is to be prevented by interlocks.

If a seismic event should occur while the EVTM was mated to a floor valve at the reactor, or other location, the design would limit the transmitted structural loads so that the reactor head or other mating facility would not be damaged. Motion of the EVTM relative to the mated facility will be limited to prevent contact with or damage to a CCP, if a CCP happens to be passing through the floor valve or the mating facility at the time of the seismic event. Cover gas release will be prevented by maintaining sealing of the EVTM to the floor valve and mating facility during a seismic event.

Development testing on the heat removal capability of the EVTM under both forced and natural convection conditions has been completed. Full-scale tests were conducted at Hanford Engineering Development Laboratory using an electrically heated 217-pin assembly. The heat removal capability of the EVTM was confirmed. In addition, tests were run to determine the change in EVTM cold-wall emissivity resulting from sodium frost and condensation. In these tests it was determined that electrical heaters in the cold wall could control the frost and condensation and thus control the cold-wall emissivity.

9.1.4.2 In-Vessel Transfer Machine

The in-vessel transfer machine (IVTM) will be refueling machine that will be mounted on the small rotating plug of the reactor head only when the reactor is shut down for refueling (see Figure 9.7). The primary function of the IVTM

will be to move core assemblies between their normal positions in the reactor core and the transfer positions outside the core but within the reactor vessel. In the transfer positions, the core assemblies will be accessible for transfer in or out of the reactor vessel by the EVTM. The IVTM will execute only vertical movements; all horizontal movements will be performed by rotating the triple rotating plugs in the reactor head.

The IVTM is a rising stem-type design and will consist of two principal subassemblies, the in-vessel section and the ex-vessel or drive section. The in-vessel section will contain the grapple, centering device, identification mechanism, holddown mechanism, and the seals that will contain the reactor cover gas. It will be installed in the small rotating plug of the reactor vessel head with the auxiliary handling machine. The drive section will contain the equipment that will power the in-vessel section and will be installed subsequently with the polar crane. The in-vessel section will be designed to operate within the reactor's sodium-filled environment during shutdown, and the drive section will provide a transition for the machine to the air-filled environment where normal power equipment, such as electric motors and pneumatic cylinders, can be located.

The grapple and drive mechanism will raise or lower core assemblies inside the reactor vessel. By moving the reactor plugs, core assemblies grappled by the IVTM can be moved to and from a core position and an in-vessel fuel transfer position. All movements will be with the assembly completely immersed in sodium. A position indication system will be provided so that the operator can tell where an assembly being handled by the IVTM is at all times.

The IVTM will be sealed to the small rotating plug by three elastomer O-ring seals with buffer gas between them. A leak detector port will be provided between each set of seals to allow for monitoring the seals' integrity before refueling. This arrangement is typical for other seal sets that involve dynamic motion, such as the grapple stem and holddown drive shafts. Static seals or seals with very small displacements, such as those for the grapple finger actuator shaft will be double, with a capability for periodic leak testing. These seals will limit the release of radioactive cover gas to the limits of 10 CFR 20. No inflatable seals will be used.

The IVTM drive mechanism has been designed to exert a maximum upward or downward force of 5,000 lb. A pneumatic load-control system and a load cell system that limits the load exerted on a core assembly to 4,300-lb pull and 3,000-lb push will be incorporated into the design. Core assemblies are to be designed to withstand the maximum IVTM push/pull forces.

The load in the pneumatic load-control system will be set to provide a normal push or pull force of 1,000 lb on a core assembly. The force of the load control system may be adjusted to provide higher load capability up to the push or pull load limits.

Adequate shielding for radiation protection is provided in the design of the IVTM. The shielding will meet the radiation protection requirements specified in 10 CFR 20.

Dropping of assemblies, misplacement of assemblies, and damage resulting from rotating plug rotation will be prevented by the following:

(1) Grapple actuation will be prevented by two independent interlocks unless the assembly has been lowered to its setdown elevation.

(2) Vertical movement of the grapple will be prevented by an interlock until positive grapple engagement or disengagement is indicated.

(3) Horizontal translation of the grapple by reactor plug rotation will be prevented by an interlock until the grapple, the grappled core assembly, and the holddown sleeve are fully raised to clear the upper core internals.

(4) Improper core assembly insertion in the core will be prevented by the configuration of the core assembly discrimination post and its receptacle in the core support structure, which will prevent the assembly from fully seating in the core if it is in the wrong core orificing zone (see Section 4.4). Proper seating will be required to satisfy IVTM interlocks, and thus allow the IVTM to release the assembly.

(5) Improper core assembly insertion also will be prevented by IVTM design features permitting unique identification of each core assembly inside the reactor vessel. Core assemblies also will be identified immediately after removal from a core position to ensure that the proper core assembly has been removed.

(6) Movement of assemblies on a loss of power will be prevented by self-actuating brakes.

9.1.4.3 Auxiliary Handling Machine

The primary function of the auxiliary handling machine (AHM) will be to install the in-vessel section of the IVTM in the reactor head before refueling and remove it afterward. The AHM also will serve a plug-handling function in removing and reinstalling the IVTM port plug from the reactor head. Additionally, the AHM will support the maintenance functions of other systems by removing and installing control rod drive lines and core inlet modules using special tools, grapples, adapters, and so forth, provided by the reactor enclosure system and the reactor system.

The AHM is a shielded, single-barrel handling machine that will be transported by the reactor containment building polar crane (see Figure 9.8). The machine will be about 60 ft high and will weigh about 100 tons. The AHM is similar in concept to the ex-vessel transfer machine (EVTM), but it has been simplified because of the less demanding requirements of the components it will handle (as compared with the irradiated core assemblies in sodium-filled core component pots handled by the EVTM). The major differences are (1) no radioactive decay heat, (2) lower radiation source strength, (3) less sodium drippage, and (4) lower use factor.

The major subassemblies of the AHM will be

(1) the grapple and hoisting system

(2) the cask body assembly, which will provide structural support and shielding

(3) the service platforms

(4) a closure-valve and drip-pan assembly

(5) an extender assembly

(6) a cask containment barrel

(7) a gas service system, which will provide gas for process systems and monitors seal leakage

(8) the electrical instrumentation and control equipment

The AHM will be moved by the polar crane and will mate by means of an extendable portion of the AHM with a special floor valve at the desired location. While over the reactor head, the weight of the AHM will be supported entirely by the polar crane.

Shielding for radiation protection will be provided in the design of the AHM. The shielding will limit radiation levels outside the AHM to the limits of 10 CFR 20.

Radioactive releases from the AHM during installation or removal of the components handled by the AHM will be prevented by proper sealing or welding of penetrations. Seals will be of double configuration with a pressurized buffer gas in between that will allow leak testing. Radioactive leakage and diffusion through seals will be limited to meet the radiation requirements of 10 CFR 20. Each inflatable seal will be provided with an independent backup gas supply that will maintains seal inflation in the event of a loss of power or normal gas supply.

Mechanical damage to the reactor caused by the AHM because of the dropping of a grappled component or toppling over or dropping of the AHM will be precluded by the design of the AHM grapple fingers, the grappling lip of the component handling sockets, and by interlocks. In addition, self-actuating brakes are to be provided to stop component motion on a loss of power. The AHM handling lugs and lifting eyes will be designed to ensure positive engagement and accommodate all normal and safe shutdown earthquake loads. A single failure of the lifting structure (polar crane) will not lead to a drop of the AHM. The potential drop height of the AHM is to be limited administratively.

The AHM will be supported from the polar crane during the entire time it will be in the vicinity of the reactor head and will be securely attached to its parking structure when not in use. Both of these components are designed for seismic Category I conditions.

If a seismic event should occur while the AHM was mated to the floor valve at the reactor, and no transfers were being made through the floor valve, the

design would limit the transmitted structural loads by means of a breakaway joint so that the reactor head would not be damaged. The ability to operate the floor valve after the seismic event is to be maintained, and no damage of major equipment located in the head access area is to be caused by the AHM during or after an operating basis earthquake or safe shutdown earthquake. If a component were being transferred through the floor valve at the time a seismic event occurred, damage to the component and cover gas release could occur. Because the time during which the plant would be in this condition would be very short, the risk of this happening is small and the consequences are considered acceptable (see Section 15.5). The AHM will be installed on the reactor only when the radioactivity concentration of the reactor cover gas will be below the limit given in the Technical Specifications, Section 16.3.10. The AHM will not have a gas cleanup system, thus, if necessary, the removal of any radioactive gas from the AHM will be by purging.

9.1.4.4 Floor Valves

The functions of a floor valve will be to provide shielding and a seal against radioactive gas when the plug is removed from a transfer port in the reactor, ex-vessel storage tank (EVST), or fuel-handling cell (FHC), by isolating the contaminated gas and radiation environment of these facilities from the ambient air of the reactor containment building and the reactor service building.

The floor valves will be portable, rotatable disc-type valves and will be used to seal the reactor, EVST, and FHC ports during various phases of the refueling operations when the transfer port plugs are removed (see Figure 9.9). Two basic sizes of floor valves will be used--one will be used with the EVTM for the transfer of core assemblies, and another with a larger opening will be used with the auxiliary handling machine (AHM) for the transfer of the in-vessel section of the IVTM and other components handled by the AHM. Both types will have the same basic design except for interior opening size and differences in thickness of the radial shielding surrounding the port.

The floor valve design concept is similar to the AHM and EVTM closure valve design. Floor valves will be installed by the building crane, bolted down, and thus will be satisfactory for safe shutdown earthquake conditions. The AHM or EVTM will be moved into position over the floor valve at the reactor, the ex-vessel storage tank, or the fuel-handling cell. After sealing, the AHM or EVTM closure and floor valves can be opened for the AHM or EVTM to access the component over which it is located.

Sufficient axial and radial shielding for the floor valves will be provided to meet the requirements of 10 CFR 20.

The floor valve (FV) will have double buffered seals to prevent excessive radioactive release to the operating floors of the reactor containment and reactor service buildings and to allow leak testing. Inflatable seals will be used on the floor valves and the gas supply for these seals will be provided by the independent supplies on the EVTM and AHM. Seal pressure will be locked into the FV seals when the AHM or EVTM are is not mated to the FVs. Accidental cover gas release through inadvertent opening of a floor valve if a mating fuel-handling machine (EVTM or AHM) is not on top of the floor valve will be prevented by interlocks. One interlock will prevent energizing the valve

operating motor unless a mating machine is on top of the floor valve (electrical power to the floor valve motor will be supplied by connection to the mating machine). Other interlocks will prevent (1) depressurizing the buffer gas zone between seals and (2) raising the closure valve extender, unless both the closure valve of the handling machine and the floor valve are in their closed positions.

9.1.4.5 Fuel Transfer Port Adapter and Cooling Inserts

The reactor fuel transfer port adapter will be positioned on top of the reactor, ex-vessel storage tank, or fuel-handling fuel transfer ports and extend from the reactor head to the bottom of the floor valve that will be located at the elevation of the operating floor. It will serve as an extension of the reactor cover gas containment and provide shielding when irradiated core assemblies are removed from those locations. The adapter also will guide cooling air from an air blower to a cooling insert inside and below the adapter.

The function of the cooling inserts will be to remove decay heat if an irradiated core assembly in a sodium-filled CCP becomes immobilized in a fuel transfer port during transfer between the reactor vessel, ex-vessel storage tank, or fuel-handling cell and the EVTM.

The upper surface of the reactor fuel transfer port adapter will consist of a flange that will be bolted to the floor valve and provide the sealing surface for the double seals on the lower surface of the floor valve. Shielding will be provided by a thick, annular lead cylinder surrounding the adapter cover gas containment tube over its entire length to limit the dose rate at the shield surface to less than the limits specified in 10 CFR 20. The lower part of the adapter will be bolted to its mating surface during fuel handling.

The reactor fuel transfer port cooling insert will extend from the top flange of the adapter to the fuel transfer port nozzle. A cold-wall cooling concept, similar to that of the EVTM, will be used for the cooling insert. The core containment pot containing a spent fuel assembly will be cooled by thermal radiation and conduction across the argon gas gap to the cold wall that form the confinement barrier for the reactor insert. Air flow from the blower will cool the outside of the cold wall and will be adequate to limit the cladding temperature of a 20 kW fuel assembly to less than 1,500°F if an assembly should get stuck in this position. Instrumentation in the cooling insert will enable the operator to determine the status of the cooling system. In the event of a loss of power that would immobilize a spent assembly in the fuel transfer port and stop forced cooling, the EVTM would have the capability to manually raise or lower the assembly to a position with adequate cooling. The operator would have 30 min to accomplish this and still maintain a cladding temperature less than that required for an anticipated event. Cladding temperature, however, will not exceed 1,500°F even if the operator fails to move the assembly and forced cooling is lost.

9.1.4.6 Rotating Guide Tube

The function of the rotating guide tube (RGT) will be to provide a means for the EVTM to insert a CCP containing a core assembly into one of four fixed

in-vessel transfer positions and remove a CCP containing an irradiated core assembly from an adjacent storage position, all without requiring an EVTM decoupling-recoupling procedure.

The RGT will consist of a straight tube with an eccentric lower end. The RGT will be permanently mounted on the top of the reactor fuel transfer port nozzle and remain in the reactor during operation. During the refueling sequence, it will be rotated 180° to locate its lower end over either of two adjacent transfer/storage positions. Following refueling, the EVTM will be used to insert a plug into the RGT. Sealing and plug holddown will be completed by a cap attached to the reactor nozzle. This cap also will serve to secure the RGT during reactor operation.

9.1.4.7 Fuel-Handling Cell

The fuel-handling cell (FHC) is an argon-inerted, shielded, steel-lined hot cell that will be located in the reactor (RSB) with powered and master/slave manipulators (see Figure 9.10). The cell service building will be normally operated at negative pressure with respect to the RSB. The primary use of the FHC will be to facilitate the handling and shipping of spent fuel. The EVTM will lower a spent fuel assembly in a CCP into one of three storage positions in a lazy-susan transfer assembly. The FHC crane will be used to attach a gas-cooling grapple to the fuel assembly. The assembly then will be removed from the CCP and transferred to a fuel-shipping container that will be sealed into a port through the FHC floor. The shipping container then will be moved off site by railroad car.

Core assemblies can also be placed in an examination fixture for inspection. Another function of the FHC will be to recover the sodium drip pots from the EVTM, dump the sodium, and make the empty pots available to the EVTM grapple. The FHC also will be used for EVTM grapple changeout and cleaning as well as various cleaning and maintenance operations on contaminated components.

The spent fuel assemblies in the FHC will be cooled by natural convection of the argon atmosphere around CCPs in the spent fuel transfer station and by blowers that will force the argon through the bare fuel assembly when it is grappled. The heated cell argon will be removed from the FHC, cooled and cleaned, and ducted back into the FHC by the recirculating gas cooling system (Section 9.16 in this SER).

Shielding will be provided for normal operation to limit the dose rates outside the cell to 10 CFR 20 values.

Radioactive releases and contamination during offnormal events from spent fuel assemblies that are being prepared for shipment in the FHC will normally be contained within the FHC by proper sealing or closure welding of penetrations and operation of the cell at negative pressure. Inflatable seals on the FHC will have check valves on their supply lines to prevent seal deflation on loss of power or supply gas. Even in the event of release of the entire fission gas inventory of a fuel assembly, and taking no credit for the FHC boundary, site boundary dose rates will be limited to well below the criteria of 10 CFR 100.

Criticality of fuel assemblies in the spent fuel transfer station in the FHC will not be possible because only three fuel assembly locations will be provided. The FHC liner and spent fuel transfer station will be designed for seismic Category I conditions. Although not designated an engineered safety feature (ESF), the liner for the FHC will be designed to the same requirement as the ESF liners. The FHC roof closure will be designed to absorb the load of the heaviest equipment handled by the RSB bridge crane over the FHC: (1) for the main hook, lowered at the maximum crane speed (5 ft per minute) and (2) for the auxiliary hook, accidentally dropping the load from the maximum handling height to which it is raised, onto the center of the roof closure. The FHC will be located so that heavy equipment not belonging to the fuel-handling and storage system will not be carried over it by the RSB bridge crane.

The spent fuel transfer station within the FHC will be designed so that movement of the lazy susan will not occur while a CCP is being inserted or withdrawn. This design condition will prevent mechanical damage to the CCP or its contents. Handling equipment inside the cell will be designed to minimize the potential for dropping of assemblies.

Monitoring instrumentation will be provided for the FHC for conditions that might result in a loss of the capability to remove decay heat and to detect excessive radiation levels.

Sufficient cooling capacity will be provided in the FHC to remove 18 kW of heat from assemblies in CCPs. Decay power of assemblies inserted into the cell will normally be limited to 6 kW each. However, the transfer station will be capable of cooling a bare assembly with up to 15 kW of decay heat (via the cooling grapple) in the event it is desired to rapidly examine an assembly after shutdown.

Cell cooling will be provided by redundant recirculating gas cooling systems, each consisting of a fan, gas heat exchanger, and a piped distribution system. The heat exchanger will remove heat from the argon gas and reject it through a recirculating Dowtherm loop, which in turn will reject it to the chilled water system. The cooling system (both cell cooling and grapple blowers) will operate during normal plant operation and be supplied with diesel power in the event of offsite electrical power failure. The argon portion of the system, including the shell of the cooler, will be designed to ANSI Std. B31.1, and Section VIII of the ASME Code, seismic Category I, and be located within a hardened structure.

The crane-handled gas cooling grapple (used for bare assembly handling) will be supplied by two redundant argon gas-cooling blowers, which will be mounted on the upper end of the gas-cooling grapple. These blowers will draw argon gas from the surrounding cell environment and blow it through the grapple and fuel assembly, discharging it back into the cell through the nozzles at the bottom of the fuel assembly.

During normal and anticipated events, fuel-cladding temperatures, when fuel is residing in the transfer station, will be well below the 1,250°F limit.

In the unlikely event of loss of the argon cooling loop for the FHC, sufficient time (>20 hr) will exist for the assemblies to be transferred to the EVST before the cladding reaches the 1,500°F limit.

The argon cooling gas flow rate through the spent fuel assemblies while being handled by the gas-cooling grapple will be sufficient to maintain the maximum steady-state cladding temperature of a 15-kW fuel assembly below 600°F. In the event of loss of argon cooling gas through a bare assembly, it will be possible for cladding temperatures to exceed 1,500°F and release fission products to the cell. If the main cell cooling system is working, the cell will be at negative pressure and no release to the RSB will occur. If main cell cooling is not operating, fission products will enter the RSB; however, doses to operating personnel will be below 10 CFR 20 limits and will allow manual action to return the assembly to a sodium-filled pot.

9.1.5 Overhead Heavy Load Handling System

The applicants' response to Question CS 410.2 and CS 410.3 (letter from J. Longenecker, DOE, to P. Check, NRC, dated June 14, 1982) provided information on the overhead crane systems of the reactor containment and reactor service buildings, as summarized below:

(1) Each crane will have two lifting systems--a main and auxiliary.

(2) Each main system will be designed for seismic Category I conditions, will be single failure proof, and will meet the criteria of NUREG-0554.

(3) Each auxiliary system will be designed for seismic Category I conditions, but not be single failure proof. Accordingly, the design of components accessed by these cranes will consider the effect of dropped loads from the auxiliary system and inadvertently lowered loads from the main system. The CRBR approach to the control of heavy loads will be consistent with NUREG-0612.

(4) Heavy loads will be moved by the main lift system, and when the reactor is operating the polar crane in the reactor containment building will have interlocks to prevent movement of the load over the reactor area. All components accessed by the cranes will be analyzed for the effect of dropped loads, and the design will be such that no dropped load will affect safe shutdown and decay heat removal or cause radiation releases in excess of allowable limits.

Summary and Conclusions

On the basis of its review of the fuel storage and handling system as described above, the staff concludes that the CRBR fuel storage and handling system is to be designed to adequately and safely store and handle fuel in a fashion to minimize the risk of damage to the assemblies from fuel-handling errors, to minimize the potential for fuel-loading errors, to prevent inadvertent criticality, to limit the spread of radioactive material and exposure of operating personnel to radiation, and to provide adequate cooling to spent fuel assemblies. In addition, the proposed design meets all applicable requirements of ANS Stds. 57.1, 57.2, and 57.3; SRP Sections 9.1.1 through 9.1.5; and RG 1.13. The applicants

have also indicated that the design will meet the requirements of NUREG-0554 and NUREG-0612.

The offnormal design-basis events considered in the analysis are worst-case or bounding events. The consequences of the events analyzed fall within acceptable guidelines. The fuel cladding temperature limits chosen for this system are consistent with those cladding temperature limits applied to the fuel at full power. The fuel storage and handling equipment and assembly movement patterns for CRBR are similar to those for the fast flux test facility (FFTF). The experience with the FFTF fuel storage and handling systems has been favorable in that they have operated successfully in a sodium environment, proved capable of cooling assemblies as designed, correctly identified and placed assemblies where desired, caught misidentified assemblies before placing them into the core, verified that the discriminator post concept on the bottom of all core assemblies works, limited the radiation exposure of operating personnel to acceptable values, and contained the spread of radioactive material and inert gas. This experience will help CRBR in verifying their design methods and provides confidence that the CRBR equipment will meet its design requirements.

Specifically, the design will accomplish the following:

(1) Provide system and machine designs that will maintain cladding temperatures below the limits specified for normal, anticipated, unlikely, and extremely unlikely events in all cases except for bare assemblies in the fuel-handling cell. For this case the assembly can be returned manually to a coolable position without the operator exceeding 10 CFR 20 dose guidelines.

(2) Provide system and machine designs that will maintain dose rates to operating personnel within 10 CFR 20 guidelines by providing adequate shielding and independent and redundant seals on the fuel-handling equipment.

(3) Provide three independent redundant means of removing decay heat from the spent fuel storage tank ex-vessel storage tank (EVST). Each of the three heat removal paths will be capable of handling the entire decay heat load of the EVST, and one of the paths will require no power or forced circulation to remove the heat.

(4) Provide spent fuel cooling system designs that can accommodate a leak in any of the three systems without causing a loss of coolant inventory sufficient to stop cooling in the remaining cooling loops.

(5) Prevent inadvertent criticality by maintaining adequate spacing between assemblies.

(6) Provide features to help prevent fuel-loading errors as described below:

    (a) The in-vessel transfer machine (IVTM) will have interlocks that will prevent releasing an assembly unless it is fully seated. This feature, in conjunction with the core assembly discriminator posts (described in Section 4.4) which prevent proper seating unless the

assembly is in its proper orificing zone, will protect against core-loading errors.

(b) As a backup to the above, the IVTM will have the capability to individually identify each assembly before insertion or after removal from the core and will have a position indication system to verify an assembly is properly seated.

(c) Monitoring of the reactor neutron count rate will be performed during refueling; thus, the capability of detecting gross loading errors will be ensured.

(7) Provide interlocks, grapple designs and fuel-handling mechanism designs that will minimize the potential for dropping assemblies or otherwise damaging them by inadvertent machine motion, loss of power, or seismic events. Interlocks will be provided so that a single failure will not cause a loss of the desired protection.

(8) Provide chemical and radiochemical limits, monitoring frequency, and criteria for cold trap replacement to ensure the ex-vessel storage tank sodium purity in the spent fuel storage pool.

(9) Provide sufficient instrumentaion to monitor the location and cooling status of all assemblies being handled.

The staff has evaluated two additional areas, and they are summarized below.

(1) The staff initiated a review of the ability of the fuel storage and handling system to withstand a 2-hour station blackout. Areas requiring cooling under this condition are those where the ex-vessel storage tank (EVST), ex-vessel transfer machine (EVTM), and fuel-handling cell (FHC) are located. Since the EVST and EVTM will have the ability to remove their heat by natural convection, a station blackout will not cause the temperature of fuel located in these areas to exceed the limits of Table 9.1. For the FHC, assemblies can be either in a sodium-filled core component pot (CCP) or bare. For assemblies in a sodium-filled CCP, a station blackout will not cause fuel cladding temperatures to exceed 900°F. For a 6-kW bare assembly (normal decay heat of assemblies in the FHC), fuel temperatures will exceed 1500°F if the station blackout lasts more than approximately 30 min. This will cause a fission gas release to the cell, but calculated doses to the cell operators will not exceed 10 CFR 20 limits and will allow manual movement of the bare assembly to a sodium-filled CCP. In addition, offsite dose limits will not be exceeded. This also will be the case for a 15-kW bare assembly. On the basis of the above considerations, the staff concludes that the fuel storage and handling system design can acceptably accommodate a station blackout. This position is further summarized in Appendix B of this SER.

(2) In evaluating the spent fuel heat removal capability of the EVST, the need for redundancy, diversity, and independence in the EVST heat removal systems equivalent to that provided for reactor residual heat removal was assessed. The staff considered that since the consequences of a loss of spent fuel cooling in the EVST could be similar to those of a loss of

reactor residual heat removal (i.e, the potential for fuel failure, motion, and recriticality), the design requirements and design of the EVST heat removal systems should provide a degree of protection against a loss of cooling equivalent to that provided for the reactor vessel. (For the loss of reactor residual heat removal, it would take an initial failure plus three additional single failures of active safety components or three separate leaks in the system.)

As stated previously in this evaluation, cooling of the EVST is to be accomplished by three independent and redundant cooling loops. One of these loops (the natural circulation loop) is to be different from the two other loops. The two loops that require electrical power will be supplied from separate Class 1E power sources. In addition, a guard vessel will be provided around the EVST and siphon breakers within the EVST to prevent a loss of EVST sodium inventory. Given this arrangement it would take an initiating event plus two single failures of any active or passive component in the system to cause a loss of EVST heat removal. In addition, flexibility will be included in the design to provide cross-connections among the EVST cooling loops or to the DHRS to establish, in the event of a loss of portions of the normal heat removal paths, an alternate heat removal path.

The consequences of a loss of all three EVST cooling loops were calculated by the applicants as a comparison with the consequences of a loss of reactor residual heat removal. The results of the calculations indicate that even if all EVST cooling loops are lost, the EVST sodium temperature will take at least 20 hours to reach its limiting temperature (1,500°F) where EVST structural integrity cannot be ensured, as compared with ~4 hours to reach a limiting temperature in the primary heat transport system for a loss of cooling to the reactor vessel. The applicants indicate that this 20-hour period will provide sufficient time to repair any conceivable failure, exclusive of a sodium or NaK leak. Additionally, the radiation level associated with EVST sodium will not preclude access to any portion of the cooling system for repairs. This would then allow repair and resumption of cooling in any loop not out of service because of a leak, or the establishment of an alternate cooling path by means of cross-connections, thus in effect establishing a fourth cooling loop.

Further considerations also tend to reduce the likelihood of having this worst-case situation occur. These are:

(1) For EVST decay heat loads less than 1,800 kW, the time for EVST heatup to 1500°F will be longer. The only time the EVST will have decay heat loads at or near 1,800 kW will be during refueling and of short duration, with the more probable long-term decay heat load being in the range of 400 kW to 800 kW.

(2) To further help ensure that the full complement of EVST decay heat removal systems will be operable during the periods of peak EVST decay heat load (i.e., refueling), the applicants have committed to include in the Technical Specifications a requirement that refueling, which will add significant decay heat to the EVST, will not begin unless all three EVST heat removal paths are operable, and that if

any one of these becomes inoperable during the refueling process, refueling operations will be terminated.

In consideration of the above, it is the staff's position that the design will provide adequate protection against a loss of all EVST heat removal, that this protection will be equivalent to that provided for the reactor, and that the redundancy, diversity, and independence of the proposed EVST heat removal systems are acceptable.

At the staff's request, the applicants have committed to treat the EVST guard vessel and siphon breaks as engineered safety features (ESFs) since their function is essential to the prevention of total loss of EVST coolant inventory. Treating these as ESFs is intended to ensure quality--that preservice and inservice inspection equivalent to that for the reactor guard vessel will be applied to these components. It is also the staff's position that as part of final design, procedures be developed for repair of EVST active components and for establishing alternate heat removal paths from the EVST, to ensure quick implementation of these options, if necessary. In addition, the capability to remove decay heat from the EVST via the natural circulation loop should be demonstrated during initial startup testing.

On the basis of the evaluation summarized above, the staff concludes that the proposed fuel storage and handling system design criteria and preliminary design meet the requirements of PDC 53, 54, and 55 and are acceptable for issuing a construction permit.

9.2 Nuclear Island General Purpose Maintenance System

The nuclear island general purpose maintenance system will provide equipment and facilities to support maintenance activities for visual inspection, handling, cleaning, decontaminating, disassembly, and repair of sodium-wetted and/or radioactive components. It will be composed of the following facilities and equipment:

(1) Primary Sodium Removal and Decontamination System

The primary sodium removal and decontamination system (PSR&D) system will be located in the reactor containment building (RCB) and will consist of the large component cleaning vessel (LCCV) and associated process equipment. The system will be used for cleaning components from the reactor and the primary heat transport system, although the highly radioactive removable core components (fuel assemblies, blanket assemblies, control assemblies, and shield assemblies) are not intended to be cleaned in the LCCV. The system will remove surface liquid metal from components using the water vapor-nitrogen (WVN) process and remove subsurface radioactive contamination using acid etching. The system will provide for control and containment of liquid metal-water reactions and reaction products. Liquid and gas effluents from these processes, which will contain radioactive materials from the liquid metal and subsurface contamination, are to be directed to the appropriate radwaste system for controlled disposal.

(2) Decontamination Facility

This facility will consist of four rooms in the radioactive waste area of the reactor service building (RSB). These rooms will be operated at negative pressure and will be used for removing trace amounts of sodium and low level radioactive contamination from small components. One room will contain the small component autoclave, which will consist of a vessel into which small components requiring liquid metal removal may be placed and cleaned using WVN process equipment. The autoclave will be similar to the PSR&D system although it will be smaller and will not provide the capability for removal of subsurface contamination. It is intended that only low level radioactive components will be cleaned in the autoclave. Liquid and gas effluents from this process, which may contain radioactive materials from the primary sodium, are to be directed to the appropriate radwaste systems for controlled disposal.

(3) Regulated Shop

The regulated shop will be a controlled section of the maintenance shop for tools used in radioactive or contaminated equipment. Waste from this shop will be directed to the appropriate radwaste system for controlled disposal.

(4) Intermediate Sodium Removal System

The intermediate sodium removal (ISR) system is a facility that will be built in the steam generator building after CRBR startup and will be used for removing sodium from the modular steam generator, intermediate system pumps, and other intermediate heat transport system components. This equipment will not contain decontamination capability, as the only source of radioactivity in the intermediate heat transport system will be from tritium.

(5) Handling Containers

Handling containers are to be provided to prevent the spread of radioactivity during removal and transport of radioactive components and to contain solidified surface liquid metal films within an inert atmosphere during removal and transport of sodium-wetted components. The containers will include plastic film bags within which components may be transported or transferred between controlled and inerted atmospheres and portable enclosures that will provide controlled air atmosphere working spaces. General-purpose shielded casks will be provided to reduce direct radiation when radioactive or radioactive sodium-wetted components will be handled.

(6) Handling Equipment

Handling equipment is to be provided to move heavy or bulky components. The major item in this group is the large component transporter (LCT), which is a platform designed to move on the EVTM rails. The LCT will be provided with seismic restraints to maintain it on the rails in case of a seismic event.

(7) Maintenance and Inspection Equipment

This equipment will consist of support stands, portable equipment for cutting, welding, and weld inspection of sodium piping and remote viewing equipment for inservice inspection of components in inerted and radioactive cells.

Scope of Review

PSAR Section 9.2 was reviewed along with PSAR Section 15.7.3.7. Facilities for the maintenance of nonsodium-wetted and/or nonradioactive equipment were not addressed in this evaluation.

Evaluation Criteria

The CRBR general purpose maintenance systems were reviewed for

(1) control of radioactive material (i.e., are the locations and practices employed consistent with minimizing the spread of radioactive material?)

(2) impact on plant and personnel safety (i.e., do the techniques, practices and plans detract from overall plant and personnel safety?)

No SRP sections were found applicable; however, CRBR PDC 4, "Protection Against Sodium Reactions," and RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Plants Will Be As Low As Reasonably Achievable," were used as criteria for the review.

Evaluation

The staff's evaluation of the nuclear island general purpose maintenance system is as follows:

(1) The proposed design adequately provides for containing and controlling the spread of radioactive material by the use of controlled, negative pressure areas for maintenance on radioactive equipment and by directing the waste from these areas to a controlled collection system. The cleaning process chosen (WVN process) has been used successfully in other facilities and has been shown to be controllable. The commitment in PSAR Chapter 12 to follow the guidance of RG 8.8 in the plant final design should enhance personnel safety while they are performing maintenance. None of the general purpose maintenance facilities or equipment is seismically qualified. The staff finds this acceptable as the applicants have committed (letter from J. R. Longnecker to P. S. Check, dated Dec. 20, 1982) to design the facilities so that (a) any failure of the system will not impact public safety and (b) failure of the system or systems will not damage safety-related components.

(2) The applicants have committed in Appendix H of the PSAR to provide adequate shielding of plant facilities so as not to preclude, in the event of an emergency, operator access to other plant safety equipment when it is in use. This will include a review of the maintenance system for adequate shielding.

(3) Processing of components by the WVN process will result in hydrogen generation. The applicants indicate that the control system will be designed to limit the amount of hydrogen gas to 4 v% by controlling the amount and rate of water vapor addition, which will prevent a hydrogen explosion by keeping the hydrogen concentration less than the lower flammability limit if an advertent release of hydrogen to the process cell should occur. Furthermore, good ventilation will be provided to the cell to prevent localized concentration of hydrogen gas. Also, all electrical equipment is to be designed and grounded to minimize any sources of ignition.

(4) The design-basis event for the systems has been defined as a series of events leading to an uncontrolled sodium-water reaction in the large component cleaning vessel (LCCV). The resultant pressure in the LCCV is calculated to be 89 psig which is less than the rupture pressure of the components in the system. However, it is further assumed that the gas will be released adiabatically into the LCCV cell. The resultant hydrogen concentration is calculated to be 2.5%, significantly lower than the 4% explosive limit of hydrogen in air.

(5) A concern regarding the propensity of caustic-induced stress-corrosion cracking of the PSR&D structural material, reactor components, and primary and intermediate coolant boundary components was investigated. An evaluation of stress corrosion cracking (Swandby, November 12, 1962) of austenetic stainless steel in sodium hydroxide shows that there will be no problem at the temperatures reached in the WVN process. The planned process temperatures (160° - 190°F) are well below the stress cracking boundary. A technical specification, limiting the WVN process to less than 190°F in the PSR&D will be incorporated at the operating license stage.

The staff concludes that the proposed design meets the evaluation criteria and that there are no outstanding issues relative to the nuclear island general purpose maintenance system that preclude issuing a construction permit.

9.3 Auxiliary Liquid Metal System

The auxiliary liquid metal system (ALMS) will consist of five subsystems designed to provide supporting functions in connection with servicing and operating plant systems that will use sodium as a coolant. These subsystems and the functions they will perform are summarized in Table 9.2.

Since the subsystems that make up the ALMS will provide various functions in several different parts of the plant, they are discussed separately in the evaluation presented below.

The principal common element of the five systems is that all are designed for sodium service. With the exception of the carbon steel vessels used for storage of primary sodium and EVST NaK (sodium-potassium), all piping, components, and vessels will be made of austenitic stainless steel.

In reviewing Section 9.3 of the PSAR, the staff assessed the proposed design of the ALMS subsystems from several standpoints. These included (1) the suitability

of the preliminary design in light of the considerable body of experience in the design and operation of similar systems, (2) the acceptability of the proposed safety (quality group) and seismic classifications of various portions of the subsystem, (3) comparability with those service conditions that can be anticipated over plant life, and (4) adherence to the applicable CRBR Principal Design Criteria set forth in Section 3.1 of this SER. In connection with its review of Section 9.3, the staff also reviewed related sections of PSAR Chapters 3, 5, 7, 9 and 15. This was done to ensure that the information in those sections was compatible with that in Section 9.3 and that the findings of other staff reviewers were consistent with those reached herein.

9.3.1 Sodium and NaK Receiving System

The sodium and NaK receiving system will be designed to receive and melt fresh solidified sodium, which will be delivered to the CRBRP site in tank cars or drums, and to transfer the sodium to primary and intermediate storage vessels.

Additionally, the system will receive and transfer to storage vessels all NaK used in the plant. The system will be located out in the open, away from any of the site structures.

This system will consist of a tank car oil heating station for meltout of sodium tank cars, a clam shell heater for melting drums of sodium, transfer piping and valves, and filters for cleanup of fresh sodium or NaK. The piping and filters for the NaK will be located on a portable rig and will be independent of the sodium system. Sodium and NaK will be transferred to system storage vessels by gravity flow.

The piping and components that will come into contact with sodium or NaK will all be fabricated from type 304 stainless steel. The design temperature of 450°F for those components that will handle sodium is appropriate since the temperature at which sodium will be transferred will be no more than about 400°F. The temperature of the piping for the portable unit that will transfer NaK will be 150°F. This temperature is appropriate since NaK will be transferred at ambient temperature. Components for both systems will be designed for 20 psig. Since transfer will be by gravity flow, there will be no pumps that can cause excessive pressurization. Although argon gas will be used to purge lines and to pressurize tank cars for sodium transfer, the likelihood of excessive pressurization is remote. However, leak detectors will be provided to monitor for leakage.

The applicants have designated the system as nonsafety class. Since the system will be used primarily during initial fill before any nuclear operations and will be disconnected by removal of spool-pieces during essentially all subsequent plant operations, the staff concurs with this classification.

The staff has considered the potential for a large sodium fire during melt and transfer of tank car loads of sodium into the plant during intial fill. Although the potential for tank car rupture or uncontrolled leakage from piping connections during actual transfer is very small, the quantity of molten sodium (80,000 lb) would be quite large, and a chemical hazard from inhalation of sodium oxide fumes by persons off site or damage to equipment on the plant site could result. The sodium would not present a radiological hazard during initial fill.

The applicants have agreed to take any necessary steps (emergency planning measures) with local officials before receiving sodium on site. In addition, the applicants have stated that precautions will be taken during liquid initial loading to limit deleterious effects to plant components. These matters can be resolved after a detailed analysis has been performed and presented in the FSAR. Sufficient flexibility will exist to provide additional protection, if necessary, to prevent the spread of fumes in the event of a fire in the unloading station. For example, an industrial building enclosing the unloading station would confine the bulk of any fumes generated by a fire, and if warranted, specific fire protection measures could be taken. Depending on the outcome of the analysis, it may be necessary to stipulate in the Technical Specifications that the presence of fresh sodium in tank cars be excluded from the site during plant operations, and that sodium and NaK loading operations be done only under certain conditions to preclude any significant potential for degrading the performance of exposed safety-related equipment (e.g., diesel air intakes, PACCs, switch gear, and air blast heat exchangers).

The staff notes that the applicants have stated that the system will provide the capability to remove sodium and NaK from plant systems for offsite disposal. It is clear that this capability would be used seldom, if ever. The use of the system for liquid metal disposal has not been described in the PSAR. However, if such use is contemplated, a full description of the conditions and the protective measures to be taken must be presented in the FSAR. Certain Technical Specification requirements may be needed in that regard.

With the exception of those items noted above, which can be resolved at the OL stage of review, the staff concludes that the preliminary design and engineering criteria for the sodium and NaK receiving system can be used with reasonable assurance to develop a final system design that will be satisfactory from the standpoint of safety.

9.3.2 Primary Sodium Storage and Processing System

The primary sodium storage and processing system will provide primary sodium purification (cold trapping) and storage for the sodium used in the reactor vessel, the PHTS loops, and the EVST; mitigate the change in reactor vessel sodium level; and accommodate thermal expansion and contraction of primary sodium. This system, working in conjunction with the ex-vessel storage (EVS) sodium processing system, also will provide a means of removing reactor decay heat in the event of a loss of all the steam generators.

With the exception of the two ex-containment primary sodium storage vessels and associated piping, all components of the system will be located within the reactor containment building. Except when sodium transfer operations will be taking place in connection with system fill or storage of sodium for system maintenance, the in-containment and ex-containment storage vessels will be isolated from other portions of the system.

During normal plant operation sodium will overflow continuously from the reactor vessel and drain by gravity into a primary sodium overflow vessel.

Elevations for the overflow nozzle on the reactor vessel and associated piping will be such that during all plant operating conditions, the sodium level in the reactor vessel will be maintained below the reactor head reflector plate. Sodium in the overflow vessels will be returned by two full-capacity electromagnetic pumps to the upper part of the reactor vessel.

A normally bypassed overflow heat exchanger (NaK cooled) will be located down stream from the pumps. This heat exchanger will be put into service only if a backup means for decay heat removal is needed (DHRS mode). Two NaK-cooled cold traps will be located in a parallel branch of the return line. Either or both of these cold traps can be bypassed, depending on plant conditions.

All components of the system, including the sodium storage tanks, will be designed to seismic Category I and to Safety Class 1 or 3 depending on the design requirements imposed by the anticipated service conditions. Review of the design conditions set forth in Table 9.3-7 of the PSAR shows those that can be anticipated over plant life. Transient temperature conditions associated with the DHRS mode of operation are discussed in detail in Section 5.6.4 of the SER. The staff has found from its review of PSAR Section 9.3 that the quality group (safety classifications) seismic classifications and design conditions for all components of the primary sodium storage and processing system are satisfactory for development of the final design and are fully consistent with the conclusions drawn in Section 3.2 of the SER.

9.3.2.1 Overflow Circuit

With respect to operation of the system in its normal mode, it appears that the design will be satisfactory from the standpoint of controlling reactor sodium level. The overflow vessel will be sized (35,000 gal) to accommodate sodium expansion and contraction over the full operating temperature range of the reactor. Although rapid cooldown, such as would accompany reactor scram, would cause overflow to temporarily stop, the sodium volume in the overflow tank would permit continuous makeup and restoration of normal sodium level in a short time. Thus, the system will be adequate for level control under all operating and shutdown conditions.

The cold traps will be designed to precipitate and filter sodium oxides and hydrides so that the oxygen and hydrogen content of the primary sodium can be maintained at 2.0 and 0.2 ppm, respectively. It is anticipated that one of these cold traps will be in operation continuously during plant operation. They can also be aligned to purify sodium in the overflow and in containment storage tanks during extended periods of plant shutdown.

In response to questions raised by the staff about cold-trap life and the possible need for replacement, the applicants have stated that the life of a cold trap is estimated at 13 years. Provisions will be made for removal, storage, and replacement of cold traps as necessary. Since the preliminary design of the cold traps and their method of operation are based on proven technology, the staff concludes that an acceptable basis exists for development of the final design.

As noted above, the system will contain, as part of the DHRS, an overflow heat exchanger that normally will be bypassed. This DHRS will be actuated manually

from the control room only if the backup mode decay heat removal capability is needed. As stated in the evaluation in Section 5.6.4, the possibility that this mode would ever have to be used is so remote that it is regarded as a faulted condition for purposes of design. The staff has found this acceptable. It has concluded that the system can satisfactorily operate to remove decay heat throughout the overtemperature transient that would occur (see Section 5). It has also been determined that the applicants' plan for development of preoperational and startup and inservice testing programs are satisfactory (see Sections 3.9 and 14.1, respectively).

In its review of these sections, the staff has noted that the transient temperatures that could be imposed on the piping and components of the overflow circuit could at some point be several hundred degrees higher than the stated design temperatures. That in itself is not unacceptable, since the mechanical effects of the design thermal transient can be evaluated with considerable confidence during the detailed design analysis. However, the system cannot be tested at the full temperature extremes and gradients involved. Consequently, care should be taken in developing the final testing programs to ensure that sufficient data are acquired to extrapolate pipe motion (from expansion and or contraction) to the extreme conditions of DHRS operation. Since the piping and components involved will not be accessible in the inerted atmosphere and because of the high Na-24 radiation levels attending operation at any significant power level, some temporary instrumentation might be needed to measure piping movement to acquire data for extrapolation. These data may be needed particularly to confirm that unexpected interferences will not occur and that essential valves and other vital components will remain operable to the extent necessary in the DHRS mode.

As previously noted, the overflow heat exchanger normally will be bypassed. To what extent the sodium trace heating system must be depended on to maintain sodium in the heat exchanger in a molten condition has not been stated. The staff notes, however, that instrumentation that will measure temperature conditions throughout the circuit, including this heat exchanger, will be of high quality and have sufficient redundancy and diversity to ensure reliable monitoring of sodium temperatures and, correspondingly, decrease the need for any corrective actions to prevent freezing and loss of availability of the overflow heat exchanger. The trace heating system and associated instrumentation are discussed in Section 9.4 of the SER.

On the basis of its review of PSAR Table 9.3-4, the staff concludes that sufficient numbers and types of sodium leak detectors will be located at or in the vicinity of these ALMS components inside the reactor containment building to ensure that timely action can be taken in the event of sodium leakage. The staff's evaluation of the leak detection system and its use in connection with safety-related actions regarding system operability and sodium fires is presented in SER Section 5.7.

9.3.2.2 Sodium Storage Operations

After intitial fill of sodium systems, the two 50,000-gal ex-containment primary sodium storage tanks are expected to remain empty and isolated from reactor systems and the spent fuel storage system by valves that will be locked closed. These tanks will be located in vaults in the steam generator building.

The vaults normally will contain an air atmosphere, but they can be provided with an inert atmosphere if at some point in the future, radioactive sodium must be transferred to the ex-containment storage tanks in connection with major maintenance activities.

The proposed provisons for isolation at the containment boundary meet the appropriate containment isolation criteria of PDC 45, 46, 47, and 48, as discussed in Section 7.3 of the SER, and are therefore acceptable. Since primary sodium becomes highly radioactive with Na-24 activity during reactor operations, inadvertent opening of isolation valves could present a severe radiation hazard to personnel. For this reason, at the OL stage of review, the staff will review the precise method for locking the isolation valves closed to ensure that inadvertent opening cannot occur.

The in-containment primary sodium storage vessel will be isolated when the reactor system is not actually in use for systems fill or sodium transfer and storage in connection with maintenance. Drain points for removal of sodium from primary loops, at the primary pumps and IHXs, will be protected by at least two locked-closed valves. Accidental drainage of a loop, therefore, would be an extremely remote possibility, but, in any case, not more than one loop would be affected. The reactor vessel cannot be accidentally drained. Thus there can be no interruption of reactor heat removal in connection with presence and use of storage tanks.

The staff considered the possibility that carbon could be transported into the plant during initial fill or in subsequent maintenance operations via oxygen-catalyzed removal of carbon from carbon-steel piping or vessels. Such transport, if there were a large enough source of dissolved carbon, could permit carbon to diffuse into and alter the properties of austenitic steel at temperatures above 800°F and possibly foul heat exchangers at lower sodium temperatures. This concern has been removed by confirmation that all piping in the sodium and NaK receiving system will be stainless steel and that temperatures in the carbon steel storage tanks will not exceed approximately 400°F. There will be a margin of several hundred degrees from temperatures high enough to cause diffusion of carbon steel at any significant rate. Also, in this low temperature regime, the solubility of carbon in sodium will be extremely low. Consequently, carbon transport from these tanks is of no concern.

The staff concludes that the preliminary design of the primary sodium storage subsystem and its proposed utilization is satisfactory. These matters that are still outstanding can be resolved at the OL stage of review.

9.3.3 Ex-Vessel Storage Sodium Processing System

The primary function of the ex-vessel storage (EVS) processing system will be to remove decay heat from spent fuel stored in the ex-vessel storage tank (EVST). Details concerning other aspects of fuel handling and storage other than decay heat removal are presented in Section 9.1 of this SER.

In addition to decay heat removal, this system will provide purification for the sodium used in the spent fuel cooling. During operation of the system, sodium will be pumped from the EVST, through a cold-trap, sodium-to-NaK heat exchanger, and back to the EVST. Oxygen and hydrogen will be removed from the

sodium at the cold trap. The spent fuel decay heat will be picked up by NaK circulating through the sodium/NaK heat exchanger and will be dumped to air through air blast heat exchangers. This system will have two forced-circulation sodium/NaK loops and a backup natural circulation sodium/NaK loop. The NaK-to-air heat exchanger in the backup loop will function by natural draft and, thus, will not require a source of electric power to operate blowers.

As stated in the discussion of the primary sodium storage and processing system (Section 9.3.2), the overflow heat exchanger, which will be normally by-passed, will provide backup means for removal of reactor decay heat. Actuation of this direct heat removal srevice (DHRS) mode involves, in part, tying the NaK cooling circuit of the overflow heat exchanger into the NaK cooling circuit of one of the two EVST air blast heat exchangers. Thus the EVS processing system not only will provide the ultimate heat sink for spent fuel in the EVST but also for the DHRS.

All components of the EVS sodium processing system will be located in the reactor service building.

With the exception of the NaK storage vessel, all components of the EVS sodium processing system, including the EVST, will be designed to meet seismic Category I requirements and have been designated as Safety Class 2 or 3 depending on service conditions. The design temperatures and pressures, set forth in PSAR Table 9.3-7, are consistent with those that can be expected during plant operations. The NaK storage vessel has been designated nonsafety class, but the applicants state that its design and construction will exceed that required by the ASME Code Section VIII. The nonsafety class designation is acceptable since the storage tank will be used infrequently and will be normally isolated from the active portions of the system. The staff concludes that the seismic and safety classifications prepared and the conditons outlined are acceptable and consistent with the conclusions drawn in Section 3.2 of this SER. Therefore, they can be used with confidence in development of the final design.

Each of the EVST sodium cooling loops will be housed in a separate shielded and nitrogen-inerted cell. This separation will limit the chance that a sodium leak could affect more than one of the three full-capacity loops. Likewise, the NaK-to-air heat exchangers and associated piping will be physically separated so that a problem with one NaK loop would be unlikely to affect the operability of the other.

Leakage from the EVST would not interrupt spent fuel cooling because of the limited volume between the EVST and its guard tank. That volume will be small enough to preclude the sodium level from dropping low enough to uncover the suction lines. Pumpout or siphoning of sodium to the point where cooling will be stopped will be prevented by judicious selection of suction line elevations in the EVST and provision of siphon breaks in the return lines. Leaks in the natural circulation loop will not affect the sodium level in the EVST because the piping and components of this loop will be at an elevation higher than the sodium level in the EVST.

Overall, the design will provide ample physical separation to protect against loss of cooling by common cause events. It will provide threefold redundancy

through the provision of three full-capacity loops and diversity in that a natural circulation loop will backup the two forced-circulation loops.

During normal EVS operation, only one of the two forced-circulation loops will be in service. The sodium pump will take suction from the EVST, will pump the sodium through the tube side of the sodium/NaK heat exchanger and back to the EVST. A portion of the sodium will be pumped through the cold trap for purification. NaK will be pumped through the shell side of the sodium/NaK heat exchanger, picking up the spent fuel decay heat through the air blast heat exchanger (ABHX), and discharging the heat to air and then back to the pump. The NaK loop will be equipped with a diffusion cold trap to control NaK purity and an expansion vessel to accommodate volumetric changes resulting from changes in NaK temperature.

In the extremely unlikely event of loss of both normal cooling circuits, the dampers on the natural-draft heat exchanger will be manually opened to initiate natural draft air flow. This will induce NaK flow through the tubes of the natural-draft heat exchanger and the shell of the backup sodium cooler. This in turn will induce sodium flow from the EVST through the tubes of the backup sodium cooler and back to the EVST.

The cold trap that purifies EVS sodium will be designed on the basis of established technology. The cold trap will be nitrogen cooled in contrast to the primary cold traps that will be NaK cooled. The expected design life is not stated, but it is clear that provision must be made for cold-trap replacement. The cold trap may become slightly radioactive if fuel with perforated cladding is stored in the EVST and it will undoubtedly retain some activation products from primary sodium, which accompanies fuel when it is transferred out of the reactor. These aspects need not be described at this stage of review, but the design provisions and precautionary measures associated with cold-trap replacement must be fully described in the FSAR. The staff will do a detailed review concerning cold-trap replacement during the OL stage of review.

Air-cooled diffusion cold traps will be used to purify NaK in the cooling circuits that transport heat to the NaK-to-air heat exchangers. Cold traps of this type are in use at FFTF and they have been used successfully at other facilities.

Because of the expectation that oxygen levels will be very low in these circuits, no provision has been made to monitor the oxygen content of the NaK or to ascertain what capacity remains in the cold traps. It is presumed that the circuits will be exceptionally clean when they are filled, that the initial NaK fill will be very pure, and that the circuits will remain essentially leaktight throughout service life. The applicants' position concerning NaK purity and their position that there is no need to monitor are reasonable. However, the circuits will provide a safety-related heat removal function and it is the staff's position that further consideration should be given to providing that capability. This can be resolved at the OL stage of review.

Either of the EVS sodium processing system's two normally forced cooling circuits will provide the capability to remove 1,800 kW of heat while maintaining

n EVST exit sodium temperature of approximately 510°F. The third (backup) ooling circuit will provide the capability to remove 1,800 kW while maintaining an exit temperature of ~ 600°F. The 1,800-kW heat load is based on the pent fuel heat release resulting from a complete core unloading occurring ith two annual loads of spent fuel assemblies already in the EVST.

he DHRS is sized to limit the average bulk primary sodium temperature to approximately 1,140°F when the DHRS is initiated one-half hour after reactor shutdown. This imposes peak heat load on the EVS cooling circuits of about 11 MW. Since the two NaK airblast heat exchangers will have a combined capacity of about 11½ MW, they would not be sufficient to remove heat from the EVST and reactor simultaneously; therefore, the applicants have committed to switch EVST cooling to the natural circulation loop whenever the DHRS is in operation.

From its review of the PSAR and associated drawings, the staff is satisfied that the design will provide Class 1E power supplies to all vital equipment of the EVS and that sufficient safety-grade instrumentation and controls are provided to monitor and control essential EVS processes. These aspects are discussed further in Sections 7 and 8 of the SER.

With adequate resolution of those matters that are to be resolved as part of final design, the staff concludes that the preliminary design of the EVS sodium processing system and its proposed utilization under normal and emergency conditions is acceptable for issuance of a CP.

9.3.4 Primary Cold-Trap NaK Cooling System

The primary cold-trap NaK cooling system will provide the capability for cooling the primary system cold traps and for transferring the heat to Dowtherm J. The system will include facilities for storage, filling, and draining of its NaK. All components of this system will be located in normally inerted cells in the reactor containment building.

During normal plant operation, when one primary sodium cold trap will be in use, NaK will be pumped by an electromagnetic pump from the NaK storage vessel through the jacket of the primary sodium cold trap, through the NaK cooler and back to the storage tank to Dowtherm J. The piping to the cold-trap jackets will be arranged in parallel to permit the operation of one or both cold traps. The storage tank will provide an inventory of NaK as well as provide for expansion and contraction of the NaK. A diffusion cold trap will maintain NaK purity. NaK pressure will be maintained at a greater value than the primary sodium to prevent radioactive sodium transfer into NaK in the event of an internal cold-trap leak.

Sufficient instrumentation will be provided to monitor temperatures, pressures, and flow rates at key locations in the circuit so that process variables can be controlled. Dowtherm J-to-NaK or NaK-to-sodium leakage will be monitored through measurement of system pressure and of the level in the NaK storage vessel. The use of Dowtherm J on the secondary side of the NaK cooler will ensure that tube leakage will not lead to a chemical reaction caused by mixing of incompatible cooling fluids. Dowtherm J is compatible with NaK at the temperatures involved.

The primary cold-trap NaK cooling system is designated by the applicants as a nonsafety class system. The PSAR states that although the system is required to meet the requirements of ASME Code Section VIII or ANSI Std. 831.1, it will be designed and constructed to the more stringent requirements of ASME Code Section III, Class 3.

This classification is acceptable, based, in part, on the fact that cold trapping can be interrupted under essentially all plant operating conditions without leading to any kind of safety problem. Experience in LMFBRs has shown that hours or days can usually pass without cold trapping, while still maintaining oxygen and hydrogen levels within acceptable limits. Sufficient monitoring, for example, through analysis of samples, will ascertain when plant shutdown might eventually be needed to prevent excessive corrosion resulting from high impurity levels at high temperatures or to prevent precipitation at shutdown temperatures. Ordinarily the cold traps would be isolated if cooling were interrupted. If for some remote reason they could not be isolated and ran sufficiently hot to redissolve the cold trap inventory of precipitated oxides and hydrides, the primary system could be maintained at sufficiently high temperatures to prevent precipitation in the reactor system, yet low enough to avoid corrosion problems while corrective measures were taken.

The staff concludes that the preliminary design, the proposed safety classification, and the envisioned operating conditions and methods are acceptable for development of the final design of the primary cold-trap NaK cooling system.

9.3.5 Intermediate Sodium Processing System

The intermediate sodium processing system will provide purification of the sodium in each of the three intermediate heat transport system (IHTS) loops. The system will not provide for storage of the IHTS sodium. This capability will be provided by the sodium dump tanks, which will be part of the steam generator system. The intermediate sodium processing system will provide the capability of transferring sodium into the loops from the dump tanks. This same piping network will allow for the filling of each dump tank with fresh sodium from tank cars or drums at the sodium receiving station. Sodium removal from the tanks into tank cars will be accomplished through the same fill lines. Additionally, the system working in conjunction with the primary heat transport system (PHTS) cold traps will limit the tritium content of the IHTS sodium. This system includes one electromagnetic pump and two cold traps in each loop together with interconnecting piping and valves. The cold traps will be air cooled.

All components of this system will be located in the steam generator building.

The system has been designated as seismic Category I and Safety Class 3. The staff concurs in these classifications, which are fully compatible with the design temperatures and pressures proposed and the need to minimize the potential for leakage or rupture in the air atmosphere surrounding the steam generator system. This will ensure that the design and construction will be of sufficiently high quality and seismic resistance that the risk of incapacitating the IHTS by a sodium leak is very small.

The loss of cold trap capability in any of the IHTS loops would not be critical to operations, but prolonged loss might ultimately force plant shutdown because of buildup of oxygen levels to an unacceptable extent in terms of corrosion or plugging by precipitates. The staff concurs with the applicants that the system is nonessential as far as safety function is concerned.

The applicants have not discussed any provisions for cold-trap replacement. It has not been stated what design life is predicted for the cold traps but it is clear that the oxide and hydride content might have to be controlled by continuous cold trapping because of very small steam generator leaks. Change out of the traps should not be difficult since the IHTS sodium will remain essentially nonradioactive throughout plant life. The staff will review the applicant's plans for replacement when final design information is presented in the FSAR.

The staff concludes that the proposed preliminary design, seismic and safety classifications, and utilization and service conditions provide an acceptable basis for development of the final design of the intermediate sodium processing system.

9.3.6 Auxiliary Liquid Metal System--Cold-Trap Technology

9.3.6.1 Introduction

The auxiliary liquid metal system (ALMS) will be made up of five subsystems. A detailed description of each subsystem is given in Section 9.3 of the PSAR. Two of the main functions of the ALMS will be to provide (1) primary sodium storage and processing and (2) intermediate sodium processing. Except for a few components, the pressure boundary material will be type 304 stainless steel (SS) to conform to ASME Code requirements. The sodium overflow line between the reactor and the overflow vessel will be made of type 316 SS. The primary sodium storage vessel, the EVST, NaK storage vessel, and the IHTS sodium cold traps will all be made of carbon steel.

The primary sodium storage and processing system will provide primary sodium purification by cold trapping, as well as providing storage for the sodium to be used in the reactor vessel, the PHTS, and the EVST to maintain sodium level and to accommodate sodium volume changes resulting from thermal differential. The cold traps will remove impurities by cooling the liquid sodium close to its melting point, thus causing most of its impurities (such as oxides, carbonates, and hydrides) to precipitate and to be filtered out by stainless steel wools inside the cold trap.

The intermediate sodium processing system will provide the capability to limit oxygen and hydrogen generation of IHTS sodium. The system, working in conjunction with the primary cold traps, will limit the tritium content of IHTS sodium. The system also will provide the capability to (1) fill the IHTS loops from the sodium dump tanks, (2) purify sodium in the dump tanks, and (3) transfer sodium from one dump tank to another.

9.3.6.2 Areas of Review and Acceptance Criteria

At the construction permit stage of review, the staff reviews the information in the applicants' PSAR on component design parameters for flow, temperature,

pressure, heat-removal capability, and impurity-removal capability of the cold traps in both the primary sodium storage and processing system and the intermediate sodium processing system, to ensure that the primary sodium and the intermediate sodium purity will be maintained. Furthermore, the instrumentation and process controls will be reviewed to ensure proper system operation by monitoring impurity removal and differential pressure across the cold traps.

The acceptance criteria of the ALMS are based on meeting similar or equivalent LWR General Design Criteria as that for the reactor coolant cleanup system. Specifically, the cold traps must be able to maintain acceptable reactor sodium purity and the intermediate sodium purity in normal operation and during anticipated operational occurrences, for example, reactor startup, refueling, and shutdown to ensure reactor coolant pressure boundary material integrity. The following criteria are used in reviewing the system:

(1) The system should be designed to maintain reactor shutdown and the intermediate sodium purity within the limits to ensure coolant pressure boundary material integrity.

(2) Redundant sodium processing systems should be provided to maintain sodium purification at normal operating temperature and flow conditions. This will permit processing of reactor sodium during periods of single active component failures or equipment downtime.

(3) The auxiliary liquid metal system should have the capability to permit processing of excess reactor sodium during startups, shutdowns, and hot standby conditions.

(4) The system should include provisions for monitoring temperature, pressure, oxygen and hydrogen content of the reactor sodium and other chemical and radiochemical limits to ensure the design limits of the cold traps are not exceeded.

9.3.6.3 Evaluation

The primary sodium processing system will consist of an overflow and makeup circuit to provide reactor coolant volume control. In addition, two liquid-cooled cold traps, arranged in parallel, are included in a bypass on the reactor makeup return line to provide sodium purification. Each trap will be rated at 60 gpm at normal system operating temperature, and at 80 gpm at 600°F (reactor hot standby temperature).. During normal plant operation, one trap will be in use and the other will be in standby. Although both traps can be operated, a single trap is sufficient to remove anticipated oxygen in leakage and maintain oxygen concentrations below 2 ppm.

The concentration of tritium in the primary sodium also will be maintained at low level by cold-trap operation. Operation of one cold trap, combined with the tritium diffusion through the intermediate heat exchanger (IHX) to the intermediate system will maintain the tritium content of the primary sodium at less than 3 μCi of tritium per gram of sodium. During shutdown for fuel handling, both cold traps may be operated to provide a maximum cleanup flow of 160 gpm. The total capacity will be designed to provide for removal of potential oxygen in leakage during fuel handling rapidly enough so that no additional

plant downtime will be required. The system also will provide direct heat-removal or decay heat-removal capability for CRBR in case of loss of all steam generators.

The intermediate sodium processing system will provide the capability to limit the oxygen and hydrogen concentrations of IHTS sodium to 2.0 ppm and 0.2 ppm, respectively. This system, working in conjunction with the primary cold traps, will limit the tritium content of IHTS sodium. Each of the three IHTS loops will be provided with a separate purification system, consisting of a pump and two cold traps. Both traps normally will be in operation, however, a single trap per IHTS loop will be sufficient to remove anticipated oxygen and hydrogen inleakage and to limit those impurities to a maximum of 2 and 0.2 ppm, respectively.

Sodium samples from the PHTS and the IHTS will be taken by the multipurpose sampler (MPS) for laboratory chemical analyses. The applicants have committed to a monthly chemical analysis schedule with the following exceptions:

(1) During periods of anticipated changes (initial fill, refueling, maintenance activities) in impurity concentrations, sampling frequency will be increased.

(2) When it has been established that the impurity concentration of a special element no longer will be changing after several years of plant operations, the frequency of analysis may be reduced.

Indication of oxygen and hydrogen concentrations by the plugging-meter technique will be obtained routinely once a day; if there is no apparent change over a long period of operation, they will be obtained less frequently.

Oxygen, hydrogen, and carbon concentrations will be determined by the equilibration method in which metal tabs of wires are exposed to flowing sodium in the MPS for a time sufficient to establish equilibrium with respect to impurities. Subsequent measurement of impurities in the wire or tab will be used to determine the impurity concentration in the sodium.

Uranium and plutonium concentrations in liquid metal will be determined fluorometrically and by alpha assay, respectively, using a vacuum distillation residue. Tritium concentrations will be measured using liquid scintillation counting techniques.

The concentrations of various elements (Ca, Co, Cr, Cu, Fe, Li, Mg, Mn, Mo, Ni, K, Cs, Rb, Si, and B) will be determined using standard techniques of atomic absorption of flame emission spectrophotometry. Several analyses will be performed with a sodium distillation residue. The basis for developing CRBR procedures for chemical and radiochemical analyses of liquid sodium is RDT Std. F3-40T, which contains typical analytical procedures.

The criterion for forced circulation cold trap replacement is an increase in cold-trap ΔP at design flow. This limiting ΔP is currently estimated to be 22 psid. When this limiting ΔP is reached, cold trapping may be continued at decreased flows, provided impurity limits are not exceeded.

9.3.6.4 Evaluation Summary

The primary sodium storage and processing system and the intermediate sodium processing system will provide processing capability to maintain reactor sodium and intermediate sodium purity within the limits to ensure coolant pressure boundary material integrity. Furthermore, the liquid sodium quality will be maintained by the use of cold traps. The applicants also provided the limits and frequencies of chemical and radiochemical analyses to be performed for liquid metal analyses and the criteria for cold-trap replacement to ensure the liquid metal processing systems will meet the requirements of PDC 13 as it relates to reactor coolant system design. The systems are acceptable for construction permit review. However, operating experience and materials data from FFTF should be provided as verification by the applicants for the OL review.

9.3.7 Summary and Conclusions

The auxiliary liquid metal system will be comprised of five separate subsystems that will provide means to (1) store and handle liquid metals used in the plant, (2) control reactor vessel sodium level, (3) purify primary, intermediate, and EVS sodium, (4) remove decay heat from spent fuel, and (5) remove reactor decay heat in a backup mode.

The staff has reviewed the preliminary design, proposed engineering criteria, functional requirements, and general arrangement of the subsystems comprising the ALMS and found that the ALMS is based on established liquid metal systems technology and a broad base of successful LMFBR experience in the design and operation of similar systems.

The staff review has included the applicants' proposed design criteria and design bases from the standpoint of deciding compliance with those principal design criteria specifically applicable to the ALMS and falling within the scope covered in this section of the SER. Based on that review the staff concludes that the design meets the applicable requirements of PDC 4, 35, 53, and 55. That conclusion is based on the following:

(1) The applicants have met the requirements of PDC 4, "Protection Against Sodium and NaK Reactions," with regard to minimizing the adverse effects of possible chemical reactions by selecting heat transfer fluids for use in interfacing systems that are compatible with sodium and/or NaK.

(2) The applicants have met the requirements of PDC 35, "Reactor Residual Heat Extraction System," with respect to provision of a passive boundary between reactor coolant and the working fluid of the heat extraction system for the DHRS mode of backup decay-heat removal and by selection of a working fluid (NaK) that is not chemically reactive with sodium.

(3) The applicants have met the requirements of PDC 53, "Fuel Storage and Handling and Radioactivity Control," with regard to provision of a residual-heat-removal capability having reliability and testability that reflects the importance to safety of decay-heat removal from stored spent fuel.

(4) The applicants have met the requirements of PDC 55, "Monitoring Fuel and Waste Storage," with regard to providing means to detect conditions that may result in loss of residual-heat-removal capability from stored spent fuel.

In addition, from its review of drawings and tabulations in the PSAR, the staff concludes that the proposed seismic and safety (quality group) classifications for the subsystems comprising the ALMS are consistent with anticipated service conditions, safety requirements, and the conclusions drawn in Section 3.2 of the SER.

From review of sections of the PSAR that interface with or are related to the ALMS and through coordination with staff reviewers for those sections, the staff concludes that related and interfacing design criteria and PDC have been fully covered by other reviewers and that the conclusions they have drawn are favorable with respect to their applicability in whole or in the ALMS.

With the exception of those matters identified above for resolution at the OL stage of review, the staff concludes that (1) the applicants' proposed preliminary design and associated design criteria can be used with reasonable assurance that the final design will be acceptable from the safety standpoint, and (2) that sufficient information has been presented to ensure that the applicants will meet the Commission's regulations and all applicable principal design criteria in the design, construction, and operation of the CRBR auxiliary liquid metal system.

9.4 Piping and Equipment Electrical Heating System

Introduction

The piping and equipment electrical heating system will provide the electrical heaters, electrical heater mounting hardware, heater power controllers, and the related temperature measuring and controlling instrumentation and equipment required to heat the following process systems and components containing sodium and sodium aerosols:

(1) reactor enclosure (guard vessel and head)
(2) primary and intermediate reactor heat transport systems
(3) steam generator dump tanks and sodium-water reaction product tanks
(4) auxiliary liquid metal systems
(5) sodium impurity monitoring system
(6) inert gas processing system
(7) ex-vessel storage tank and its associated sodium piping

Electrical heat is required to preheat these sodium systems before fill to prevent sodium from freezing when system heat sources such as reactor decay heat and pumping heat become insufficient and to maintain preestablished temperatures in the system.

Scope of Review

PSAR Sections 5.2.1.5, 5.2.1.6, and 9.4 were reviewed along with CRBR PDC 7, "Sodium Heating Systems." No SRP sections, regulatory guides, or NUREG documents were found to apply.

Evaluation Criteria

The staff reviewed the piping and equipment heating system

(1) to determine compliance with CRBR PDC 7.

(2) to determine acceptable impact on component integrity (i.e., heater failures will not cause failure of the components to which they are mounted)

(3) to determine the following for those systems requiring electrical heat to perform their safety functions:

    (a) redundancy and separation in temperature monitoring instrumentation so that a single failure will not remove monitoring capability

    (b) adequate heating coverage (including monitoring capability and Technical Specification limits)

    (c) adequate power supply

Evaluation

The agreed upon design resulting from the staff's review is described below.

The safety-related components that will have heaters are

(1) reactor guard vessel
(2) ex-vessel storage tank (EVST) guard vessel
(3) primary heat transport system

    (a) main piping
    (b) pump guard vessels
    (c) intermediate heat exchanger guard vessels
    (d) appendage piping

(4) intermediate heat transport system

    (a) main piping
    (b) pumps
    (c) expansion tanks
    (d) appendage piping

(5) steam generator system appendage piping

(6) auxiliary liquid metal system

    (a) overflow heat exchanger
    (b) primary sodium overflow vessel
    (c) in-containment sodium storage vessel
    (d) primary and EVST cold traps
    (e) overflow line

(f) piping in reactor and EVST cavities
(g) piping in primary and EVST cold trap cells

(7) inert gas processing system--all safety Class 1, 2, 3 components, piping, and equipment

(8) impurity monitoring and analysis system--all safety Class 1, 2, 3 components, piping, and equipment

Although these components will be electrically heated, not all will require heat to perform their safety functions. The components that will require heat in order for them to perform their safety functions are:

(1) During Power Operation

(a) Control rod drive mechanisms--to prevent sodium vapor from condensing and freezing in cooler regions of mechanisms*

(b) primary and intermediate system cover gas piping (including that leading to the system relief valves)--to prevent sodium vapor from condensing and freezing thus blocking these lines*

(c) sodium portion of ex-vessel storage tank standby heat removal loop--to prevent sodium from freezing

(d) direct heat removal system overflow heat exchanger and associated piping and valves--to prevent sodium from freezing

(2) During Plant Shutdown

(a) sodium portion of ex-vessel storage tank standby heat removal loop
(b) direct heat removal system overflow heat exchanger and associated piping and valves

The heater physical mounting arrangement and the electrical protection of the heater circuitry are to be designed to preclude damage to the components being heated.

Heaters and the associated mounting hardware that will be applied to safety-related components are to be designed so that the ability of these components to perform their safety function during or after a design-basis event up to and including the safe shutdown earthquake will not be impaired. They also will be designed so that the function of other instrumentation mounted on or near the equipment (i.e., leak detectors and loose parts monitors) will not be impaired. In addition, all components of the piping and equipment electrical heating system are to be designed and qualified as seismic Category II** to ensure their function will be performed if plant operation continues after an operating basis earthquake. All power to the heaters and their instrumentation will come from offsite sources (i.e., no diesel power will be supplied).

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*Only of concern when sodium temperature is more than ~450°F.
**Seismic Category II is an applicant designation for equipment designed to withstand an operating basis earthquake (OBE).

During loss of offsite power, the plant will shut down and the ex-vessel storage tank standby loop and direct heat removal system overflow heat exchanger and associated piping and valves will require the addition of heat. The applicants have proposed that in this event, procedures will require that sodium flow through these loops be established, thus using plant decay heat to maintain system temperature. In all cases of loss of heating, the response time to establish flow, reestablish heating, or shut down the plant will be in the range of several hours, since it will take this amount of time for the systems to cool to the point where the sodium will freeze or vapor will condense.

Heater sizes will vary depending on the component being heated; however, they will all be metal sheathed, MgO insulated with a central heating element. For the safety-related components shown above, the heaters will not be in direct contact with the component surface but rather will be mounted on standoffs. Ground fault interrupt (GFI) circuits will be provided for all heater power supplies to protect against heater arcing. Heaters will generally be arranged in groups (called zones) with all of the heaters in a zone having a common power supply and control system.

The heat rates required by different components will be controlled by using control thermocouples to monitor piping and component temperatures and to automatically adjust the power supplied to the heaters. The location of the control thermocouples will be chosen to be representative of the component temperature in that heater or heater zone location (i.e., not located directly next to a heating element) and each control thermocouple will control all of the heaters in a zone. Thermocouples for strictly monitoring purposes also will be provided and will be physically separate and independent from the control thermocouples. Sufficient monitoring thermocouples will be provided to preclude the presence of an undetected cold spot.

Control stations for the heaters will be located locally in the reactor containment building, reactor service building, and steam generator building, and a master control panel will be in the main control room.

Summary and Conclusions

The concept of electrically heated components and piping through the application of metal sheath heaters has been applied in other plants. The concepts chosen by CRBR to apply the heaters are, in fact, very similar to those used in the fast flux test facility (FFTF). The electrical heating system is not considered by the applicants as important to safety or safety related. In view of the fact that (1) redundant and separate monitoring instrumentation is being provided for those components that require heat to perform their safety functions, (2) Technical Specification will be established on critical temperatures, (3) procedural actions can be taken on loss of power to the heaters to shut down the plant and maintain the operability of affected decay heat removal systems, and (4) the heating system design requirements will ensure that the heaters will not affect the function of the safety-related equipment to which they will be attached, the staff finds this acceptable. The components and systems that are heated are the correct ones. Failures of the trace heat system were reviewed and can take the following forms:

(1) electrical shorting with possible arcing to the component
(2) loss of power to one or more heaters
(3) heaters fail on (overheating)
(4) loss of offsite power
(5) seismic events

Item (1) will be prevented by the installation of GFIs circuits and by operating the heaters at less than their rated capacity. Items (2) and (3) will be accounted for in the component design process by the establishment of limits on the number of failed heaters and by the analysis of the components for failed-on heaters. In addition, Item (2) will be addressed by establishing Technical Specification limits on critical temperatures. Item (4) will be accommodated by having the plant procedures establish flow in those loops requiring the addition of heat during loss of offsite power. Item (5) will be accommodated in the design by requiring that the heaters not impair the safety functions of the components to which they will be mounted and by designing and qualifying the heating system to the operating basis earthquake (OBE) to support continued plant operations in the event the OBE occurs.

The applicants have committed to establishing Technical Specifications on minimum temperature of the following components:

(1) primary heat transport system (PHTS) and intermediate heat transport system (IHTS) cover gas equalization lines

(2) PHTS argon lines leading to relief valves

(3) overflow heat exchanger

On the basis of the staff's review, two additional areas requiring electrical heat also require Technical Specification. These are:

(1) The reactor head around the control rod drive mechanism (CRDM) penetrations and above the head portion of the CRDMs should be kept hot (>250°F) to ensure scram capability of the control rods. If not kept hot, sodium aerosol from the reactor cover gas has the potential for condensing in the drive line annular regions between the CRDM bellows and the bottom of the reactor head and impeding drive line motion. FFTF experience has shown that temperatures in this region can drop without the addition of heat. Therefore, a Technical Specification on temperatures in this region should be prepared.

(2) Certain portions of the direct heat removal system (DHRS) and ex-vessel storage tank decay heat removal systems will stay in standby during normal reactor operation in the event they are required for reactor decay heat removal. Electrical heat must be applied to those sodium lines and valves in standby to prevent them from freezing. Therefore, a Technical Specification on temperatures of these lines should be prepared.

With incorporation of the above items into the Technical Specifications, the design of the piping and equipment electrical heating system will meet the evaluation criteria and is considered satisfactory for issuance of a construction permit.

9.5 Inert Gas Receiving and Processing System

Introduction

The purpose of the inert gas receiving and processing (IGR&P) system is to supply inert gas (either argon or nitrogen) to those plant systems requiring this gas. This system, in conjunction with the radioactive argon processing system (RAPS) and the cell atmosphere processing system (CAPS), will monitor and control the gas purity to meet the requirements of the systems it serves and to ensure that the inert gas released to the atmosphere will comply with 10 CFR 20 and as low as is reasonably achievable (ALARA) guidelines.

Scope of Review

PSAR Section 9.5 was reviewed as well as interfacing portions of Section 9.16 and 11.3. SRP Section 9.3.1 and RG 1.29 were used as guidance for the review. This review concentrated only on the argon distribution system and the nitrogen distribution system. The cell atmosphere processing system (CAPS) and the radioactive argon processing system (RAPS) are evaluated in SER Section 11 and the recirculating gas cooling system in SER Section 9.16.

Evaluation Criteria

The staff reviewed the inert gas receiving and processing system

(1) to determine compliance with the following CRBR Principal Design Criteria (PDC):

  (a) PDC 4, "Protection Against Sodium and NaK Reactions"

  (b) PDC 34, "Reactor and Intermediate Coolant and Cover Gas Purity Control"

  (c) PDC 45, "Piping Systems Penetrating Containment"

  (d) PDC 47, "Primary Containment Isolation"

  (e) PDC 48, "Closed Systems Penetrating Containment"

(2) to determine compliance with RG 1.29, "Seismic Design Classification"

(3) in general, to determine

  (a) acceptability of design criteria

  (b) provision for supplying adequate normal and backup inert gas supplies

(c) conformance with appliable requirements of SRP Section 9.3.1 (Those requirements of SRP Section 9.3.1 considered applicable were those not unique to an LWR compressed air system.)

Evaluation

The agreed upon design resulting from the staff's review is described below.

(1) Argon Distribution System

The argon distribution system will be composed of liquid argon dewars with vaporizers, gaseous argon bottles, piping, valves, vapor traps, filters, vessels, and so forth, necessary to supply argon gas to the CRBR systems for

(a) liquid sodium and NaK cover gas

(b) filling and purging the primary and intermediate heat transport systems before sodium fill

(c) buffered and inflatable seals in the HTS and EVST

(d) atmosphere control in the fuel-handling cell

(e) supplying pressure for secondary control rod drive mechanism latching

(f) services connected with fuel handling, sampling, and major maintenance operations

In general, the argon distribution system is seismic Category III* with the exception of the recycle argon storage vessel and the containment isolation valves and their penetrations which are seismic Category I. Also, those portions of the argon system that are extensions of the primary system boundary or the EVST system boundary or that will supply inert gas to any component or system containing sodium or NaK, up to and including the first isolation valve, will be seismic Category I.

Vessels within the system will be designed to ASME Code, Section VIII, with the exception of the recycle argon storage vessel, which will be designed to Section III. Most of the system will be made of carbon steel; those portions that could be exposed to sodium or cryogenic temperatures will be made of stainless steel. The piping is to be designed to ANSI Std. B31.1.

The design capability supply rate for the argon system will be sufficient to meet plant argon needs for normal and offnormal operation. The system will include the capability to recycle argon from the RAPS and use it for the reactor cover gas argon supply, thus reducing the total argon usage.

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*Seismic Category III is an applicant designation for certain nonseismic Category I equipment as discussed in Section 3.7 of this SER.

Redundant argon supplies will exist to maintain reactor and EVST inert atmospheres in the event of a loss of the normal argon supply. Argon bottles will be used for the secondary control rod drive mechanism and as a backup supply to each intermediate heat transport loop. A dewar will provide a backup supply of LN_2 (liquid nitrogen) to the RAPS cold box in the event of containment isolation. An evaluation of the reactor vessel pressure control and its effect on safety is included in Section 15.7 of this SER.

Except for the fuel-handling cell, argon purity will be monitored and controlled by the RAPS. The fuel-handling cell will have its own gas purification system, which will monitor and control the cell's argon purity.

(2) Nitrogen Distribution System

The nitrogen distribution system will be composed of liquid nitrogen dewars, vaporizers, gaseous nitrogen bottles, piping, valves, filters, vessels, and so forth, necessary to supply nitrogen gas for the following purposes:

(a) inerting and monitoring the atmosphere of the cells and pipeways containing radioactive sodium

(b) cooling the primary control rod drive stepping motor

(c) activating pneumatically operated valves

(d) supplying cover gas for the Dowtherm tank in the chilled water system

(e) purging of the CAPS and RAPS cold boxes

(f) supplying cover gas for the sodium-water reaction pressure relief system

(g) miscellaneous handling and maintenance services

The nitrogen distribution system will be seismic Category III except for the containment penetrations which will be seismic Category I. Vessels within the system will be designed to the ASME Code, Section VIII. The piping materials will be carbon steel, and stainless steel will be used where cryogenic temperatures will exist. The piping is to be designed to ANSI Std. B31.1. The design capability supply rate for the system will be sufficient to meet plant nitrogen needs for normal and offnormal operation. The nitrogen supply requirements for the reactor containment building (RCB), reactor service building (RSB), steam generator building (SGB), and sodium-water reaction pressure relief system will be on independent systems. Plans to install supplementary high-pressure gas bottles (designed as safety-related equipment) in the RCB and RSB will ensure nitrogen supply gas for the essential valve operation function. These will be designed to serve as safe shutdown protection in the event of an interruption or loss of the principal gas supply.

Nitrogen sampling and analysis units will be part of the nitrogen distribution system to monitor and control gas purity. Both the RSB and RCB supply systems will contain a sampling and analysis unit, which will be seismic Category III. Each unit periodically will sample the nitrogen gas in each inerted cell of either the RSB or RCB and analyze it for radioactivity, oxygen, and water. Single sensors capable of calibration will be used in each unit. The oxygen and water in the inerted cell atmosphere will be controlled to within prescribed units (0.5-2.0% O_2; less than 1,000 ppm water) by the nitrogen distribution system by automatically purging with fresh nitrogen or manually adding oxygen, if needed. All RCB inerted cells will discharge their effluent to the CAPS. RSB inerted cell effluents will be diverted to the CAPS only if they are radioactive; otherwise, they will discharge to the RSB exhaust. This sensing and diversion function will be carried out by the nitrogen sampling and analysis unit. This is considered an ALARA function since the RSB exhaust monitors are the safety-grade protection to limit offsite doses.

(3) Instrumentation

Instrumentation for the argon and nitrogen supply systems will be provided to monitor supply level, valve position, system pressure and flow rates and to control inerted cell purging and exhaust paths. No safety or seismic category is stated for this instrumentation.

(4) General

The argon and nitrogen supply system are not in themselves required to actively function for the safe shutdown of the CRBR or to support the decay heat removal after shutdown (except for backup nitrogen supplies to certain essential valves). However, portions of the systems will serve as extensions of the primary coolant boundary and/or the reactor containment boundary or supply nitrogen for functions important to safety (cell and SWRPS inerting).

All penetrations of the containment by the argon and nitrogen distribution systems, with the exception of an LN_2 line, will be protected by double isolation valves that will prevent the escape of radioactive material when high activity levels exist within this containment. The LN_2 line will be protected by an isolation valve on the outside of containment and a check valve on the inside. This is consistent with PDC 45, 47, and 48.

Summary and Conclusions

The inert gas receiving and processing system is similar to that used in FFTF. Except for the backup nitrogen supply for certain essential valves, this system is not classified as important to safety since loss of the argon or nitrogen supply will not in itself lead to an accident condition or affect safe shutdown or decay heat removal. In addition, the effect of a rupture and subsequent release of radioactive gas from this sytem has been analyzed and found not to exceed allowable release limits or to affect safe shutdown. However, since portions of the system will connect with systems containing radioactive material, act as barriers to radiation, or provide nitrogen for functions important to safety, the design with regard to these areas is emphasized. For these areas the system design and the staff's conclusions are as follows:

(1) In addition to designing the entire piping system to ANSI Std. B31.1, "Power Piping," and vessels to ASME Code, Section VIII, requirements, those portions of the system that will connect with systems containing radioactive material (up to the first isolation valve) or that will act as a barrier to the release of radiation are to be designed to seismic Category I conditions and vessels containing large inventories of radioactive material are to be ASME Code Section III. These additional requirements are considered acceptable and consistent with heat transport system design requirements and ALARA concerns.

(2) The valves not provided with safety-grade backup supplies and secondary CRDMs, which rely on an inert gas supply for normal operation, will go to a safe configuration under a loss of supply pressure. This is acceptable.

(3) The loss of inerting in the cells, in the sodium-water-reaction product system (SWRPS), or in cover gas spaces will not in itself cause an accident. A sodium or NaK leak will be required in conjunction with the deinerted condition to lead to an accident not analyzed in PSAR Chapter 15. Therefore, the applicants should provide sufficient instrumentation to detect such a loss of inerting so that extended plant operation without the conditions assumed in the safety analysis will not occur. For the argon system the gas purity will be monitored by the RAPS (see Section 11 in this SER) except for the fuel-handling cell, which will rely on its own atmosphere control unit. For the nitrogen system the nitrogen analysis and sampling units will provide this function. Since these units are basically industrial-grade instruments, with single sensors and a seismic Category III designation, it will be necessary to monitor the inerted cell atmosphere purity through the Technical Specifications and to establish calibration requirements and frequencies that will ensure that the plant will not be operated for an extended period of time with a failed sensor which would contribute significantly to the risk of CRBR operation. If such calibration requirements cannot be established, redundant sampling and analysis units will be required. In addition, if a seismic event, should occur, calibration of these units also would be required. Implementation of such requirements and limits for this system in place of safety-grade sensors is acceptable to the staff since loss of inerting in the cells will not in itself cause an accident.

With incorporation of the above considerations into final design, the staff concludes that the proposed inert gas receiving and processing system meets the requirements of the evaluation criteria and is acceptable for issuance of a construction permit.

9.6 Heating, Ventilating, and Air Conditioning System

9.6.1 Control Building HVAC System

The heating, ventilating, and air conditioning (HVAC) system in the control building will be composed of three separate subsystems: the control room HVAC system, the control building and electrical equipment building (EEB) emergency HVAC system, and the control building and EEB nonessential HVAC system. The control room HVAC system will be a safety-related system designed to provide filtered and conditioned air to the control room as required to permit continuous occupancy of the control room and to ensure the operability of all control room equipment and instrumentation under all conditions. The control

building and EEB emergency HVAC system will be a safety-related system designed to provide filtered and conditioned air under all operating conditions to the control building, battery compartments, battery maintenance area, upper and lower cable spreading rooms, vital ac/dc rooms, and the EEB Class 1E switchgear rooms for the division 1 and 2 diesel generators. The system will provide the required environment to permit personnel access during normal plant operation and to ensure operability of the equipment under all conditions. The control building and EEB nonessential HVAC system will be a nonsafety-related system designed to provide filtered air during normal plant operation to the control building control rod drive mechanism rooms, unit substation area, motor control center area, the EEB non-Class 1E switchgear room and switchgear bus area, and the control building and EEB motor generator (MG) set rooms. The system will provide the required environment during normal plant operation to permit personnel access and to ensure operability of the equipment. These systems are described in Section 9.6.1 of the PSAR, Amendment 45. Supplemental information was provided by the applicants in a meeting on December 13, 1982. This supplemental information will be incorporated in a future amendment to the PSAR.

The control building HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.1, "Control Room Area Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the proposed system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR Principal Design Criterion (PDC) 2. Acceptance is based on meeting the guidance of RG 1.29, "Seismic Design Classification," Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," with respect to maintaining environmental conditions in the control room compatible with the design limits of essential equipment located therein during normal, transient, and accident conditions. GDC 4 is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 19, "Control Room," as it relates to providing adequate protection to permit access and occupancy of the control room under accident conditions. GDC 19 is identical to CRBR PDC 17 except that PDC 17 includes sodium reaction but does not include loss-of-coolant accidents in the accident conditions to be protected against. Acceptance is based on meeting the guidance of RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Plant Control Room During a Postulated Hazardous Chemical Release," relating to the instrumentation to detect and alarm any hazardous chemical release in the plant vicinity, to the system's capability to isolate the control room from such releases, and to the system's capability to meet the single-failure criterion (Positions C.3, C.7, and C.14, respectively). Acceptance is also based on meeting the guidance of RG 1.95, "Protection

of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," relating to the system's capability to limit the accumulation of chlorine within the control room and the system's capability to meet the single-failure criterion (Positions C.4a and C.4d, respectively). Because of the potential for sodium reactions at CRBRP, acceptance is also based on meeting the guidance of RG 1.78, Positions C.3, C.7, and C.14, as extrapolated to include sodium reaction products.

(4) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, "Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Position C.2), and RG 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Positions C.1 and C.2), as related to design, testing, and maintenance criteria for atmosphere cleanup system and normal ventilation exhaust system air filtration and absorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.1 is applicable to the review of the CRBR control building HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.1, except as noted, provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

The control room HVAC system will be composed of redundant trains of air conditioning and filtration equipment including two 100% capacity air conditioning units, two 100% capacity control room filter units and filter unit supply fans, a toilet exhaust fan, a kitchen exhaust fan, and associated dampers and ductwork. Redundant missile-protected roof air intake structures will provide a supply of outside air for the control room. During normal operation, one control room air conditioning unit will operate to maintain a positive pressure in the control room and appropriate levels of sound, cleanliness, temperature, and relative humidity to permit continuous occupancy and operability. Temperature will be maintained during normal operation by cooling coils in the air conditioning units served by the normal plant service water system and by duct-mounted reheat coils. Pre- and after-filters in the air conditioning units will maintain the cleanliness of the air supply during normal operation. During accident conditions that could introduce airborne radioactivity, chlorine, or other toxic chemicals into the control room, sensors in the outside air intake ducts will provide signals to initiate closure of the contaminated intake duct, start a control room filter unit supply fan, and line up the appropriate dampers to provide filtration and recirculation of the control room air mixed with a minimum makeup of outside air introduced through the uncontaminated duct. In the event of loss of normal chilled water, the air conditioning units will be served by redundant emergency chilled water system trains.

The essential HVAC equipment serving the control room will be designed to seismic Category I requirements. All of the control room HVAC system equipment will be housed within the control building, which will be a flood-protected,

tornado-hardened, seismic Category I building. The toilet exhaust and kitchen exhaust systems will be the only nonessential portions of the control room HVAC system. They will be separated from the essential portions of the control room HVAC system by two automatic dampers in series. In a meeting on December 13, 1982, the applicants verified that these dampers will be designed to seismic Category I requirements. The exhaust duct from the toilet and kitchen exhaust fans will connect with the control building missile-protected exhaust structure. The exhaust fans along with the associated ductwork running through safety-related areas will be seismic Category III,* and as such will be designed so that failure of these components in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Thus, the staff concludes that the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C-1 and C-2, have been met.

Two 100% capacity trains will be provided to maintain the required conditions in the control room to ensure continuous occupancy and operability of control room equipment. All essential control room HVAC equipment will be located in a tornado-hardened building, and the outside air intake and exhaust structures will be protected against missiles. The redundant air conditioning units, filter units and fans, and control room return fans will be located in separate compartments to preclude common-mode failures from missiles or pipe break. All control room HVAC safety-related equipment will be served by Class 1E power. Failure of an operating air conditioner supply, filter unit, or control room return fan will automatically start the redundant fan. Failure of the advance mechanism for the roll-type filters will be annunciated by a high differential pressure alarm. The setpoint of the alarm will be selected to provide 72 hours to correct the failure without significantly deviating from the system design parameters. There will be no single active failure that will prevent operation of both trains of control room HVAC equipment in normal or emergency modes of operation. The redundant air conditioning unit chillers will be served by redundant emergency plant service water trains in the event of loss of normal plant service water. Thus, the requirements of PDC 5 (GDC 4) are satisfied.

Outside air will be supplied to the control room from two separate air intake structures. Each of the redundant air supply ducts will be provided with redundant detectors to detect airborne radioactivity, smoke, chlorine, and other toxic chemicals. A containment isolation signal or the detection of radioactivity or high levels of smoke, chlorine, or other toxic chemicals in either of the outside air intake ducts will initiate closure of the control room outside air intake and exhaust isolation valves, open the redundant filter unit outside air isolation valves, and start one of the filter unit fans to allow a minimum amount of outside air, required for pressurization, to pass through the operating filter unit. Air will be drawn from the less contaminated of the two outside air intakes. The operators also will be able to manually initiate control room isolation if higher than normal levels of toxic chemicals or radioactivity are detected in other areas of the plant by toxic chemical detectors or by the radiation monitoring system. As discussed in Section 6.3.1.6.1 of the PSAR, the sodium leak detection system will provide the first warning to

*Seismic Category III is an applicant designation for certain nonseismic Category I equipment as discussed in Section 3.7 of this SER.

the control room of a probable sodium/NaK combustion product release to the atmosphere. If the sodium/NaK leak occurs in a cell containing an air atmosphere and the leak starts a fire, the sodium fire detection system will provide the second early warning to the control room of a probable sodium/NaK combustion product release to the atmosphere. It is also stated that the detectors in the control room air intake ducts will be able to detect sodium or NaK combustion products. As discussed in Sections 6.3.1.6.1 and 6.3.1.6.2 of the PSAR, the applicants will evaluate the effects on control room habitability, in accordance with RG 1.78, of release of toxic chemicals stored on site, release of toxic chemicals at nearby transportation accidents, and release of sodium or NaK combustion products. Thus, the guidelines of RG 1.78 are satisfied. No gaseous or liquid chlorine will be stored on site and, therefore, the guidelines of RG 1.95 are not applicable. In a meeting on December 13, 1982, the applicants verified that the toilet and kitchen exhaust fan dampers, which must shut for control room isolation, will be designed to Quality Group C requirements and will be controlled by safety-related controls. Thus, the staff concludes that the requirements of PDC 17 (GDC 19) and the guidelines of RG 1.78 are satisfied.

The control room HVAC system will not be required for control of releases of radioactive materials to the environment, and thus, the requirements of PDC 52 (GDC 60) and guidelines of RGs 1.52 and 1.140 are not applicable.

The control building and EEB emergency HVAC system will be separate from the control room HVAC system. The control building and EEB emergency HVAC system will operate normally to ensure personnel access and operability of equipment in the spaces served by the system. During emergency conditions the system will be required to ensure operability of equipment. The system will be composed of two 100% capacity air conditioning units, which will combine a minimal amount of outside air with recirculated air provided by two 100% capacity return fans. The amount of outside air used by the system will be based on ventilation requirements and battery room hydrogen dilution requirements. Outside air will be supplied to each air conditioner through separate ducting from separate missile-protected intake structures.

Each train of the control building and EEB emergency HVAC system will serve essential equipment of the corresponding safety division in the redundant cable spreading rooms, battery rooms, vital ac/dc rooms, and EEB Class 1E switchgear rooms. Each HVAC train will be supplied with emergency power from the corresponding division of Class 1E power supplies. Temperature of the spaces served will be maintained by chiller coils in the redundant air conditioning units which will be supplied by the corresponding division of the emergency chilled water system on loss of the normal chilled water system. Cleanliness will be maintained by the air conditioning unit filters. Each battery room will be provided with a separate exhaust fan to prevent an accumulation of hydrogen. Exhaust flow will be directed to the missile-protected exhaust structure. By letter dated June 1, 1982, the applicants verified that the battery room exhaust fans will be powered from Class 1E power supplies and that each fan will be provided with a low flow switch that will alarm in the control room. On a low flow alarm, any battery charging activities in the affected compartment will be discontinued and temporary ventilation provided to exhaust any residual hydrogen. Redundant air conditioning units and return fans will be located in separate control building HVAC rooms. Each pair of battery room fans also will be

in separate compartments. No single active failure of the control building and EEB emergency HVAC system, including loss of a division of Class 1E power, will prevent operation of both trains of essential equipment served by the HVAC system. All the HVAC system components will be located within the tornado-hardened, flood-protected control building and EEB. Thus, the requirements of PDC 5 (GDC 4) are satisfied.

All the control building and EEB emergency HVAC system equipment will be designed to seismic Category I requirements. The system will be located within the seismic Category I control building and EEB. Thus, the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C-1 and C-2, are satisfied.

Because the control building and EEB emergency HVAC system will not be required to ensure occupancy or personnel access to the spaces served during emergency conditions, the requirements of PDC 17 (GDC 19) and the guidelines of RGs 1.78 and 1.95 are not applicable. Also, this system will not be required for control of releases of radioactive materials to the environment, and thus, the requirements of PDC 52 (GDC 60) and the guidelines of RGs 1.52 and 1.140 are not applicable.

The control building and EEB nonessential HVAC system will be a nonsafety-related system designed to ensure operability of nonessential equipment and permit personnel access to nonessential equipment spaces during normal operation. The system will be composed of two 50% capacity supply fans, two 50% capacity return fans, cooling coils, filters, ductwork, and dampers. This air-handling equipment will serve the motor-generator set rooms, switchgear rooms, control building CRDM rooms and EEB nonessential spaces. A separate nonsafety-related unit cooler will be provided for each motor-generator set to remove internally generated heat. The air-handling coolers will be served by the normal chilled water system, and the motor-generator set coolers will be served by the normal plant service water system. The outside air intake and exhaust structures used by the system will be protected against missile damage. All of the control building and EEB nonessential HVAC system components will be located within the control building and EEB, which will be flood-protected, tornado-hardened, seismic Category I structures. The nonessential HVAC system is designated applicants' seismic Category III and, as such, will be designed so that failure of the system or its components in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Thus, the proposed control building and EEB nonessential HVAC system meet the requirements of PDC 2 (GDC 2) and PDC 5 (GDC 4) and the guidelines of RG 1.29, Positions C-1 and C-2.

Because the control building and EEB nonessential HVAC system will not be required to ensure occupancy or personnel access to the spaces served during emergency conditions, the requirements of PDC 17 (GDC 19) and the guidelines of RGs 1.78 and 1.95 are not applicable. Also, this system will not be required for control of releases of radioactive materials to the environment, and thus, the requirements of PDC 52 (GDC 60) and the guidelines of RGs 1.52 and 1.140 are not applicable.

On the basis of the above, the staff concludes that both the proposed control building and EEB emergency HVAC system and nonessential HVAC system are in

conformance with the requirements of PDC 2 and 5 relating to protection against seismic events, floods, tornadoes, missiles, and environmental effects and the ability to maintain an acceptable environment for people and equipment, and the guidelines of RG 1.29, Positions C.1 and C.2, relating to the system's seismic classifications, and are, therefore, acceptable.

The staff also concludes that the proposed control room HVAC system is in conformance with the requirements of PDC 2, 5, and 17 relating to protection against natural phenomena, maintenance of proper environmental limits for equipment operation, and protection to permit access to the control room under accident conditions, and the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.78, Positions C.3, C.7, and C.14 relating to system seismic classification, system design for emergency operation, design for protection against hazardous chemical release, protection of personnel against sodium/Nak combustion product release and system design for normal operation and is, therefore, acceptable.

9.6.2 Reactor Containment Building/Annulus HVAC and Cleanup Systems

9.6.2.1 Reactor Containment Building HVAC System

The reactor containment building HVAC system will provide filtered and conditioned air to the operating floor of the reactor building as well as to the normal atmospheric (noninerted) areas below the operating floor. Systems components used only to provide personnel access and equipment operability in the above-mentioned areas during normal operation will be nonsafety related. The essential instrumentation and control cubicle unit coolers and the supply and exhaust duct containment isolation valves will be safety related.

The reactor containment building HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the proposed system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, "Seismic Design Classification," Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions. GDC 4 is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 17, "Electric Power Systems," as it relates to ensuring proper functioning of the essential electric power system. GDC 17 is identical to CRBR PDC 15. Acceptance is based on meeting the guidance of NUREG/CR-0660, Recommendations A.2 and C.1, relating to the protection of essential electrical components from failure resulting from the accumulation of dust and particulate materials.

(4) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, "Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Position C.1, and RG 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, relating to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.5 is applicable to the review of the CRBR reactor containment building HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.5, except as noted, provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

The reactor containment building HVAC system will be designed to maintain a maximum temperature of 75°F in the essential instrumentation and control (EI&C) cubicles and 95°F in the remainder of the reactor containment building. The safety-related EI&C cubicle unit coolers will be designed to maintain a maximum temperature of 120°F under emergency conditions. The system will maintain a negative pressure in the building, dilute any cover gas leakage on the operating floor, and dilute radioactive gas leakage from the inerted cells below the operating floor. The system will also be designed to provide isolation of the building in the event of high levels of radioactivity in the containment building atmosphere. Other system functions are discussed below.

During normal operation, filtered and conditioned air will be supplied to the operating floor ducts by the outside air-handling unit, which will supply air from the missile-protected air intake structure. The outside air-handling unit will contain heating and cooling coils, filters, and two 100% capacity supply fans. The filters in the air-handling unit along with the air intake being located 21 ft above grade level will ensure cleanliness of the supply air to the building. Ducting in the building will distribute the air to the space above the operating floor and to the appropriate spaces below the operating floor. Two recirculation fans will be provided in the containment upper dome to prevent air stagnation in this area. Air from above the operating floor will be chilled by three nonsafety-related unit coolers and ducted back into the supply ducts to mix with the supply air to supplement the cooling provided by the outside air-handling unit coolers during normal operation. Two air conditioning units will be provided for the cooling and ventilation of noninerted spaces below the operating floor during normal operation. A

mixture of outside supply air and recirculated air from the areas below the operating floor will be cooled and filtered by the air conditioning units. A return fan will be provided for each air conditioning unit to supply the recirculated air. The large component cleaning vessel (LCCV) cell and the head access area will each be provided with a unit cooler to maintain indoor design conditions during normal operation. A purge exhaust duct, with flanged connections, located below the operating floor, and connected to the main exhaust duct will be used for deinerting the inerted cells. Deinerting will be performed manually by a portable filter fan unit that will purge the inerted cell and discharge into a connection with the purge duct. All air exhausted from the reactor containment building will pass through a single exhaust duct that will direct the air to a missile-protected exhaust structure. Two 100% capacity exhaust fans will be provided to remove air from the building and maintain a negative pressure within the building. The supply and exhaust ducts will each be provided with three containment isolation valves, two immediately outside containment and one inside.

With the exception of the supply and exhaust isolation valves, all the equipment discussed above will be nonsafety related and either seismic Category I or the applicants' seismic Category III. Seismic Category III equipment will be designed so that failure of these components in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (see Section 3.7 of the SER). The containment isolation valves will be Quality Group B and seismic Category I. All of the equipment discussed above will be located in tornado-hardened, flood-protected, seismic Category I buildings.

All of the air conditioners and unit coolers discussed above will be provided with cooling water from the normal chilled water system and, therefore, will not be available during loss of offsite power or loss of the normal chilled water system. Also, both the supply of outside air and the exhaust path will be isolated in the event of a containment isolation signal. However, considerable redundancy will be provided to maintain design conditions during normal operation. The standby supply and exhaust fans will start automatically on failure of the operating fan. The supply filter units will be provided with differential pressure sensors to automatically advance the roll-type filter and to alarm on higher than normal differential pressure. The alarm setpoint will be selected to provide 72 hours to correct the filter failure before system design parameters deviate significantly from normal. The automatic dampers for air conditioners below the operating floor will be provided with remote position indicators and manual operators. A single active failure of an HVAC system component will not result in excessively high temperatures.

The only safety-related air coolers will be the three unit coolers serving the safety-related equipment located in the EI&C cubicles. These coolers will supplement the nonsafety-related operating floor unit coolers to maintain a maximum temperature of 75°F in the EI&C cubicle during normal operation. In the event of a single failure of the reactor containment building HVAC system, loss of normal chilled water, or loss of offsite power or containment isolation, the coolers will maintain the EI&C at a maximum temperature of 120°F. The EI&C cubicle unit coolers will be served by the safety-related emergency chilled water system and Class 1E power supplies. The coolers will be designed to Quality Group C and seismic Category I requirements. Each unit cooler will be located in the EI&C cubicle that it will serve. In the event of a loss of

one emergency power train in conjunction with the loss of offsite power, more than one unit cooler could be lost. However, in a meeting on December 13, 1982, the applicants verified that under these conditions, even with containment isolation, the temperature would not exceed the upset design temperature of 120°F in any EI&C cubicle.

On the basis of the above, the staff concludes that the reactor containment building HVAC system design will satisfy the requirements of PDC 2 (GDC 2), PDC 5 (GDC 4), and PDC 15 (GDC 17) and the guidelines of RG 1.29, Positions C.1 and C.2, and NUREG/CR-0660, Recommendations A.2 and C.1.

During normal operation, the exhaust from areas served by the reactor containment building HVAC system will be ducted to the atmosphere with no exhaust filtration. The ventilation air flow rates will be selected to maintain the radioactive gas concentrations below acceptable limits. Sources of airborne radioactivity from the areas below the operating floor will be probable outleakage of inert gases from normally inert cells. The inert gases will be continuously purified by the cell atmospheric processing system (see Section 9.5 of this SER). These cells will be sealed and maintained at lower pressures than the adjacent air-filled cells during normal operation. The low leakage from these cells and the cover gas leakage at the head access area and primary pump seals will be diluted by the normal ventilation flow rates.

In the event of excessive radiation levels in the reactor containment building exhaust, the supply and exhaust duct isolation valves will be shut. The containment building HVAC system will not be used for atmospheric cleanup after isolation caused by excessive radiation levels. Radiation monitors will be provided in the containment building HVAC exhaust duct upstream of a time-delay duct section that will be located upstream of the inboard containment exhaust isolation valve. An excessive radiation level signal from either of the exhaust air radiation monitors will cause automatic isolation of the reactor containment building HVAC system and shut down the supply and exhaust fans before radiation is released from the building. The radiation monitors will be located and the exhaust duct velocity will be determined in accordance with the closure time of the isolation valves.

As discussed in Section 9.6.3 of this SER, the annulus filtration system will provide postaccident atmosphere cleanup for the containment building.

Portable filter-fan units will be used in conjunction with the purge exhaust duct to purge the inerted cells. Purge exhaust will be ducted into the main exhaust duct upstream of the main exhaust duct radiation monitors. In addition, a radiation monitor will be provided at the discharge side of the portable filter-fan unit to stop the purging operation if high radiation levels are sensed.

During open hatch refueling, the containment building exhaust fans will be stopped. The containment building negative pressure differential and atmospheric filtration will be provided by the annulus filtration system during open hatch refueling (see Section 9.6.3 of this SER).

As shown above, the proposed reactor containment building HVAC system satisfies the requirements of PDC 52 (GDC 60) with respect to control of releases of

radioactivity. Because this system will not be required for filtration during normal operation, the guidelines of RG 1.52, Position C.2, and RG 1.140, Positions C.1 and C.2, are not applicable.

The staff concludes that the proposed reactor containment building HVAC system's design criteria and bases and preliminary design are in conformance with the requirements of PDC 2, 4, 15, and 52 as they relate to protection against natural phenomena, maintaining proper environmental limits for equipment operation, and controlling the release of gaseous radioactive effluents to the environment, and the guidelines of RG 1.29, Positions C.1 and C.2, relating to system seismic design classification and NUREG/CR-0660, Recommendations A.2 and C.1, relating to the protection of essential electrical equipment from failure resulting from the accumulation of dust and particulate materials and are, therefore, acceptable.

9.6.2.2 Annulus Filtration System

The annulus filtration system will provide filtered exhaust to maintain a minimum 1/4-in. water gage negative pressure in the containment/confinement annulus during normal and accident conditions. The system will be safety related.

The annulus filtration system design criteria and bases and the preliminary design were reviewed in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the proposed system design criteria and bases and the preliminary design meet applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, Position C.1, and RG 1.140, Positions C.1 and C.2, relating to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.3 is applicable to the review of the CRBR annulus filtration system. Conformance with the acceptance criteria of SRP Section 9.4.3, except as noted, provides the bases for concluding that the proposed system design criteria and bases and preliminary design are acceptable.

The annulus filtration system will be designed to maintain continuous pressure control and exhaust air filtration from the annulus during normal and emergency

operating modes. It will consist of two redundant full-capacity annulus pressure maintenance fans and filter fans, one pair associated with each of the two redundant full-capacity filter units. Each filter unit will contain a moisture separator, heating coil, prefilter, high-efficiency particulate air (HEPA) filter, charcoal adsorber, and afterfilter.

One pressure maintenance fan normally will be operating to maintain a minimum 1/4-in. water gage negative pressure in the annulus while exhausting air through its associated filter unit. The exhaust air will be passed through redundant radiation monitors before discharging to the atmosphere. On receipt of a containment isolation signal, redundant containment supply and exhaust isolation valves will close, reactor containment building supply fans will stop, and the redundant annulus pressure maintenance fan and both annulus filter fans will start. However, only one pressure maintenance and filter fan will be required to ensure proper emergency system operation, thereby providing necessary safety function in the event of a single failure. During emergency conditions, only a portion of the total air flow will be exhausted to the atmosphere and the remainder will be returned to the annulus.

Each annulus filter train will be powered by a separate Class 1E emergency power supply. The system will be safety-related, seismic Category I, and Quality Group C. It will be housed in the seismic Category I, flood- and tornado-protected reactor service building and containment annulus (see Sections 3.4.1 and 3.5.2 of this SER). As described above, the proposed annulus filtration system design criteria satisfy the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2, concerning seismic classification, and the requirements of PDC 52 (GDC 60) and the guidelines of RG 1.52, Position C.1, and RG 1.140, Positions C.1 and C.2, concerning atmospheric cleanup and normal ventilation exhaust filtration for gaseous radioactive waste control during normal and emergency conditions.

On the basis of the above, the staff concludes that the proposed annulus filtration system's design criteria and bases and the preliminary design meet the requirements of PDC (GDC 2) and PDC 52 (GDC 60) with respect to protection against natural phenomena and capability to control gaseous radioactive releases and meet the guidelines of RGs 1.29, 1.52, and 1.140 concerning seismic classification and design, testing and maintenance criteria for atmospheric cleanup, and normal ventilation exhaust system air filtration and adsorption units. The system design criteria and bases and preliminary design are, therefore, acceptable.

9.6.3 Reactor Service Building HVAC Systems

9.6.3.1 Fuel-Handling Area and Refueling Communications Center HVAC Systems

The fuel-handling area HVAC system will provide filtered and conditioned air to the atmospheric areas of the reactor service building (RSB) fuel-handling areas and will be composed of safety-related and nonsafety-related components. The system will be designed to control the release of radioactivity, cool essential equipment, permit continuous routine personnel access during normal operation, and control the release of radioactivity during fuel-handling operations and accident conditions. The refueling communications center HVAC system will be a nonsafety-related system designed to provide filtered and conditioned air

to the refueling communications center, permit personnel access, and ensure operability of equipment during normal operations.

The fuel-handling area and refueling communications center HVAC system's, design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.2, "Spent Fuel Pool Area Ventilation System" (NUREG-0800). Although CRBRP will not use an open spent fuel pool, the functions of the fuel-handling area and refueling communications center HVAC systems will be identical in function to the standard LWR fuel-building HVAC system. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the proposed systems, design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the systems' capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the systems' capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, Position C.2, and RG 1.140, Positions C.1 and C.2, relating to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

(3) GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the systems' capability to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the fuel storage facility under normal and postulated accident conditions. GDC 61 is identical to CRBR PDC 53. Acceptance is based on meeting the guidance of RG 1.13, "Fuel Storage Factility Design Basis," Position C.4, as it relates to the design of the ventilation system for the spent fuel storage facility.

SRP Section 9.4.2 is directly applicable to the CRBR fuel-handling area and refueling communications center HVAC systems. Conformance with the acceptance criteria of SRP Section 9.4.2 provides the bases for concluding that the proposed systems' design criteria and bases and preliminary design are acceptable.

The HVAC system of the RSB fuel-handling area will be a minimum outside air system with a supply portion consisting of an air-handling unit, two 50% capacity supply fans, and two 50% capacity exhaust/return fans with associated ducts, dampers, valves, instrumentation, and controls. The supply portion will be for the most part nonsafety related. Two 100% capacity RSB cleanup filter/fan units will be provided to filter and discharge air from the fuel-handling area during normal and accident conditions. The cleanup filter/fan unit and associated ductwork and dampers will be safety related.

During normal operation, air from the outside air intake structure will be directed to an air-handling unit and then to the inlet of the RSB supply fans. Heating and cooling coils and filters in the air-handling unit will provide clean air at the appropriate temperature for equipment operability and personnel access during normal operation. The RSB supply fans will distribute the air to the atmospheric areas in the RSB fuel-handling area through the supply branch ducts at a flow rate calculated to dilute inert and radioactive gas leakage to levels acceptable for occupancy. The RSB exhaust/return fans will draw air from the ventilated areas and recirculate most of the flow to the air-handling unit inlet duct. A portion of the exhaust/return fan discharge flow will be ducted to the inlet of the RSB cleanup filter/fan units where the air will be filtered before being discharged to the atmosphere. The exhaust/return fan inlet vane dampers will be controlled to maintain a negative pressure in the RSB fuel-handling area during normal operation except when the railroad door is open. Pressure will be monitored in selected areas with alarms provided locally and in the control room for loss of negative pressure.

The refueling communications center will be a separate compartment within the fuel-handling area and will be provided with a separate air conditioning unit. The air conditioning unit filters and cooling coils will maintain the appropriate conditions for equipment operability and personnel occupancy during normal operation. Air from the refueling communications center will be recirculated through the air conditioning unit, which will be equipped with manual dampers that will be controlled to maintain positive pressure in the room. Outleakage of air will be to the fuel-handling area. A makeup air supply duct will be connected to an outside air intake structure.

During open hatch refueling operations, the outside air dampers to the RSB will be closed until negative pressure conditions in the RSB and reactor containment building are regained by use of the RSB cleanup filter/fan units, the RSB exhaust/return fans, and the annulus filter system fans. After negative pressure conditions are attained, the RSB supply dampers will be opened and the fuel-handling area HVAC system will operate as for normal operating conditions. The railroad door will not be opened during refueling operations.

During accident conditions, the RSB fuel-handling area HVAC system will be switched automatically to a recirculation mode, the outside air supply damper will be closed, and the normal supply and exhaust air flow to the areas where the accident has occurred will be isolated. The cleanup filter/fan unit will draw exhaust air direct from the areas where the accident has occurred by the use of a separate exhaust duct system. A portion of the cleanup filter/fan unit discharge will be exhausted outside the building, the remainder will be returned to the RSB fuel-handling area operating floor. The cleanup filter/fan unit with the associated ducts, valves, and dampers will maintain a greater negative pressure within the affected areas than the rest of the building to limit the spread of airborne radioactivity or sodium combustion products. Redundant cleanup filter/fan units and redundant filtration paths will be provided to ensure that a single active failure will not prevent emergency filtration. In addition, the automatic roll filters will be provided with high differential pressure alarms to indicate failure of the filter advance mechanism. Failure of an operating filter fan will initiate an alarm and start the redundant filter fan. Although operation of the system in the emergency

mode could result in loss of ventilation cooling to essential equipment in the reactor service building, these essential components will be provided with separate safety-related unit coolers (see Section 9.6.3.3 of this SER).

As discussed in Section 6.2.6 of the PSAR, the emergency filtration system will be designed and tested in accordance with the guidelines of RG 1.52. The filter fan units and associated ductwork, dampers, and controls will be designed to Quality Group C requirements. Quality Group C ductwork and redundant dampers will be provided to selectively isolate the ventilation supply to potentially contaminated areas and realign the exhaust path. The cleanup filter fans will be powered from separate divisions of Class 1E power supplies and will be operable on loss of offsite power. Redundant radiation monitors will be provided in the exhaust ducts of potentially contaminated areas.

As discussed above, the proposed RSB fuel-handling area and refueling communication center HVAC systems satisfy the requirements of PDC 52 (GDC 60), relating to the acceptance criteria of RG 1.52, and PDC 53 (GDC 61) and meet the guidelines of RG 1.13, Position C.4. Because the proposed system satisfies the guidelines of RG 1.52 for cleanup filtration system, it also satisfies the guidelines of RG 1.140, Positions C.1 and C.3.

The components of the RSB fuel-handling area and refueling communication center HVAC systems will be located within the flood-protected, tornado-hardened, and missile-protected seismic Category I portion of the reactor service building. The outside air intake and exhaust vents will be located in missile-protected structures. The redundant safety-related components will be located in separate compartments and all essential components and ducts will be seismic Category I. The remaining portions of these HVAC systems will be nonseismic Category I but will be designed so that failure of the components or ductwork in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (applicants' seismic Category III) (see Section 3.7 of this SER).

Thus, the proposed RSB fuel-handling area and refueling communications center HVAC systems satisfy the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2.

The staff concludes that the proposed RSB fuel-handling area and refueling communications center HVAC systems' design criteria and bases and preliminary design are in conformance with the requirements of PDC 2, 52, and 53 as they relate to protection against natural phenomena, radioactive releases, and fuel storage radioactivity control and the guidelines of RG 1.13 (Position C.4), RG 1.29 (Positions C.1 and C.2), RG 1.52 (Position C.2), and RG 1.140 (Positions C.1 and C.2) relating to protection against radioactive releases, seismic classification, and system design for radioactivity control and are, therefore, acceptable.

9.6.3.2 Cell Atmosphere Processing System Exhaust and Unit Cooler System

The nonsafety-related cell atmosphere processing system (CAPS) exhaust and unit cooler system will be designed to maintain a negative pressure in the reactor service building (RSB) compartments housing CAPS equipment and to provide cooling for three of these compartments. Provisions will be incorporated in the system for discharging CAPS effluent, for purging inerted RSB cells, for

purging CAPS compartments, and for isolating individual CAPS compartments and the main exhaust header to contain abnormal radioactive gaseous leakage. The system will be designed to operate during normal operation and maintenance conditions. This system is described in Section 9.6.3 of the PSAR, Amendment 59. Supplemental information was provided by the applicants in a meeting on December 13, 1982. The applicants have stated that this supplemental information will be incorporated into a future PSAR amendment.

Because the CAPS equipment will contain and process radioactive gases and because of the similarity to LWR gaseous radwaste systems, the CAPS exhaust and unit cooler system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria and bases and preliminary design meet applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, "Seismic Design Classification," Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the guidance of RG 1.140, Positions C.1 and C.2.

SRP Section 9.4.3. is directly applicable to the CRBR CAPS exhaust and unit cooler system. Conformance with the acceptance criteria of SRP Section 9.4.3 provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

Three unit coolers will serve three of the CAPS compartments. The RSB filter exhaust fans will maintain a negative pressure in the CAPS equipment compartments during normal operation. Air will be supplied to the compartments by infiltration from other atmospheric areas in the RSB fuel-handling area. The CAPS exhaust duct will be connected to the two 100% capacity, redundant RSB exhaust filtration units upstream of the first safety-related isolation damper. Radiation monitors will be located in the exhaust duct downstream of the equipment compartments. An abnormal radioactivity signal from the monitor will shut the associated quick-closing valve in the exhaust duct, isolating the compartment ducting from the main exhaust headers. Any excessive radioactivity in the effluent from the CAPS cells will also be detected by the safety-related radiation monitors provided in the RSB exhaust filtration system. CAPS exhaust will then be automatically isolated from the filtration unit exhaust fans by safety-related dampers. The RSB exhaust cleanup units filtration system will be designed and tested in accordance with the guidelines of RG 1.52 (see Section 9.6.3.1 of this SER). The instrumentation for inert gas receiving and processing system will determine which compartment has a radiation leak; the

noncontaminated cells can then be manually realigned to the exhaust fan inlet (see Section 9.5 of this SER). Only one exhaust fan will operate to maintain negative pressure during normal operation. In the event of loss of normal power, the individual cells will be isolated by means of the quick-closing valve, and the main exhaust header will be isolated from the RSB exhaust filtration system by safety-related isolation valves in the RSB HVAC system.

A separate branch deinerting exhaust duct, connected to the main exhaust header, will be provided with capped inlets at each level below el 816 ft. This duct will be used, in conjunction with a portable filter/fan unit and flexible ducting, to purge inerted RSB compartments. A radiation monitor will be provided at the discharge side of the portable filter/fan unit to stop the purging operation if high radiation levels are sensed. Connections also will be provided at the main exhaust header for effluent from CAPS and bleedoff from the inert gas receiving and processing system (see Section 9.5 of this SER). These effluents will be monitored before entering the connection to the CAPS exhaust system.

Three unit coolers and associated ducting and manual dampers will cool three potentially contaminated CAPS compartments. Ducting connections will be provided from these compartments to the CAPS exhaust system; negative pressure will be maintained by the CAPS exhaust system. The unit coolers will not be supplied from a Class 1E power supply. Cooling water to the unit cooler will be supplied from the normal chilled water system.

On the basis of the above, the proposed CAPS exhaust and unit cooler system satisfies the requirements of PDC 52 (GDC 60) as they apply to nonsafety-related HVAC systems. Because the exhaust system will use the filters of the RSB exhaust cleanup unit filtration system, the guidelines of RG 1.140 are satisfied in that the RSB HVAC system filters satisfy the appropriate guidelines of RG 1.52.

The components of the CAPS exhaust and unit cooler system will be located within the flood-protected, tornado-hardened, and missile-protected seismic Category I portion of the reactor service building. The system will be nonseismic Category I but will be designed so that failure of the components or ductwork in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (applicants' seismic Category III) (see Section 3.7 of this SER). Thus, the proposed CAPS exhaust and unit cooler system satisfies the requirements of PDC 2 (GDC 2) relating to nonsafety-related systems and the guidelines of RG 1.29, Position C.2.

The staff concludes that the proposed CAPS exhaust and unit cooler system's design criteria and bases and preliminary design are in conformance with the requirements of PDC 2 and 52 as they relate to protection against natural phenomena and radioactive releases and the guidelines of RG 1.140, Positions C.1 and C.2, and are, therefore, acceptable.

9.6.3.3 Reactor Service Building Cooling Systems

The reactor service building (RSB) cooling systems will be individual safety-related air recirculation unit coolers for the following plant areas: reactor service building and annulus filter cells, electrical equipment cells, contain-

ment cleanup pump cells, containment cleanup scrubber cell, and containment cleanup pipe chase valve areas.

The RSB cooling systems' design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the systems' design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the systems' capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions. GDC 4 is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 17, "Electric Power Systems," as it relates to ensuring proper functioning of the essential electric power system. GDC 17 is identical to CRBR PDC 15. Acceptance is based on meeting the guidance of NUREG/CR-0660, Recommendations A.2 and C.1, relating to the protection of essential electrical components from failure resulting from the accumulation of dust and particulate materials.

(4) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the systems' capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, Position C.1, and RG 1.140, Positions C.1 and C.2, as they relate to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.5 is applicable to the review of the CRBR RSB cooling systems. Conformance with the acceptance criteria of SRP Section 9.4.5, except as noted, provides the bases for concluding that the proposed systems' design criteria and bases and preliminary design are acceptable.

The RSB cooling systems will provide air cooling for individual safety-related equipment cells to ensure the fuctioning of essential equipment during normal and emergency conditions. Individual full-capacity unit coolers, each with a disposable filter, chilled water cooling coil, and fan, will be provided for the two redundant RSB filter unit cells, annulus filter cell, electrical equipment cells, containment cleanup pump cells, containment cleanup scrubber cells,

and containment cleanup pipe chase valve areas. Cooling water to the coils will be supplied by the normal chilled water system under normal conditions and automatically from the emergency chilled water system under accident conditions. The unit coolers will be located within the cells they will serve and will start automatically on start of their associated equipment. Ventilation requirements for the cells will be provided by the safety-related HVAC system serving the RSB fuel-handling area.

Each unit cooler will be powered by the associated redundant Class 1E emergency power supply for the equipment it will serve. Failure of a unit cooler will automatically start the redundant cell cooler. Thus, proper cooling for essential equipment will be maintained in the event of a single active failure. The unit coolers will be safety related, seismic Category I, Quality Group C, and will be located in the seismic Category I, tornado-missile and flood-protected reactor service building (see Sections 3.4.1 and 3.5.2 of this SER). As described above, the proposed RSB cooling systems' design criteria satisfy the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2, concerning seismic classification, and PDC 5 (GDC 4) concerning the capability to maintain a proper environment for equipment under all operating conditions. The requirements of PDC 15 (GDC 17) and the guidelines of NUREG/CR-0660 are not applicable because the systems will not serve the diesel generators. The requirements of PDC 52 (GDC 60) and the guidelines of RGs 1.52 and 1.140 also are not applicable because the systems will not be used to control radioactive gases.

On the basis of the above, the staff concludes that the proposed RSB cooling systems' design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) and PDC 5 (GDC 4) with respect to protection against natural phenomena and capability of providing a proper operating environment for essential equipment and the guidelines of RG 1.29 concerning their seismic classification. The proposed systems' design criteria and bases and preliminary design are, therefore, acceptable.

9.6.3.4 Radioactive Waste Treatment Area HVAC System

The radioactive waste treatment area HVAC system will provide filtered and conditioned air and filtered exhaust for the radioactive waste treatment areas during normal operation only. The system is, therefore, nonsafety related.

The radioactive waste treatment area HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on

meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.140, Positions C.1 and C.2, as it relates to the design, testing, and maintenance criteria for normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.3 is applicable to the review of the CRBR radioactive waste treatment area HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.3, except as noted, provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

The radioactive waste treatment area HVAC system will be designed to provide a suitable normal operating environment for personnel access and equipment operation in the radioactive waste treatment area, to maintain a negative pressure within decontamination areas during normal operation, and to limit the spread of radioactive materials during normal operation. It will consist of one full-capacity air-handling unit with two 50% capacity supply fans. Both supply fans normally will operate and draw outside air through the air-handling unit, which will condition the air through filters, heating coils, and cooling coils before distributing it throughtout the radwaste treatment areas. Separate nonsafety-related unit heaters also will be provided to maintain comfort in the railroad and truck areas.

The radwaste treatment areas also will be served by two 50% capacity exhaust fans and a full-capacity exhaust filter unit and fan. The filter unit will provide cleanup of potentially contaminated air before it will be released to the environment. A radiation monitor will be provided in the exhaust duct and will open a bypass damper to direct exhaust air through the filter unit on detection of abnormal radiation. An additional radiation monitor will be provided at the roof exhaust unit. Both exhaust fans normally will operate and will maintain the proper negative pressure within the radwaste treatment areas.

Failure of a supply or exhaust fan will result in stopping its associated supply or exhaust fan. Environmental conditions in the radwaste treatment areas will then be maintained by the operable exhaust and supply fan.

The radioactive waste treatment area HVAC system will serve no safety function. It will be located in the radioactive waste treatment area of the reactor service building away from essential equipment areas. Failure of the system will, therefore, not affect plant safety or result in unacceptable release of radioactivity because activity levels will be low within the radwaste areas. As described above, the proposed system's design criteria satisfy the requirement of PDC 2 (GDC 2) and the guidelines of RG 1.29, Position C.2, concerning seismic classification for nonsafety-related systems, and the requirements of PDC 52 (GDC 60) and the guidelines of RG 1.140, Positions C.1 and C.2, concerning normal ventilation exhaust filtration for gaseous radioactivity control during normal operating conditions.

On the basis of the above, the staff concludes that the proposed radioactive waste treatment area HVAC system design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) and PDC 52 (GDC 60) with respect to protection against natural phenomena and capability to control gaseous radioactivity releases and the guidelines of RGs 1.29 and 1.140 concerning seismic classification and design and testing and maintenance criteria for normal ventilation exhaust air filtration units. The proposed system's design criteria and bases and preliminary design are, therefore, acceptable.

9.6.4 Turbine Generator Building HVAC System

The turbine generator building HVAC system will provide filtered and conditioned air to the turbine generator building. The system will be nonsafety related as it will serve no safety-related equipment, and its failure will not affect safe plant shutdown or result in release of unacceptable radioactivity.

The turbine generator building HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.4, "Turbine Area Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the proposed system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.140, Positions C.1 and C.2, as they relate to design, testing, and maintenance criteria for normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

SRP Section 9.4.4 is applicable to the review of the CRBR turbine generator building HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.4 will provide the bases for concluding that the system's design criteria and bases and preliminary design are acceptable.

The turbine generator building HVAC system will be designed to maintain a suitable environment for personnel and nonessential equipment operation in those areas to be served during normal plant operation. The system will have no safety function. The system will consist of 5 supply fans, filters, ducts, dampers, 11 exhaust fans and unit coolers for the steam generator feed pump area, condensate feed pump area, and low-pressure feedwater heater area. The system will be separated from safety-related plant systems and potentially radioactive areas; therefore, failure of the system will not compromise the

operation of essential plant systems, or result in an unacceptable release of radioactivity. Thus, the proposed system's preliminary design meets the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Position C.2, concerning seismic classification of nonsafety-related systems. Further, the requirements of PDC 52 (GDC 60) and the guidelines of RG 1.140 are not applicable, as the system will not provide a gaseous radioactivity control function and will not be located in a radioactivity area.

On the basis of the above, the staff concludes that the proposed nonsafety-related turbine generator building HVAC system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) with respect to the need for protection against natural phenomena because its failure will not affect safety system functions or result in release of radioactive material, and the guidelines of RG 1.29 concerning its seismic classification. The system's design criteria and bases and preliminary design are, therefore, acceptable.

9.6.5 Diesel Generator Building HVAC System

The diesel generator building HVAC system will provide conditioned air for the various essential and nonessential equipment areas of the diesel generator building. It will consist of the safety-related diesel generator building emergency HVAC system, nonsafety-related diesel generator building nonessential HVAC system, and safety-related diesel generator rooms HVAC system. The diesel generator building emergency HVAC system, which includes the diesel generator building division 1 and 2, Class 1E switchgear rooms HVAC system, and the diesel generator building nonessential HVAC system, which includes the diesel generator building non-Class 1E switchgear rooms and motor generator rooms HVAC system, and diesel generator building motor generator sets unit cooler system, are discussed in Section 9.6.1 of this SER. Ventilation for the division 3 diesel generator Class 1E switchgear is discussed in Section 9.6.6 of this SER. The diesel generator rooms HVAC system is discussed below.

The diesel generator rooms HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions. GDC 4 is identical to CRBR PDC 5 except that

PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 17, "Electric Power Systems," as it relates to ensuring proper functioning of the essential electric power system. GDC 17 is identical to CRBR PDC 15. Acceptance is based on meeting the guidance of NUREG/CR-0660, Recommendations A.2 and C.1, relating to the protection of essential electrical components from failure resulting from the accumulation of dust and particulate materials.

(4) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, Position C.1, and RG 1.140, Positions C.1 and C.2, as they relate to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.5 is applicable to the review of the proposed CRBR diesel generator rooms HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.5, except as noted, will provide the bases for concluding that the system's design criteria and bases and preliminary design are acceptable.

The diesel generator rooms HVAC system will be designed to maintain a suitable environment for essential equipment operation during normal and emergency operating modes when the diesel is required. It will consist of one full-capacity nonessential heating and ventilating unit and two 50% capacity essential emergency supply fans for each of the three diesel generator rooms. In addition, a continuously operating full-capacity essential exhaust fan will be provided for the day tank cell associated with each diesel. The day tank cell exhaust fan also will exhaust air from its associated diesel fuel oil transfer pump cell. The heating and ventilation unit will be nonsafety related as it only will provide suitable temperature for personnel and equipment when the diesel is not operating. Its failure will not affect proper diesel generator operation. The emergency supply fans will start automatically on actuation of the associated diesel. The temperature in each diesel generator room will then be controlled bypassing recirculated air through a cooling coil. Water to the cooling coil will be supplied by the emergency plant service water system. Air will be relieved from its diesel generator rooms through separate exhaust dampers.

Each of the diesel generator room emergency supply fans and day tank cell exhaust fans will be powered from the emergency (Class 1E) power supply corresponding to the diesel it will serve. Thus, operation of at least two diesel generators will be ensured in the event of a single failure. The entire diesel generator room HVAC system will be safety related, seismic Category I, Quality Group C, with the exception of the nonsafety-related heating and ventilation unit which will be designed to maintain its integrity in a safe shutdown earthquake (SSE) so that its failure will not affect essential equipment. The entire diesel generator room HVAC system will be located in the seismic Category I, flood- and tornado-protected room housing its associated diesel generator (refer

to Sections 3.4.1 and 3.5.2 of this SER). The outside air intakes and exhaust openings will be provided with tornado-missile hardened structures. The bottom of the outside air intake will be 19 ft above grade, which, although meeting the 20-ft criterion of the SRP, in the staff's judgment, will be sufficient to provide protection against airborne particulate material accumulation within the diesel generator room. Further, air supplied to the diesel generator room normal heating and ventilating unit and the emergency supply fans will be filtered through replaceable filters to provide additional protection against dust. Diesel engine combustion air will be drawn through a separate outside air intake. As described above, the proposed diesel generator rooms HVAC system's design criteria and bases and preliminary design satisfy the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2, concerning seismic classification for safety-related and nonsafety-related portions of the system; the requirements of PDC 5 (GDC 4) concerning the capability to maintain a proper equipment environment under all operating conditions; the requirements of PDC 15 (GDC 17); and the guidelines of NUREG/CR-0660, Recommendations A.2 and C.1, concerning ensuring proper function of essential electrical equipment by providing protection against dust. The requirements of PDC 52 (GDC 60) and the guidelines of RGs 1.52 and 1.140 will not be applicable as the system will not provide a control of gaseous radioactivity function and will not be located in a radioactivity area.

On the basis of the above, the staff concludes that the proposed diesel generator rooms HVAC system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2), PDC 5 (GDC 4), and PDC 15 (GDC 17) with respect to protection against natural phenomena, its capability to provide a proper operating environment for essential equipment and protection of essential electrical equipment, and meet the guidelines of RG 1.29 and NUREG/CR-0660 concerning its seismic classification and protection against dust accumulation. The proposed system's design criteria and bases and preliminary design are, therefore, acceptable.

9.6.6 Steam Generator Building HVAC System

The steam generator building HVAC system will provide filtered and conditioned air to the various essential and nonessential equipment area in the steam generator building. It will consist of the safety-related steam generator and auxiliary bay HVAC system, the steam generator building intermediate bay HVAC systems, and the nonsafety-related steam generator building maintenance bay HVAC system.

The steam generator building HVAC system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions. GDC 4 is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 17, "Electric Power Systems," as it relates to ensuring proper functioning of the essential electric power system. GDC 17 is identical to CRBR PDC 15. Acceptance is based on meeting the guidance of NUREG/CR-0660, Recommendations A.2 and C.1, relating to the protection of essential electrical components from failure resulting from the accumulation of dust and particulate materials.

(4) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. GDC 60 is identical to CRBR PDC 52. Acceptance is based on meeting the applicable guidelines of RG 1.52, Position C.1, and RG 1.140, Positions C.1 and C.2, as they relate to design, testing, and maintenance criteria for atmospheric cleanup system and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants.

Except for the differences noted above, SRP Section 9.4.5 is applicable to the review of the CRBR steam generator building HVAC system. Conformance with the acceptance criteria of SRP Section 9.4.5, except as noted, provides the bases for concluding that the proposed system's design criteria and bases, and preliminary design are acceptable.

The steam generator and auxiliary bay HVAC system and steam generator building intermediate bay HVAC system will be designed to maintain a suitable environment in the steam generator loop cells, auxiliary bay cells, intermediate bay intermediate heat transport system (IHTS) cells, and other intermediate bay areas for personnel access during normal operation and proper equipment operation during all operating modes including emergencies. The steam generator and auxiliary bay HVAC system will consist of three redundant trains each with an air-handling unit and duct system containing two full-capacity redundant supply fans. One air-handling unit will serve each of the three redundant combined loop cell trains (steam generator loop cells, IHTS cells, and auxiliary bay cells). The steam generator building intermediate bay HVAC system will consist of two full-capacity redundant trains, each with an air-handling unit and duct system containing two full-capacity redundant supply fans. Each train will serve nonessential and essential redundant equipment areas of the intermediate bay. Each of the air-handling units will contain replaceable filters and cooling coils to condition the outside supply air. Two full-capacity redundant exhaust fans and duct systems also will be provided for each of the above three combined loop cell area trains and each of the two intermediate

bay trains. Nonsafety-related unit heaters also will be provided for the system, but will not be essential for proper equipment operation, since they will serve only to provide an environment for personnel. Chilled water to the cooling coils will be supplied by the nonsafety-related normal chilled water system. Under accident conditions when normal chilled water is not available, both supply and exhaust fans for each train will operate to provide the increased air flow necessary to compensate for loss of chilled water to ensure sufficient essential equipment cooling. Failure of any single fan will result in loss of cooling to only one of the redundant system trains and thus adequate ventilation will be provided to essential equipment for safe shutdown.

The system air-handling units will draw outside air and mix it with air returned from the building cells by the exhaust fans. A portion of the exhaust also will be normally discharged to the atmosphere. Failure of an exhaust damper in the closed position will affect air flow to only one of the redundant system trains and thus adequate ventilation will be provided to essential equipment for safe shutdown. In addition, alarms will be provided to alert the operator to take action to manually open the closed damper. Sufficient time will be available for this action before unacceptable conditions are reached in the affected loop cells.

The system will include one full-capacity safety-related unit cooler for each of the two redundant motor-driven auxiliary feedwater pump cells, two redundant emergency chilled water system equipment rooms, and three instrumentation and control panel cells. Two full-capacity safety-related redundant unit coolers will be provided for the turbine-driven auxiliary feedwater pump cell. The unit coolers will start on initiation of their associated pump and chillers. Separate nonsafety-related outside air supply fans, filters, and electric duct heaters will be provided for the motor-driven and turbine-driven auxiliary feedwater pump cells to maintain a suitable environment for personnel during normal operation. A nonsafety-related unit cooler will be provided for the nonessential primary sodium storage tank cell, and nonsafety-related heating coils will be provided for maintaining a suitable environment for personnel in the intermediate bay areas.

The division 3 diesel generator Class 1E switchgear will be located in the steam generator building. It will be served by a separate full-capacity supply air cooling system, which will draw outside air. The fans will be powered from the associated division 3, Class 1E power source.

Redundant safety-related components in the steam generator building HVAC system will be powered from separate emergency (Class 1E) power supplies. Thus, operation of at least one train of essential equipment will be ensured in the event of a single failure. Safety-related portions of the system will be seismic Category I, Quality Group C. The nonsafety-related portions will be seismically supported to retain their integrity in an SSE so that their failure will not affect essential equipment. The system will be housed within the seismic Category I, flood- and tornado-protected steam generator building (refer to Sections 3.4.1 and 3.5.2 of this SER). The outside air intake and exhaust openings will be provided with tornado-missile-hardened structures. Chilled water to the safety-related unit coolers will be supplied from redundant trains of the safety-related emergency chilled water system. As described above, the proposed steam generator building HVAC system's design criteria satisfy the

requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2, concerning seismic classification for safety-related and nonsafety-related portions of the system, and the requirements of PDC 5 (GDC 4) concerning the capability to maintain a proper environment for essential equipment under all operating conditions. The requirements of PDC 15 (GDC 17) and the guidelines of NUREG/CR-0660 are not applicable as the system will not serve the diesel generators. The requirements of PDC 52 (GDC 60) and the guidelines of RGs 1.52 and 1.140 are also not applicable because the system will not provide a control of gaseous radioactivity function.

On the basis of the above, the staff concludes that the proposed steam generator building HVAC system design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) and PDC 5 (GDC 4) with respect to protection against natural phenomena and the capability to provide a proper operating environment for essential equipment, and meet the guidelines of RG 1.29 concerning its seismic classification. The system's design criteria and bases and preliminary design are, therefore, acceptable.

9.7 Chilled Water Systems

9.7.1 Normal Chilled Water System

The normal chilled water system (NCWS) will be designed to provide chilled water for all nonsafety-related air conditioning and recirculating gas cooling system cooling coils and primary cold-trap heat exchangers located throughout the plant. Chilled water will satisfy area temperature requirements and remove heat from confined areas and equipment during normal plant operation.

The NCWS design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.2.1, "Station Service Water System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria and bases and preliminary design meet applicable portions of GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions. SRP Section 9.2.1 is directly applicable to the CRBR normal chilled water system. Conformance with the acceptance criteria of SRP Section 9.2.1 provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

The NCWS will be a nonsafety-related, nonseismic Category I system with the exception of those portions located in the reactor containment building (RCB) that will be designed to Quality Groups B and C and seismic Category I requirements. Cooling loads to be served by the NCWS will be the unit coolers and the coolers and the air-handling and air conditioning units located in the turbine generator building, control room building, steam generator building, reactor service building, reactor containment building, plant service building, and shop and warehouse building. The NCWS also will supply chilled water to the safety-related air conditioning and recirculating gas cooling unit coolers and

cooler units during normal plant operation. In addition, the NCWS will have one intermediate Dowtherm J loop to provide cooling to the primary cold-trap NaK cooler (located in the RCB). The Dowtherm J will be cooled via a heat exchanger by normal chilled water. (See Section 9.7.4 for further discussion.)

The NCWS will consist of six 20% capacity electric motor-driven mechanical refrigeration water chillers, six 20% capacity chilled water circulation pumps, an air separator, an expansion tank, and associated piping, valves, and instrumentation. The NCWS will not be required to cool any safety-related equipment during accident conditions. The safety-related equipment that the NCWS will serve during normal plant operations will be isolated during emergency conditions. All required cooling loads in the event of a loss of normal chilled water will be provided by the emergency chilled water system (see Section 9.7.2 for further discussion). Failure of the system will not affect the safe shutdown of the plant or result in an unacceptable release of radiation.

Those portions of the system entering areas or components possibly containing sodium or NaK will be provided with isolation valves that automatically will close on a signal from water leak detectors located in either the area or the component. (See Section 9.7.3 for further discussion.)

On the basis of its review, the staff concludes that the proposed design criteria and bases and preliminary design for the normal chilled water system satisfy the requirements of PDC 2 with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2, and are, therefore, acceptable.

9.7.2 Emergency Chilled Water System

The emergency chilled water system (ECWS) will be designed to provide sufficient chilled water to essential coolers and HVAC units in the event of an accident that results in a loss of the normal chilled water system or a loss of offsite ac power.

The emergency chilled water system's design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.2.1, "Station Service Water System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria and bases and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to structures housing the system and the system's capability to withstand the effects of natural phenomena. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29.

(2) GDC 44, "Cooling Water," as it relates to transferring heat from structures, systems, and components important to safety to an ultimate heat sink. GDC 44 is identical to CRBR PDC 38, "Additional Cooling Systems," except that PDC 38 does not require that CRBR additional cooling systems be

capable of removing reactor decay heat. Otherwise, the acceptance criteria of SRP Section 9.2.1 to meet GDC 44 are directly applicable to the CRBR emergency chilled water system.

(3) GDC 45, "Inspection of Cooling Water System," as it relates to design provisions to permit inservice inspection of safety-related components and equipment. GDC 45 is identical to CRBR PDC 39 except that "additional cooling systems" has been substituted for "cooling water system." The acceptance criteria of SRP Section 9.2.1 to meet GDC 45 are directly applicable to the CRBR emergency chilled water system.

(4) GDC 46, "Testing of Cooling Water Systems," as it relates to design provisions to permit operational functional testing of safety-related systems and components. GDC 46 is comparable to CRBR PDC 40; the latter criterion, however, is more general so as to include the variety of cooling system functions at CRBR. The intent of the two criteria is identical; therefore, the acceptance criteria of SRP Section 9.2.1 to meet GDC 46 are directly applicable to the CRBR emergency chilled water system.

SRP Section 9.2.1 is directly applicable to the review of CRBR emergency chilled water system. Conformance with the acceptance criteria of SRP Section 9.2.1 provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

Emergency chilled water will be supplied by two independent loops, each loop capable of meeting the total chilled water demand during an accident. Each loop will contain one electric motor-driven, mechanical refrigeration water chiller, circulation pump, air separator, expansion tank, piping, valves, and instrumentation. The system will provide chilled water for the chilled water coils of the air handlers, cooler units, and unit coolers that will be located in the control building, reactor service building, reactor containment building, and steam generator building. The system will be designed to operate during accident conditions without loss of function. The ECWS will be designed to seismic Category I, Quality Group C, requirements and will be powered by a Class 1E ac power supply. All equipment will be located in the seismic Category I, tornado-missile- and flood-protected steam generator building. On loss of the normal chilled water supply, the ECWS isolation valves located in the ECWS header will isolate the NCWS from the ECWS, and the emergency plant service water system and the ECWS will automatically start. Thus, the requirements of PDC 2 (GDC 2) and PDC 38 (GDC 44) and the guidelines of RG 1.29 have been satisfied.

Those portions of the system entering areas or components possibly containing sodium or NaK will be provided with isolation valves that will automatically close on receipt of a signal from water leak detectors located in either the area or the components (see Section 9.7.3 for further discussions.)

The ECWS will not operate during normal plant operations. However, the system, including all active components, will be tested periodically during normal plant operation or scheduled shutdowns. Inservice inspections will be conducted according to ASME Code, Section XI. Thus, the requirements of PDC 39 (GDC 45) and PDC 40 (GDC 46) will be satisfied.

On the basis of its review, the staff concludes that the proposed ECWS design criteria and bases and preliminary design meet the requirements of CRBR PDC 2, 38, 39, and 40 with respect to the system's protection against natural phenomena, capability for transferring the required heat loads, inservice inspection and functional testing, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.7.3 Prevention of Sodium-Water or NaK-Water Interactions

Since portions of the normal and emergency chilled water systems will provide chilled water to components where sodium-water or NaK-water interactions may occur, both systems will incorporate features to maintain three barriers (three passive or two passive and one active) between water and sodium or NaK. The barriers will be the chilled water pipe wall, the cell boundary, the sodium or NaK pipe wall, and guard piping or drain pans and spray shields.

The facility's design criteria for the prevention of sodium-water or NaK-water interactions are based on the requirements of CRBR PDC 4, "Protection Against Sodium and NaK Reactions." PDC 4 does not have a GDC counterpart since the use of sodium or NaK as coolants is unique to the CRBR.

To prevent water contact with sodium or NaK equipment through leaks in the piping, the following measures have been incorporated into the system design:

(1) Chilled water piping will not be routed through areas and cells containing radioactive sodium or NaK.

(2) Where chilled water will be routed through areas containing nonradioactive sodium or NaK, the barriers between the chilled water and sodium or NaK will be (a) the chilled water pipe wall, (b) guard piping or drain pans and spray shields, and (c) the sodium or NaK pipe walls.

(3) All areas and cells containing chilled water piping will be provided with a floor drain system (see Section 9.15 for further discussion). For areas where sodium or NaK will be present (or areas adjacent to these), water leak detectors will be provided.

(4) Appropriate isolation valves in the normal and emergency chilled water systems will close on receipt of a signal from the water leak detectors.

(5) Where liquid metal is the fluid to be cooled, a secondary cooling loop of Dowtherm J will be provided to separate the sodium or NaK and the chilled water. The working fluid in the secondary loop will not be reactive to either water or liquid metal. The pressure in the secondary loop will be higher than in both the water loop and the loop containing liquid metal to prevent contact between water and liquid metal during a heat exchanger tube leak. (See Section 9.7.4 for further discussion.)

To prevent moisture or water carryover from the HVAC system cooling coils resulting from water condensation or coil tube leakage, every HVAC unit used to cool areas and cells containing sodium or NaK will be provided with a moisture eliminator. The cooling coils and moisture eliminators will be located above drain pans in the HVAC unit casings. Two drain connections will be provided for

each drain pan. The overflow drains will have leak detectors; activation of these detectors will indicate failure of the primary drain or excessive leak rate caused by coil failure. The same detectors will initiate an alarm, stop the affected HVAC unit fan, and close the chilled water coil isolation valves.

On the basis of its review, the staff concludes that the proposed design criteria and bases and preliminary design for the prevention of sodium-water or NaK-water interactions meet the requirements of CRBR 4 and are, therefore, acceptable.

9.8 Liquid Metal Impurity Monitoring and Analysis

9.8.1 Introduction

The liquid metal impurity monitoring and analysis system will consist of three subsystems: (1) primary heat transport system (PHTS) sodium characterization, (2) ex-vessel storage tank (EVST) sodium characterization, and (3) intermediate heat transport system (IHTS) sodium characterization. These subsystems will be designed for automatic, online impurity monitoring and for sample collection of aliquots of sodium (or equilibration devices of particulate samples) for laboratory analysis. These sample streams will be monitored, and samples analyzed, to verify that the circulating sodium meets the allowable oxygen and hydrogen levels. Analytical results from these subsystems will provide information for the operation of the sodium purification units of the auxiliary liquid metal system (ALMS). All components and piping of the system in contact with sodium will be constructed of type 304 austenitic stainless steel. This includes sampling inserts in the multipurpose sampler in the sodium-sampling package. Argon cover gas sampling and monitoring piping and components will be constructed of type 304 stainless steel.

The primary sodium-sampling package (PSSP) of the PHTS will provide the following information pertinent to primary coolant chemistry control:

(1) total sodium impurity levels
(2) oxygen, hydrogen, and carbon activities
(3) tritium level
(4) fission product level
(5) corrosion product level
(6) particulate impurity levels
(7) level of other impurities as considered necessary

The intermediate heat transport system (IHTS) will be a three-loop system, sampled and monitored by three intermediate sodium characterization packages (ISCPs), each containing a plugging temperature indicator module and a multipurpose sampler. The ISCP inlet will be connected to the outlet of each IHTS cold-trap pump, and the return line will be connected to the return line of the IHTS loop. Sampling monitoring can be performed only when the IHTS cold-trap is functioning. Since the sodium in the IHTS loops will not be radioactive, the ISCP is not designed for remote operation.

9.8.2 Areas of Review and Acceptance Criteria

There is no standard review plan (SRP) equivalent to that for the light-water reactors (LWRs) that can be used for reviewing the liquid metal impurity monitoring and analysis system. However, SRP Section 5.4.8 (reactor cleanup system (BWR)) can be used as the guideline for reviewing the CRBR liquid metal impurity monitoring and analysis system. Some of the areas of review and acceptance criteria defined in SRP Section 5.4.8 are directly applicable to the CRBR review.

The following specific areas have been reviewed in the applicants' PSAR:

(1) design of components, design features that influence system availability and reliability, and interconnections with the reactor primary sodium (monitoring of chemical impurities and fission products and their analysis have been considered)

(2) quality group and seismic design criteria

(3) instrumentation and process controls provided to ensure proper system operation and system isolation

The acceptance criteria for the liquid metal impurity monitoring and analysis system are based on meeting the relevant requirements of the following CRBR PDC:

(1) PDC 4 as it relates to the protection against sodium reactions
(2) PDC 12 as it relates to reactor boundary integrity
(3) PDC 34 as it relates to reactor and intermediate coolant and cover gas purity control

9.8.3 Review Evaluation

9.8.3.1 Primary Heat Transport System Sodium Characterization Subsystem

The primary heat transport system (PHTS) sodium characterization subsystem, described in Section 9.3.2.1 of the PSAR, will provide a continuous sampling capability for the primary sodium from the overflow vessel or the storage vessel tanks. The piping will be designed to accept sodium from the discharge side of the makeup pumps. The flow rate through the sampling loop will be controlled at 10 gpm and returned either to the primary sodium overflow vessel or the primary sodium storage vessel tanks, whichever will be the source of the sample. The sampling loop will be equipped with a sodium-plugging temperature indicator (PTI). The PTI is a device to determine the saturation temperature of impurities dissolved in the sodium. The applicants are aware that the results obtained by the PTI will not be specific for a given impurity, but will give an indication of impurity levels which are consistent with the saturation temperature. The specific impurity concentration will be determined by chemical analysis.

In parallel with the primary PTI will be the primary sodium sampling package (PSSP), which will provide samples for several types of laboratory analyses. A portion of the primary sodium sample stream can be diverted to the multipurpose samplers (MPSs), which

(1) will provide a sodium sample for laboratory analyses of impurities

(2) can be used to expose foils and/or wires of selected materials which will equilibrate with oxygen, hydrogen, or carbon dissolved in the sodium and which can be analyzed in the laboratory

(3) will filter known quantities of sodium to provide a measure of particulate impurities in the sodium

The PHTS sodium characterization subsystem will perform no control functions, but analytical results will be used to indicate out-of-range conditions and also will be used by the auxiliary liquid metal system to determine primary sodium cold-trap operational requirements. This subsystem will not be required for the safe shutdown of the reactor. Sodium leak detectors will signal the release of sodium. Piping and components of this subsystem will be designed to accepted industrial and nuclear standards.

9.8.3.2 Intermediate Heat Transport System Sodium Characterization Subsystem

The sodium in this subsystem will not be radioactive; therefore, hands-on manual operation is planned. Continuous operation of this subsystem will not be required for the operation of the IHTS or the leak detection instrumentation. It also will not be required for the operation of the IHTS during an emergency. Isolation valves will be provided in case of piping or component failure. This subsystem will perform no control functions, but analytical results will be used to indicate out-of-range conditions, and also will be used by the auxiliary liquid metal system (ALMS) to determine intermediate sodium cold trap operational requirements.

9.8.3.3 Ex-Vessel Storage Tank Sodium Characterization Subsystem

This subsystem will not be required for the operation of the ex-vessel storage tank (EVST) during an emergency. Isolation valves will be provided to separate failed components from this sodium-sampling loop piping. Sodium leak detectors will be provided to separate failed components from the sodium-sampling loop piping. Sodium leak detectors will be provided on bellows and seal valves. The cells containing this equipment will have sodium aerosol leak detectors to signal the release of sodium to the inerted cell.

If the EVST PTI is being repaired or has malfunctioned, monitoring of the sodium impurity level can be accomplished by using the EVST sodium sampling packages (SSPs) and performing sodium impurity analysis. This subsystem will perform no control functions, but results will be used to indicate out-of-range conditions and will be used by the ALSMS to determine ex-vessel storage sodium cold trap operational requirements. Furthermore, sufficient shielding will be provided to prevent radiation overexposure of operating personnel under normal and anticipated faulted conditions.

9.8.3.4 Cover Gas Sampling and Monitoring

Primary cover gas samples will be provided by the failed fuel monitoring system. Sample connections also will be provided for sampling EVST, fuel-handling cells, and IHTS cover gas. Sample bottles whose contents will be processed to determine

impurities such as helium, hydrogen, oxygen, nitrogen, methane, and carbon monoxide will be attached to these connections. In the case of primary cover gas samples, tritium content will also be determined. Sufficient shielding will be provided to prevent radiation overexposure of operating personnel under normal and anticipated faulted conditions. This subsystem will perform no control functions and will not be required to function during an emergency, nor will it be required for the safe shutdown of the reactor.

9.8.4 Evaluation Summary

The impurity monitoring and analysis system (IMAS) will be used to aid in maintaining the sodium purity in the PHTS, IHTS, and the EVST. The staff's review has included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the systems design and operation contained in Section 9.8 of the PSAR.

The staff concludes that the proposed design of the IMAS is acceptable and meets the relevant requirements of CRBR PDC 4, 12, and 34. This conclusion is based on the following:

(1) The applicants have met the requirements of PDC 4 by having the subsystems, components, and structures containing sodium designed to limit the consequences of sodium chemical reactions resulting from a sodium spill. Special features such as inerted cells will be provided as appropriate for the subsystems and components. Means to detect sodium leakage or sodium reaction products will be provided to limit and control the extent of such reactions to protect plant personnel and to avoid undue risk to the public health and safety.

(2) The applicants have met the requirements of PDC 12 by having the primary heat transport system coolant boundary designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

(3) The applicants have met the requirements of PDC 34 by providing for reactor and intermediate coolant and cover gas purity control systems to monitor and to maintain the cover gas purity within specified design limits. These limits shall be based on consideration of (a) chemical attack, (b) fouling and plugging of passages, (c) radioisotope concentrations, and (d) detection of sodium-water reactions.

(4) The applicants have also provided for proper instrumentation and controls to monitor variables and systems over their anticipated ranges for normal operation.

(5) Proper quality groups and seismic category were assigned to subsystems and components commensurate with their design functions.

(6) The applicants have committed to a postaccident sampling capability. The applicants provided, in Appendix H of the PSAR, their evaluation and resolution of the requirements delineated in NUREG-0718. NUREG-0718 defines requirements of NUREG-0737 applicable to applications for construction permits. Item II.B.3 of NUREG-0737 is addressed in Appendix H of the PSAR.

Means to assess the degree of core damage in the event of an accident can be defined at the operating license stage. Therefore, the staff concludes that the proposed liquid metal impurity monitoring and analysis system is acceptable.

9.9 Service Water Systems

9.9.1 Normal Plant Service Water System

The normal plant service water system will be a nonsafety-related system designed to provide cooling water for the normal chilled water system chiller condensers, the secondary service closed cooling water system, and other nonessential components during normal plant operation and planned outages.

The normal plant service water system's design criteria and bases and preliminary design were reviewed in accordance with those portions of SRP Section 9.2.1, "Station Service Water System" (NUREG-0800) pertaining to nonsafety-related service water systems. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that nonsafety-related service water systems meet the applicable portions of LWR GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the structures housing the system and the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, if any portion of the system is considered safety related, and Position C.2 for nonsafety-related functions. SRP Section 9.2.1 is directly applicable to the CRBR normal plant service water system. Conformance with the acceptance criteria of SRP Section 9.2.1 provides the bases for concluding that the proposed system's design criteria and bases and preliminary design are acceptable.

The normal plant service water system will be a nonsafety-related, nonseismic Category I system, completely independent of the emergency plant service water system (refer to Section 9.9.2 of this SER). The cooling loads served by the normal plant service water system will be the normal chilled water system chillers, the secondary service closed water heat exchangers, the intermediate activity level liquid (IALL) and the low activity level liquid (LALL) evaporator chillers, the primary and intermediate sodium pump drive motor generator set lube oil coolers, the motor generator set HVAC unit coolers, and the auxiliary steam system blowdown tank. Cooling for the above equipment will not be required for safe shutdown.

The normal plant service water system will consist of two 100% capacity (approximately 26,000 gpm) electric motor-driven vertical, wet-pit circulating water pumps and the associated pipes, valves, and instrumentation. The normal plant service water will be pumped from the basin of the circulating water system cooling tower to the equipment to be cooled and will be returned to the cooling tower return header. The pumps will be located in the nonseismic Category I circulating water pumphouse. Normally, one pump will be operating with the second pump in an auto-standby mode. The operation of the pumps will be rotated to equalize wear. The normal plant service water header pressure will be indicated in the control room with annunciation for low header

pressure. The standby pump will be automatically started when the operating pump motor trips or is inadvertently stopped.

The normal plant service water system is classified by the applicants as seismic Category III, and as such will be designed so that failure of the system or its components in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Although the normal plant service water system will serve the potentially radioactive IALL and LALL evaporator chillers, leakage from this part of the liquid radwaste system into the normal plant service water will not have unacceptable consequences (refer to Section 11.2 of this SER). In addition, radiation monitors will be provided in the normal plant service water system piping downstream of the IALL and LALL evaporator chillers, and valves will be provided to isolate leakage paths. As described above, failure of the system will not affect the safe shutdown capability of the plant or result in unacceptable radiation release. Therefore, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2, will be satisfied.

On the basis of the above, the staff concludes that the proposed nonsafety-related normal plant service water system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2 with respect to the system's seismic classification, and the system is, therefore, acceptable.

9.9.2 Emergency Plant Service Water System

The emergency plant service water system will be designed to provide sufficient cooling water to achieve and maintain safe shutdown of the plant in the event of an accident resulting in the loss of the normal service water system or loss of plant or offsite ac power supplies. The system will provide cooling for the emergency chilled water system chiller condensers and the standby diesel generators but will not provide for removal of shutdown decay heat from the reactor core. The emergency service water system also will provide fire fighting water for seismically qualified fire pumps of the nonsodium fire protection system. The emergency plant service water system will include the emergency cooling towers and emergency cooling tower basin as discussed in Section 9.9.4 of this SER. The emergency plant service water system is described in Section 9.9.2 of the Preliminary Safety Analysis Report (PSAR), Amendment 59. Supplemental information was provided at a meeting with the applicants on December 13, 1982. This supplemental information will be incorporated in a future amendment to the PSAR.

The emergency plant service water system's design criteria and bases and preliminary design reviewed in accordance with SRP Section 9.2.1, "Station Service Water System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system meets applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to structures housing the system and the systems capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, for safety-related portions, and Position C-2 for nonsafety-related portions.

(2) GDC 44, "Cooling Water," as it relates to transferring heat from structures, systems, and components important to safety to an ultimate heat sink. GDC 44 is identical to CRBR PDC 38, "Additional Cooling Systems," except that PDC 38 does not require that additional cooling systems at CRBRP be capable of removing reactor decay heat. Otherwise, the acceptance criteria of SRP Section 9.2.1 to meet GDC 44 are directly applicable to the CRBRP emergency plant service water system.

(3) GDC 45, "Inspection of Cooling Water System," as it relates to design provisions to permit inservice inspection of safety-related components and equipment. GDC 45 is identical to CRBR PDC 39 except that "additional cooling systems" has been substituted for "cooling water systems." The acceptance criteria of SRP Section 9.2.1 to meet GDC 45 are directly applicable to the CRBRP emergency plant service water system.

(4) GDC 46, "Testing of Cooling Water Systems," as it relates to design provisions to permit operational functional testing of safety-related systems and components. GDC 46 is comparable to CRBR PDC 40; the latter criterion is more general so as to include the variety of cooling system functions at CRBRP. The intent of the two criteria is identical; therefore, the acceptance criteria of SRP Section 9.2.1 to meet GDC 46 are directly applicable to the CRBRP emergency plant service water system.

SRP Section 9.2.1 is directly applicable to the review of the CRBRP emergency plant service water system. Conformance with the acceptance criteria of SRP Section 9.2.1 provides the bases for concluding that the proposed system's criteria and bases and preliminary design are acceptable.

The emergency plant service water (EPSW) system will consist of two 100% capacity, fully redundant cooling loops and a third loop that will just cool the division 3 diesel generator. Each cooling loop will include one circulating pump, one emergency cooling tower and the associated piping, valves, instrumentation, and controls. The EPSW circulating pumps will be located in the emergency cooling tower pumphouses. The circulating pumps will take suction from the emergency cooling tower storage basin under the emergency cooling towers. The EPSW circulating pumps and the cooling tower fans will start automatically on loss of normal chilled water or start of the standby diesel generators. The EPSW equipment will not operate during normal plant operation.

Two EPSW pumps will supply cooled water at a maximum temperature of 90°F to the emergency chilled water system chillers in the steam generator building and two standby diesel generators in the diesel generator building. The third EPSW pump will supply cooled water to the division 3 diesel generator. After cooling the emergency chillers and the standby diesel generators, warm water will be returned to the emergency cooling towers. To account for seasonal temperature variations, temperature control valves will operate automatically

to divert a part of the warm return water to the EPSW circulating pump's suction to maintain supply water temperature above the minimum of 55°F required for emergency chiller operation. The system will be designed to operate automatically and will be normally controlled from the control room. Redundant controls also will be provided at a panel in the diesel generator building.

The two full EPSW loops will be separate and independent with 100% redundancy. All electric motors serving the system will be connected to Class 1E power supplies. In case of loss of plant and offsite power, these motors will be switched automatically to the standby diesel generators. Each EPSW loop will be supplied with power from the standby diesel generator that it will serve.

EPSW loop A will be supplied from Class 1E, division 1; loop B will be supplied from Class 1E, division 2; and loop C, will be supplied from Class 1E, division 3. The piping and equipment for each redundant loop will be physically separated or protected with a barrier. The emergency service water system will be designed to seismic Category I and Quality Group C requirements. Components of the system will be located in flood-protected, tornado-hardened, seismic Category I structures. In a telephone conversation on August 20, 1982, the applicants verified that the underground system piping between the emergency cooling tower and the diesel generator building will be adequately protected from tornado missiles by a minimum cover of 10 ft of Class A fill. The supply connection to nonsafety-related piping of the nonsodium fire protection standpipe system will be isolated by a normally shut Quality Group C, seismic Category I valve. The relief connection from the standpipe system back to the emergency service water system will be through a Quality Group C, seismic Category I check valve. Pressure temperature, flow rate, and water level indications will be provided to ensure proper operation of the system. A flow switch, located in the return line from each diesel generator and emergency chiller, will detect an abnormal low flow condition and energize an annunciator in the control room. Thus, the requirements of PDC 2 (GDC 2) and PDC 38 (GDC 44) and the guidelines of RG 1.29, Positions C.1 and C.2, are satisfied.

The emergency plant service water system will not operate during normal plant operations. However, the system, including all active components, will be operated periodically during the year in conjunction with the standby diesel generator testing program. The system can be proven operable at any time by manual initiation. Inservice inspections will be conducted according to ASME Code, Section XI. In addition, isolation valves and pressure test connections on the supply and return headers will permit inservice inspection of the buried piping by hydrostatic testing. Adequate instrumentation will be provided to test flow rate and heat removal capacity of the system. Thus, the requirements of PDC 39 (GDC 45) and PDC 40 (GDC 46) are met.

On the basis of the above, the staff concludes that the proposed emergency plant service water system's design criteria and bases and preliminary design meet the requirements of CRBR PDC 2, 38, 39, and 40 with respect to the system's protection against natural phenomena, capability for transferring the required heat loads, inservice inspection, and functional testing, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.9.3 Secondary Service Closed Cooling Water System

The secondary service closed cooling water (SSCCW) system will provide cooling for power generation equipment and auxiliary equipment to be located in the turbine building. This nonsafety-related system will be used only during startup, normal operation, and normal plant shutdown.

Because the SSCCW system will be a nonsafety-related closed cooling water system, the guidance of NUREG-0800 was not used in the review of this system. SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," applies only to safety-related closed cooling water systems. The SSCCW system was, however, reviewed to verify that applicable portions of LWR GDC 2 are satisfied relating to the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance will be based on meeting the guidance of RG 1.29, Position C.2, for the nonsafety-related function.

The SSCCW system will consist of a single closed loop with two 100% capacity pumps arranged in parallel. The system will use two 100% capacity heat exchangers which will be cooled by the normal plant service water system. A surge tank, located above the SSCCW pump suction lines, will accommodate system volume changes and maintain static head on the pumps. Makeup water to the SSCCW system will be provided by the demineralized water system; tank level will be maintained automatically by means of locally mounted level transmitters and controllers. Indicators will be provided for system temperature and pressure, and high and low pump discharge pressure will be alarmed in the control room.

The SSCCW system is classified by the applicants as seismic Category III, by which the applicants mean that the system will be designed so that failure of the system or its components in a safe shutdown earthquake will not damage seismic Catgeory I structures, systems, and components (refer to Section 3.7 of this SER). Therefore, the requirements of GDC 2 and the guidelines of RG 1.29, Position C.2, are satisfied.

On the basis of the above, the staff concludes that the proposed nonsafety-related normal plant service water system's design criteria and bases and preliminary design meet the requirements of PDC 2 with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Position C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.9.4 Emergency Cooling Towers and Emergency Cooling Tower Basin

The emergency cooling towers will operate as part of the emergency plant service water system to provide cooling water for the emergency chilled water system chiller condensers and for the standby diesel generators (refer also to Section 9.9.2 of this SER). Operation of the emergency cooling towers will be required to achieve and maintain the safe shutdown condition of the plant in the event of loss of the normal plant service water system. The emergency cooling tower basin will provide the storage capacity for the emergency plant service water. The ultimate heat sink, consisting of the emergency cooling towers and emergency cooling tower basin, will be used to reject essential equipment heat but will not be used to reject reactor decay heat from the plant. This

equipment is described in Section 9.9.4 of the PSAR, Amendment 59. Supplemental information was provided by the applicants in a meeting on December 13, 1982. The applicants have stated that this supplemental information will be incorporated in a future amendment to the PSAR.

The design criteria and bases and preliminary design of the emergency cooling towers and emergency cooling tower basin were reviewed in accordance with SRP Section 9.2.5, "Ultimate Heat Sink" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the design criteria and bases and preliminary design of the emergency cooling towers and emergency cooling tower basin meet applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to structures housing the system and the system's capability to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. GDC 2 is identical to CRBR PDC 2. Acceptance will be based on meeting the guidance of RG 1.29, Position C.1, and RG 1.27, Positions C.2 and C.3.

(2) GDC 44, "Cooling Water," as it relates to transferring heat from safety-related structures, systems, and components to the heat sink. GDC 44 is identical to CRBR PDC 38, "Additional Cooling Systems," except that PDC 38 does not require that additional cooling systems CRBRP be capable of removing reactor decay heat. Acceptance is based on meeting the guidance of RG 1.27, Positions C.2 and C.3. Because the emergency cooling towers and emergency cooling tower basin will not remove reactor decay heat, they need not meet the guidance of BTP ASB 9.2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." Also, the acceptance criteria of RG 1.72, "Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin," do not apply to the CRBRP emergency cooling towers and emergency cooling tower basin.

(3) GDC 45, "Inspection of Cooling Water System," as it relates to design provisions to permit inservice inspection of safety-related components and equipment. GDC 45 is identical to CRBR PDC 39 except that "additional cooling systems" has been substituted for "cooling water systems." The acceptance criteria for SRP Section 9.2.5 to meet GDC 45 will be directly applicable to the CRBRP emergency cooling towers and emergency cooling tower basin.

(4) GDC 46, "Testing of Cooling Water Systems," as it relates to design provisions to permit operational functional testing of safety-related systems and components. GDC 46 is comparable to CRBR PDC 40; the latter criterion is more general so as to include the variety of cooling system functions at CRBRP. The intent of the two criteria is identical; therefore, the acceptance criteria of SRP Section 9.2.5 to meet GDC 46 will be directly applicable to the CRBRP emergency cooling towers and emergency cooling tower basin.

SRP Section 9.2.5 will be directly applicable to the review of the CRBRP emergency cooling towers and emergency cooling tower basin. Conformance with the acceptance criteria of SRP Section 9.2.5, with the exception of those pertaining to the reactor decay heat removal, provides the bases for concluding that the design criteria and bases and preliminary design of the emergency cooling tower and emergency cooling tower basin are acceptable.

The emergency cooling tower structure will consist of three emergency cooling towers, three pumphouses, and one below-grade cooling water storage basin. The pumphouses will be arranged to provide adequate separation between the redundant emergency plant service water pumps. The emergency cooling towers will be located above the storage basin.

The storage capacity of the emergency cooling tower basin will be determined using historical regional temperature, humidity, and wind measurements to calculate the maximum 30-day evaporative and drift losses. Normal component leakage and losses from a postulated pipe rupture will also be considered in the design. The storage basin capacity will include 30,000 gal of water for fire protection, in addition to the amount required for emergency heat removal. Instrumentation on the fire pumps will automatically terminate fire water usage greater than that specifically reserved for fire protection. There will be no drains in the storage basin or connections that would permit draining below acceptable levels. The emergency plant service water circulating pumps will take water from the storage basin to cool the emergency heat loads and return the warmer water to the cooling towers.

The emergency cooling towers will be of a counter-flow, wet-type, mechanically induced draft design. They will not operate under normal plant conditions except for routine testing. Cooling tower fans will start automatically on loss of the normal chilled water system. Each emergency cooling tower and its associated equipment will be designed to remove the maximum heat load associated with a postulated design-basis accident. The towers will be designed on the assumption that the heat load for the first 24 hours of the design-basis accident will be dissipated under average ambient wet-bulb temperature conditions for the worst day of record. Electrical power for the emergency cooling tower fans, pumps, and control equipment will be provided from Class 1E ac power supplies. Each cooling tower division will be powered from a separate Class 1E power supply division. The design described above will ensure that adequate heat removal capability to maintain plant safety will be provided by the emergency cooling towers and emergency cooling tower basin for a postulated design-basis accident with a single active failure. The staff, therefore, concludes that the proposed system satisfies the requirements of PDC 38 (GDC 44) and the guidelines of RG 1.27, Positions C.2 and C.3.

The emergency cooling towers, pumphouses, operating basins, and storage basin will be designed to withstand the most severe natural phenomenon postulated for the plant. The entire structure will be designed to seismic Category I requirements and to withstand tornado wind forces and missiles. Each cooling tower will be enclosed in a seismic Category I structure, which will protect the cooling towers and associated equipment from the effects of tornado wind forces and tornado missiles. The piping associated with operation of the cooling towers will be designed to Quality Group C requirements and will be protected against tornado missiles by being placed inside the tower structure or under-

ground. The emergency cooling towers, pumphouses, and top level of the storage basin will be above the level of the probable maximum flood. In a meeting on December 13, 1982, the applicants stated that the storage basin will be covered and that an analysis has been performed to verify that the system will be protected against the effects of freezing. The staff, therefore, concludes that the proposed system satisfies the requirements of PDC 2 (GDC 2) and the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.27, Positions C.2 and C.3.

The applicable emergency cooling tower components will be tested periodically in conjunction with the emergency plant service water system. Instrumentation will be available to monitor system operation, and the components will be accessible to permit inservice inspection as required. Thus, the requirements of PDC 39 (GDC 45) and PDC 40 (GDC 46) are met.

The staff concludes that the proposed design criteria and bases and preliminary design of the emergency cooling towers and emergency cooling tower basin meet the requirements of CRBR PDC 2, 38, 39, and 40, with respect to protection against natural phenomena, missiles, and pipe break effects, heat dissipation capability, inservice inspection and functional testing, and the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.27, Positions C.2 and C.3, with respect to seismic classification and the capability to remove heat in the event of a postulated design-basis accident, and are, therefore, acceptable.

9.9.5 River Water Service System

The river water service system supplies water from the Clinch River to the main cooling tower and plant water treatment facility during normal operation. This is not a safety-related system and will not be required during an emergency shutdown of the plant.

The river water service system's design criteria and basis and preliminary design were reviewed in accordance with those portions of SRP Section 9.2.1, "Station Service Water System" (NUREG-0800) pertaining to nonsafety-related service water systems. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that nonsafety-related service water systems meet the applicable portions of LWR GDC 2, "Design Bases for Protection Against Natural Phenomena" as it relates to the structures housing the system and the system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, if any portion of the system is deemed to be safety related, and Position C.2 for nonsafety-related functions. SRP Section 9.2.1 is directly applicable to the CRBR river water service system. Conformance with the acceptance criteria of SRP Section 9.2.1 provides the basis for concluding that the system's design criteria and bases and preliminary design are acceptable.

The two river water service pumps will be located in an intake structure on the Clinch River and will take suction from the river via two perforated pipe inlets. The river water service pumps will provide makeup water to the main cooling tower basin and the plant water treatment facility as needed during normal operation. Neither function will be required for safe shutdown of the plant. Design flow

rate for the system will be 9,000 gpm, which can be supplied by either pump. Pump discharge flow will be regulated by level control of the main cooling tower basin. Two backwash lines will be provided to clear debris from the intake pipes and a recirculation line will be provided for each pump to preclude low flow problems with the pumps.

The river water service system is classified by the applicants as seismic Category III, by which the applicants mean that the system will be designed so that failure of the system or its components in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). In addition, failure of the system will not affect safe shutdown capability of the plant or result in unacceptable radiation release. Therefore, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2, are satisfied.

On the basis of the above, the staff concludes that the proposed nonsafety-related river water service system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2) with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.10 Compressed Gas Systems

9.10.1 Service Air and Instrument Air Systems

The service air system will provide clean, oil-free, compressed air for various maintenance functions, for personnel breathing stations where respiratory protection is required, and for the instrument air system. The instrument air system will provide filtered, dry, compressed air to various instruments and control devices throughout the plant.

The service air and instrument air systems' design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.3.1, "Compressed Air System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of this SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the systems' design criteria and bases and preliminary design meet the applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 1, "Quality Standards and Records," as it relates to systems and components important to safety being designed, fabricated, and tested to quality standards and commensurate with the importance of the safety functions to be performed. GDC 1 is identical to CRBR PDC 1. Acceptance is based on meeting ANSI Std. MC 11.1-1976 (ISA S7.3), "Quality Standards for Instrument Air," as it relates to minimum instrument air quality standards and meeting the guidance of RG 1.68.3, "Preoperational Testing of Instrument Air Systems," as it relates to testing of the instrument air system. Compliance with the other requirements of GDC 1 are evaluated separately in Section 3.2 of this report.

(2) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the safety-related compressed air system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, if any portion is deemed to be safety-related, and Position C.2 for nonsafety-related functions.

SRP Section 9.3.1 is directly applicable to the CRBR service air and instrument air systems. Conformance with the acceptance criteria of SRP Section 9.3.1 provides the basis for concluding that the systems' design criteria and bases and preliminary design are acceptable.

The service air and instrument air systems' equipment will be located in the turbine generator building. Compressed air for these systems will be provided by three 100% capacity oil-free compressors arranged in parallel and supplying air to a common discharge header and parallel air receivers. The system will be maintained at a pressure of 165 ± 15 psig by a pressure control valve in the distribution header. Each compressor will include an intake filter-silencer, intercooler, aftercooler, moisture separator, and automatic control equipment. Under normal plant use, one compressor will be in operation, a second compressor will be on automatic standby for large load swings, and the third compressor will serve as backup during maintenance down time. The air compressors will be operated from the normal plant power supply. A distribution header, downstream of the air receivers, will supply air to the service air stations and to the instrument air system.

Portable equipment will be provided to process service air to the necessary pressure and quality for breathing air use. The equipment will supply breathable air through a hose connection to respiratory apparatus worn by personnel.

Service air piping to the instrument air system will pass through one of two full-capacity prefilters, one of two full-capacity air dryers, and one of two full-capacity afterfilters. The instrument air will then be distributed to various stations located throughout the plant as required.

Although instrument air will be required for operation of safety-related equipment, the service air, breathing air, and a portion of the instrument air function are considered nonessential. Where loss of instrument air could result in the loss of safety function of another component, local air accumulators with dual automatic isolation valves will be provided. The capacity of the accumulators will be based on the time during which operation is required, air losses, minimum pressure required to operate the valve or instrument, and adequate margin.

The compressed air supplied to the instrument air system will be processed by components of the instrument air system for use by safety-related valve operators and instruments. Air dryers will be provided to dry the instrument air to a -40°F dewpoint. Prefilters and afterfilters will be located in the inlet and outlet piping of the instrument air dryers and remove all particles approximately 3 μm or larger in size. Periodic maintenance checks of all filters will be performed. In a meeting on December 13, 1982, the applicants verified that the instrument air system will be designed to supply clean, dry,

oil-free air in accordance with ANSI Std. MC-11-1 (ISA 57.3). Therefore, the staff concludes that the requirements of PDC 1 (GDC 1) related to instrument air quality are satisfied.

The service air and instrument air systems will be observed, tested, and maintained during normal operation according to plant operating procedures. Inservice inspections for all ASME Code, Section III, Class 3 (Quality Group C) components will be conducted according to ASME Code, Section XI, Division 1. For compliance of the systems with the requirements of PDC 1 (GDC 1), as related to system quality standards and records, refer to Section 3.2 of this SER.

Service air equipment and piping and the nonsafety-related portions of the instrument air system will be nonsafety class and nonseismic Category I up to the isolation valves outside containment. The isolation valves outside and inside the reactor containment and the piping between them will be Quality Group B and seismic Category I. Distribution piping inside the reactor containment, downstream of the isolation valves, will be nonsafety class. The nonsafety-class portions of the service air and instrument air systems have been designated seismic Category III by the applicants and, as such, will be designed so that their failure in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Thus, the proposed nonsafety-related portions of the service air and instrument air systems satisfy PDC 2 (GDC 2) and the guidelines of RG 1.29, Position C-2.

The safety-related portions of the instrument air system will be the accumulators and associated piping and redundant automatic isolation valves that will be provided to ensure operability of the safety function of other safety-related equipment. The safety-related portions of the instrument air system will be Quality Group C and seismic Category I. These portions of the system will be located within seismic Category I, flood- and tornado-protected structures (refer to Sections 3.4 and 3.5 of this SER). The safety-related portions of the instrument air system also will be protected from the effects of missiles and pipe breaks (refer to Sections 3.5 and 3.6 of this SER). Thus, the proposed system satisfies the requirements of GDC 2 and the guidelines of RG 1.29, Position C.1.

A discussion of preoperational testing of the service and instrument air systems and compliance with RG 1.68.3 is contained in Section 14 of this SER.

On the basis of the above, the staff concludes that the proposed service and instrument air systems' design criteria and bases and preliminary design meet the requirements of PDC 1 (GDC 1) and PDC 2 (GDC 2), with respect to ensuring that minimum instrument air standards will be met and the systems' protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the systems' seismic classification and are, therefore, acceptable.

9.10.2 Hydrogen System

The nonsafety-related hydrogen system will provide for the storage and distribution of the supply and makeup hydrogen gas required for generator cooling.

The hydrogen systems' design criteria and bases and preliminary design were reviewed in accordance with those portions of SRP Section 9.3.1, "Compressed Gas System" (NUREG-0800) pertaining to nonsafety-related compressed gas systems. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that nonsafety-related compressed gas systems will meet the applicable portions of LWR GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the safety-related compressed gas system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, if any portion of the system is deemed to be safety-related, and Position C-2 for nonsafety-related functions. SRP Section 9.3.1 is directly applicable to the CRBR hydrogen system. Conformance with the acceptance criteria of SRP Section 9.3.1 provides the bases for concluding that the systems' design criteria and bases and preliminary design are acceptable.

The two hydrogen storage equipment and pressure-reducing stations will be located outdoors, adjacent to the turbine generator building. Hydrogen will be distributed only in the turbine generator building. Under normal operating conditions one storage cylinder skid and its pressure-reducing station will be active and the second station will be held in reserve. When the active supply system will reach a preset low-pressure value, the reserve storage skid and its pressure-reducing station automatically will start operating to supply the hydrogen required by the generator. Suitable alarms and pressure, flow, and temperature indicators will be provided to ensure proper operation of the hydrogen system components. Manual switchover from normal to reserve hydrogen supply will be provided in addition to the automatic capability.

Hydrogen supply piping in the turbine generator building, up to the connection to the generator, will be protected by a guard pipe. Hydrogen system equipment, piping, and components have been classified by the applicants as seismic Category III by which the applicants mean that the system will be designed so that its failure in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Therefore, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2, are satisfied.

On the basis of the above, the staff concludes that the proposed nonsafety-related hydrogen system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2), with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.10.3 Carbon Dioxide System

The nonsafety-related carbon dioxide system will provide liquid carbon dioxide storage and gaseous distribution to generator purging in the turbine generator building.

The carbon dioxide system's design criteria and bases and preliminary design were reviewed in accordance with those portions of SRP Section 9.3.1,

"Compressed Gas System" (NUREG-0800) pertaining to nonsafety-related compressed gas systems. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that nonsafety-related compressed gas systems meet the applicable portions of LWR GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the safety-related compressed gas system's capability to withstand the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, Position C.1, if any portion of the system is deemed to be safety related, and Position C.2 for nonsafety-related functions. SRP Section 9.3.1 is directly applicable to the CRBR carbon dioxide system. Conformance with the acceptance criteria of SRP Section 9.3.1 provides the bases for concluding that the system's design criteria and bases and preliminary design are acceptable.

Liquid carbon dioxide for generator purging will be stored under pressure in an outdoor storage tank. Carbon dioxide distribution piping and valves for generator purging will be located in the turbine generator building. A vaporizer heater will be provided to convert the liquid carbon dioxide to a gas. The liquid carbon dioxide storage vessel normally will be kept full at the required temperature and pressure. Pressure, flow, and temperature indicators will be provided to ensure proper operation. A temperature alarm will be provided to protect the generator against purging with carbon dioxide at too low a temperature. The nonsafety-related carbon dioxide system's equipment, piping, and components have been classified by the applicants as seismic Category III, by which the applicants mean that the system will be designed so that its failure in a safe shutdown earthquake will not damage seismic Category I structures, systems, and components (refer to Section 3.7 of this SER). Therefore, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2, are satisfied.

On the basis of the above, the staff concludes that the proposed nonsafety-related carbon dioxide system's design criteria and bases and preliminary design meet the requirements of PDC 2 (GDC 2), with respect to the system's protection against natural phenomena, and the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification, and are, therefore, acceptable.

9.11 Communication Systems

The communication systems will be designed to provide reliable intraplant and interplant (or plant-to-offsite) communications under both normal plant operation and accident conditions.

9.11.1 Intraplant Systems

The intraplant communication systems will provide sufficient equipment of various types so that the plant will have adequate communications to start up, continue safe operation, or safely shut down. The intraplant systems include:

(1) Public Address Intraplant Communication (PA-IC) System

The public address system will be designed to provide effective communication between plant personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels, and will consist of an integrated system of speakers and handset paging units located in designated plant areas. The PA-IC system will also be used to broadcast alarm signals over the entire plant area for building evacuation, radiation emergency, and fire alarm. Power for the PA-IC system will be supplied by the 120-V ac instrument bus, or by inverter from the non-Class 1E station battery.

(2) Private Automatic Exchange (PAX) System

The PAX system will consist of a network of dial-type telephone handsets located throughout the administrative and operating areas of the plant. The PAX system will be interconnected with the PA-IC, powerline carrier, and microwave communication systems. Power for the PAX system will be supplied by a non-Class 1E, uninterruptible power source.

(3) Maintenance Communications Jacking (MCJ) System

The MCJ system will consist of sound-powered headset/microphones and jack stations, and will be designed to facilitate testing and calibration of equipment instrumentation and to provide fixed communication for response to emergencies. The MCJ system will cover all safety-related areas in the plant, and may also be used for the support of remote plant shutdown. No power source will be required for this system.

(4) Portable Radio System

This system will consist of a number of selective call portable radios, satellite receivers, and a base station. The system will be designed to allow two-way communications between the base station and any part of the plant, and between portable units on an alternate frequency. Power for base station operation will be supplied by a non-Class 1E uninterruptible power source. Individual portable units will be powered with rechargeable batteries.

All intraplant communication systems will be designed to provide adequate communications under the maximum anticipated noise levels in the plant areas served. The adequacy of the communication systems will be verified by actual test following plant startup, and any systems adjustments will be made at that time. Communication equipment located in seismic Category I structures will be seismically supported.

9.11.2 Interplant (Plant-to-Offsite) Communication Systems

The design basis for interplant communications is to provide dependable communications for reliable operation. The interplant communication systems include:

(1) Microwave Communications System

A microwave radio station operating on the 7,125-8,400 MHz government band frequency range will be provided at the CRBR site. This system will be the primary, Tennessee Valley Authority- (TVA-) operated, interplant communications link, and will be used for transmitting data on generated power for control purposes, and for voice communications with other TVA facilities. The microwave system will be powered from a non-Class 1E, 120-V ac source, and will have its own battery backup for 8 hours of operation in the event of loss of ac power.

(2) Powerline Carrier Communications System

The powerline carrier communications system will transmit voice and plant operations data. This system will use low-voltage, high-frequency signals superimposed on the 161-kV transmission lines. Power for this system will be from a non-Class 1E, 120-V ac source, with battery backup for 8 hours of operation in the event of loss of ac power.

(3) Very-High-Frequency (VHF) Radio Station System

A VHF radio station will be provided at CRBRP. The purpose of the VHF radio station is to transmit and receive emergency voice communications between the CRBR control room and the TVA power production emergency staff operations office. This system will be powered by a non-Class 1E, uninterruptible 120-V ac power source.

(4) Commercial Telephone System

Plant-to-offsite communications will also be available through the commercial telephone system at the CRBR site.

(5) Security Communications System

An offsite law enforcement radio, an intercom, and a portable radio system will be provided for plant security.

The scope of review included assessment of the number and types of communication systems that will be provided, assessment and adequacy of the power sources, and verification of functional capability of the communication systems under all conditions of operation.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the proposed communication systems to the acceptance criteria and guidance of SRP Section 9.5.2 (NUREG-0800). Other bases for acceptance were conformance to industry standards and the ability of the systems to provide effective communications from diverse means within the CRBRP during emergency conditions under maximum potential noise levels.

On the basis of its review, the staff concludes that the proposed communication systems at the CRBRP conform to the cited standards, criteria, and design bases; they will perform their design functions, and are, therefore, acceptable.

Communication systems requirements for fire protection are discussed in Section 9.5.1 of this SER.

9.12 Lighting System

The lighting system for the CRBRP will be designed to provide adequate lighting in all areas of the plant. The system will consist of normal and standby ac lighting systems and an emergency lighting system using battery packs. The system design will provide for lighting levels under normal and emergency operating conditions in accordance with applicable sections of NUREG-0700, Illuminating Engineering Society (IES) Lighting Handbook, and/or NFPA 101 (National Fire Protection Association).

The normal ac lighting system will provide illumination under all normal plant operating conditions, as well as accident and transient conditions, with offsite power available. The normal lighting system will use fluorescent and high-pressure sodium fixtures and will draw power from the plant, preferred, or reserve power supply systems. All lighting fixtures and raceways in safety-related structures will be seismically supported to meet seismic Category I requirements, and fixtures will be seismically qualified to maintain structural integrity.

The standby lighting system will provide illumination under all normal plant operating and accident conditions, including loss of offsite power. The standby lighting system will use fluorescent and high-pressure sodium fixtures and will draw power from the plant, preferred, reserve, or Class 1E onsite ac power system. Lighting fixtures and raceways will be seismically qualified and/or supported, as will be the normal lighting system. Standby lighting will be provided in all plant areas containing safety-related equipment.

The emergency lighting system will provide illumination at points of egress, at remote shutdown locations, and at all locations required for access to safety-related equipment. The emergency system will use self-contained, individual 8-hour-rated battery-powered units with sealed beam lamps, and self-contained 8-hour-rated battery-powered exit signs. In safety-related areas, the emergency lighting units will be seismically supported.

Operation of both the standby and emergency lighting systems will be automatic in the event of loss of offsite power. The emergency lights will be designed to illuminate immediately on loss of ac power, using the integral batteries for power. Standby lighting will be restored automatically with the availability of onsite ac power, except for the control room (and other selected safety-related areas). Control room standby lighting will be powered from a Class 1E uninterruptible power source, and the Class 1E station batteries will provide backup. Standby lighting in safety-related areas will be powered from the Class 1E power train associated with the safety-related equipment in that area. Where more than one safety train is represented (i.e., control room, remote shutdown panel), standby lighting will be divided into two sections, each of which will be powered from a separate Class 1E source.

The scope of the review of the lighting system for the CRBRP included assessment of all components that will be necessary to provide adequate lighting during both normal and emergency operating conditions, the adequacy of the power

sources for the normal and standby lighting systems, and verification of functional capability of the lighting system under all conditions of operation.

The basis for acceptance in the staff's review was conformance of the design bases and criteria and proposed design of the lighting systems and necessary auxiliary supporting systems to the acceptance criteria and guidance of SRP Section 9.5.3. Other bases for acceptance were conformance to industry standards and the ability to provide effective lighting in all areas of the CRBRP under all conditions of operations.

On the basis of its review, the staff concludes that the various lighting systems proposed for the CRBRP will be in conformance with the above-cited standards, criteria, and design bases. They will perform their design function, and are, therefore, acceptable.

Lighting system requirements for fire protection are discussed in Section 9.5.1 of this SER.

9.13 Fire Protection

9.13.1 Conventional Fire Protection

On May 23, 1980, the Commission indicated that the essential elements for an acceptable fire protection program are defined in Appendix R to 10 CFR 50 and Appendix A to BTP ASB 9.5-1. The Standard Review Plan (NUREG-0800, dated July 1981) consolidates these fire protection guidelines. The staff has reviewed the applicants' PSAR through Amendment 64 for conformance with these guidelines.

The applicants compared the fire protection program only with Appendix A to BTP ASB 9.5-1. This comparison indicated that areas of the program deviate from the staff's guidelines, but the applicants did not provide justification for these deviations.

At the staff's request by letter dated June 29, 1982, the applicants committed to meet the current fire protection guidelines set forth in Section 9.5-1 of NUREG-0800 or provide an equivalent level of safety in all areas of the plant. Deviations from these guidelines will be specifically identified and justified.

On the basis of the applicants' commitment, the staff finds the proposed fire protection program for the construction permit acceptable and in conformance with CRBR PDC 3.

9.13.2 Sodium Fire Protection System

Introduction

The sodium fire protection system will consist of those features provided in the plant to detect, extinguish, and mitigate the consequences of a sodium or NaK leak and fire. The primary safety concern will be the prevention of sodium or NaK reactions with concrete, which release hydrogen and cause deterioration of the structural concrete. The basic features of the system will consist of steel liners and an inert atmosphere for those cells containing radioactive

sodium; steel catch pans, drains, and fire suppression decks for those cells containing nonradioactive sodium or NaK; fire detection instrumentation; instrumentation for limiting aerosol release and portable fire extinguishers for small sodium or NaK spills. No water sprinkler systems or any self-initiating extinguisher system will be located in any cell containing a sodium or NaK system.

Scope of Review

PSAR Sections 9.13.2, 3.8, 3.8B, 3.8C, 3A, 6.4, 6.5, 15.6, and 15.7 were reviewed along with draft ANS Std. 54.8, "Proposed Standard for Liquid Metal Fire Protection in LMFBR Plants," and CRBR PDC 4. The Standard Review Plan (which applies to conventional fire protection) was reviewed for background information only. No regulatory guides or NUREG documents were found to apply. The systems that will provide inert gas and cooling to the cells were not part of this review. They are discussed in Sections 9.5, 9.16, and 11 of this SER.

Evaluation Criteria

The sodium fire protection system was reviewed for

(1) Compliance with CRBR PDC 4

(2) Compatibility with draft ANS Std. 54.8. Because of the draft nature of ANS Std. 54.8, compliance with this standard was not looked for, but rather it was used as a checklist to help identify major areas for consideration in the design. The way major areas were addressed by CRBR was evaluated and a determination was made regarding the acceptability of the CRBR approach. This determination was based on how well the design met the following:

    (a) Do the assumed sodium spills bound all reasonably possible spills?

    (b) Are adequate features provided in the design to mitigate the consequences of a liquid metal spill? Have they been used in other plants? Have they been tested for CRBR?

    (c) Is the design basis of these features adequate and consistent with the assumed spills?

    (d) Are fire-extinguishing capabilities required/provided?

    (e) Are the consequences of the assumed liquid metal spills acceptable (i.e., site boundary dose rate within limits, effect on other plant safety equipment acceptable, effect on the ability to safely shut down and maintain the plant shutdown acceptable)?

Evaluation Cell Liner System

Each cell containing radioactive sodium will be inerted with nitrogen (except for the fuel handling cell (FHC), which will be inerted with argon), will be provided with a steel liner system (with specified leak rate requirements), and will be operated at a slightly negative pressure. The liner system for the

nitrogen inerted cells will be considered an engineered safety feature (ESF) for the plant and as described in Section 6.4 of the PSAR. The liner system for the FHC will not be designated an ESF because of the small volume of sodium in the cell (~50 gal) and the fact that this sodium will be stored in three core component pots at atmospheric pressure. These factors would tend to minimize the potential for and consequences of a sodium spill in the FHC. The FHC liner, however, will be designed to the same requirements as the ESF liners. There will be a total of 22 lined cells in the reactor containment building and 16 in the reactor service building (including the FHC). The purpose of the cell liner system will be to protect the structural concrete from high temperatures during a sodium spill condition, prevent contact between the sodium and concrete (and the resulting sodium-water reaction that will release hydrogen and weaken the concrete), and provide a controlled leakage barrier for the cell to aid in maintaining an inert atmosphere in the cell (thus limiting the available oxygen to support combustion in the cell) and limiting the leakage of any radioactive material from the cell.

These liners (see Figure 9.11) will be constructed of 3/8-in.-thick, 1% carbon steel plates welded together to line the cell walls, floor, and ceiling. The liners will be anchored to the structural concrete of the cell wall and ceiling with Nelson studs, which will be welded to the liner plate and imbedded in the cell wall's structural concrete. The liners on the floor will be supported by steel beams imbedded in the structural concrete. A layer of insulating concrete (in most cases 4 in. thick) will be provided between the cell structural concrete and the back side of the liner plates to protect the structural concrete from high temperatures in the event of a sodium spill. In addition, a 1/4-in. gap along the cell walls and ceiling (1/8 in. on the cell floor) will be provided between the back side of the liner plate and the insulating concrete to allow the venting of steam (produced when the concrete behind the liner plate will be heated to more than 212°F during a sodium spill condition) to other noninerted regions of the plant, thus avoiding an external pressure buildup behind the liner plate (for design purposes a 5-psig pressure behind the liner plate will be assumed). The liners for the in-containment cells will be designed for an internal pressure equal to the design pressure of the cell structural concrete (see PSAR Table 3A.1-3 for a listing of in-containment cell design pressures), a steady-state temperature of 120°F, and for spills with a sodium temperature up to 830°F.* The liners for the ex-containment cells will be designed for an internal pressure of 12 psig, a steady-state temperature of 120°F, and for spills with a sodium temperature of 600°F (except for cell 351A, which will be based on 400°F sodium). In all cases structural concrete temperatures are to be kept within the limits specified in Paragraph CC-3440 of ASME Code, Section III, Division 2 (i.e., 150°F steady state, 350°F transient unless higher temperature can be justified).

The volume of sodium assumed spilled under accident conditions will be the maximum volume available to spill from the affected system, and the spill rate will be based on the break size defined in BTP MEB 3-1 for moderate-energy fluid system piping. This will result in spill rates from 6 to 950 gpm. The liner system will be designed as seismic Category I and, since it will be

*The reactor cavity and primary heat transport system (HTS) cell liners will be designed for a sodium spill temperature of 1,015°F.

designated as an ESF, will receive periodic inservice inspection (discussed in more detail later on in this section). The maximum allowable leak rate of the lined system is specified as 0.36% volume per day inleakage at 2.5-in. water negative pressure. Corrosion of the liner plate material is assumed in the design and a 1/16-in. allowance has been provided. In addition, a coating to retard corrosion will be applied to both sides of the liner material. The design analysis of the liner system is to demonstrate liner integrity during postulated accident conditions and will be based on meeting specified stress and strain criteria, which are discussed in Section 3.8 of the SER. Penetrations through the liner also will be designed to maintain liner integrity and penetration integrity during normal and accident conditions.

To verify the design, a sodium spill test on a liner section prototypic of the CRBR design was run at the Hanford Engineering Development Laboratory (HEDL) in 1979. The results of this test (documented in report HEDL-TME-79-35, "Large Scale Liner Sodium Spill Test, LT-1," dated Dec. 1980) indicate that under spill conditions more severe than those postulated for CRBR, the liner remained intact and no damage to the structural concrete behind the liner was evident. Some liner attachment studs did, however, detach from the liner. Back pressure behind the liner wall resulting from steam venting was measured to be negligible. The effect of this test on the liner design and structural criteria is discussed in Section 3.8 of this SER. In addition, tests with intentionally failed liners will be run to determine the magnitude and effect of the resulting sodium-concrete reaction in the event liner integrity is not maintained. In support of the full-scale tests, the following tests were also run:

(1) "Selection of a Sodium and Radiation Resistant Sealant for LMFBR Equipment Cell Penetrations" (Westinghouse, WARD-D-0252), January 31, 1978

(2) "Base Material Tests for Cell Liner Steels," February 1980

The cell liner plate seam welds will be fabricated as full penetration welds and will be radiographically examined in accordance with Article CC-5500 of the ASME Code, Section III, Division 2. Acceptance standards for the ESF liner welds are to be in accordance with Subarticle CC-5540 of the ASME Code, Section III, Division 2. Where radiography cannot be performed on welds because of location or lack of access, the following nondestructive examination (NDE) will be done on the welds before the cell is placed into service:

(1) The entire length of liner welds will be examined visually.

(2) The entire length of all liner butt welds will be examined by the vacuum-box method using either a bubble solution or gas detector technique.

(3) The entire length of all liner welds will be examined by liquid penetrant or magnetic particle methods.

(4) The entire length of all attachment welds will be examined by either the magnetic particle or liquid penetrant methods.

(5) A minimum of 10% of the total length of all liner seam welds will be examined ultrasonically.

Preselected areas in the cell liner floor and wall plates below the postulated pool depth in three reactor containment building (RCB) and three reactor service building (RSB) cells will be examined ultrasonically in accordance with Article 5, "Ultrasonic Examination," of the ASME Code, Section V, to determine the reference plate thickness to be used in future monitoring of cell liner plate corrosion. Stud welds will be visually inspected in accordance with Subarticle CC-5548 of the ASME Code, Section III, Division 2. After all liner panels, doors, and hatches have been completely installed and openings and penetrations for system piping, electrical lines, and heating, ventilating, and air conditioning have been blanked off, the cell will be pressurized to 2 psig. Leaktightness of the liner system will be acceptable if the rate of leakage is less than or equal to 0.36% volume per day measured under atmospheric pressure as standard cubic feet per minute.

Inservice inspection is planned for selected liner welds and plate material. The applicants have stated that liners in three cells in the RCB and three cells in the RSB will be inspected twice every 10 years. Selected welds will be inspected visually and with a magnetic particle examination. Generally, the welds chosen for inspection will be those predicted to have the highest stress during steady-state or transient operation. Selected areas of the liner plate, characterized during construction by ultrasonic (UT) examination, will also be examined for corrosion by measuring the plate thickness by UT examination.

Preliminary analysis of the liner system was performed for steady-state and accident conditions. Sodium leaks were assumed to result in a spray fire for the duration of the spill. The SPRAY 3B code was used to calculate the cell pressures and temperatures during this condition. After the total volume of sodium available to spill was discharged, the remainder of the accident was calculated by the SOFIRE-II code. Preliminary analysis by the applicants shows that all proposed liner structural criteria have been met and structural concrete temperatures have been calculated to be within their allowable values. The structural capability of the cell liners, as supported by the structural analysis, the LT-1 liner spill test, and the liner structural design criteria, is evaluated in Section 3.8 of this SER.

It should be noted that if leak rates higher than the design-basis leak rate should occur, higher cell temperatures and pressure also would occur. For the primary HTS cells, leak rates up to that equivalent to full flow (33,500 gpm) were analyzed by the applicants with the result that cell pressure limits were still not exceeded.

In addition, the cell pressure upon cooldown after a sodium spill (resulting in negative pressure in the cell) was conservatively calculated for the primary HTS spill case and the resulting back pressure did not exceed the 5-psig cell back-pressure limit.

All of the large leaks analyzed by the applicants will result in some release of radioactive aerosol to the building (calculated by the HAA-3B code) in which the cell will be located; however, the release of radioactive material to the outside atmosphere will be well within allowable releases for all sodium spill events. The results from the bounding (most severe) sodium spill conditions

analyzed are presented in Section 15.6 of the PSAR and are evaluated in Section 15.6 of the SER.

The computer codes used by the applicants (SPRAY 3B, SOFIRE-II, and HAA-3B) have been verified against experimental data.

Conventional fires in inerted cells are not expected because of the lack of oxygen.

Catch Pans and Fire Suppression Decks

Catch pans and fire suppression decks (which are also considered engineered safety features) will be provided in all cells containing NaK-filled components or piping or nonradioactive sodium-filled components or piping. There will be a total of 25 cells with catch pans, and 10 of these also will have fire suppression decks. Six of these will be located in the RSB with the remaining 19 located in the steam generator buildings. Those cells without fire suppression decks either will be inerted (provided with drains that will direct any spilled sodium or NaK to a cell with a fire suppression deck) or will allow open pool burning as long as no safety-related equipment or building structures are affected (limited to cells with a small spill volume potential). The purpose of the catch pan and fire suppression deck system is to prevent a sodium-concrete or NaK-concrete reaction in the event of a spill, to protect the cell structural concrete from high temperatures in the event of a spill, and to collect and isolate the spilled liquid metal in a fashion so as to limit the amount and rate of combustion. The cells containing the catch pans and fire suppression decks will have no other liquid metal fire suppression or extinguishing devices or systems. The catch pans (see Figure 9.12) will be made from 3/8-in., 1% carbon steel plates and of an all-welded construction (full penetration welds). To allow for free thermal expansion they will not be fastened to any structural concrete but will sit on a bed of compacted, insulating MgO aggregate. Insulating concrete also will be provided behind the vertical sides of the catch pan to protect the structural concrete from high temperatures. An air gap will be provided between the vertical walls of the catch pan and the insulating concrete to allow for venting of steam from hot concrete during a spill condition, thus avoiding any back-pressure buildup behind the catch pans. The vented gases will be discharged into the same cell as the catch pans. The catch pans will be designed for seismic Category I conditions, for a steady-state temperature of 120°F, and for spills with a liquid metal pool temperature up to 1,350°F for catch pans without fire suppression decks and 1,000°F for catch pans with fire suppression decks. No design pressure is associated with the catch pan or fire suppression decks, since they will not be pressure-retaining components. Any pressure built up in the cells containing the catch pans will be relieved to surrounding areas, since the unlined cells will not be designed as pressure-retaining structures. Releases of combustion products to the outside atmosphere will, however, be controlled so as not to affect other safety-related equipment (letter, J. R. Longenecker to P. S. Check, dated Jan. 7, 1983, #HQ:S:83:171). Control of combustion product release has already been provided for in the steam generator buildings (SGBs) (SGB aerosol release mitigation system--described in Section 9.13.2.3); if similar features are determined to be needed elsewhere in the plant, they will be provided.

The size of the catch pans is determined by calculating the depth of the liquid metal after the spill is complete and adding an additional 1 ft to the height of the catch pan walls. If a fire suppression deck is located in the catch pan, the elevation of its lower side will be 4 in. above the top of the pool and the sides of the catch pans will be at least 22 in. above the bottom of the fire suppression deck.

The spill volumes assumed under accident conditions will be the maximum volumes available to spill from the affected system, and the spill rate will be based on the break size defined in BTP MEB 3-1 for moderate-energy fluid system piping. This will result in spill rates from 45 to 1,000 gpm. Where it is not practical or feasible to install a catch pan with a capacity sufficient to contain the entire spill volume, a drain will be provided to drain the spilled liquid to a larger catch pan of sufficient size. The drains will be sized to handle the design-basis spill rate.

For those cells with catch pans, where burning of the liquid metal pool after a spill could pose a threat to safety-related equipment in the cell or to the structural integrity of the cell, fire suppression decks will be provided. The purpose of the fire suppression deck will be to allow the sodium to enter the catch pan and then to provide a cover for the sodium pool in the catch pan to minimize its exposure to air and thus limit the pool burning. The fire suppression deck (see Figure 9.13) will be constructed of commercial corrugated steel with 1½-in. drain pipes extending from the deck to ½ in. above the bottom of the catch pan to allow the liquid metal to flow into the catch pan. Vents also will be provided in the deck to expel air as the sodium flows into the catch pan. With this arrangement the effective pool burning surface will be reduced to the surface of the sodium exposed in the pipes (1/250 of the floor surface area). Burning will be terminated when the pipes become plugged with combustion products (36 hours assumed in the design analysis). The decks will not be supported by or attached to the catch pans but rather will be supported separately from embedments in the cell concrete. This is done to reduce the loading on the catch pan walls during accident conditions. The deck will be designed for 1,000°F accident conditions and seismic Category I conditions. Corrosion of the catch pan plate material will be assumed in the design, and a 1/16-in. corrosion allowance will be provided. In addition, a coating to retard corrosion will be applied to both sides of the catch pan material.

The design analysis will demonstrate catch pan integrity during postulated accident conditions. The stress/strain criteria to be applied in the design analysis of the catch pans are discussed in Section 3.8 of the SER.

Structural concrete temperatures during postulated spill conditions are to be kept within the limits specified in Paragraph CC-3440 of ASME Code, Section III, Division 2 (i.e., 150°F steady state, 350°F transient unless higher temperature can be justified.).

Penetrations through the catch pan and fire suppression deck system will be designed to accommodate all loads caused by the system without failure and will not affect the integrity of the system. In addition, in those cells that will contain both sodium and water-steam piping and components, features will be provided to minimize the potential for impingement of leaks from one onto the

other. Features also will be provided to limit the collection of water in the catch pans resulting from condensation.

The NDE requirements for the catch pans are to be identical to those for the cell liners except that the cells containing the catch pans will not be pressurized and checked for leaks, since they are not intended to be controlled leakage areas. Inservice inspection requirements on the catch pans will be similar to those for the cell liners. The NDE and inservice inspection requirements for the fire suppression decks will consist only of visual examination, since they will not be load-bearing structures.

The concept of fire suppression decks has been proven in small-scale tests run previously at HEDL and in Germany. Although the designs tested were not prototypic of the CRBR design, the effectiveness of fire suppression decks was confirmed. A full-scale test of the CRBR catch pan and fire suppression deck is planned for 1983. In this test approximately 22 tons of sodium at 1,000°F will be spilled and the performance of the system observed.

A preliminary spill analysis has been performed on all catch pan and fire suppression deck systems. In this analysis all liquid metal discharged from the pipe was assumed to be in the form of small droplets (spray condition) and pool fire conditions were assumed when it gathers in the catch pan. The SPCA code was used for the spray fire and pool fire analysis and the HAA-3B code was used for aerosol generation. The SPCA code will be verified using the results from the sodium spill test planned for 1983.

Cell pressures and concrete temperatures have been analyzed and found to be within allowable limits. Aerosol release rates were predicted by the applicants to be within desired values. Results from the most limiting cases are presented in Sections 15.6 and 15.7 of the PSAR and are evaluated in the corresponding sections of the SER. The structural criteria of the catch-pan/fire-suppression deck system is evaluated in Section 3.8 of this SER. It should be noted that analysis of fires and aerosol releases from air-filled cells other than the steam generator building were not provided. The applicants have committed, however, to analyze these and to control any release of aerosol to a value that will not affect safety-related equipment (letter, J. R. Longenecker to P. S. Check, dated Jan. 7, 1983, #HQ:S:83:171).

Conventional fires in air-filled cells containing sodium or NaK systems will, if not too large, be fought by the application of dry extinguishment. Large fires in these cells will be left to burn themselves out. A fire in a cell is expected to be contained in that cell, and since plant equipment necessary for safe shutdown and decay heat removal will be redundant and physically separated, a fire in a cell will not detract from the ability to shut down and maintain the plant in a safe condition.

Instrumentation

Permanent instrumentation will be provided to detect and alarm a liquid metal fire in noninerted areas of the plant. This instrumentation will not be part of the leak detection instrumentation, discussed in SER Section 7.5.2.5, that

will monitor inerted areas. This fire detection instrumentation will be the same instrumentation as that used to detect and alarm a conventional fire and will meet the requirements of Appendix R of 10 CFR 50 and BTP CMEB-9.5-1. Seismic Category III has been specified for this instrumentation by the applicants, and the applicants have committed, in the event of an earthquake, to establish a fire watch in the plant until the instrumentation is verified operable. Fire alarms will be individually investigated by the operators to determine the specific location and cause (i.e., liquid metal, electrical, and so forth). The extinguishment methods will be determined on the basis of the results of this investigation and established fire-fighting practices.

The types of detectors to be used are ionization chamber detectors (detect smoke or aerosols), photoelectric detectors (detect smoke or aerosols) and thermal radiation detectors (detect the presence of heat). These detectors will provide an alarm locally as well as at a local panel and in the control room. The locations and types of detectors are listed in PSAR Table 9.13-11. For temporary monitoring of the inerted cells, when not inerted, receptacles located both outside and inside the cells for plugging portable or temporary ionization chamber-type detectors into the appropriate area fire zone panels will be provided. When the cell is being deinerted, a detector will be plugged into the outside receptacle and will monitor the exhaust of the cell. When the cell is deinerted, the detectors will be plugged into the receptacle inside the cell and will monitor the internal cell atmosphere.

Each steam generator cell will contain engineered safety feature aerosol detectors in the H&V (SGB aersol release mitigation system). The purpose of these detectors is to provide a signal to the dampers in the SGB H&V inlet and exhaust ducts to close after 5 minutes in the event of a leak in an intermediate heat transport system loop. A signal also will be provided to close other plant H&V inlets. This will allow 5 minutes of venting for the SGB to avoid overpressurizing and opening other exhaust paths in the building. This action will limit the amount of aerosol discharged from the SGB to 630 lb--a value selected by the applicants to limit aerosol ingestion by other plant components important to safety.

The 630-lb release limit will be met by releasing through a controlled vent area a maximum of 440 lb of aerosols during the first 5 minutes of the accident, 90 lb of aerosols through building cracks between 5 minutes and 5,000 sec, and an additional 100 lb of aerosols through building cracks after 5,000 sec. Each steam generator cell will have two sets (three detectors in each set) of these aerosol detectors powered from Class 1E batteries. Evaluation of the instrumentation and control portion of this system is described in SER Section 7.3.2.4. The operation of this system is to be verified as part of the sodium spill test planned for 1983.

Fire Extinguishment and Piping Insulation

The only extinguishment capability (other than the inerted cells and fire suppression decks) will be portable hand-held and wheel-cart fire extinguishers to be used on small sodium or NaK fires. These will be filled with NaX (sodium carbonate) which is noncorrosive to stainless steel piping. They will be placed in locations convenient to areas containing sodium and NaK equipment.

Protective clothing and self-contained breathing apparatus also will be provided.

Alumina silicate insulation will be used on the sodium and NaK piping. This insulation is nonreactive with sodium or NaK and is compatible with the stainless steel piping. It also meets the requirements of RG 1.36, "Non-Metallic Thermal Insulation for Austenitic Stainless Steel."

Summary and Conclusions

In general, the applicants will use methods that are conventional in the mitigation of liquid metal fires. The amount of oxygen available for combustion of the liquid metal will be limited by controlling the atmosphere in the lined cells. Concrete-liquid metal reactions will be prevented because a nonreactive barrier in the form of a liner or catch pan will be furnished. The concrete structure will be protected from high temperatures by the use of insulating concrete and an air gap, which also will be used as a means of venting hot gases from behind the liner to prevent overpressurization during accident conditions.

In the air-filled cells, features will be provided, where necessary, to minimize the exposure of spilled sodium to the air atmosphere. Aerosol detection and mitigation features also will be provided, where necessary.

Specific items which should be emphasized are:

(1) The use of cell liners and covered catch pans to suppress pool burning is a well-established concept and has been developed to varying degrees in several countries. The use of liners and catch pans at CRBRP will be similar to that at FFTF. A prototypic cell liner has been tested at HEDL under conditions more severe than those expected at CRBRP. No liner failure occurred.

(2) Covered catch pans have been tested under simulated sodium spill conditions at HEDL and at the Karlsruhe Nuclear Research Center and, although not prototypic of CRBR, were shown to be effective in suppressing pool burning. The catch pan fire suppression deck design for CRBRP will incorporate features to reduce the burning rate to approximately 1/250 of that of an open pool and to terminate burning with the sodium pool at an elevated temperature. A full-scale test of a catch pan and fire suppression deck system, prototypic of CRBRP, is to be performed by the applicants, and the results factored into the design and analysis.

(3) Spill volumes assumed in the analysis are the maximum available to spill.

The applicants also prepared a preliminary Fire Hazards Analysis Report for CRBRP. This report summarized the fire hazard in each area of the plant and indicated that a fire in any area would not affect safe shutdown capability. The applicants plan to expand and complete this report in support of the final design.

During sodium spill accident conditions, the cell pressures and concrete temperatures are not predicted to exceed design values. The maximum values calculated by the applicants during these conditions are cell pressures approximately 80% of their design values and concrete temperatures of ~275°F, with most cell pressures being approximately 50% of their design pressure.

To verify the applicants' analysis methods, the staff (by contract with Brookhaven National Laboratory) performed independent analyses of certain of the sodium spills postulated by the applicants. The purpose of these independent analyses was to review selected sodium fire analyses performed by the applicants. Exact agreement with the applicants' results was not considered necessary but rather what was looked for was assurance that the applicants' analysis methods are conservative, that the design conditions of the system will not be exceeded during postulated accident conditions, and that the temperature trends during the transient shown in the staff's and applicants' calculations are similar.

Two cases were selected for analysis:

(1) the spill of 35,000 gal of 1,015°F primary sodium from a primary HTS loop into an inerted cell

(2) the spill of 32,000 gal of 400°F primary sodium from the primary sodium in-containment storage tank into an air-filled cell open to the containment

These cases were chosen since, according to the applicants' analysis, they represent the worst challenge to the HTS cells and containment building, respectively, for design-basis accidents. The code used by the staff in the PHTS analysis (NACOM) was independent of any code used by the applicants and had a basic difference from the applicants' codes in that in the NACOM code the treatment of spray fires assumed a normal distribution of the droplet size, whereas in the codes used by the applicants the droplets were all of a fixed size. In addition, the input to the NACOM code was independently checked so as to maintain independence from that used by the applicants. The NACOM code was verified using results from actual sodium fire experiments, as were the applicants' codes. The code used in the containment analysis (SOFIRE-II) was not used by the applicants for pool fires in air-filled cells; however, it has been verified by comparison with experimental results for such analysis. The results from the independent analysis were:

(1) A spill of 35,100 gal of 1,015°F sodium in an inerted primary heat transport system (PHTS) cell was analyzed. A peak cell pressure of 18.4 psig and a peak cell atmosphere temperature of 950°F were calculated. This calculated pressure and temperature are higher than those calculated by the applicants (14.4 psig and 680°F, respectively) but are well within the design capability of the PHTS cells (30 psig and 1,015°F, respectively). Structural concrete temperatures also stayed below their 350°F transient limit.

(2) A spill of 32,000 gal of 400°F sodium in an air-filled in-containment cell was analyzed. The calculated peak in-containment pressure and temperature were lower than those calculated by the applicants and are well within the design capability of the containment building (10 psig and 250°F, respectively).

In both cases analyzed, the cell pressure and temperature trends during the transient were similar to those calculated by the applicants.

On the basis of the results of its overcheck calculations, the staff concludes that the analysis methods used by the applicants provide a reasonable estimate of the effect of a sodium fire on plant temperatures and pressures. Since the staff's calculation of the response to a sodium spill in an inerted cell indicated that the applicants were underpredicting cell pressure and temperature (as compared with the staff's calculations), it will be necessary for the applicants to demonstrate during final design (1) that the design pressure and temperature of all inerted cells are sufficiently greater than the predicted pressure and temperature during a sodium spill so that there is margin to accommodate this underprediction or (2) that the applicants' analysis methods are, in fact, sufficiently conservative.

In the analysis of the hazard of sodium or NaK fires, primary attention is given to the offsite radiological consequences and to the onsite effect on plant equipment. However, sodium and NaK combustion products can, in sufficient quantity, represent a chemical hazard to the population. The staff has reviewed these hazards, and the results and conclusions follow.

Sodium reacts with the oxygen in air to form principally Na_2O and with the smaller amounts of carbon dioxide and water vapor universally present in air to form NaOH, H_2, $NaHCO_3$, and Na_2CO_3. The sodium combustion products rapidly form aerosol particles, which tend to drop out of air suspension.

If the fire involved NaK, the combustion products would contain analagous potassium compounds as well as the sodium compounds listed above.

In general, the combustion products of a sodium or NaK fire in air at CRBR would consist of a combination of the above compounds.

All of the sodium combustion products are not inherently toxic. Most animal tissue, however, will react with and be decomposed by NaOH.

A person exposed to the above combustion products in sufficient quantity would experience irritation of the eyes, nose, throat, and lungs. It is probable that exposure to a high concentration of these compounds for sufficient time would be lethal. Any fatalities would probably be due to the combustion product reaction on the respiratory system, since no systemic effects have been recognized. These conclusions are based on the following.

Some experimental data relevant to the health affects of exposure to sodium combustion products have been developed and are summarized below.

(1) Lethal concentrations of nonradioactive sodium combustion products to laboratory animals (rats, guinea pigs, mice, hamsters) have been studied at Battelle Northwest Laboratory and documented in reports PNL-2500, PNL-2850, PNL-3300, and PNL-3700. Their experiments were all for a 2-hour exposure and indicated that the lethal concentration (as characterized by the death of 50% of the exposed animals a short time after exposure) for the 2-hour exposure ranged from 570 to 6,200 mg/m^3 of combustion products in the atmosphere. Death was due to the affect of the combustion products on the

animal's respiratory system, not to systemic effects. For concentrations less than 50 mg/m^3 for a 2-hour duration, no tissue damage was evidenced. These exposures were not translated to an exposure applicable to humans was not done; however, the equivalent human exposure would be expected to be lower.

(2) Data involving human reactions to high concentrations of sodium combustion products (up to 100 mg/m^3) for short durations are offered in Hanford Engineering Development Laboratory report HEDL-TME-79-17. The results indicated that although there was some irritation to the eyes and lungs, exposures up to 100 mg/m^2 were tolerable for short durations.

It should also be noted that the American Conference of Governmental Industrial Hygienists has set a limit of 2 mg/m^3 of NaOH in the atmosphere as the occupational exposure limit (40 hr per week). They have also recommended a 25 mg/m^3 occupational limit for Na_2O. An occupational limit of 2 mg/m^3 has been set for KOH by F. A. Patty (1963).

To determine whether a sodium fire at CRBRP will present any chemical hazard to the public, the predicted combustion product concentrations at the site boundary must be known. The applicants have calculated the worst-case combustion product concentration at the site boundary from the design-basis accidents. This concentration was 6.7 mg/m^3, which lasted for approximately 5 minutes and was assumed to be all NaOH. The accident evaluated was an assumed sodium spill in the steam generator building (SGB), which resulted in the release of 630 lb of combustion products through the ESF aerosol mitigation system (440 lb in the first 5 min and then smaller release rates for the duration of the event). The applicants also have committed (letter, J. R. Longenecker to P. S. Check, dated Jan. 7, 1983, #HQ:S:83:171) to ensure that the design of other air-filled cells in the plant will limit aerosol releases so as not to affect safety-related equipment and to take supply precautions to limit combustion product release in the event of a sodium fire during sodium fill operations.

On the basis of the above considerations, the staff concludes that the risk of permanent injury to the public from the chemical effects of a sodium or NaK fire at CRBR is very low. Some temporary irritation may, however, occur and the emergency planning for CRBRP should address such an occurrence (see SER Section 13.3.3).

If future analysis indicates that a site boundary concentration in excess of 6.7 mg/m^3 could occur, justification must be provided on the acceptability of the higher concentration.

The staff's review of the PSAR and preliminary Fire Hazards Summary Report has identified several areas requiring additional documentation, corrections, or justification in the final design.

These are:

(1) The basis for the limit on the allowable release of sodium combustion products (630 lb of aerosol) needs to be provided. This basis should demonstrate why 630 lb is the correct number.

(2) The predicted combustion product release (both to the outside atmosphere and to other internal areas of the plant) from all cells where a sodium or NaK spill is possible should be documented. Plant features necessary to ensure that the release will be within established limits should also be documented along with documentation that demonstrates that any affected safety equipment will be qualified for the resulting environment.

(3) Qualification of the SGB aerosol release mitigation system to perform its function under the conditions it would experience in the plant must be demonstrated and documented. This also applies to any other aerosol mitigation features added to the plant to maintain aerosol releases at values that will not affect operation of plant safety equipment.

(4) Documentation on the affect of vented steam from behind the cell liners on plant safety equipment should be provided. Specifically, the location of the vented steam, the safety equipment subjected to the steam, and the documentation that demonstrates that this equipment will be qualified for the resulting environment should be provided.

(5) The protection of cable trays passing through or near catch pans from the effects of a sodium or NaK spill should be documented.

(6) The analysis of a spill of intermediate heat transport system sodium into a primary heat transport system cell should be documented. If this spill should result in higher cell temperatures and pressure than a primary sodium spill, it should be included in FSAR Chapter 15.

(7) The techniques to be used to accommodate liquid metal jet impingement on the concrete walls of the cells containing catch pans should be documented. The applicants committed to the investigation of these techniques in the response to Question CS760.158.

(8) Additional documentation that addresses, for each cell containing liquid metal components, the potential spill volumes, temperatures, and results of the fire analysis, and that demonstrates that the cell and concrete temperature and pressure limits will not be exceeded for the event should be provided.

(9) All cells with catch pans connected by drains should be included in the same fire area. The final Fire Hazard Summary Report should clearly document this.

(10) All boundaries of fire areas (walls, ceilings, floors, doors, and penetrations) should have a fire rating associated with them that should be documented in the final Fire Hazard Analysis Report.

With proper resolution of the above items during final design, the staff concludes that the plant design criteria and preliminary design are acceptable and that no open items remain in the sodium fire protection area which would preclude issuing a construction permit.

9.14 Diesel Generator Auxiliary Systems

Except for the diesel generators' exhaust stacks and the fuel oil storage tanks (including fill, vent, stick, and transfer lines), the diesel generators will be housed in a separate, dedicated seismic Category I building which will provide protection from the effects of tornadoes, tornado missiles, and floods. Therefore, the requirements of CRBR principal design criteria (PDC) 2, "Design Bases for Protection Against Natural Phenomena," and 5, "Environmental and Missile Design Basis," and the recommendations and guidance of RGs 1.115 and 1.117 will be met. PDC 2 and 5 are the same as GDC 2 and 4 of Appendix A of 10 CFR 50.

Protection from the effects of tornadoes, tornado missiles, and floods is evaluated in Section 3 of this SER. Protection of the fuel oil storage tanks and the diesel generators' exhaust stacks from the effects of natural phenomena is discussed in Sections 9.14.1 and 9.14.5, respectively, of this SER.

There will be three emergency diesel generators at CRBRP and each diesel engine will have the following auxiliary systems, which are addressed in detail in the SER sections indicated.

(1) fuel oil storage and transfer system (Section 9.14.1)
(2) cooling water system (Section 9.14.2)
(3) starting system (Section 9.14.3)
(4) lubrication system (Section 9.14.4)
(5) combustion air intake and exhaust system (Section 9.14.5)

Each diesel generator and its auxiliary systems will be independent of one another. Thus, the requirements of PDC 6, "Sharing of Structures, Systems, and Components," will be met. PDC 6 is identical to GDC 5 of Appendix A of 10 CFR 50.

The diesel engine and engine-mounted components, as well as separately skid-mounted portions of the auxiliary support systems piping and components normally furnished with the diesel generator package, will be designed to seismic Category I requirements and will follow the guidelines of the Diesel Engine Manufacturers Association (DEMA) standards. The diesel engine and engine-mounted auxiliary support systems piping and components will conform to the requirements of IEEE Std. 387-1977, "Standard Criteria for Diesel-Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations," which endorses the DEMA standards and guidelines of RG 1.9. The diesel engine and its auxiliary support systems will also meet the quality control requirements of 10 CFR 50, Appendix B. The quality assurance program is evaluated in Section 17 of this SER.

Accumulation of dust, including dust generated from concrete floors and walls, on the electrical equipment associated with starting of the diesel generators (e.g., auxiliary relay contacts, control switches) will be limited by the design of the combustion air intake system, use of filters in the ventilation system of the diesel generator building, a dust-proof enclosure for electrical equipment, and surface treatment of concrete floors and walls to minimize creation of concrete dust.

A training program for operating and maintenance personnel will be developed and implemented for the CRBRP. The training program will be similar to the program currently at use by TVA. Diesel engine manufacturers' recommendations will be used in developing the training program, and the final product will be comparable to vendor training.

Except for sensors and other equipment that must be directly mounted on the engine and associated piping, the controls and monitoring instrumentation will be installed on a freestanding floor-mounted panel separate from the engine skids and located in a vibration-free floor area.

Preventive maintenance at the CRBRP will go beyond the routine adjustments, servicing, and repair of components when a malfunction occurs. The preventive maintenance program will encompass investigation of components that have a history of repeated malfunctioning and require constant attention and repair. The applicants will also be reviewing operating experiences of other utilities through vendor programs to aid in identifying problems.

Administrative procedures will be implemented so that on completion of repairs or maintenance, and before an actual start, run, and load test, a final equipment check will be made to ensure that all electrical circuits are functional; that is, fuses are in place, switches and circuit breakers are in their proper positions, no wires are loose, all test leads have been removed, and all valves are in the proper position to permit a manual start of the equipment. After the unit has been satisfactorily started and load tested, the unit will be returned to automatic standby service and under the control of the control room operator.

The applicants discussed their procedures for no-load and light-load operation of the diesel generators. The applicants committed to implement the following procedures before startup.

(1) During periodic testing, the diesel will be loaded to a minimum of 25% of full load.

(2) Administrative procedures will be implemented to limit no-load or light-load operation of the diesel generators.

(3) Provisions will be made to adequately load the diesel generators during load testing.

The proposed designs of the diesel engine auxiliary support systems were evaluated with respect to the recommendations and guidelines of BTP ASB 3-1, "Prevention Against Postulated Piping Failures in Fluid System Piping Outside Containment," and BTP MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Evaluation of protection against dynamic effects associated with the postulated pipe system failures is covered in Section 3.6 of this SER.

The adequacy of the proposed fire protection for the emergency diesel generator and associated auxiliary support systems with respect to the recommendations and guidelines of BTP CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," is evaluated in Section 9.5.1 of this SER.

The proposed designs of the diesel generator auxiliary support systems also have been evaluated with respect to the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." This report made specific recommendations on increasing the reliability of nuclear plant emergency diesel generators.

Requests for information concerning these recommendations were transmitted to the applicants during the review process. The applicants responded in the amendments to the PSAR, stating how they will meet the recommendations of NUREG/CR-0660.

The staff has reviewed these responses and has determined that the applicants' conformance to the recommendations is as follows.

Recommendation	Conformance	SER Section
(1) Moisture in air starting system	Yes	9.14.2
(2) Dust and dirt in diesel generator room	Yes	9.14
(3) Turbocharger gear drive problem	Yes	Design modified to eliminate problem
(4) Personnel training	Yes	9.14
(5) Automatic prelube	Yes	9.14.4
(6) Testing, test loading, and preventive maintenance	Yes	9.14
(7) Improve the identification of root cause of failures	Yes	9.14
(8) D/G ventilation and combustion air systems	Yes	9.14.5
(9) Fuel storage and handling	Yes	9.14.1
(10) High-temperature insulation	*	9.14
(11) Engine cooling water	Yes	9.14.3
(12) Concrete dust control	Yes	9.14
(13) Vibration of instruments	Yes	9.14

The design of the proposed diesel generator meets the requirements of PDC 2; 5; 6; 15, "Electric Power Systems"; 16, "Inspection and Testing of Electric Power Systems"; and 19, "Protection System Reliability and Testability." PDC 15, 16, and 19 are the same as GDC 17, 18, and 21 of Appendix A of 10 CFR 50. The proposed design is also in conformance with the recommendations of NUREG/CR-0660 for enhancement of diesel generator reliability and the related NRC guidelines and criteria. The staff, therefore, concludes that the design, when implemented,

*The staff considers explicit conformance unnecessary in view of the equivalent provided by the design, margin, and qualification testing requirements that are normally applied to emergency standby diesel generators.

will provide reasonable assurance of diesel generator reliability throughout the design life of the plant.

9.14.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

The design function of the emergency diesel engine fuel oil storage and transfer system is to provide a separate and independent fuel oil supply train for each diesel generator and to permit operation of the diesel generator at engineered safety feature load requirements for a minimum of 7 days without replenishment of fuel. The system will be designed to meet the requirements of PDC 2, 5, and 6, as discussed in Section 9.14 of this SER.

There will be three emergency diesel generators at the CRBRP. Each diesel engine fuel oil storage and transfer system will consist of an 1,100-gal day tank, a fuel oil storage tank (two 35,000-gal tanks in parallel for each of the division 1 and 2 diesel generators, and one 35,000-gal tank for the division 3 diesel generator) with adequate capacity to power the respective diesel generator for 7 days at fully loaded capacity, two motor-driven transfer pumps, two fuel oil booster pumps, and associated valves, piping, strainers, filters, instrumentation, and controls.

The fuel oil storage tanks will be under ground, made of reinforced concrete with steel liners, and will be in conformance with the requirements of ASME Code, Section VIII. Each tank will have fill, vent, test, suction, and return line connections. All fill connections will be fitted with filters to minimize the entrance of deleterious materials, and inlet lines to the storage tanks will be fitted with perforated diffusers to minimize turbulence during filling. Each tank will also be fitted with a manhole to facilitate periodic inspection and cleaning. Test connections will be provided for taking fuel samples, testing for water, or for removing water and/or contaminated fuel. All piping to and from the tank will be protected from tornado missiles, either by an appropriate thickness of earth cover, or by concrete missile shields. Fill, vent, and test connections will be located above the maximum flood level.

Two fuel oil transfer pumps will be provided each diesel generator to transfer fuel from the storage tanks to the day tanks. The pumps will be redundant, and each pump will be capable of delivering fuel at approximately four times that required by the diesel generator. The pumps will be ac motor driven and powered from their respective diesel generator Class 1E bus. A shaft-driven booster pump powered by the diesel engine and a motor-driven backup booster pump will be provided to deliver fuel from the day tank to the engine for each diesel generator unit. Power for the booster pump will be provided by its respective diesel generator Class 1E bus. Strainers and filters will be provided at appropriate locations in the system, and instrumentation and controls will be provided for monitoring, controlling, and alarming the system.

All piping and components between the storage tanks and the diesel engine-mounted piping and components, including the day tank, will be designed to ASME Code, Section III, Class 3. Diesel engine-mounted piping and components will conform to ANSI Std. B31.1, and manufacturers' standards. Each diesel

generator fuel oil storage and transfer system will be completely independent of the other systems and will be designed to accept a single active failure without loss of function. All systems, including diesel generators and engine-mounted piping and components, will be designed seismic Category I. The systems will meet the recommendations of RGs 1.26 and 1.29.

The design of the emergency diesel engine fuel oil storage and transfer systems will conform to ANSI Std. N195, "Fuel Oil Systems for Standby Diesel Generators." In addition, the fuel oil quality and tests will conform with the guidelines of RG 1.137.

The scope of review of the diesel engine fuel oil storage and transfer system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.14.1 of the PSAR for the system and auxiliary support systems essential to its operation.

The basis for acceptance in the staff's review was conformance of the design criteria and bases and proposed design of the diesel engine fuel oil storage and transfer system to the requirements of PDC 15, with respect to redundancy and physical independence, the guidance of the cited regulatory guides and SRP Section 9.5.4, the recommendations in NUREG/CR-0660, and industry codes and standards.

On the basis of its review, the staff concludes that the emergency diesel engine fuel oil storage and transfer system will meet the requirements of PDC 2, 5, 6, and 15, will meet the recommendations of NUREG/CR-0660, and is, therefore, acceptable.

9.14.2 Emergency Diesel Engine Cooling Water System

The design function of the emergency diesel engine cooling water system is to maintain the temperature of the diesel engine within a safe operating range under all load conditions and to maintain the engine coolant preheated during standby conditions to improve starting reliability. The system will be designed to meet the requirements of PDC 2, 5, 6, 15, 38, 39, and 40. PDC 38, 39, and 40 are identical to GDC 44, 45, and 46 of Appendix A of 10 CFR 50. Meeting the requirements of PDC 2, 5, and 6 is discussed in Section 9.14 of this SER.

For all three diesel generators at Clinch River, the cooling water system will be a closed loop, which provides capability for heat rejection from the engine jackets, lube oil coolers, governor lube oil cooler (divisions 1 and 2, only), and turbocharger aftercoolers. The major components of these systems will include an engine-driven circulating water pump (two engine-driven pumps for the division 3 diesel generator), an auxiliary motor-driven pump, heat exchangers for jacket water, lube oil, governor oil, and turbochargers, a standpipe or expansion tank, a thermostatic three-way valve, and associated piping and valves to connect the equipment. During operation, the heat generated by the diesel engines will be rejected to the emergency service water system by means of the cooling water heat exchangers. Also during operation, the temperature of the diesel engines will be regulated automatically through the action of the temperature-sensing three-way thermostatic valve. The auxiliary circulating pumps will operate

continuously during diesel engine operation and will serve as a backup to the engine-driven pumps. Power for the auxiliary pump motors will come from their respective diesel generator Class 1E buses.

A keep-warm system will be provided for each diesel generator. This system will be designed to maintain the temperature of the diesel engine jacket water at approximately 120°F when the diesel generator's are in the standby mode. Use of this keep-warm system will enhance the engines capability for starting and accepting load within the prescribed time interval. The system will consist of a motor-driven circulating pump, a thermostatically controlled electric immersion heater, and associated valves and piping. The system will be designed to start and stop automatically when the diesel generators are shut down and started, respectively.

The diesel generators will be capable of operating fully loaded without secondary cooling for a minimum amount of time. Sufficient water will be contained in the engine and stand pipe of the expansion tank to absorb the heat generated during this period. This time will be in excess of the time needed to restore emergency service water to the diesel generators in the event of a loss of offsite power. Alarms will be provided to enable the control room operator to monitor the cooling of the diesel generators while the unit is in the standby mode or in operation.

The three emergency diesel generators for CRBRP will be completely independent of each other. The division 1 and 2 generators will be redundant, and only two of the three generators will be required for safe plant shutdown. The cooling water systems also will be independent, and each will have a backup circulating pump. Therefore, the requirements of PDC 15, "Electric Power System," and PDC 38, "Additional Cooling Systems," as they relate to physical independence, redundancy, and single-failure criteria are met.

Piping and components of the diesel engines' cooling water system up to the diesel engine interface, including auxiliary skid-mounted piping, will be designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and will meet the recommendations of RGs 1.26 and 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and the associated components, such as valves, fabricated headers, and fabricated special fittings, will be designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI Std. B31.1, "Code for Pressure Piping"; ANSI Std. N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR 50, Appendix B. The engine-mounted cooling water piping and associated components are intentionally overdesigned (subjected to low-working stresses) for the application, and thereby result in high operational reliability. The design of the engine-mounted cooling water piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The diesel engine cooling water system will conform to RG 1.9, Position C.7, as it relates to engine cooling water protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this SER.

The diesel engine cooling water system will have provisions to permit periodic inspection and functional testing during standby and normal modes of power plant operation as required by PDC 39, "Inspection, of Additional Cooling Systems," and PDC 40, "Testing of Cooling Water System."

The scope of review of the emergency diesel engine cooling water system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.14.2 of the PSAR for the system and auxiliary support systems essential to its operation.

The basis for the acceptance in the staff's review was conformance of the design criteria and bases and proposed design of the diesel engine cooling water system to PDC 15 and 38 with respect to redundancy and physical independence, PDC 39 and 40 with respect to inspection and testability of the system, the guidance of the cited regulatory guides and SRP Section 9.5.5, the recommendations of NUREG/CR-0660 and industry codes and standards, and the ability of the system to maintain conditions.

On the basis of its review, the staff concludes that the emergency diesel engine cooling water system will meet the requirements of PDC 2, 5, 6, 15, 38, 39, and 40; will meet the guidance of the cited regulatory guides; will meet the recommendations of NUREG/CR-0660 and industry codes and standards; will be capable of performing its design safety function; and is therefore acceptable.

9.14.3 Emergency Diesel Engine Starting Systems

The design function of the emergency diesel engine starting system is to provide a reliable method for automatically starting each diesel generator so that the rated frequency and voltage will be achieved and the unit will be ready to accept required loads within 10 sec. The system will be designed to meet the requirements of PDC 2, 5, 6, and 15. Meeting the requirements of PDC 2, 5, and 6 is discussed in Section 9.5.4.1 of this SER.

There will be three emergency diesel generators at CRBRP. Each diesel generator will have an independent and redundant air-starting system consisting of two separate full-capacity air-starting subsystems, each with sufficient air capacity to provide a minimum of five consecutive cold-engine starts.

Further redundancy will be provided by the three diesel generators, each of which will be independent of the others, and any two of which will provide power for safe plant shutdown. A malfunction in an air-starting system, or a malfunction of one diesel generator, will not impair the ability of the other diesel generators to start. Therefore, the requirements of PDC 15, "Electric Power Systems," are met.

Each subsystem will include an air compressor, a receiver, a desiccant-type air dryer, air filters and strainers, starting air valves, and interconnecting piping and valves. Each subsystem will be provided with instrumentation, controls, and alarms. The air compressors will be automatically started and stopped as a function of air pressure in the receivers. Provisions will be made for local monitoring of system conditions, and low-pressure alarms will be annunciated locally and in the control room. Each redundant air-starting system will have a separate dc power supply for starting controls.

The piping and component of the diesel engines' air-starting system up to the diesel engine interface, including auxiliary skid-mounted piping, will be designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and will meet the recommendations of RGs 1.26 and 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and the associated components, such as valves, fabricated headers, and fabricated special fittings, will be designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI Std. B31.1, "Code for Pressure Piping"; ANSI Std. N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR 50, Appendix B. The engine-mounted air-starting piping and associated components are intentionally overdesigned (subjected to low-working stresses) for the application, thereby resulting in high operational reliability. The design of the engine-mounted air-starting piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The diesel generator air-starting system will conform to RG 1.9, Position C.7, as it relates to diesel engine air-starting system protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this SER.

The scope of review of the emergency diesel engine air-starting system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.14.3 of the PSAR for the system and auxiliary support systems essential to its operation.

The basis for acceptance in the staff's review was conformance of the design criteria and bases and proposed design of the diesel engine air-starting system to the requirements of PDC 15 with respect to redundancy and physical independence, the additional acceptance criteria and guidance of SRP Section 9.5.6, the recommendations of NUREG/CR-0660 and industry codes and standards, and the ability of the system to start the diesel generator within a specified time period.

On the basis of its review, the staff concludes that the proposed emergency diesel engine air-starting system will meet the requirements of PDC 2, 5, 6, and 15; will meet the guidance of the cited regulatory guides, SRP Section 9.5.6, and industry codes and standards; will meet the recommendations of NUREG/CR-0660; and is, therefore, acceptable.

9.14.4 Emergency Diesel Engine Lubricating Oil System

The design safety function of the emergency diesel engine lubricating oil system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine, including pistons and bearings. The system will be designed to meet the requirements of PDC 2, 5, 6, and 15. Meeting the requirements of PDC 2, 5, and 6 is discussed in Section 9.14 of this SER.

Major components of the division 1 and 2 emergency diesel engine lubricating oil system will include an engine-driven lube oil circulating pump; a lube oil cooler, strainers, and filters; an electric immersion heater; a motor-driven keep-warm pump; instrumentation; controls; and alarms. Provisions will be made to minimize crankcase explosion hazards, and alarms and protective devices will be provided to alert the control room operator to any abnormal condition in a lube oil system during standby, startup, or operating status.

During operation, the division 1 and 2 engine-driven lube oil circulating pump will supply oil to the vital parts of the diesel engine via the lube oil cooler. Excess heat will be rejected to the jacket water in the lube oil cooler. During standby, the motor-driven pump will provide continuous lubrication to the diesel engine wearing parts, except for the upper cylinder (valve train) area and the turbocharger sleeve bearings, to preclude accumulation of lube oil in the exhaust or in one or more cylinders. Oil temperature will be maintained during standby with a thermostatically controlled immersion heater located in the lube oil sump.

Major components of the division 3 emergency diesel engine lubricating oil system will include an engine-driven scavenging pump, two engine-driven lube oil circulating pumps, a lube oil cooler, strainers and filters, a motor-driven turbocharger circulating pump, a motor-driven keep-warm pump, instrumentation, controls, and alarms. Provisions will be made to minimize crankcase explosion hazards, and alarms and protective devices will be provided to alert the control room operator to any abnormal conditions in the lube oil system during standby, startup, or operating status.

During operation, the engine-driven scavenging pump will take lube oil from the engine sump and pump it through the lube oil cooler to the inlet of the engine-driven circulating pumps. These pumps will supply oil to the vital parts of the diesel engine. Excess heat will be rejected to the jacket water in the lube oil cooler. During standby, one motor-driven pump will provide continuous lubrication to the engine at all levels below the camshaft. This will prevent the accumulation of lube oil in the diesel engine exhaust. The second motor-driven pump will provide continuous lubrication to the turbocharger during operation and standby. To improve diesel engine starting, lube oil on standby will be preheated by circulating the lube oil through the lube oil cooler where it will pick up heat from the preheated jacket water.

The emergency diesel engine lube oil system will be an integral part of the diesel engine and thus meets the requirements of PDC 15 with regard to system independence and single-failure criteria.

The piping and components of the diesel engine lubrication oil system up to the diesel engine interface, including auxiliary skid-mounted piping, will be designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and will meet the recommendations of RGs 1.26 and 1.29.

The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and will be seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and associated components, such as valves, fabricated headers, and fabricated special fittings, will be designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI Std. B31.1,

"Power Piping," and 10 CFR 50, Appendix B. The engine-mounted lube oil piping and associated components are intentionally overdesigned (subjected to low-working stresses) for the application, thereby resulting in high operational reliability. The design of the engine-mounted lube oil piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The diesel generator lubricating oil system will conform to RG 1.9, Position C.7, as it relates to diesel engine lubrication system protective interlocks. The protective interlocks in the diesel generator are discussed in Section 8.3 of this SER.

The scope of review of the diesel generator lubricating oil system included piping and instrumentation diagrams, and descriptive information in Section 9.14.4 of the PSAR for the system and auxiliary support systems essential to its operation.

The basis for acceptance in the staff's review was conformance of the design criteria and bases and proposed design of the diesel engine lubricating oil system to the requirements of PDC 15 with respect to redundancy and physical independence, the additional acceptance criteria and guidance of SRP Section 9.5.7, and the recommendations of NUREG/CR-0660 and industry codes and standards.

On the basis of its review, the staff concludes that the emergency diesel engine lubricating oil system will meet the requirements of PDC 2, 5, 6, and 15; will meet the guidance of the cited regulatory guides and SRP Section 9.5.7; will meet the recommendations of NUREG/CR-0660 and industry codes and standards, and will perform its design safety function, and is, therefore, acceptable.

9.14.5 Emergency Diesel Engine Combustion Air-Intake and Exhaust System

The design function of the emergency diesel engine combustion air-intake and exhaust system is to supply filtered air to the engine for combustion and to dispose of engine exhaust to the atmosphere. The system will be designed to meet the requirements of PDC 2, 5, 6, and 15. Meeting the requirements of PDC 2, 5, and 6 is discussed in Section 9.14 of this SER.

Each diesel generator will be located in a separate room of the diesel generator building. Combustion air for each diesel generator will be drawn from outside into an intake plenum and will pass through a filter, silencer, and connecting piping to the diesel engine turbocharger inlet. Exhaust gas will be routed through the turbocharger turbine to the exhaust silencer and into the exhaust stack. There will be no interconnection between diesel generator combustion air-intake and exhaust systems. Therefore, the requirements of PDC 15, "Electric Power Systems," with regard to system independence, redundancy, and single-failure criteria, are met.

The exhaust stack will terminate on the roof of the diesel generator building and will be protected from tornado missiles. The exhaust and intake will be separated horizontally by a distance of 70 ft to preclude short-circuiting of exhaust gas into the combustion air. No gaseous fire-extinguishing agents will

be used in or near the diesel generator building. This will preclude possible lowering of the oxygen content of combustion air and resultant degradation of diesel generator performance.

The diesel generator combustion air-intake and exhaust system conforms to RG 1.9, Position C.7, as it relates to diesel engine combustion air-intake and exhaust system protective interlocks. The protective interlocks of the diesel generator system are discussed in Section 8.3 of this SER.

The piping and components of the diesel engine combustion air-intake system up to the diesel engine interface, including auxiliary skid-mounted piping, will be designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and will meet the recommendations of RG 1.26.

The piping and components of the diesel engine exhaust system, from the diesel engine interface up to the diesel generator building boundary, will be designed to seismic Category I requirements and will meet or exceed the recommendations of RG 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine block to the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. The engine-mounted piping and associated components, such as fabricated headers and fabricated special fittings, including air intake and exhaust piping beyond engine interface, will be designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI Std. B31.1, "Code for Pressure Piping"; ANSI Std. N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR 50, Appendix B. The engine-mounted intake and exhaust piping and associated components are intentionally overdesigned (subjected to low working stresses) for the application, thereby resulting in high operational reliability. The design of the engine-mounted intake and exhaust piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The scope of review of the diesel generator intake and exhaust system included layout drawings, piping and instrumentation diagrams, and descriptive information in Section 9.14.5 of the PSAR for the system and auxiliary support systems essential to its operation.

The basis for the acceptance in the staff's review was conformance of the design bases and proposed design of the diesel engine air-intake and exhaust system to PDC 15 with respect to redundancy and physical independence, the guidance of the cited regulatory guides, the additional acceptance criteria and guidance of SRP Section 9.5.8, the recommendations of NUREG/CR-0660 and industry codes and standards, and the ability of the system to provide sufficient combustion air and release of exhaust gases to enable the emergency diesel generator to perform on demand.

On the basis of its review, the staff concludes that the proposed emergency diesel engine combustion air-intake and exhaust system will meet the requirements of PDC 2, 5, 6, and 15; and the recommendations of the cited regulatory guides and industry codes and standards. It will be capable of performing its design safety function, and is, therefore, acceptable.

9.15 Equipment and Floor Drainage System

The plant equipment and floor drainage system will provide drainage for plant equipment such as pumps, tanks, and coolers as well as drainage for the floor areas of the plant during normal operating conditions. In addition, the drainage system will be sized to accommodate the maximum postulated flooding event such as a pipe rupture, tank rupture, or sprinkler discharge.

The equipment and floor drainage systems' design criteria and bases and preliminary design were reviewed in accordance with SRP Section 9.3.3, "Equipment and Floor Drainage System" (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the systems' design criteria and bases and preliminary design meet the applicable portions of the following LWR GDC of 10 CFR 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the system's capability to withstand earthquakes (identical to CRBR PDC 2)

(2) GDC 4, "Environmental and Missile Design Bases," with respect to the system's capability to withstand the effects and to be compatible with the environmental conditions (flooding) associated with normal operation, accidents (pipe break, tank ruptures) (comparable to CRBR PDC 5)

(3) GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to providing a means for controlling suitably the release of radioactive materials in liquid effluent, including anticipated operational occurrences (identical to CRBR PDC 52)

The acceptance criteria of SRP Section 9.3.3 also include the guidance of RG 1.29, Position C.1, if any portion of the system is deemed to be safety related, and Position C.2 for nonsafety-related functions.

Except for the difference between LWR GDC 4 and CRBR PDC 5, SRP Section 9.3.3 is directly applicable to the CRBR equipment and floor drainage system. The SRP review was augmented by a review to determine compliance of the system with CRBR PDC 4, "Protection Against Sodium Reactions," as it relates to system design provisions to minimize possible contacts between sodium and water. Conformance with the acceptance criteria of SRP Section 9.3.3 and the applicable portions of CRBR PDC 4 provide the bases for concluding that the system's design criteria and bases and preliminary design are acceptable.

The equipment and floor drainage system will not be safety related, except for the piping and valves required for containment isolation. Separate equipment and floor drainage system sumps will be provided for radioactive, potentially radioactive, and nonradioactive areas of the plant. Each sump will contain two vertical sump pumps with one pump serving as a full-capacity spare. Radioactive fluids in floor drains will go through a common pipe and will be routed to the radioactive liquid waste treatment sump. Potentially radioactive fluids in drains will be routed or pumped to a main collection sump in the reactor service building radwaste area and monitored for level of radioactive contamination.

Contaminated sump influent will be pumped to the radioactive liquid waste sump where it then will be processed through the radwaste treatment system. Sump contents not radioactively contaminated will then be pumped to the waste water disposal system. In a meeting on December 13, 1982, the applicants verified that a normally closed, manually operated valve would be provided in the piping from the main collection sump to the waste water disposal system. Before this valve is opened to permit transfer of water from the sump to the waste water treatment ponds, the contents of the sump will be manually sampled to verify that contamination levels are acceptably low. Treated water and other process water treatment wastes that do no have the potential to be radioactively contaminated will be routed to separate sumps for transport to the waste water treatment system. Therefore, the staff concludes that the requirements of PDC 52 (GDC 60) will be satisfied.

Cells or compartments housing equipment containing sodium will not be provided with floor drains. Thus, the floor drain system will not provide a path for water backflow into compartments containing sodium. Therefore, the staff concludes that the requirements of PDC 4 will be satisfied.

Equipment and floor drainage system piping within areas containing safety-related equipment will be supported by seismic Category I supports. The piping and other components, except for piping and valves required for containment isolation, will be nonseismic Category I. The system will be designed to accommodate the maximum postulated flooding event, such as a pipe rupture, tank rupture, or sprinkler discharge, and to limit water accumulation to no more than 3½ in. All safety-related equipment will be mounted on pads at least 4 in. high. Equipment pad heights of 6 or 8 in. will be used where water can accumulate to levels above 4 in. where postulated complete blockage of drainage system piping could occur. The higher pad height was determined by a flooding study performed by the applicants. In some cases (also determined by the flooding study), gaps under doors have been increased to provide an alternative drainage path from compartments housing safety-related equipment. The applicants' flooding study also verified that backflow between connected cells will not occur. Each drain sump will contain two vertical sump pumps with one pump serving as a full-capacity spare. Each sump will be provided with automatic level controls to stop, start, and alternate operation of the sump pumps. Level detection for the sumps to automatically start the pumps when needed and to alarm potential sump overflow in the control room will be provided. The above described features will satisfy the requirements of PDC 2 (GDC 2) and PDC 5 (GDC 4) and the guidelines of RG 1.29, Position C.2.

The staff concludes that the proposed equipment and floor drainage system's design criteria and bases and preliminary design meet the requirements of PDC 2, 4, 5, and 52, with respect to the system's protection against natural phenomena, provisions to minimize possible contact between water and sodium, protection against environmental effects (flooding), and provisions to prevent release of radioactive material to the environment, and the guidelines of RG 1.29, Position C.2, with respect to seismic classification, and are, therefore, acceptable.

9.16 Recirculating Gas Cooling System

Introduction

The recirculating gas cooling system (RGCS) will consist of 13 independent subsystems whose purpose will be to provide cooling for those components and cells that will be normally inerted or cooled by inert gas.

A typical RGCS loop cell subsystem will consist of a fan, cooler, isolation valves (all located outside the inerted cells in accessible areas), and distribution ducts within the cell. The supply and return gas will be drawn through piping embedded in the cell shielding concrete. Twelve of the thirteen systems will use nitrogen as the coolant, and the thirteenth (fuel-handling cell subsystem) will use argon. Heat from the cells containing safety-related equipment that will require cooling to perform its safety function will be rejected to the emergency chilled water system. Heat from the cells containing equipment that will not require cooling to perform its safety function will be rejected to the normal chilled water system. Heat from the argon system will be rejected to Dowtherm J loops and then to two emergency chilled water systems. Inerting and deinerting connections for the cells will be connected to the RGCS, and cell negative pressure will be maintained by connection of the RGCS to the cell atmosphere processing system.

Scope of Review

PSAR Section 9.16 was reviewed as well as those portions of Sections 9.5 and 11.3 that support the recirculating gas cooling system. SRP Section 9.4.5, "Engineered Safety Feature Ventilation System," was also reviewed.

Evaluation Criteria

(1) The recirculating gas cooling system was reviewed to show compliance with

- (a) CRBR PDC 38, "Additional Cooling Systems"
- (b) CRBR PDC 39, "Inspection of Additional Cooling Systems"
- (c) CRBR PDC 40, "Testing of Additional Cooling Systems"
- (d) Applicable requirements of SRP Section 9.4.5. Those requirements considered applicable are those not unique to an LWR system.

(2) In general, the review assessed

- (a) adequacy of design criteria
- (b) potential for radioactive release
- (c) potential for affecting decay heat removal

Evaluation

The agreed upon design resulting from the staff's review is described below.

The design basis for the subsystems is to maintain cell temperatures below a level that would be deleterious to the concrete, electrical wiring, instrumentation, components, and equipment in the cell. For normal operation this has been defined to mean

(1) 120°F nominal cell gas temperature
(2) 150°F maximum concrete temperature

For offnormal operation the RGCS must maintain safety-related component temperatures low enough so that they can perform their safety functions.

For some offnormal conditions, such as a sodium leak in the cell, the RGCS will be shut off. However, the design of the cells is to be such that the cell and RGCS design temperature and pressure will not be exceeded, nor will cell structural concrete limiting temperature be exceeded for any of the design-basis events.

Additional required capabilities will include

(1) preventing leaked water from the cooling coils from entering the cells containing sodium or NaK

(2) maintaining the independence of redundant systems served by the RGCS

(3) isolating individual cells for maintenance

(4) direct cooling capability to primary sodium makeup pumps, ex-vessel storage tank (EVST) sodium cooling pumps, EVST sodium cold trap, and primary control rod drives

(5) isolating the fans, coolers, and gas supply during sodium or NaK spills

The 13 subsystems and the facilities or components to be serviced are listed below. The letters in brackets are the identification symbols for each of the 13 separate systems.

(1) Primary Heat Transport Systems (PHTSs) [PA, PB, PC] will cool the PHTS cells, the associated hot-leg pipeways, and check valve cells up to and including the reactor cavity bellow seals. Nitrogen atmosphere will be at slightly negative pressure.

(2) Control Rod Drive Mechanism [CR] will cool the primary control rod drive mechanisms. Nitrogen system pressure will be high (~100 psig).

(3) Sodium Makeup Pump and Vessel [MA] will cool one primary sodium makeup pump, makeup pump cell, overflow and primary sodium storage vessel cell, and the plugging temperature indicator cell. Nitrogen atmosphere will be at slightly negative pressure.

(4) Sodium Makeup Pump and Pipeways [MB] will cool one primary sodium makeup pump, makeup pump cell, and the sodium fill and drain pipeway. Nitrogen atmosphere will be at slightly negative pressure.

(5) Primary Sodium Cold Trap, NaK Cooling System, Plugging Temperature Indicator, Sodium Sampling Package [CT] will cool the primary sodium cold traps, cold-trap valve galleries, NaK electromagnetic pump and cells, plugging temperature indicator and cell, and the sodium sampling package and cell. Nitrogen atmosphere will be at slightly negative pressure.

(6) Reactor Cavity Subsystem [RC] will cool the reactor cavity, the three flowmeter cells, and the primary loop pipeways up to the reactor cavity seals. Nitrogen atmosphere will be at slightly negative pressure.

(7) Ex-Vessel Storage (EVS) Loop 1 [EA] will cool one of the EVS sodium pumps and cell, the sodium sampling package cell, and the EVS sodium pipeway cells. Nitrogen atmosphere will be at slightly negative pressure.

(8) Ex-Vessel Storage Loop 2 [EB] will cool one of the EVS sodium pumps and cell, the EVS cold trap and cell, and plugging temperature indicator cell 387. Nitrogen atmosphere will be at slightly negative pressure.

(9) Ex-Vessel Storage Loop 3 [EC] will cool the EVS natural circulation sodium loop cell. Nitrogen atmosphere will be at slightly negative pressure.

(10) Ex-Vessel Storage Tank (EVST) Cavity [ET] will cool the EVST cell and support ledge. Nitrogen atmosphere will be at slightly negative pressure.

(11) Fuel-Handling Cell [FH] will cool the fuel-handling cell and equipment. Argon atmosphere will be at slightly negative pressure. No isolation valves will be included in this system to isolate the cell from the cooling systems, since the sodium inventory of this cell will be small (at most the sodium contained in three core component pots) and contained in pots at atmospheric pressure; thus, the potential for a sodium-water reaction will be minimized.

Table 9.3 lists each of the subsystems and their seismic and safety classification. Figure 9.14 provides a typical RGCS arrangement. The RGCS will be designed in accordance with ASME Code, Section III, Class 3, seismic Category I requirements, supplied with Class 1E electrical power, and cooled by emergency chilled water whenever it serves a cell containing safety-related equipment that needs cooling to perform its function. In addition, the piping from the cell liners up to and including isolation valves for nonsafety-related subsystems will be designed to seismic Category I requirements. Any part of the RGCS that could be exposed to liquid sodium during a sodium spill condition will also be designed to ASME Code, Section III, Class 3 requirements. The cooling system design pressures for all subsystems will be equal to or greater than the design pressure of the cells they serve. Thus, failure of the isolation valves will not cause a loss of cell integrity. The design temperatures will be consistent with the maximum expected temperatures in the cells during accident conditions. Subsystems CR and FH will contain redundant cooling loops for plant availability and as low as is reasonably achievable considerations. Subsystems MA, MB, EA, and EB have been designated Safety Class 3 because they will provide cooling to equipment required for decay heat removal (EVST and direct heat removal system equipment) which would not perform its safety function if not cooled.

Loss of cooling in the other subsystems will not affect the ability of the equipment being cooled to perform its safety function.

Instrumentation for the RGCS will be provided as follows:

(1) cell temperatures indicators
(2) cooler inlet and outlet temperature indicators
(3) water leakage sensors
(4) fan status and ΔP
(5) local monitoring and alarms and alarms in the control room

Instrumentation for the safety-related subsystems is evaluated in Section 7 of this SER.

For the safety-related subsystems (EA, EB, MA, and MB) the instrumentation will be designed in accordance with IEEE Std. 279 and will be Class 1E. For these systems, the isolation valves will close automatically on indication of high cell outlet temperature (which would indicate a fire) or on indication of water leakage into the RGCS (which would indicate a leak in the water cooling system). This action will be taken to minimize the spread of sodium combustion products throughout the cooling system and to minimize the potential for water entering the cell containing sodium-filled components. The other sensors, however, will provide alarms to the control room, and the operator will have the option of isolating the cell. In addition, the sodium leak detection system will also provide an alarm to the operator, in the event of a sodium leak, allowing manual isolation of the cell if a leak is confirmed and automatic isolation has not occurred. For the nonsafety-related systems, commercial grade instrumentation will be used; however, the isolation valves will close automatically on the same indications as those for the safety-related subsystems. Subsystem CR will not have automatic closure of its isolation valves since it will not be connected to a cell containing sodium, and subsystem FH will not have isolation valves.

All isolation valves will be air operated and fail on loss of power or air pressure.

The system will be tested during initial startup, and individual components will be periodically inspected and have the capability for testing throughout the life of the plant.

Summary and Conclusions

The functions of the RGCS will be consistent with the plant needs and will be similar in concept to the cell cooling systems used at the FFTF. The safety-related components that need cooling to perform their functions will be supplied with a safety-grade cooling system via the RGCS. Since the safety-related components will be redundant, a single cooling system for each component will be adequate.

All cells serviced by the RGCS will be steel lined and have leak-rate requirements. This, in conjunction with operation of these cells at slightly negative pressure and the design of the RGCS to the cell design pressure and corresponding temperatures, will ensure no radioactive releases from the cells during normal operation and limited releases during accident conditions.

It should be noted that if a spill of radioactive sodium occurs in an inerted cell, the radioactive combustion products will be carried outside the cell up to the isolation valves, or in the event of an isolation valve failure, they

will be present throughout the affected RGCS subsystem. In accordance with the applicants' commitment in Appendix H of the PSAR to provide adequate shielding to permit access to vital areas, the need for shielding on the RGCS accessible equipment should be evaluated and documented as part of the final design.

To ensure that the systems are functioning properly they are to be designed for periodic testing and inspection. The safety-related subsystems will be part of the formal plant inservice inspection program as required by ASME Code, Section XI.

On the basis of its evaluation, the staff considers the proposed design criteria acceptable and in conformance with CRBR PDC 38, 39, and 40. The proposed design is in conformance with the applicable sections of SRP Section 9.4.5 and is considered acceptable for issuance of a construction permit.

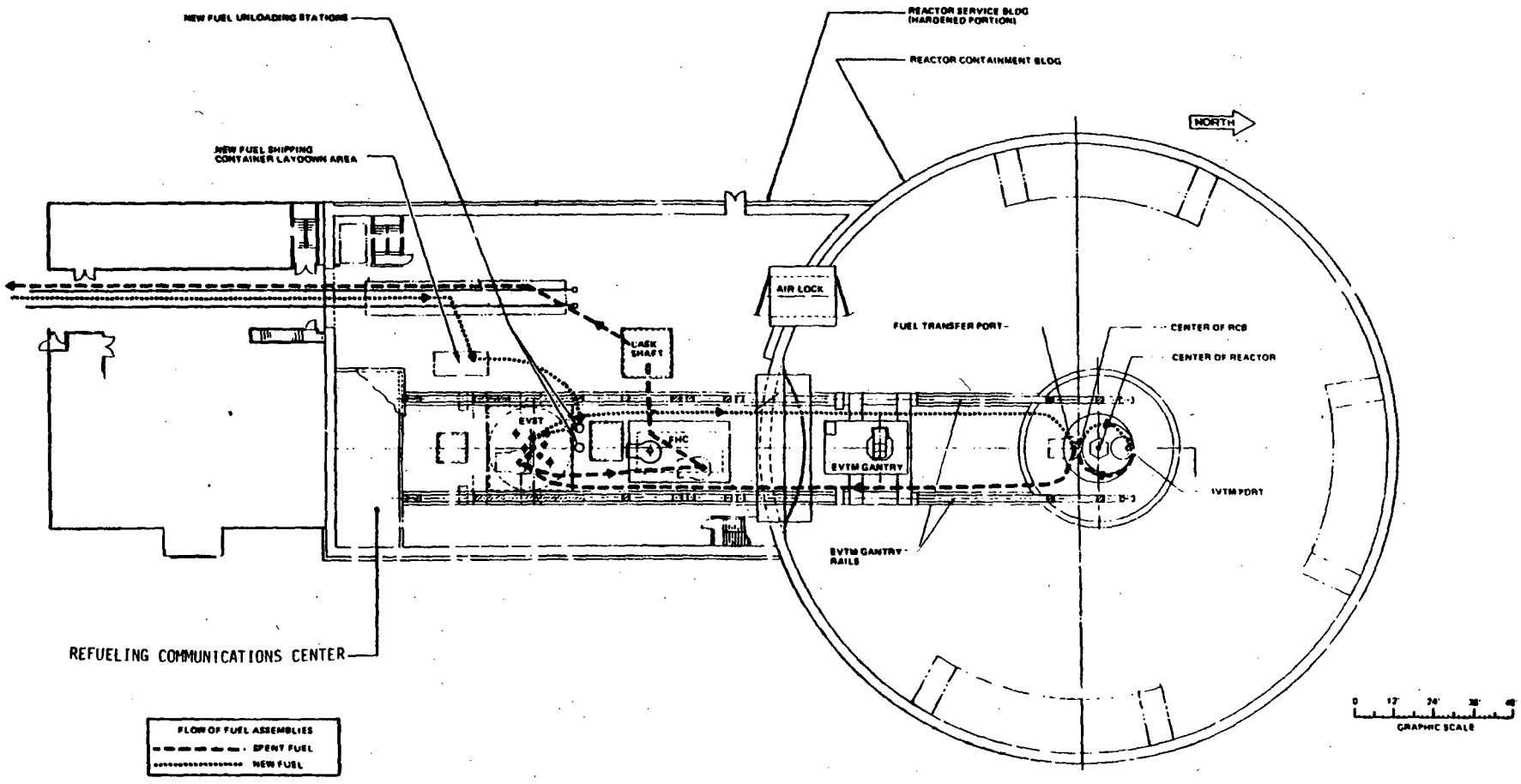


Figure 9.1 General arrangement of fuel storage/handling equipment and facilities

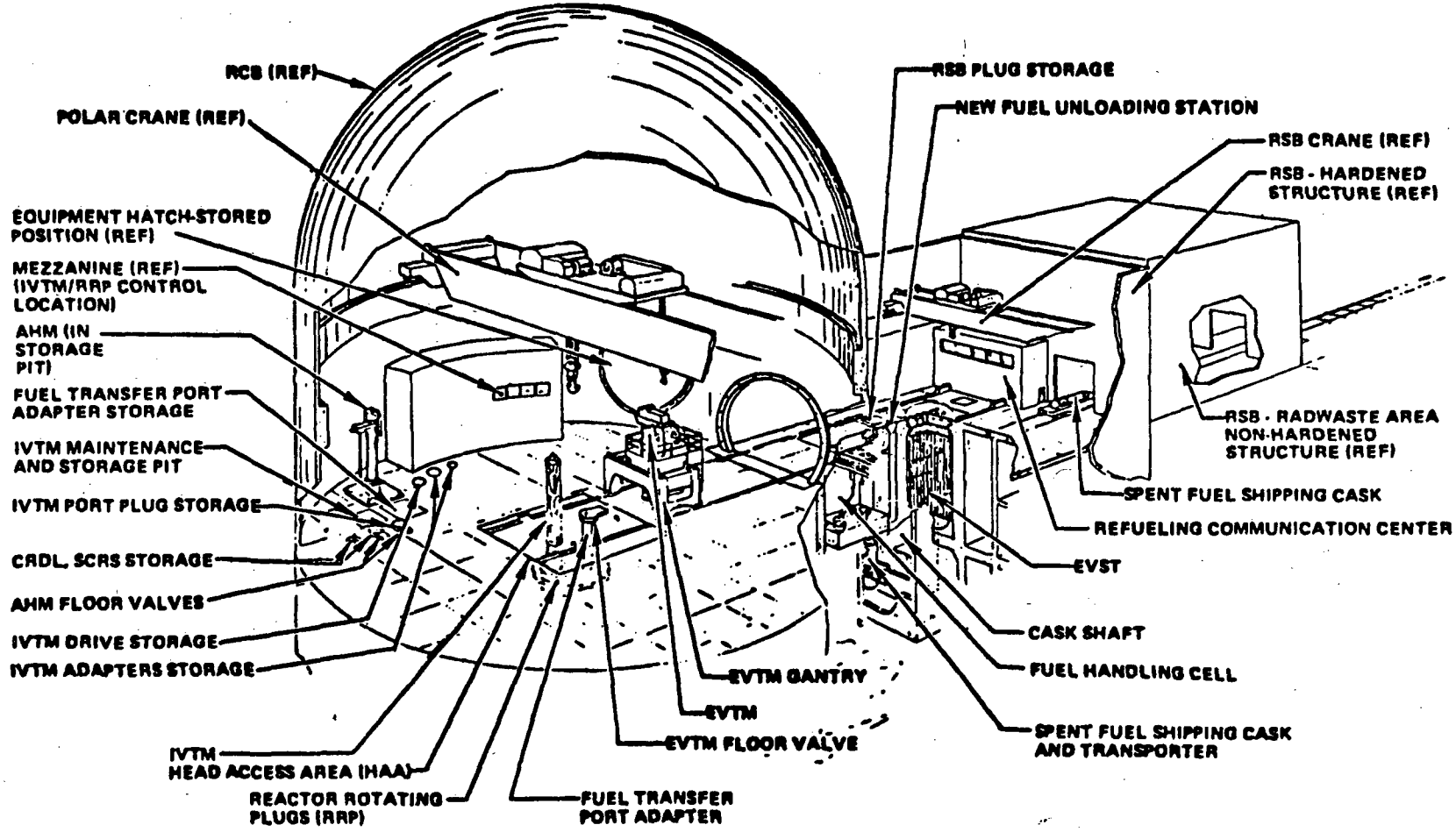


Figure 9.2 Arrangement of fuel storage/handling equipment and facilities (perspective view)

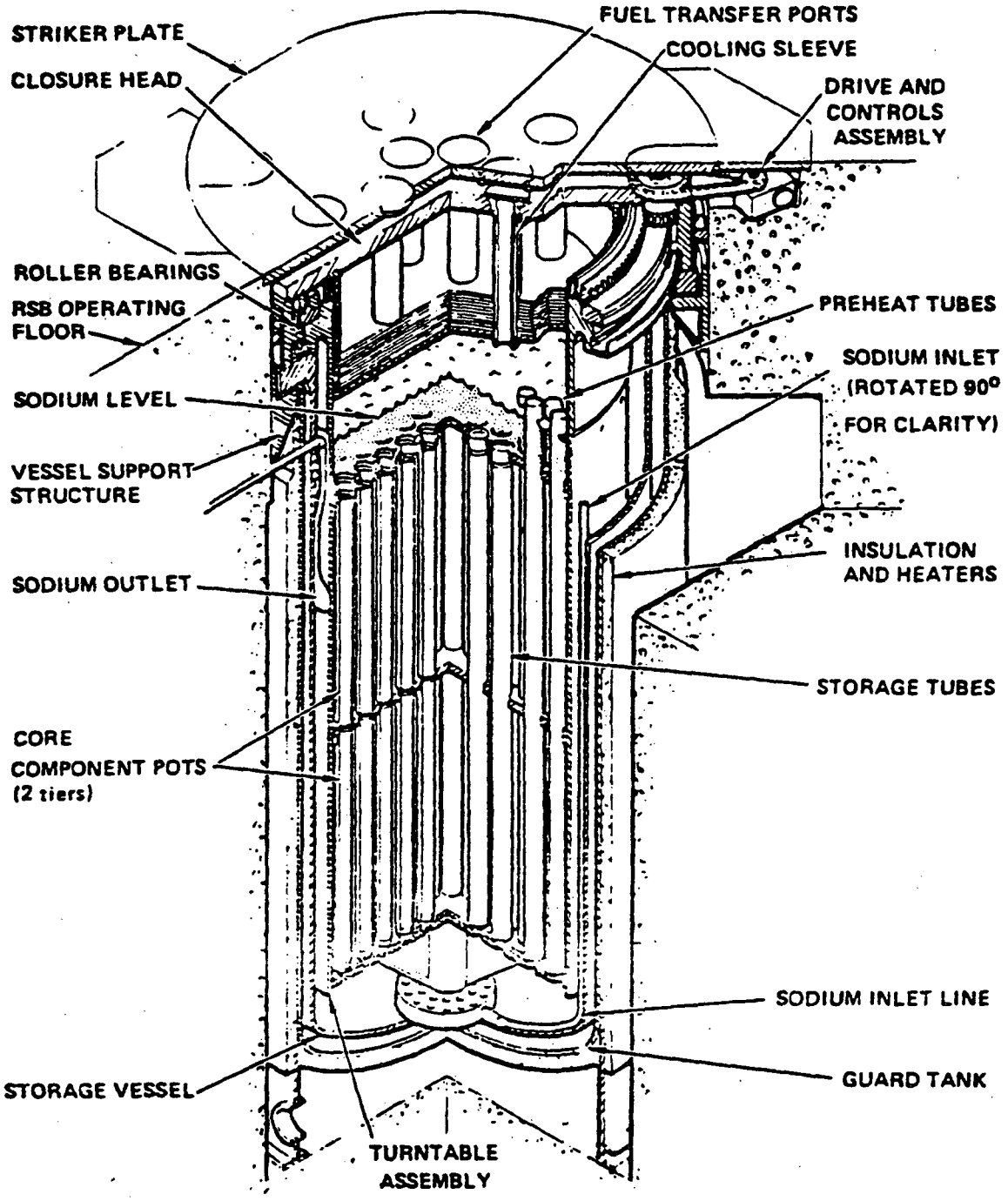


Figure 9.3 Ex-vessel storage tank

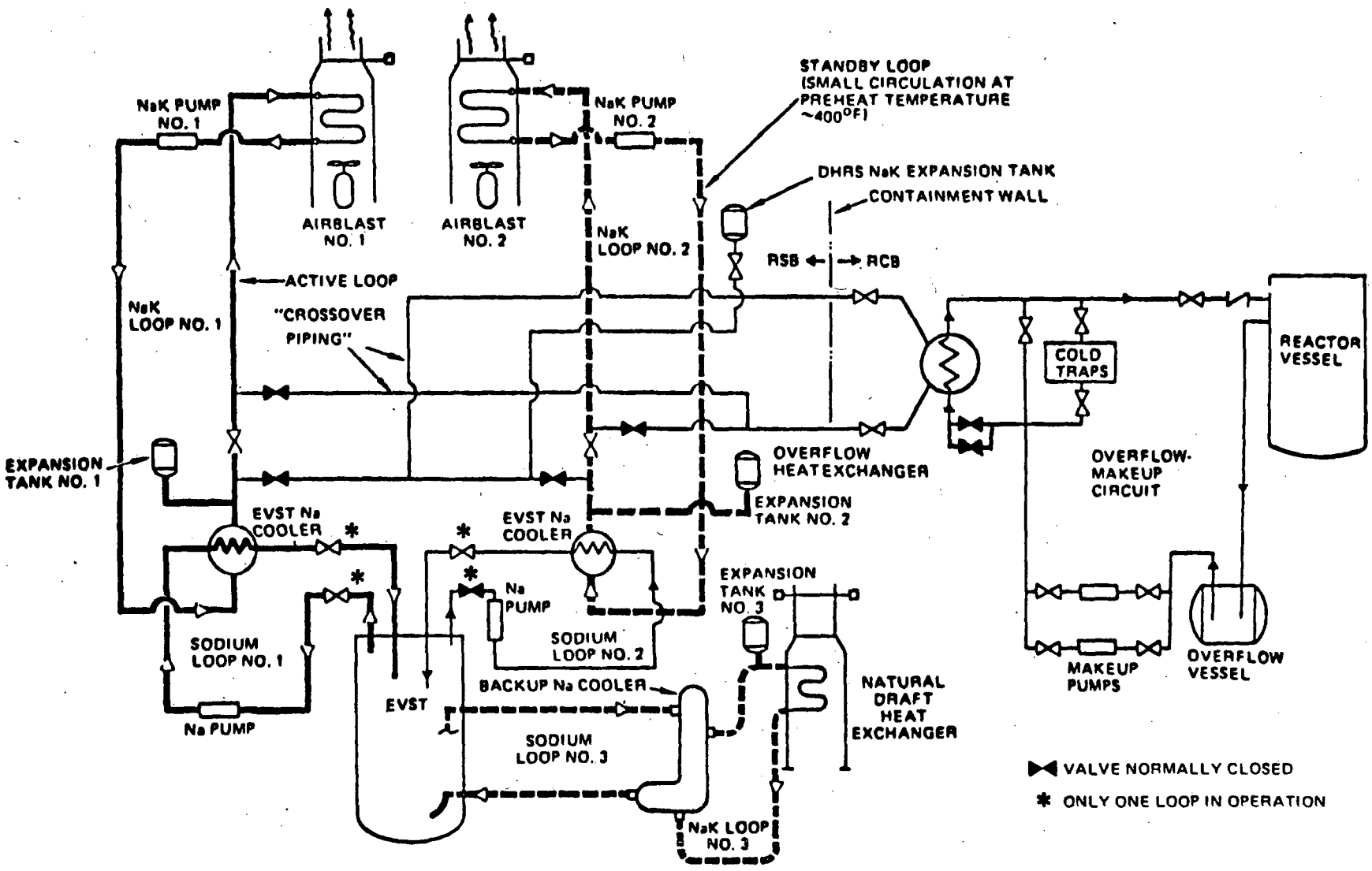


Figure 9.4 Sodium-NaK flow paths--ex-vessel storage tank cooling

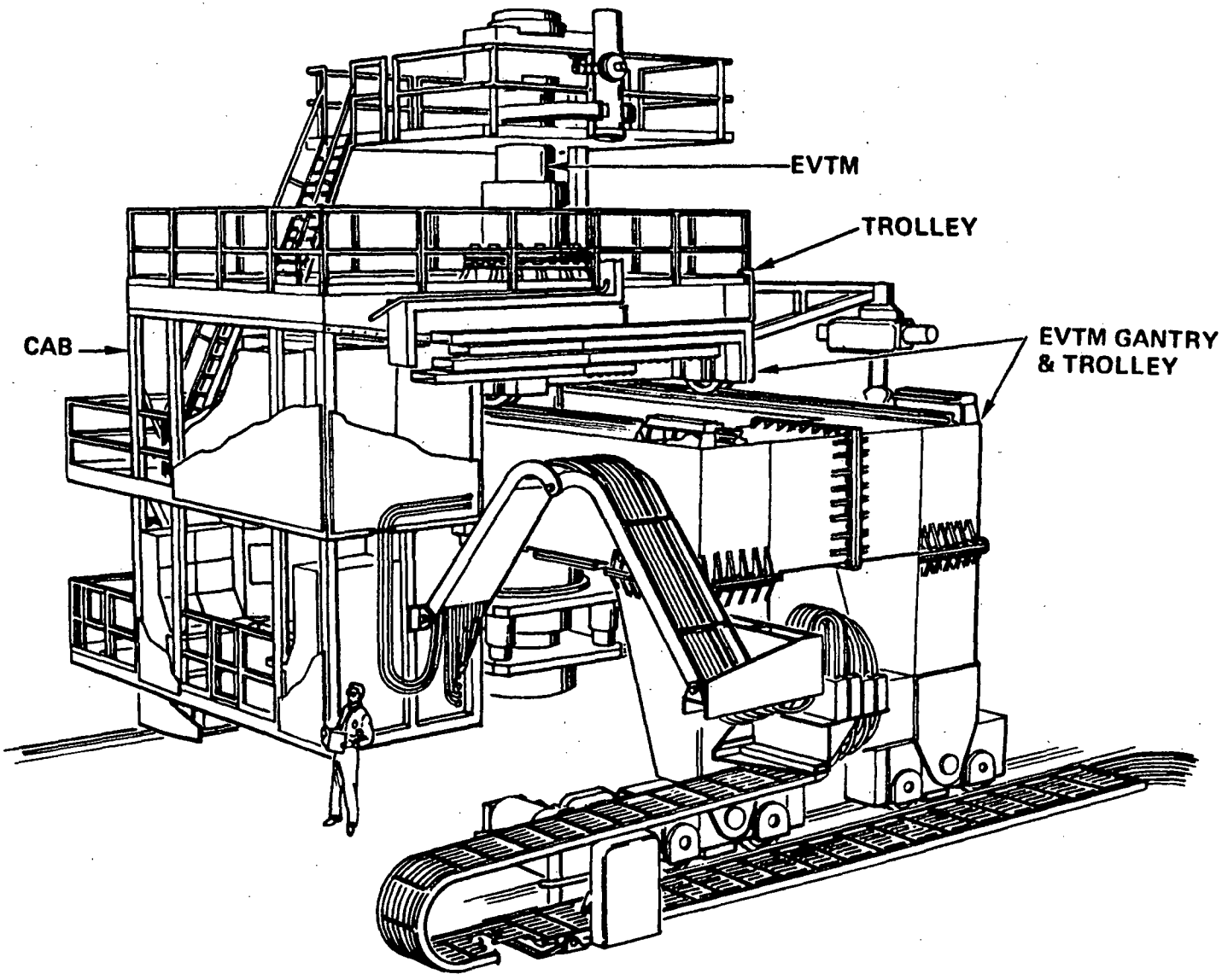


Figure 9.5 Ex-vessel transfer machine on gantry

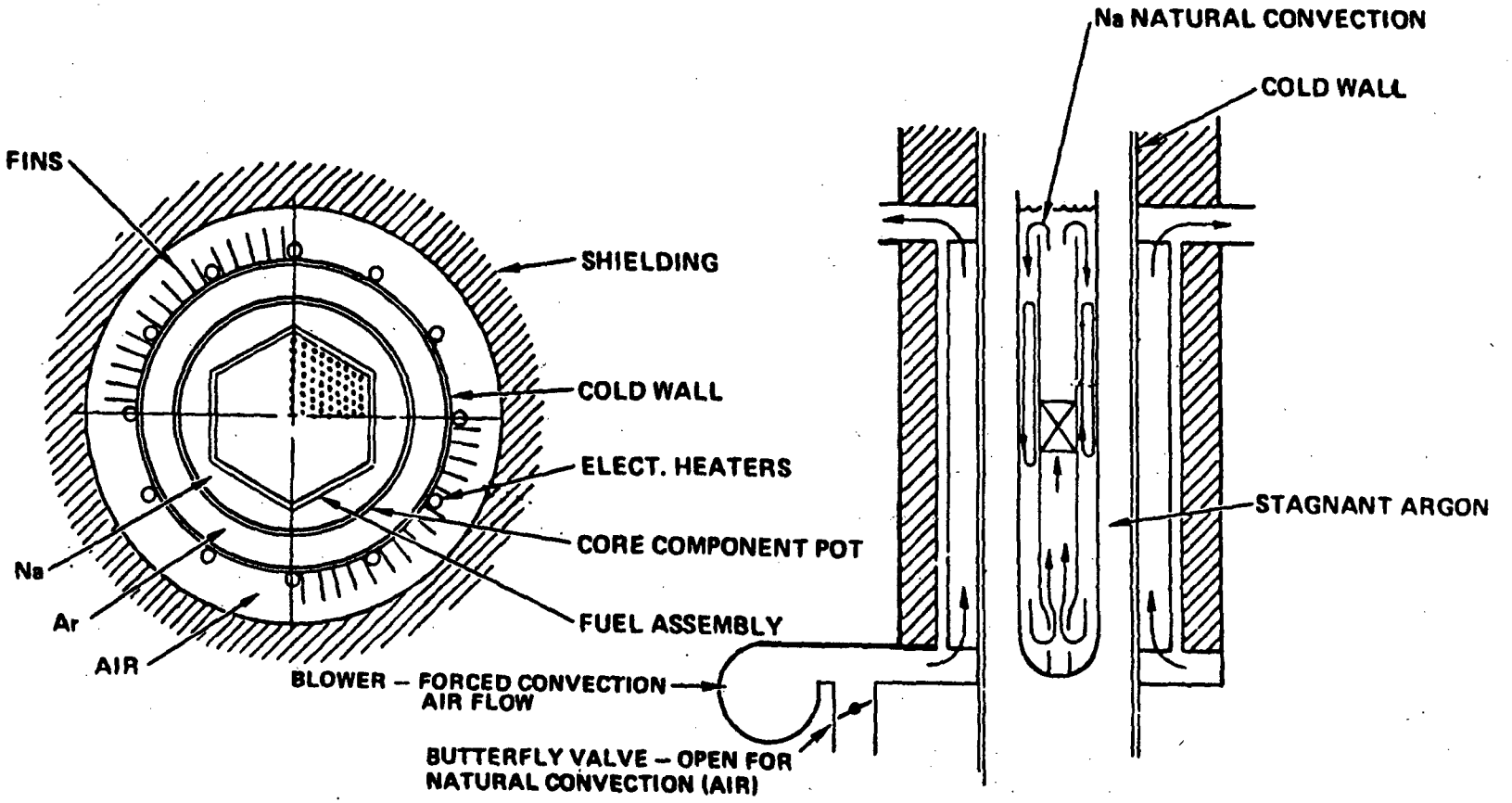


Figure 9.6 Ex-vessel transfer machine cold-wall cooling concept

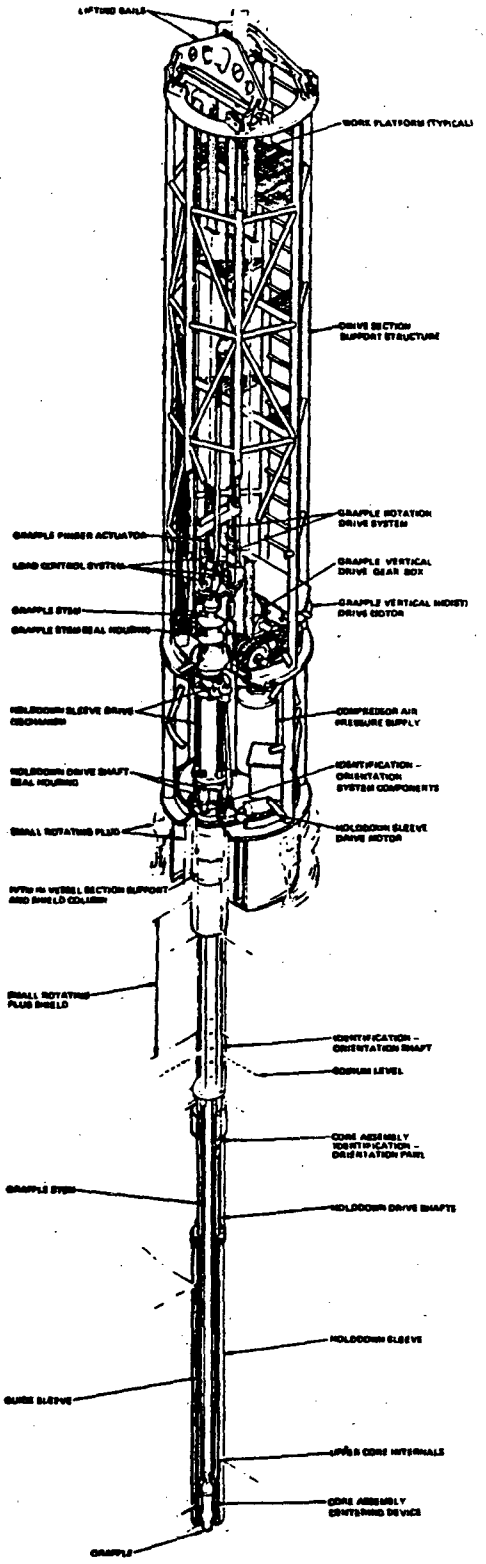


Figure 9.7 In-vessel transfer machine

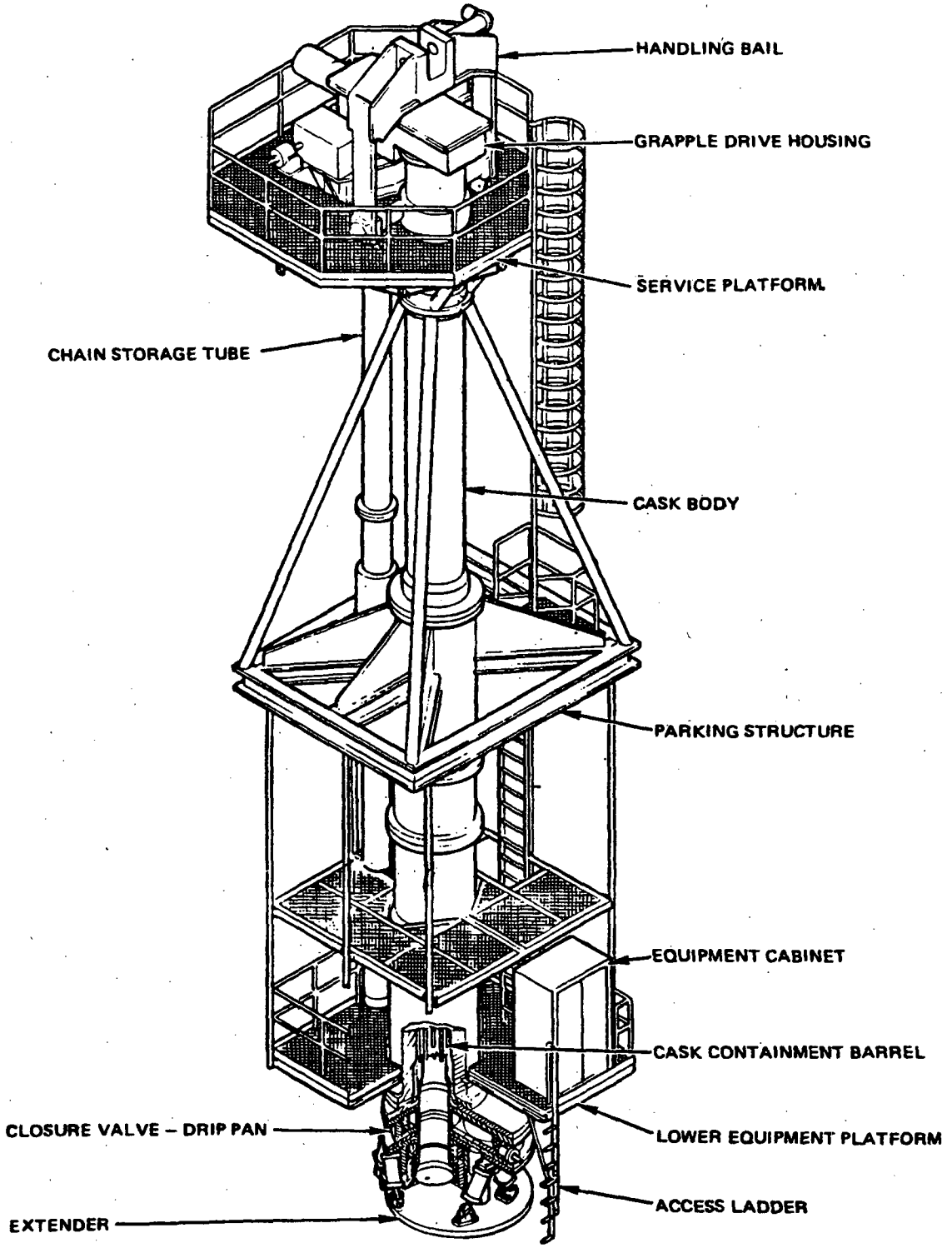


Figure 9.8 Auxiliary handling machine

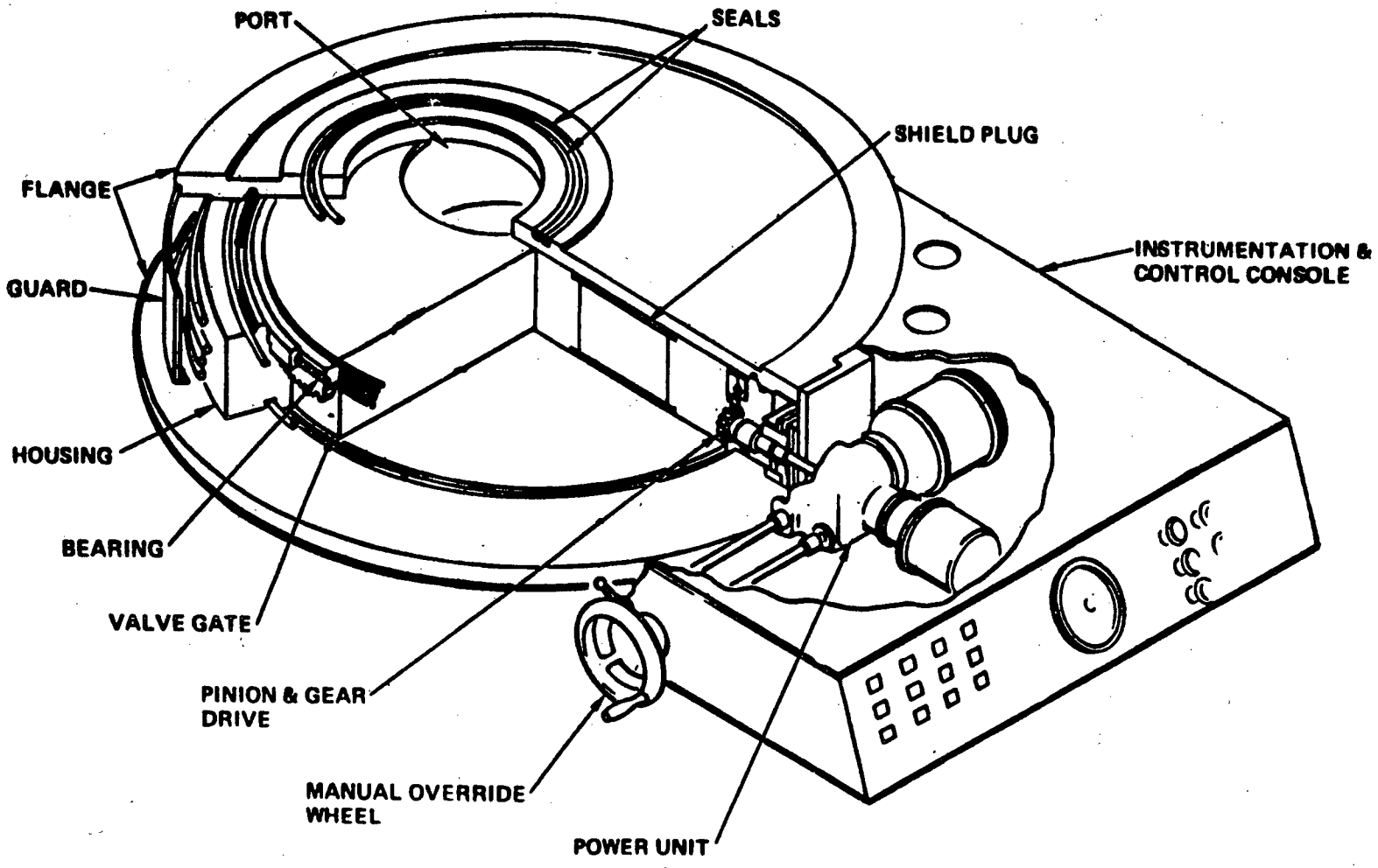


Figure 9.9 Floor valve

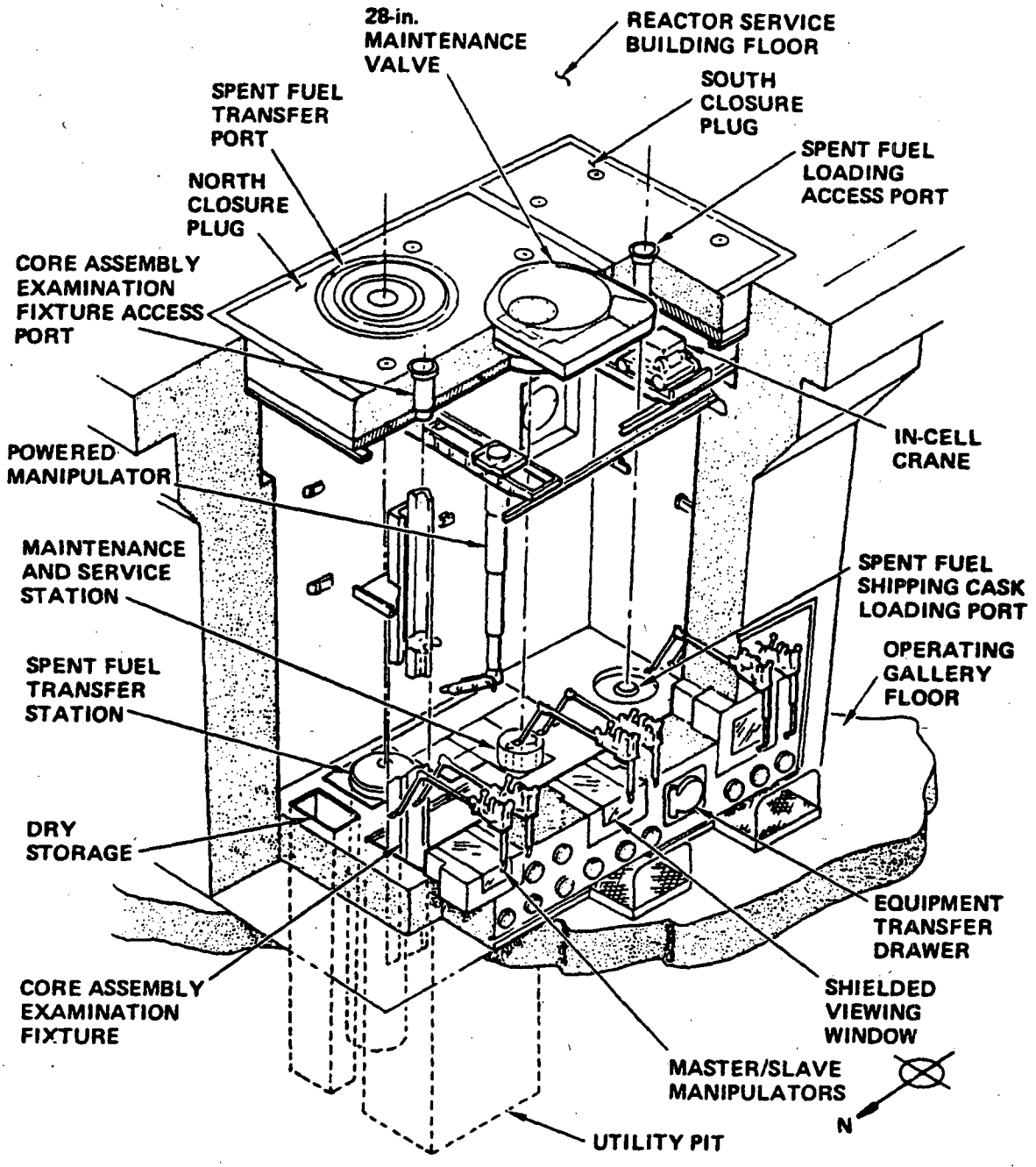


Figure 9.10 Fuel-handling cell

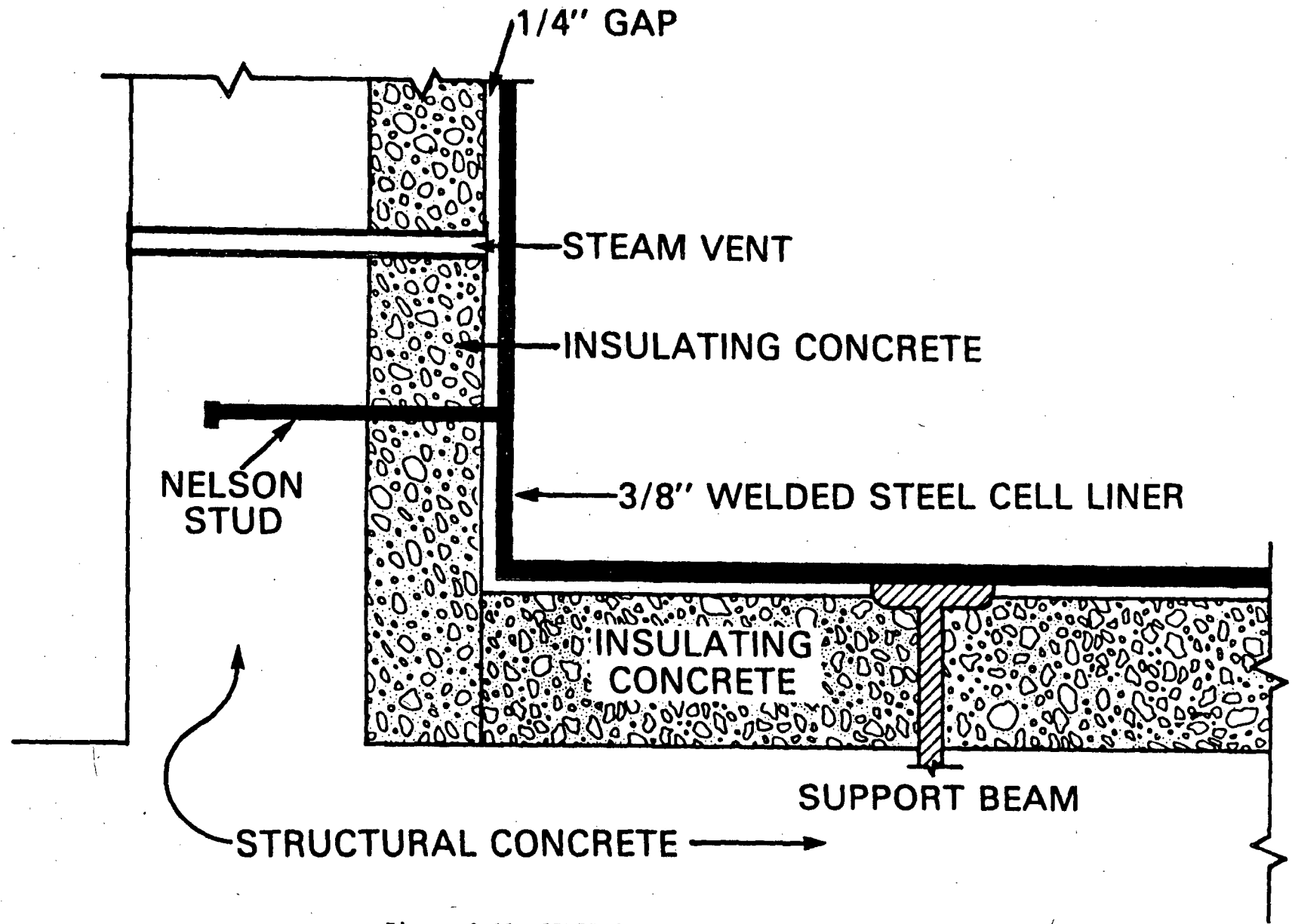


Figure 9.11 CRBRP inserted cell liner system

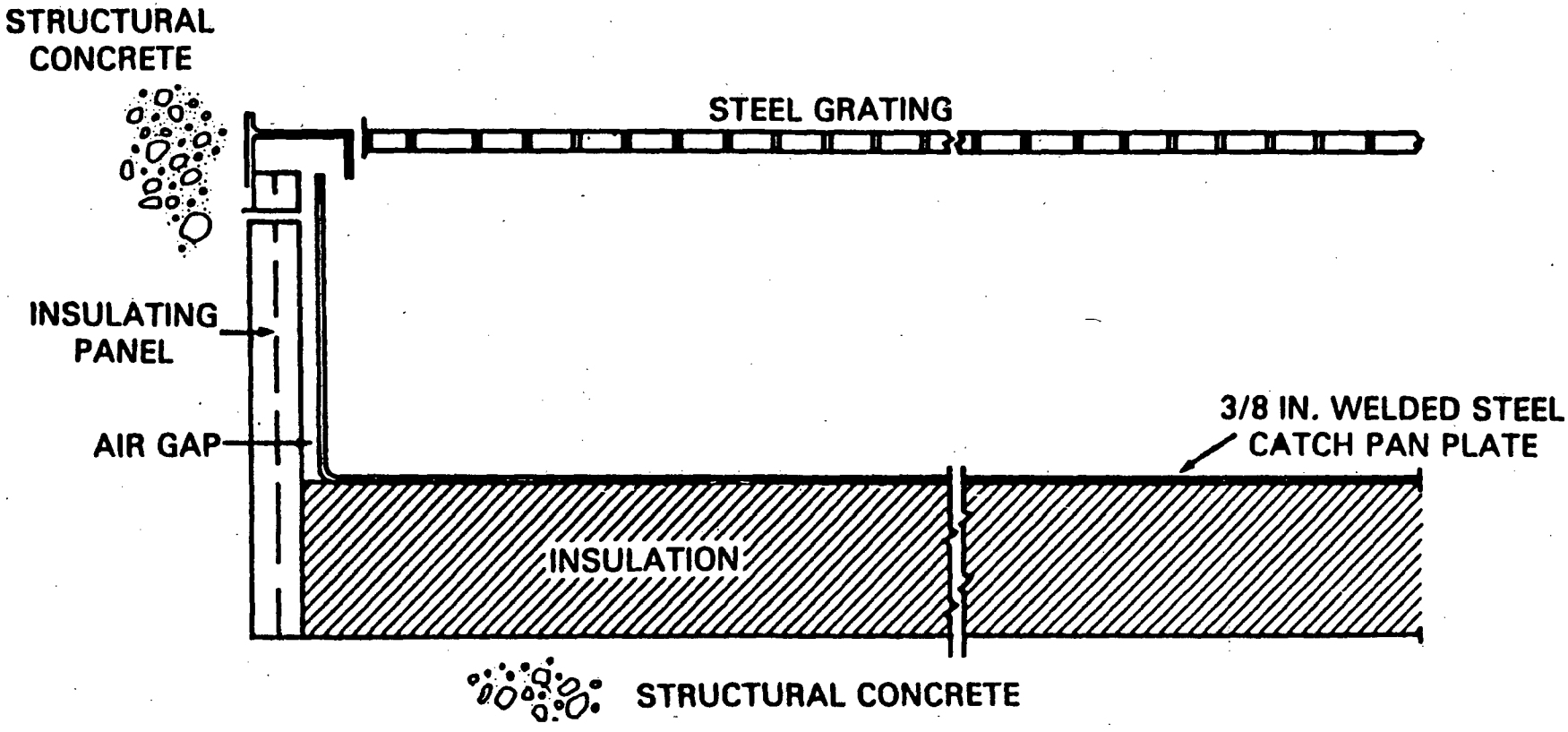


Figure 9.12 CRBRP open catch pan for air-filled cells

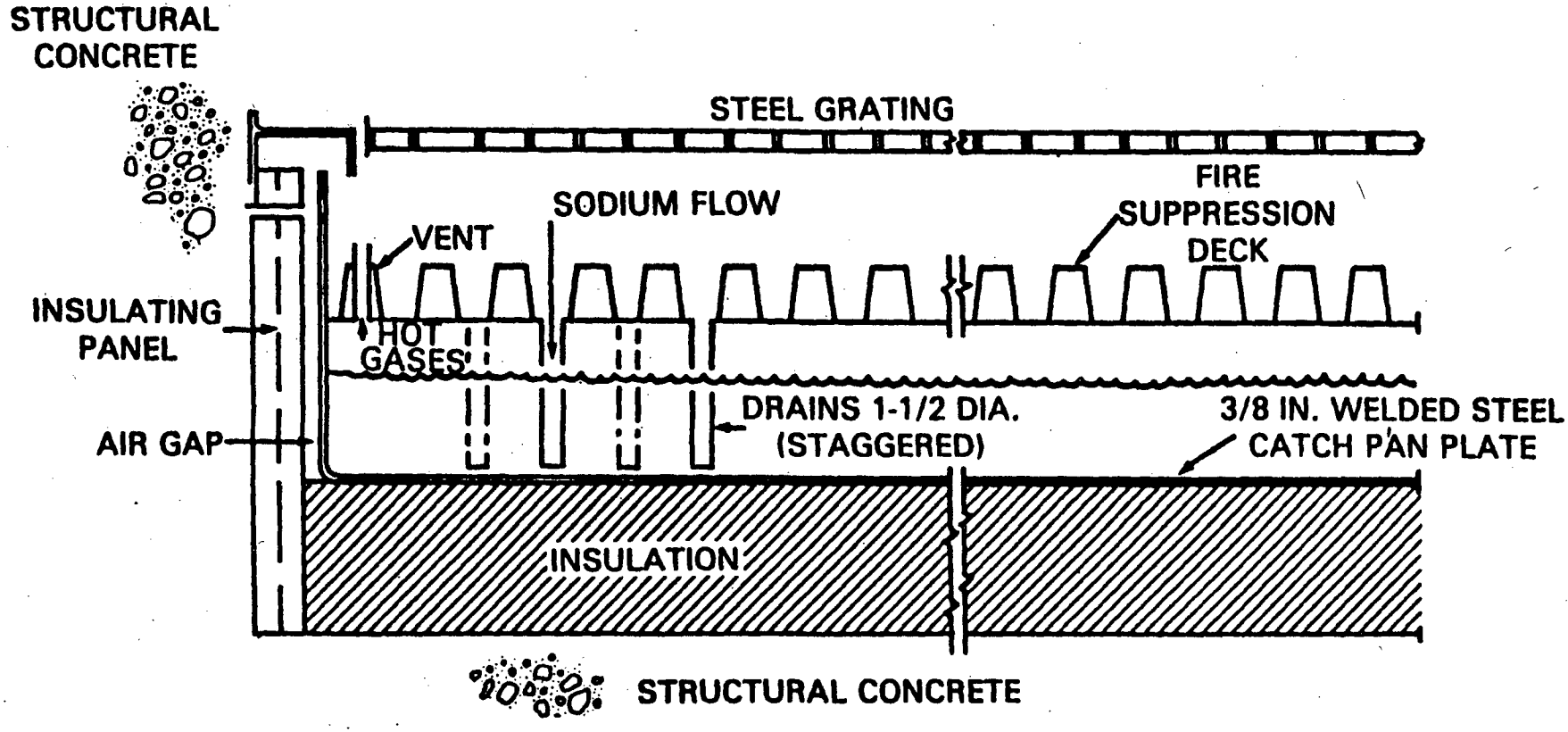


Figure 9.13 CRBRP catch pan and fire suppression deck system for air-filled cells

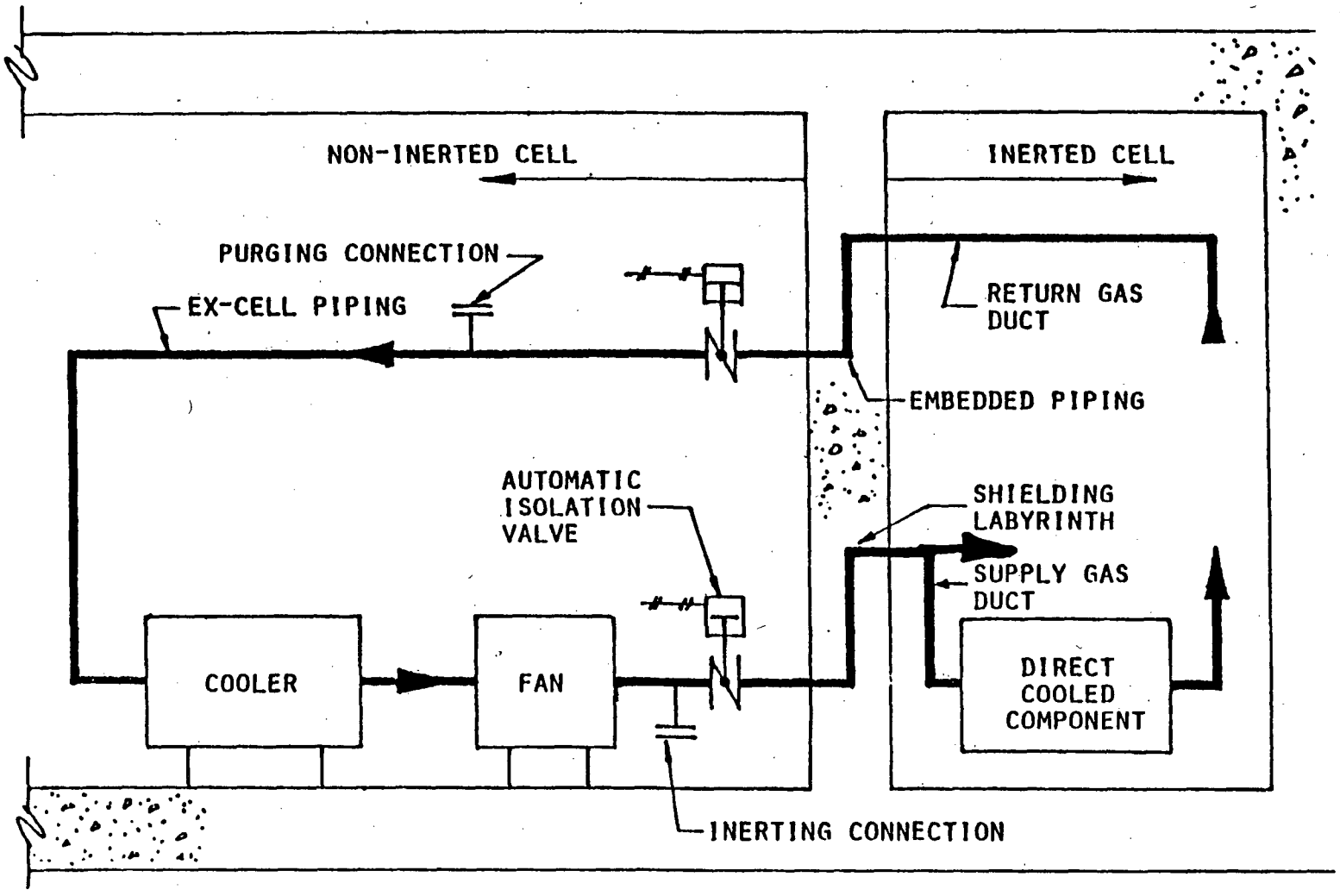


Figure 9.14 Typical subsystem of recirculating gas cooling system

Table 9.1 Irradiated fuel assembly steady-state design temperature limits

Frequency class or description of operation	Cladding or sodium temperature limit (°F)
Normal operations and anticipated events	
Handling	1,250--8-hr duration
Short-term storage (fuel-handling cell)	1,150--1-wk duration
Long-term storage (ex-vessel storage tank)	1,000--indefinite
Unlikely and extremely unlikely events	1,500
Reactor reinsertion or offsite postirradiation examination	1,175--1.5-hr duration 1,125--8-hr duration 1,000--indefinite

Table 9.2 Auxiliary liquid metal system

Subsystem	Function
Sodium and NaK receiving	Loading of sodium and NaK to the plant
Primary sodium storage and processing	Reactor vessel level control
	Storage of primary and ex-vessel storage tank sodium
	Purification of primary sodium
	Backup means for reactor decay heat removal (decay heat removal system mode)
Ex-Vessel storage processing	Heat removal from spent fuel stored in ex-vessel storage tank
	Ex-vessel storage tank sodium purification
	Dissipation of reactor decay heat (decay heat removal system DHRS mode)
Primary cold trap NaK coding	Primary cold-trap temperature control
Intermediate sodium processing	Purification of intermediate heat transport system sodium

Table 9.3 Safety and seismic classification

Subsystem	Area served	Seismic category*	Safety class
PA	Primary heat transport system (PHTS) loop #1	III	**
PB	PHTS loop #2	III	**
PC	PHTS loop #3	III	**
CR	Control rod drive mechanisms (CRDMs)	III	**
MA	Sodium makeup pump and vessels	I	SC-3
MB	Sodium makeup pump and pipeways	I	SC-3
CT	Cold trap, NaK cells	III	**
RC	Reactor cavity	I	**
EA	Ex-vessel storage (EVS) loop #1	I	SC-3
EB	EVS loop #2	I	SC-3
EC	EVS loop #3	III	**
ET	Ex-vessel storage tank (EVST) cavity	III	**
FH	Fuel-handling cell	I	**

*All subsystems' (except subsystem CR) isolation valves and piping between the automatic isolation valves and the liner of served cells shall be seismic Category I.

**These subsystems have no designated safety classification.

10 STEAM AND POWER CONVERSION SYSTEMS

10.1 Summary Description

The steam and power conversion system will be designed to convert the heat produced in the reactor to electrical energy. The CRBRP will be a three-loop-concept plant. Heat generated in the reactor will be transferred from the primary loop to an intermediate coolant loop via intermediate heat exchangers. Heat from the intermediate coolant loop will be transferred to the main steam system via three steam generators. Superheated steam from the steam generators will be used to drive a turbine generator consisting of one high-pressure and three low-pressure double-flow units. Steam exhausting from the low-pressure units will be cooled and deaerated in the main condenser. Condensate from the main condensers will be returned to the steam generators as heated feedwater.

A turbine bypass system will be provided to route steam directly to the main condenser and to the deaerator in order to protect the steam generators from overpressurization in the event of a turbine trip. The condenser circulating water system will be designed to dissipate 65% of total reactor heat in the form of bypassed steam. The deaerator will be designed to dissipate an additional 15% of reactor heat to steam generator feedwater.

10.2 Turbine Generator

The turbine generator will convert steam power into electrical power and will have a turbine control and an overspeed protection system. The design function of the turbine control and overspeed protection system will be (1) to control turbine action under all normal or abnormal conditions, (2) to ensure that a full-load turbine trip will not cause the turbine to overspeed beyond acceptable limits, and (3) to minimize the probability of generation of turbine missiles in accordance with the requirements of PDC 5, "Environmental and Missile Design Bases." The turbine control and overspeed protection system is therefore essential to the overall safe operation of the plant.

The turbine generator will be a tandem-compound type (single shaft) with one double-flow high-pressure turbine and three double-flow low-pressure turbines. The turbine generator will operate at 3,600 rpm and will be designed to deliver 375 MW of electricity.

The turbine generator will be equipped with an electrohydraulic control (EHC) system. The EHC system will consist of an electronic governor using solid-state control techniques in combination with a high-pressure hydraulic actuating system. The system will include electrical control circuits for speed control, trip anticipator, mechanical overspeed trip, and backup overspeed trip. Overspeed protection will be provided using four systems: normal speed control, trip anticipator, mechanical overspeed, and electric backup overspeed protection. The speed control unit will respond to speed error signals for positioning of the turbine control valves. The trip anticipator circuit will function to avoid a turbine overspeed following a load rejection which is accompanied by

failure of one or more control valves, and to prevent a turbine trip when the speed control system is functioning properly. The trip anticipator circuit will use a comparison of rotor speed with a load-dependent reference to generate closing signals to the stop valves. The mechanical overspeed will trip both the stop and control valves if turbine speed reaches 110% of rated speed.

The electric overspeed backup will operate at approximately 111% of rated speed. This system will generate an electric trip signal to the stop and control valves using magnetic speed pickups and 2-out-of-3 logic. The stop and control valves will be designed to close in 0.2 sec after receipt of a signal, and will fail closed on loss of hydraulic pressure. Provisions will be made for testing the speed control and trip functions while the unit is on-line. Therefore, the requirements of PDC 5 with regard to providing a turbine overspeed protection system will be met.

An inservice inspection program for the main steam stop and control valves will be developed. This program will include periodic cycling of the stop and control valves.

The turbine generator system will meet the recommendations of BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and BTP MEB 3-1, "Postulated Break and Leakage Locations in Fluid Systems Outside Containment." Evaluation of protection against dynamic effects associated with the postulated pipe system failure is covered in Section 3.6 of this SER.

The scope of review of the turbine generator included descriptive information in Section 10.2 of the PSAR flow charts and diagrams. The basis for acceptance in the staff's review was conformance of the design criteria and bases and design of the proposed turbine generator system to PDC 5 with respect to the prevention of the generation of turbine missiles, the additional guidance in Section II of SRP Section 10.2, and industry codes and standards. This system was reviewed in accordance with SRP Section 10.2.

On the basis of its review, the staff concludes that the proposed turbine generator overspeed protection system will meet the requirements of PDC 5 and the guidance of SRP Section 10.2. It will be capable of performing its designed safety functions, and is, therefore, acceptable.

10.3 Main Steam Supply System (Downstream of Main Steam Isolation Valves)

The main steam supply system will be designed to deliver steam from the steam generators to the high-pressure turbine, the turbine bypass system, the deaerator, and the turbine gland sealing steam system. The main system from the main steam isolation valve (MSIV) to the first shutoff valves, and all branch lines in between these valves up to and including the first valve, will be designed to the requirements of ANSI Std. B31.1 and will not be seismic.

The scope of review of the main steam supply system (between the MSIV and up to and including the turbine stop valves) included descriptive information in Section 10.3 of the PSAR and flow charts and diagrams. The basis for acceptance in the staff review was conformance of the design criteria and bases and proposed design of main steam supply system to the acceptance criteria in Section II of SRP Section 10.3.

On the basis of its review, the staff concludes that the main steam supply system between the MSIV and up to and including the turbine stop valves will be in conformance with the criteria and design bases cited, will be capable of performing its design functions, and is, therefore, acceptable.

10.4 Other Features of Steam and Feedwater Systems

10.4.1 Main Condenser

The main condenser will be designed to function as a heat sink for the turbine exhaust system, turbine bypass steam, and other turbine cycle flows, and to receive and collect condensate flows for return to the steam generators. Heat from the main condenser will be transferred to the circulating water system which will operate in conjunction with a mechanical draft cooling tower to dissipate the rejected heat to the atmosphere.

The main condenser will not be required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. The main condenser will be a single-shell, single-pass, deaerating type of condenser and will be designed to produce a turbine back pressure of 3.35 in. Hg (absolute) when operating at turbine-rated output. The main condenser design will include provisions for condensate deaeration and hot-well surge storage of the condensate. Offgas from the main condenser will be processed in the main condenser evacuation system which is described and evaluated in Section 10.4.2 of this SER.

The main condenser will be designed to accept full load exhaust steam from the main turbine, up to 65% of the main steam flow from the turbine bypass system, and other cycle steam flows. The main condenser will also be designed to deaerate the condensate to the required water quality. Copper-nickel tubes will be used to minimize corrosion and erosion of condenser tubes. Leakage of condenser tubes could degrade the quality of the feedwater and could corrode secondary system components. The applicants will provide for continuous monitoring of condensate, feedwater, and steam drum drains to ensure that proper water quality will be maintained, and that sources of contamination, such as circulating water inleakage to the condenser, will be identified quickly.

Monitoring equipment will include alarms to indicate out-of-specification conditions. The adequacy of the secondary sampling system for leak detection is evaluated in Section 9.3.2 of this SER.

Flooding of the turbine building and adjacent building as a result of condenser failure or circulating water system failure is discussed and evaluated in Section 10.4.5 of this SER.

The scope of review of the main condenser included layout drawings and descriptive information of the condenser in Section 10.4.1 of the PSAR.

The basis for acceptance in the staff review was conformance of the design criteria and bases and proposed design of the condenser to the acceptance criteria in Section II of SRP Section 10.4.1 and industry standards.

On the basis of its review, the staff concludes that the main condenser will be in conformance with the criteria and design bases cited, will be able to perform its designed function, and is, therefore, acceptable.

10.4.2 Condenser Air Removal System

10.4.2.1 Introduction

Section 10.4.2 of the PSAR contains information pertaining to the condenser air removal system, the system design bases, and the applicable acceptance criteria.

10.4.2.2 Acceptance Criteria

The staff has reviewed the applicants' design, design criteria, and design bases for the condenser air removal system for the CRBRP. The acceptance criteria used in its evaluation are those contained in Section II of SRP Section 10.4.2 (NUREG-0800); and the applicable CRBR PDC, which have replaced the general design criteria of Appendix A to 10 CFR 50; and Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers." Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides referenced in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the proposed condenser air removal system meets the requirements of 10 CFR 50.

10.4.2.3 Method of Review

The condenser air removal system was reviewed in accordance with the SRP. The results of the review are discussed below.

10.4.2.4 Review Discussion

The condenser air removal system will be designed to establish and maintain main condenser vacuum by transferring noncondensable gases from the condenser to the turbine building ventilation system. The system is not safety related and the components will be designed to Quality Group D and to a nonseismic design classification. The condenser air removal system will consist of two full-capacity mechanical vacuum pumps. Air and noncondensable gases removed from the main condenser will be sampled continuously by a radiation monitor before they are released to the environment.

One of the full-capacity pumps will operate during normal operation, the other will be on standby. The standby pump will start automatically if the suction mainfold pressure increases to 3.5 in. Hg (absolute), or if the running pump trips. If either of these mechanisms is not sufficient to maintain the condenser vacuum, the turbine will automatically trip on high back pressure. The above provisions meet the criteria in CRBR PDC 52 and 56.

The applicants' provisions for quality assurance for the design, construction, and operational phases of the condenser air removal system were reviewed to determine that they conform to RGs 1.33 and 1.123, as provided in the SRP. No exceptions to the criteria were noted. Equipment quality group classifications were reviewed to determine that they conform to RG 1.26, as provided in the SRP. No exceptions were noted.

10.4.2.5 Evaluation Findings

The condenser air removal system will include equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The scope of the staff's review included the system's capability to transfer radioactive gases to the turbine building ventilation exhaust system, and design provisions incorporated to monitor and control releases of radioactive materials in effluents. The staff has reviewed the applicants' system descriptions, piping and instrumentation diagrams, and design criteria for the components of the condenser air removal system.

The staff concludes that the design of the condenser air removal system will be acceptable in that the applicants have met the requirements of CRBR PDC 52 and 56 with respect to the control and monitoring of releases of radioactive materials to the environment by providing a controlled and monitored condenser air removal system. The applicants have met the requirements of Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers," which has been reviewed by the staff and determined to be appropriate for this application.

10.4.3 Turbine Gland Sealing System

10.4.3.1 Introduction

Section 10.4.3 of the PSAR contains information pertaining to the turbine gland sealing system, the design bases, and applicable acceptance criteria.

10.4.3.2 Acceptance Criteria

The staff has reviewed the applicants' design, design criteria, and design bases for the turbine gland sealing system for the CRBRP. The acceptance criteria used as the basis for the staff's evaluation are set forth in Section II of SRP Section 10.4.3 (NUREG-0800), CRBR principal design criteria, which have replaced the general design criteria of Appendix A to 10 CFR 50. Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides identified in Section II of the SRP.

Conformance to the acceptance criteria provides the bases for concluding that the proposed turbine gland sealing system meets the requirements of 10 CFR 50.

10.4.3.3 Method of Review

The design of the turbine gland sealing system for CRBRP was reviewed in accordance with SRP Section 10.4.3. The results of the review are discussed below.

10.4.3.4 Review Discussion

The turbine gland sealing system (TGSS) will be designed to control radioactive steam leakage from, and air in-leakage into, the turbine. The components of the system will be designed to Quality Group D and to a nonseismic design classification. The TGSS will consist of the steam seal regulator and the steam packing exhauster. Steam will be supplied to the system from the auxiliary

boiler system during startup and from the main steam supply system or the extraction steam system during load operations. The steam packing exhauster will evacuate the steam/air mixture from the packings, condense the steam, and discharge the air to the atmosphere through the turbine generator building vent system. Before release to the environment, the steam packing exhauster effluent will be monitored continuously by a radiation monitor.

The staff's review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with CRBR PDC 52 and 56. The staff has reviewed the applicants' design criteria and system descriptions and finds them consistent with RG 1.26.

10.4.3.5 Evaluation Findings

On the basis of its evaluation, the staff finds the proposed TGSS acceptable. The basis for acceptance has been conformance of the applicants' design, design criteria, and design basis for the TGSS to the applicable regulations referenced above.

10.4.4 Turbine Bypass System

The turbine bypass system will be designed to bypass up to 65% of main steam flow to the main condenser. The turbine bypass will be used in two ways. First, during power operation (above 40%) the system will be controlled by a load error signal (TG power versus reactor power) to minimize the effect of large load reduction transients on the reactor, and second, at less than 40% power, the system will be responsive to steam header pressure and will operate automatically to maintain steam header pressure at a preset point.

The bypass system will consist of four pressure-reducing valves, four desuper heaters, and associated valves and piping. The valves will be 6-in., globe-type, air-operated, normally closed, and designed to pass 16% of rated flow per valve. The valves will be designed to fail closed on loss of air pressure. The turbine bypass system is not a safety-related system and will not be required for plant shutdown following an accident. The turbine bypass valves will be designed to close on loss of main condenser vacuum.

The turbine bypass system will meet the recommendations of BTP ASB 3-1, "Protection Aganist Piping Failures in Fluid System Piping Outside Containment," and BTP MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Evaluation of protection against dynamic effects associated with the postulated pipe system failures is covered in Section 3.6 of this SER.

The scope of review of the turbine bypass system included drawings, piping and instrumentation diagrams, and descriptive information of the system in Section 10.4.4 of the PSAR.

The basis for acceptance in the staff review was conformance of the design criteria and bases and proposed design of the turbine bypass system to the acceptance criteria in Section II of SRP Section 10.4.4 and industry standards.

On the basis of its review, the staff concludes that the turbine bypass system will be in conformance with the above cited criteria and design bases, it will be capable of performing its design function, and is, therefore, acceptable.

10.4.5 Circulating Water System

In the draft SER input of February 15, 1983, the staff indicated that the applicants have not provided sufficient information concerning the consequences of flooding from failures in the circulating water system. The applicants provided this information in a letter dated February 23, 1983. Additional information was provided regarding instrumentation to detect flooding and the quality group classification of the system.

The nonseismic Category I circulating water system is designed to meet the Quality Group D requirements. The applicants have addressed adequately the failure of large diameter piping of circulating water system within the turbine generator building (TGB). The volume of water that could drain into the TGB as a result of a failure in the circulating water piping or condenser expansion joints is approximately equivalent to the inventory of the condenser (51,300 gal) plus the circulating water contents of the cooling tower basin (about 1,272,000 gal), that is, a total of 1,323,000 gal before the circulating water pumps would cavitate (assuming no operator action). There is no safety-related equipment located in the TGB. There are two potential pathways (a personnel door and a personnel access corridor) below the maximum circulating water system flood level from which water could enter the nuclear island buildings where safety-related equipment is located. The intended function of the safety-related equipment will not be impaired since these potential pathways will be provided with watertight doors or will be located so that they are above the potential flood level. The applicants should address the operation of these watertight doors in the plant Technical Specifications.

Since no safety-related equipment will be affected by a postulated failure in the circulating water system, the requirements of PDC 5, "Environmental and Missile Design Bases," with respect to protection of safety-related systems from failure of nonsafety-related systems are satisfied.

Indication of leakage or potential failure in circulating water system components will be provided to operators in the control room. Waste water treatment instrumentation will actuate main control board alarms to signify simultaneous high water levels in sumps located adjacent to the main condenser. These alarms, coupled with an alarm denoting a simultaneous low pressure in the circulating water pump discharge header and/or loss of main condenser vacuum will advise the operator of the flood incident initiation so that the circulating water pumps can be shut down, eliminating the water source.

On the basis of review of system design criteria and bases and preliminary design, the staff concludes that the circulating water system meets the requirements of PDC 5 with respect to protection of safety-related systems and is, therefore, acceptable. The circulating water system meets the acceptance criteria of SRP Section 10.4.5.

10.4.6 Secondary Water Chemistry Monitoring and Control

The CRBRP is a three-loop concept plant. Sodium coolant in the primary heat transport system (PHTS) will enter the reactor where it will be heated and pumped to intermediate heat exchangers (IHXs). The IHXs will transfer heat to the intermediate sodium coolant, which will be pumped through the steam generator system. Heat will be transferred to the main steam system by three steam generator modules in each of the three loops. Sufficient superheated steam will be provided to drive a tandem compound steam turbine with auxiliaries that will operate in a closed condensing cycle with six stages of regenerative feedwater heating. Exhaust steam will be condensed in one surface-type steam condenser and returned to the steam generators as heated feedwater with a major portion of its gaseous, dissolved, and particulate impurities removed by condensate demineralizer ion exchangers.

Steam generator experience in operating PWRs indicates that less than thoroughly effective water treatment, operational procedures, and design factors have led to the degradation of steam generator tubing, as evidenced by an extensive history of stress-corrosion cracking, wastage, and denting of steam generator tubing in operating PWRs. Consequently, the CRBR water chemistry was defined to reflect the experience gained from the occurrence of the degradation problems in operating PWR steam generators.

A critical examination of the available data led to the selection of an all-volatile treatment (AVT) of the water, full-flow condensate polishing (with the particular goal of achieving low sodium content), and a steam drum high drain (10% blowdown) rate. This leads to a water purity level approaching that required for once-through steam generators.

10.4.6.1 Area of Review and Acceptance Criteria

One of the specific features of the CRBR steam generator operation that required attention was the occurrence of departure from nucleate boiling (DNB) permitted by the design of the evaporator modules. The concern is to what extent the evaporator tubing integrity will be affected by DNB. Experimental results obtained under the DNB conditions will be reviewed and evaluated. Because of the stringent requirements imposed on water quality, analytical and monitoring procedures for feedwater control also will be reviewed.

Effective, long-term, reliable operation of CRBR steam generators will require that operational procedures, design, and selection of materials be such that there will be no leakage across the steam generator tubes and that the barrier between the intermediate sodium and the secondary side water will maintain its integrity under operating, maintenance, or testing conditions and during design-basis accident conditions as stated in CRBR PDC 12, 13, and 26.

The reactor coolant pressure boundary should be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Furthermore, the reactor coolant system and associated auxiliary control and protection system should be designed with sufficient margin to ensure that design conditions of the reactor coolant boundary will not be exceeded during any condition of normal operation, including anticipated operational occurrences. In addition to

these criteria, CRBR applicable requirements of BTP MTEB 5-3 on monitoring of secondary side water chemistry in PWR steam generators should be met.

10.4.6.2 Review Evaluation

In CRBR PSAR Section 5.5.3.11.4, the applicants present the feedwater and steam drum purity concentrations established to protect the steam generator tubes against excessive corrosion and deposition. The applicants also provided information regarding chemical limits, monitoring and control procedures, and the basis for establishing the chemistry limits.

(1) Condensate Chemistry Limits and Control

To meet the evaporator water chemistry requirements to protect the evaporator tubing and other component material, requirements (given in Table 10.1) for condensate system demineralizer resin regeneration and/or replacemnt, continuous monitoring/recording, and grab sampling of the condensate polishing effluent have been established by the applicants.

(2) Feedwater Chemical Limits

The basic structural materials in the CRBR feedwater system will be carbon steel and 2¼% Cr-1% Mo steel. In water and steam, these alloys are susceptible to caustic gouging and possibly caustic stress-corrosion cracking. By maintaining the feedwater and steam drum purity concentrations, as given in Table 10.2, occurrences of these forms of degradation will be minimized. For normal operation and other than startup conditions, Table 10.2 provides the feedwater and steam drum water chemistry limits.

The applicants stated that limited-duration operation with impurity concentrations above specified limits will be allowable for periods not to exceed 24 hours in special instances. This is consistent with PWR steam generator operating experience, where special instances are defined to include condensate polishing system perturbations, such as those immediately associated with termination of regeneration.

(3) Water Chemistry Monitoring Frequency and Location

To determine the feedwater quality, continuous analyzers with alarms will be provided to sample conductivity, dissolved oxygen, hydrazine, turbidity, pH, sodium, chloride, and silica. Continuous samples of steam drum downcomer water and periodic samples of drum drain (blowdown) water will be monitored for sodium, silica, pH, and conductivity. The downcomer continuous sample monitors also will be alarmed if out-of-specification conditions occur. The condenser hotwell will be monitored for conductivity and sodium ions to indicate condenser leakage. The breakthrough of impurities from the demineralizer effluent will be minimized by in-line measurements of silica, conductivity, and sodium. Finally, the feedwater train will be monitored downstream of the deaerator for pH and oxygen content to indicate the presence of corrosive conditions in this portion of the steam system. Another special precautionary measure instrumented by the applicants will be an alarm coupled with the most critical in-line measurements to indicate departure from specified concentrations.

(4) Steam Generator Blowdown

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To minimize damage from such sources, the feedwater and steam drum water will be maintained at concentrations within stated limits by full-flow demineralization and continuous steam drum flow (blowdown) at a nominal rate of 10% full-power steam flow.

(5) Basis for Establishing the Chemistry Limits

Because of the relatively low evaporator recirculation ratio in the CRBR, the applicants recognized that the CRBR water chemistry limits would need to be similar to those limits imposed on the once-through steam generator designs used in fossil-fired boilers and PWRs. Basically, this will require the use of an all-volatile treatment (AVT) consisting of a pH adjustment agent (typically ammonium hydroxide) and an oxygen scavenging agent (typically hydrazine). The applicants indicated the principles for CRBR chemistry control would be established on the basis of experiences with the existing LWR industry AVT-chemistry requirements. This commitment satisfies BTP MTEB 5-3 in SRP Section 5.4.2.1 of NUREG-0800.

The CRBR chemistry requirements were further refined to address the particular needs of CRBR relative to materials; that is, because of the 90/10 copper-nickel condenser, a 0.002-ppm copper concentration was specified. This low limit will minimize the potential for transport of copper to the evaporator tube internal surfaces where it would cause excessive tube corrosion.

The low recirculation ratio in the evaporators results in DNB in the 2¼% Cr-1% Mo evaporator tubes. The applicants have committed to a close control of the sodium ions to prevent stress-corrosion-cracking problems and to tight control of chloride and sulfate ions to prevent "under deposit" corrosion. For example, the applicants indicated by controlling the feedwater sodium ion to less than 0.001 ppm, the sodium ion concentration in the recirculation loop will be maintained at less than 0.006 ppm. Similarly, the chloride and sulfate ions will be maintained at low values in the feedwater to achieve a 0.015 ppm maximum for species in the recirculation loop.

10.4.6.3 Evaluation Findings

The applicants have described and committed in the CRBR PSAR to monitor and control secondary water chemistry, and they have indicated the location and frequency of measurements that will be performed to monitor feedwater purity and the need for condensate cleanup system demineralizer and filter replacement. These actions will establish an effective, long-term, reliable operation of CRBR steam generators to maintain no leakage across the steam generator tubes under operating, maintenance, or testing conditions and during design-basis accidents.

On the basis of its evaluation, the staff concludes that the proposed secondary water chemistry monitoring and control meet the requirements of CRBR PDC 12, 13, and 26 and the suitable portions of BTP MTEB 5-3 for CRBR and are, therefore, acceptable.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system is designed to provide a reliable source of high purity feedwater for the steam generators during both normal operation and anticipated transient conditions. The system is designed so that feedwater will not be delivered to the steam generators when feedwater isolation is required. The system will include the piping and components from the condenser hotwell, through the condensate polishing unit, low-pressure feedwater heaters, deaerating feedwater heater, deaerator storage tank, steam generator main feedwater pumps, high-pressure feedwater heaters, topping heaters, and containment isolation valves to the steam generators.

The condensate and feedwater system's design criteria, bases, and preliminary design were reviewed in accordance with SRP Section 10.4.7, "Condensate and Feedwater System." An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. The "Review Procedures" provide guidelines to verify that the system's design criteria, bases, and preliminary design meet applicable portions of the following LWR General Design Criteria (GDC) of 10 CFR Part 50:

(1) GDC 2, "Design Bases for Protection Against Natural Phenomena," as related to the system being capable of withstanding the effects of earthquakes. GDC 2 is identical to CRBR PDC 2. Acceptance is based on meeting the guidance of RG 1.29, "Seismic Design Classification," Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4, "Environmental and Missile Design Bases," as related to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions (PWRs only). Acceptance is based on meeting the guidance contained in BTP ASB 10-2 on reducing the potential for water hammers in steam generators with top feedring designs. GDC 4 is identical to CRBR PDC 5 except that PDC 5 does not include loss-of-coolant accidents as a condition to be protected against.

(3) GDC 5, "Sharing of Structures, Systems, and Components," as related to the capability of shared systems and components important to safety to perform required safety functions. GDC 5 is identical to CRBR PDC 6.

(4) GDC 44, "Cooling Water," as it relates to:

    (a) The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

    (b) Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure (this may be coincident with the loss of offsite power).

    (c) The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained. GDC 44 is identical to CRBR PDC 38, "Additional Cooling Systems" except that PDC 38 does not require that CRBR additional cooling systems be capable of removing reactor decay heat.

(5) GDC 45, "Inspection of Cooling Water System," as related to design provisions to permit periodic inservice inspection of system components and equipment. GDC 45 is identical to CRBR PDC 39 except that "additional cooling systems" is subsituted for "cooling water systems."

(6) GDC 46, "Testing of Cooling Water Systems," as related to design provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions. GDC 46 is comparable to CRBR PDC 40; the latter criterion is more general to include the variety of cooling system functions at CRBR. The intent of the two criteria is identical.

Except for the differences noted above, SRP Section 10.4.7 is applicable to the review of the CRBR condensate and feedwater system. Conformance with the acceptance criteria of SRP Section 10.4.7, except as noted above, provides the bases for concluding that the system's design criteria and bases and preliminary design are acceptable.

The condensate and feedwater system from the condenser hotwell up to but not including the feedwater piping anchor at the steam generator building penetration will not be safety related (Quality Group D, nonseismic Category I). The system from that anchor to the steam generator steam drum is a safety-related part of the steam generator system and is designed in accordance with ASME Code, Section III, Class 2 or 3.

Normally, two 50% capacity condensate pumps will be in service; the third one will be on standby for automatic starting should one of the operating pumps fail. Therefore, the system will be capable of supplying the maximum required flow. The pumps will take suction from the condenser hot well which is designed for a storage capacity of 3 minutes. Another source of water for the condensate system will be the condensate storage tank which will be connected to the main condenser hot well. Also, interconnections will provide makeup water from the makeup water treatment system and the demineralized water system to either the condenser or condensate storage tank.

Two 50% capacity steam generator feed pumps will normally be in service. Should one of the two operating feedwater pumps fail, the third will automatically start. The source of water for the feed pumps will be the deaerator storage tank which is designed to have 7 minutes' storage capacity. When either the normal feedwater sources will not be available, or the normal heat removal path through the main condenser will become unavailable because of loss of outside power or other malfunctions, the heat removal will be taken over by the auxiliary feedwater system, which is a subsystem of the steam generator auxiliary heat removal system (SGAHRS), as described in Section 5.6. Instrumentation will be provided to initiate operation of the SGAHRS if the condensate and feedwater system becomes inoperable.

Under normal conditions, no radioactive contaminants other than tritium will be present in the condensate and feedwater system. The tritium will be present because of diffusion from the intermediate sodium loop. Even in the highly unlikely event of simultaneous leakage in the intermediate heat exchanger and

in the steam generator, the pressure in the steam generator water/steam side, which is at least five times greater than that on the sodium side, will prevent leakage into the water side. For the detection of any possible radioactivity which might be introduced into the system, the condenser vacuum pump discharge will be continuously monitored before release into the atmosphere. Tritium discharges from the steam-water cycle are described in Sections 11.2 and 11.3.

All safety-related portions of the main feedwater system will be located within a seismic Category I, tornado- and flood-protected steam generator building, and therefore will be protected against natural phenomena in accordance with GDC 2 (PDC 2). The requirements of GDC 4 (PDC 5) regarding missiles, pipe whip, and jet impingement are evaluated and discussed in Sections 3.5 and 3.6 of this SER. However, the staff cannot conclude that the requirements of GDC 4 have been met with respect to water hammer effects. The potential of waterhammer in the system has not been addressed. The applicants should address the guidelines of BTP ASB 10-2, "Design Guidelines For Water Hammers in Steam Generators with Top Feedring Design." The requirements of GDC 5 (PDC 6) do not apply, since the system will not be shared between units.

There are two safety-grade main feedwater isolation valves and a check valve on each feedwater line. Thus, feedwater isolation is ensured in the event of a single failure in any isolation valve. The feedwater system design, therefore, meets the requirements of GDC 44 (PDC 38) by providing a redundant and isolable system capable of transferring heat loads from the reactor system to a heat sink under both normal and accident conditions.

The system is designed to permit periodic inservice inspection and operational functional testing; thus the requirements of GDC 45 (PDC 39) and 46 (PDC 40) are met.

Based on the review, the staff concludes that the system's design criteria and bases meet the requirements of GDC 2, 5, 44, 45, and 46 (PDC 2, 6, 38, 39, and 40) as they relate to protection against natural phenomena, missile and environmental effects (excluding water hammer), decay heat removal, and inservice inspection and testing and guidelines of RG 1.29 with respect to seismic classification and are, therefore, acceptable. However, the staff cannot conclude that the requirements of GDC 4 (PDC 5) are met until the applicants address the generic water hammer concern for the main feedwater system. (See Appendix B of this SER.)

Table 10.1 Condensate chemistry and monitoring and control

Maximum allowable impurities	Design limit operation above 5% power	Monitoring	Grab sample
Total suspended solids	16 ppb	None	Yes
Silica (SiO_2)	5 ppb	Continuous	Yes
Iron (Fe)	5 ppb	None	Yes
Copper (Cu)	1.5 ppb	None	Yes
Sodium (Na)	1 ppb	Continuous	Yes
Chloride (Cl)	2.5 ppb	Continuous	Yes
Cation conductivity at 77°F	0.2 mho/cm	Continuous	Yes

Table 10.2 Feedwater and steam drum chemical limits

Feedwater impurities	Feedwater	Steam drum
Suspended solids, ppm	-	0.1
Dissolved oxygen, ppm	0.005	-
Silica, ppm	-	0.1
Iron as Fe, ppm	0.01	-
Copper as Cu, ppm	0.002	-
Hydrazine, ppm	0.005-0.015	-
Chlorides, ppm	-	0.015
Sodium, ppm	0.001	0.006
Sulfate ppm	-	0.015
pH at 77°F	8.8-9.2	8.8-9.2
Conductivity (after cation removal) at 77°F μmho/cm	0.2	1.0

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Introduction

The radioactive waste management systems will provide for controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radioactive waste system will process wastes that are produced from sodium spillage, component decontamination operations, laboratory operations, and plant personnel showers. The gaseous radioactive waste system will provide holdup capacity to allow decay of short-lived noble gases that have migrated to the primary sodium system cover gas or leaked into the cells housing primary system components and piping. The solid radioactive waste system will provide for the solidification, packaging, and storage of radioactive wastes generated during station operation before shipment off site to a licensed facility for burial.

In its evaluation of the liquid and gaseous radioactive waste systems, the staff has considered

(1) the capability of the systems for keeping the levels of radioacitivity in effluents "as low as is reasonably achievable" on the basis of expected radwaste inputs over the life of the plant

(2) the capability of the systems to maintain releases below the limits specified in 10 CFR 20 during periods of fission product leakage at design levels from the fuel

(3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences

(4) the quality group and seismic design classification applied to the equipment, components, and structures housing these systems

(5) the design features that will be incorporated to control the release of radioactive materials in accordance with CRBR PDC 52.

In its evaluation of the solid radioactive waste system, the staff has considered:

(1) system design objectives in terms of expected types, volumes, and activities of waste processed for offsite shipment,

(2) waste packaging and conformance to applicable Federal packaging regulations and provisions for controlling potentially radioactive airborne dusts during baling operation

(3) provisions for onsite storage before the wastes are shipped

In its evaluation of the process and effluent radiological monitoring and sampling systems, the staff has considered the system's capability (1) to monitor

all normal and potential pathways for release of radioactive materials in the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor performance of process equipment and detect radioactive material leakage between systems.

In the Final Environmental Statement for the CRBRP (Feb. 1977), the staff assumed that the design objectives of Appendix I to 10 CFR 50 are applicable for determining whether CRBRP radioactive releases are "as low as is reasonably achievable." In the Final Environmental Statement (FES), the staff performed an evaluation to determine if the waste management systems meet the design objectives of the Annex to Appendix I. Since publication of the FES, the staff has refined and supplemented the Appendix I evaluation contained in the FES by (1) using additional site-related meteorological information and (2) performing a cost-benefit analysis for the addition of augments to the liquid and gaseous radioactive waste systems.

The quantities of radioactive materials that will be released in liquid and gaseous effluents and the quantity of radioactive waste that will be shipped off site are provided in Section 3.5 of the FES and Section 3.5 of the Supplement to the FES (NUREG-0139). In making these determinations, the staff considered waste flows, activity levels, and equipment performance that are consistent with expected normal plant operation, including anticipated operational occurrences, for an assumed 30 years of normal plant operation.

The estimated releases of radioactive materials in liquid and gaseous effluent were calculated using the PWR-GALE Code described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors (PWRs)," April 1976, modified to apply to liquid metal fast breeder reactors. The principal parameters used in these calculations are given in Table 3.2 of the FES and Section 3.5 of the Supplement to the FES.

The staff evaluated potential radwaste system augments based on a study of the applicants' system designs, the doses to the population within 50 mi of the reactor, an interim value of $1,000 per total body man-rem and $1,000 per man-thyroid-rem for reductions in dose by the application of augments, and the cost of potential radwaste system augments as presented in RG 1.110.

11.2 Acceptance Criteria

The staff has reviewed the applicants' design, design criteria, and design bases for the radioactive waste management systems for the CRBRP. The acceptance criteria used as the basis for its evaluation are set forth in Section II of SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5 (NUREG-0800); the applicable CRBR PDC, which have replaced the GDC of Appendix A to 10 CFR 50; 10 CFR 20.106; Appendix I to 10 CFR 50; and ANSI Std. N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." Guidelines for implementation of the requirements of the acceptance criteria are provided in the ANSI standards, regulatory guides, and other documents identified in Section II of the respective SRP sections. Conformance to the acceptance criteria provides the bases for concluding that the proposed radioactive waste management systems meet the requirements of 10 CFR 50.

11.3 Method of Review

The CRBRP radioactive waste management systems and associated radiation monitoring systems have been reviewed according to the applicable portions of SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5. The results of the review are discussed below.

11.4 System Description and Evaluation

11.4.1 Liquid Radioactive Waste Treatment System

The liquid radioactive waste treatment system will consist of process equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes. All potentially radioactive liquids generated in the plant, except for the bleed from the steam-water system, will be collected and processed through the liquid radioactive waste treatment system before they are released to the cooling tower blowdown stream and subsequently to the Clinch River. This system is discussed in detail in Section 11.2 of the PSAR.

All liquid waste will be processed on a batch basis to permit optimum control of releases. Before the wastes are released, samples will be analyzed to determine the types and amounts of radioactivity present. On the basis of the results of the analyses, the waste will be recycled for reuse, retained for further processing, or released under controlled conditions to the environment.

The liquid radioactive waste treatment system will consist of two subsystems: (1) the intermediate activity level liquid system and (2) the low activity level liquid system.

The liquid radioactive waste treatment system will be designed to collect and process wastes on the basis of the origin of the waste in the plant and the expected levels of radioactivity.

The intermediate activity level liquid (IALL) system will process the aqueous radioactive wastes generated from the washing of contaminated plant components in the large component cleaning cell and the small component cleaning autoclave. The IALL system will consist of two 20,000-gal collection tanks, a 10-gpm evaporator, two 10-ft^3 mixed-bed demineralizers, and two 20,000-gal distillate storage tanks. The staff estimates that the IALL system input flow will be approximately 340 gpd and the system design capacity will be approximately 14,400 gal per day, based on the evaporator flow rate. The staff assumed that 10% of the processed IALL system waste will be released to the environment through the cooling tower blowdown line.

The low activity level liquid (LALL) system will process the aqueous waste effluents from the floor drains, shower drains, and laboratory drains located in the plant and reactor service building. The LALL system will consist of two 2,400-gal collection tanks, a 10-gpm evaporator, two 10-ft^3 mixed-bed demineralizers, and two 2,400-gal monitor tanks. The staff estimates that the LALL system input flow will be approximately 850 gpd and the system design capacity will be approximately 14,400 gpd, based on the evaporator flow rate. The staff assumed that all of the processed LALL system waste will be released to the environment through the cooling tower blowdown line.

For the IALL and LALL systems, the difference between the expected flows and design capacity will provide adequate reserve for processing surge flows. The staff considers the system capacity and system design adequate for meeting the demands of the station during anticipated operational occurrences.

Tritium will enter the steam-water system by diffusion from the primary to intermediate heat transport system (IHTS) and from the IHTS to the steam-water system. To control the buildup of tritium in the steam-water system, the applicant will provide a 1-gpm bleed from the condensate and feedwater system to the cooling tower blowdown line.

The liquid radioactive waste treatment system will be located in the reactor service building. The design capacities of principal components considered in the liquid radwaste evaluation are listed in Table 11.1. The staff finds the applicants' proposed liquid radioactive waste treatment system design acceptable in accordance with SRP Section 11.2. The system design also will include measures intended to control the release of radioactive materials resulting from potential overflows from indoor storage tanks. Tank levels will be monitored either locally or in the control room and high level alarms will be activated should preset levels be exceeded. Overflow provisions such as sumps, dikes, and overflow lines will permit the collection and subsequent processing of tank overflow. The staff considers these provisions adequate to control the release of radioactive materials to the environment.

In its evaluation of the liquid radioactive waste management system, the staff considered

(1) the capability of the system to keep the levels of radioactivity in effluents "as low as is reasonably achievable" on the basis of the expected radwaste inputs over the life of the plant

(2) the capability of the system to maintain releases below the limits in 10 CFR 20 during periods of fission product leakage at design levels from the fuel

(3) the capability of the system to meet the processing demands of the station during anticipated operational occurrences

(4) the quality group and seismic design classification applied to the equipment, components, and structures housing the system

(5) the design features that are incorporated to control the releases of radioactive materials in accordance with CRBR PDC 52

The estimated releases of radioactive materials in liquid effluents were calculated using a modified version of the PWR-GALE Code. The PWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from pressurized-water reactors (PWRs). The code has been in used since 1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations (July 7, 1975), (4)

(4) release and transport mechanisms that result in the appearance of radioactive material in liquid streams, and (5) the CRBRP radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters used in these calculations are given in the Final Environmental Statement (FES) and the Supplement to the FES for CRBRP. The liquid source term is given in the FES and its Supplement.

Evaluation Findings

On the basis of its evaluation, the staff concludes that the radioactive materials in liquid effluents will not result in total body doses to an individual in an unrestricted area greater than 3 mrems per year or any organ dose greater than 10 mrems per year in accordance with Section II.A of Appendix I to 10 CFR 50. Also, the calculated release of radioactive material in the liquid effluents exclusive of tritium and dissolved gases will be less than 5 Ci per year and the total body and any organ dose will be less than 5 mrems per year from the site, in accordance with the option to Section II.D of Appendix I as provided in the Annex to Appendix I.

The calculated doses from liquid releases to the population within a 50-mi radius of the reactor are approximately 0.12 total body man-rem and 0.15 man-thyroid-rem. To assess the need for augmenting the radwaste system, values of $1,000 per total body man-rem and $1,000 per man-thyroid-rem are prescribed in Section II.D of Appendix I to 10 CFR 50. In view of the expected low population doses, it is clear that the maximum justified annual expenditure for any augments would be substantially less than $1,000. Potential radwaste system augments and their costs are listed in RG 1.110. When this list is examined, it becomes obvious that there are no available augments whose costs do not far exceed $1,000. Thus, the staff concludes that the liquid radwaste treatment system will reduce liquid radioactive effluents to as-low-as-is-reasonably-achievable levels in accordance with 10 CFR 50.34a, Appendix I to 10 CFR 50, and the Annex to Appendix I to 10 CFR 50, and is in accordance wit CRBR PDC 52 and 53.

The staff has determined that the liquid radwaste treatment system will be capable of reducing the release of radioactive materials in liquid effluents to concentrations below the limits in 10 CFR 20 during periods of fission product leakage from the fuel at design levels. The design of the system and structures housing these systems will meet the criteria set forth in RG 1.143.

11.4.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste treatment system will consist of equipment and instrumentation designed to collect, process, store, monitor, recycle, and dispose of radioactive gaseous wastes generated during normal operation, including anticipated operational occurrences. The radionuclides associated with these wastes will be principally noble gases. The gaseous radioactive waste treatment system will consist of two subsystems: (1) the radioactive argon processing system and (2) the cell atmosphere processing system. A detailed description of the systems is given in Section 11.3 of the CRBRP PSAR.

The radioactive argon processing system (RAPS) will process the primary sodium system cover gas (argon). The argon cover gas, contaminated with noble gases

and small quantities of tritium, will be collected in a 261-ft^3 vacuum vessel and transferred by a compressor to a 500-ft^3 surge vessel. From the surge vessel, the gases will be directed to a cryogenic still containing liquid argon in the still bottoms. The cryogenic still will condense the radioactive krypton and xenon that will collect in the still bottoms. Krypton and xenon will be separated periodically from the argon in the still bottoms by draining and transferring the gases to a 260-ft^3 noble gas storage vessel. The purified argon will be directed to the vacuum vessel as recirculation throughput and to a recycle argon vessel for reuse in the plant. The applicants propose to process gases from the noble gas storage vessel through the cell atmosphere processing system. The staff assumed in its evaluation model that the contents of the storage vessel will be released to the environment.

The cell atmosphere processing system (CAPS) will collect and process the gaseous radioactivity that may leak or diffuse into the inerted cells (nitrogen), which house the reactor, primary heat transfer system (PHTS), and reactor overflow vessel. The major input to the CAPS will consist of nitrogen containing trace quantities of contaminated argon cover gas and tritium that has diffused through PHTS piping and components. The nitrogen/air gas bleeds and purges from the cells will be collected in a 260-ft^3 vacuum vessel and tranferred by a compressor to a 698-ft^3 surge vessel. From the surge vessel, the gases will be treated in one of two tritium oxidizer units where the tritium will be converted to tritiated steam, condensed, and sent to the solid radioactive waste system for solidification and ultimate offsite disposal in a series of two cryogenically cooled charcoal delay beds, each containing 625 lb of charcoal. Although the flow rate into the CAPS will be variable, the flow rate through the beds will be maintained at a constant 50 scfm by an automatically controlled variable-flow recirculation loop. The charcoal beds will be operated at 35 psig and an average temperature of -140°F. After holdup of approximately 0.6 day for krypton, 40 days for xenon, and 0.03 day for argon, the gases will be discharged to the environment through the reactor service building heating and ventilation system.

The proposed seismic design and quality group classification and design capacities of the principal components in the RAPS and CAPS are listed in Table 11.1. The staff finds that the applicants' proposed gaseous radwaste management system is in conformance with SRP Section 11.3 and is, therefore, acceptable.

In addition, the staff has examined the possibility of nitrogen and oxygen combining in the CAPS to form potentially corrosive nitrogen oxides. Because of the low concentration (2%) of oxygen in the cell atmospheres and the fact that the reactions are unfavorable at the operating conditions of the CAPS unit, this will not be a problem.

Ventilation exhaust air from the reactor containment building, reactor service building, turbine building, and steam generator building will be exhausted to the atmosphere without treatment. Offgases removed by the mechanical vacuum pumps of the condenser air removal system will be discharged to the atmosphere without treatment through the exhaust plenum of the turbine generator building lube oil exhaust duct. The plant ventilation systems will be designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination.

The estimated releases of radioactive materials in gaseous effluents were calculated using a modified version of the PWR-GALE Code. The PWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from PWRs. The code has been in use since 1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations (July 7, 1975), (4) release and transport mechanisms that result in the appearance of radioactive material in gaseous streams, and (5) the CRBRP radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters used in these calculations are given in the FES and the Supplement to the FES for CRBRP. The gaseous source term is also given in the FES and its Supplement.

Evaluation Findings

On the basis of its evaluation, the staff finds that the radioactive materials in gaseous effluents will not result in an annual air dose to an individual in an unrestricted area that is greater than 10 mrads for gamma radiation or 20 mrads for beta radiation in accordance with Section II.B of Appendix I of 10 CFR 50 and the Annex to Appendix I. The calculated annual total quantity of all radioactive iodine and particulate material in gaseous effluents will not result in an annual dose or dose commitment to any individual in an unrestricted area that is greater than 15 mrems to any organ in accordance with Section II.D of Appendix I and the Annex to Appendix I.

The calculated area doses from gaseous releases to the population within a 50-mi radius of the reactor are approximately 0.17 total body man-rem and 0.17 man-thyroid-rem. Values of $1,000 per total body man-rem and $1,000 per man-thyroid-rem are prescribed in Section II.D of Appendix I to 10 CFR 50 and are used to assess the need for augmenting the radwaste system. Because of the expected low population doses, it is evident that the maximum justified annual expenditure for any augments will be substantially less than $1,000. Potential radwaste system augments and their costs are listed in RG 1.110. Upon examination of this list, it is obvious that there are no available augments whose costs do not far exceed $1,000. Thus, the staff concludes that the gaseous radwaste treatment system will reduce gaseous radioactive effluents to levels that are as low as is reasonably achievable (ALARA) in accordance with 10 CFR 50.34a, Appendix I to 10 CFR 50; thus, the criteria of CRBR PDC 52 and 53 have been satisfied.

The staff has determined that the proposed gaseous radioactive waste treatment system and plant ventilation systems will be capable of reducing the release of radioactive materials in gaseous effluents to concentrations that are below the limits of 10 CFR 20 during periods of fission product leakage from the fuel at design levels.

The staff has reviewed the applicants' quality assurance provisions for the gaseous radioactive waste treatment system, the quality group classifications used for the system components, and the seismic design applied to the design of the system and the structures housing the system and determined that they meet the criteria as set forth in RG 1.143.

11.4.3 Solid Radioactive Waste Treatment System

The solid radioactive waste treatment system will be designed to handle, collect, and process five types of waste: (1) concentrated liquids, (2) low-activity noncompactible solids, (3) metallic sodium, (4) sodium-contaminated components, and (5) compactible solids. A detailed description of the solid radioactive waste treatment system is given in Section 11.5 of the CRBRP FSAR. Concentrated liquids will consist of evaporator bottoms from the liquid radwaste treatment system and tritiated waste from the CAPS tritium oxidizer units. This waste will be combined with cement and an additive in a 55-gal drum to form a solid matrix. The drum will be subsequently capped, decontaminated, and placed in temporary storge before it is shipped off site to a licensed burial facility. Low-activity noncompactible solids will include tools, contaminated filters, metal component parts, valves, and vapor traps. This waste will be placed in drums and the drums will then be capped, decontaminated, and placed in temporary storage before they are shipped off site. Metallic sodium will be generated from fuel-handling operations and will be either processed on site into a form suitable for offsite shipment or stored on site. Sodium-contaminated components will include the primary, intermediate, and ex-vessel storage tank cold traps. The handling of the cold traps will include placing the trap into a removal cask for subsequent offsite shipment in a special container. The final disposition of the cold traps has not yet been determined; however, the applicants will use the research and development efforts of the fast flux test facility concerning the packaging, transport, and disposition of sodium-contaminated waste. Compactible solids will consist of rags, paper, and rubber seals. The waste will be placed in 55-gal drums and compacted by a hydraulic machine before it is shipped off site. All wastes will be packaged and shipped in accordance with 10 CFR 20, 10 CFR 71, and 49 CFR 170-178.

The solid radwaste system and structures housing the system will conform to the design, construction, and testing criteria of RG 1.143. The solid radwaste system will be located in the reactor service building. In addition, the solid radwaste system will incorporate a process control program and provide for waste storage in accordance with BTP ETSB 11-3. The applicants' process control program will be reviewed in detail at the operating license stage. The staff finds the applicants' proposed system design to be acceptable.

The staff has determined that the following solid wastes will be shipped off site annually: (1) approximately 1,100 ft^3 of solidified concentrated liquids containing 2,800 Ci of activity, (2) approximately 800 ft^3 of noncompactible solid waste containing 300 Ci of activity, (3) approximately 15 ft^3 of processed metallic sodium containing 40 Ci of activity, (4) approximately 750 ft^3 of sodium-contaminated waste containing 16,000 Ci of activity, and (5) approximately 1,000 ft^3 of compacted waste containing 5 Ci of activity. Storage facilities will include an area in the radwaste building for 214 drums (55-gal each). The staff finds the storage capacity adequate for meeting the demands of the station for normal operation.

Evaluation Findings

The staff has reviewed the applicants' solid waste management system and concludes that the applicants will use components, system designs, and design

criteria for the solid radwaste systems that are consistent with components and systems used in operating plants that have demonstrated their efficiency, capacity ratings, and availability factors in extensive operational use.

The applicants have not yet submitted a process control program for the purpose of providing assurance that waste solidification will meet the requirements for packaging, handling, shipping, and disposal. Although such a program is not addressed in this report, it will be part of the Technical Specifications as required by BTP ETSB 11-3 and will be provided at the operating license stage.

The applicants will provide adequate storage space for onsite storage of solidified and dry wastes to meet the requirements of BTP ETSB 11-3. The capability of the proposed system to process the types and volumes of wastes generated during normal operation, including anticipated operational occurrences, is in accordance with CRBR PDC 52.

Provisions for handling the wastes are in accordance with 10 CFR 20, 10 CFR 70, and 49 CFR 171-179 (which are Department of Transportation regulations). The applicants' quality group classification and seismic design have been found acceptable. The proposed system meets the requirements of RG 1.143.

The basis for the staff's acceptance of the solid waste management system has been conformance of the system design criteria to the regulations and guides referenced above. On the basis of the foregoing evaluation, the staff concludes that the proposed solid radwaste system is acceptable.

11.4.4 Process and Effluent Radiological Monitoring Systems

The process and effluent radiological monitoring systems will provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environment.

Table 11.4-1 of the CRBRP PSAR provides the proposed locations and descriptions of continuous monitors and sampling points. Monitors on certain effluent release lines will automatically terminate discharges should radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be sampled and analyzed periodically in the plant laboratory.

Evaluation Findings

The staff has reviewed the locations and types of effluent and process monitoring provided. On the basis of the plant design and continuous monitoring and intermittent sampling locations, the staff has concluded that all normal and potential release pathways will be monitored. It has also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes that affect radioactivity releases. On this basis, the staff considers that the monitoring and sampling provisions meet the requirements of CRBR PDC 52, 55, and 56 and the guidelines of RG 1.21.

11.5 TMI Items Related to Waste Management

11.5.1 Item II.F.1, Attachment 1, Noble Gas Effluent Monitor (NUREG-0737)

Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements included in a revision to RG 1.97 and in other regulatory guides, which will be promulgated in the near term.

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper-range capacity of 10^5 $\mu Ci/cm^3$ (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition concentrations (ALARA) to a maximum of 10^5 $\mu Ci/cm^3$ (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of 10.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed, taking into consideration

(1) the use of this information by an operator during both normal and abnormal plant conditions

(2) integration into emergency procedures

(3) integration into operator training

(4) other alarms during emergency

(5) need for prioritization of alarms

Discussion and Clarification

NUREG-0578, Section 2.1.8b, provides the basic requirements for this item. Letters dated September 27 and November 9, 1979, provide clarification, and NUREG-0660, Item II.F.1, provides the action plan for additional accident-monitoring instrumentation by noble gas effluent radiological monitor requirements. Additional clarification was provided by letters dated September 5 and October 31, 1980.

By summary clarification, the following guidelines were established:

(1) Applicants shall provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant. Monitors of gaseous effluents shall meet the requirements specified in Table II.2. Typical plant effluent pathways to be monitored are also given in the table.

(2) The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

(3) Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steamline upstream of the valves are acceptable with procedures to correct for the low-energy gamma radiation that the external monitors would not detect. Isotopic identification is not required.

(4) Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

(5) The design description shall include the following information:

(a) system description, including:

- instrumentation to be used, including range of sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable
- monitoring locations (or points of sampling), including description of methods used to ensure representative measurements and background correction
- location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data
- assurance of the capability to obtain readings at least every 15 min during and following an accident
- the source of power to be used

(b) description of procedures or calculative methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

(5) Applicants should have available for review the final design description of the as-built system, including piping and instrumentation diagrams together with either (a) a description of procedures for system operation and calibration or (b) copies of procedures for system operation and calibration.

Applicants will submit the above details in accordance with the proposed review shedule, but in no case less than 4 months before the issuance of an operating license.

Evaluation and Conclusion

The applicants' initial submittal did not contain sufficient information to conduct a review and was unacceptable. The applicants submitted supplemental information in May 1982. On the basis of the information received, the staff concludes that the applicants have taken adequate steps toward meeting the requirements of Item II.F.1, Attachment 1, of NUREG-0737. More detailed information will be supplied at the operating license stage.

11.5.2 Item II.F.1, Attachment 2, Sampling and Analysis of Plant Effluents (NUREG-0737)

Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to RG 1.97 and in other regulatory guides, which will be promulgated in the near term.

Because it is not considered practical at this time to provide iodine gaseous effluent monitors for the accident condition, capability for such radioiodine monitoring for the accident condition shall be provided via sampling conducted by adsorption on charcoal or on other media, followed by onsite laboratory analysis.

Discussion and Clarification

(1) Applicants shall provide continuous sampling of gaseous effluents for postaccident releases of radioactive iodines and particulates to meet the requirements of Table 11.3. Applicants shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize interference from noble gas.

(2) The shielding design basis is given in Table 11.3. The design of the sampling system should permit plant personnel to remove samples, replace sampling media, and transport the samples to the onsite analysis facility, receiving no radiation exposures in excess of the criteria of GDC 19 (5-rems whole-body exposure and 75 rems to the extremities), during the duration of the accident.

(3) The design of the systems for the sampling of particulates and iodines should provide for sample nozzle-entry velocities which are approximately isokinetic (same velocity) with expected in-duct and in-stack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow-control devices which have

the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ±20%. Further departure from the isokinetic condition need not be considered in design. Corrections for anisokinetic sampling conditions, as provided in Appendix C of ANSI Std. N13.1-1969, may be considered on an ad hoc basis.

(4) Effluent streams which may contain air with entrained water shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

(5) Applicants should have available for review the final design description of the as-built system, including piping and instrumentation diagrams, together with either (a) a description of procedures for system operation and calibration or (b) copies of procedures for system operation and calibration.

(6) Changes to Technical Specifications will be required.

Evaluation and Conclusions

The applicants did not supply enough information to conduct a review, so the applicants' initial submittal was unacceptable. The applicants submitted additional information in May 1982. On the basis of the information that has been supplied, the staff concludes that the applicants have taken adequate steps toward meeting the requirements of Item II.F.1, Attachment 2, of NUREG-0737. More detailed information will be supplied at the operating license stage.

11.5.3 Item III.D.1.1, Integrity of Systems Outside Containment Likely To Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors (NUREG-0737)

Position

Applicants shall implement a program to reduce to as-low-as-practical levels leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. This program shall include the following:

(1) Immediate Leak Reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction

Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Discussion and Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

(1) Systems that should be leak tested are as follows (any other plant system which has similar functions or postaccident characteristics, even though not specified herein, should be included):

(a) residual heat removal (RHR)
(b) containment spray recirculation
(c) high-pressure injection recirculation
(d) containment and primary coolant sampling
(e) reactor core isolation cooling
(f) waste gas (includes headers and cover gas system outside containment in addition to decay or storage system).

A list of systems containing radioactive materials which are excluded from the leak-test program should be included, and justification for their exclusion should be provided.

(2) Testing of gaseous systems should include helium leak detection or equivalent testing methods.

(3) A program should be considered to reduce the leakage potential of release paths that result from design and operator deficiencies, as discussed in the staff's letter of October 17, 1979 to all operating nuclear power plants regarding North Anna and related incidents.

(4) This requirement shall be implemented by applicants for operating license before issuance of a full-power license.

(5) Applicants shall submit the requested information in the "Discussion and Clarification" section of this item (above) at least 4 months before issuance of a fuel-loading license.

(6) Changes to Technical Specifications will be required.

Evaluation and Conclusion

By features of design and operational controls, CRBRP will confine primary coolant fluids to within containment and will not process them outside the containment. Systems containing these primary coolant fluids would leak to the containment only.

Because of these design features, the staff concludes that the CRBRP will be in compliance with the concern of Item III.D.1.1 of NUREG-0737.

Table 11.1 Design parameters of principal components considered in the evaluation of the liquid and gaseous radioactive waste treatment systems

Component	Number	Capacity each
Liquid Systems		
Intermediate-activity system*		
Collection tank	2	20,000 gal
Evaporator	1	10 gpm
Demineralizer	2	10 ft^3
Monitor tank	2	20,000 gal
Low-activity system*		
Collection tank	2	2,400 gal
Evaporator	1	10 gpm
Demineralizer	2	10 ft^3
Monitor tank	2	2,400 gal
Gaseous Systems		
Radioactive argon processing system*		
Vacuum vessel	1	261 ft^3
Compressor	2	10 scfm
Surge vessel	1	500 ft^3
Cryogenic still	1	10 scfm
Storage vessel	1	260 ft^3
Cell atmosphere processing system*		
Vacuum vessel	1	260 ft^3
Compressor	2	50 scfm
Oxidizer	2	50 scfm
Charcoal bed adsorber	2	625 lb charcoal

*Quality group and seismic design in accordance with RG 1.143.

Table 11.2 High-range noble gas effluent monitors

Requirement

Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.

Purpose

To provide the plant operator and emergency planning agencies with information about plant releases of noble gases during and following an accident.

Design-Basis Maximum Range

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter ($\mu Ci/cm^3$) of air at standard temperature and pressure (STP) for monitors employing beta radiation detectors. (NOTE: 1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source.) Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gas changes) should be taken into account.

10^5 $\mu Ci/cm^3$ - undiluted containment exhaust gases (e.g., PWR reactor building purge; BWR drywell purge through the standby gas treatment system).

- undiluted PWR condenser air removal system exhaust.

10^4 $\mu Ci/cm^3$ - diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).

- BWR reactor building (secondary containment) exhaust air.

- PWR secondary containment exhaust air.

10^3 $\mu Ci/cm^3$ - buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).

- PWR steam safety valve discharge, atmospheric steam dump valve discharge.

10^2 $\mu Ci/cm^3$ - other release points (e.g., radwaste buildings, fuel handling/storage buildings).

Redundancy

Not required; monitoring the final release point of several discharge inputs is acceptable.

Table 11.2 (Continued)

Specifications

(None) Sampling design criteria per ANSI Std. N13.1.

Power Supply

Vital instrument bus or dependable backup power supply to normal ac.

Calibration

Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/Hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.

Display

Continuous and recording as equivalent Xe-133 concentrations or $\mu Ci/cm^3$ of actual noble gases.

Qualification

The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.

Design Considerations

Off-line monitoring is acceptable for all ranges of noble gas concentrations.

In-line (in-duct) sensors are acceptable for 10^2 $\mu Ci/cm^3$ to 10^5 $\mu Ci/cm^3$ noble gases. For less than 10^2 $\mu Ci/cm^3$, off-line monitoring is recommended.

Upstream filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident.

For externally mounted monitors (e.g., PWR main steamline), the thickness of the pipe should be considered in accounting for low-energy gamma radiation.

Table 11.3 Sampling and analysis or measurement of high-range radioiodine and particulate effluents in gaseous effluent streams

Equipment

Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant's gaseous effluents during and following an accident. The capacity to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.

Purpose

To determine quantitative release of radioiodines and particulates for dose calculation and assessment.

Design-Basis Shielding Envelope

10^2 $\mu Ci/cm^3$ of gaseous radioiodines and particulates, deposited on sampling media; 30-min sampling time, average gamma energy (E) of 0.5 MeV.

Sampling Media

Iodine >90% effective adsorption for all forms of gaseous iodine.

Particulates >90% effective retention for 0.3-micron (μ)-diameter particles.

Sampling Considerations

Representative sampling according to ANSI N13.1-1969.

Entrained moisture in effluent stream should not degrade adsorber.

Continuous collection required whenever exhaust flow occurs.

Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

Analysis

Design of analytical facilities and preparation of analytical procedures shall consider the design-basis sample.

Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

12 RADIATION PROTECTION

The Preliminary Safety Analysis Report (PSAR) provides information on the methods for radiation protection including the facility design and layout, equipment design, and a description of the health physics program. An estimate of occupational radiation exposure to plant personnel is included. Shielding is provided to reduce radiation levels. Ventilation is arranged to control the flow of potentially contaminated air. Radiation monitoring is employed to measure levels of radiation and contamination in potentially occupied areas, and to measure airborne radioactivity throughout the plant. A health physics program is provided for plant personnel and visitors, during reactor operation, maintenance, refueling, radwaste handling, and inservice inspections. The staff reviewed and evaluated the description and analysis of the radiation protection program included in the PSAR, and responses to staff requests for additional information.

The principal criteria used to determine acceptability of the program are that doses to personnel will be maintained within the established limits of 10 CFR 20, "Standards for Protection Against Radiation," and that design and program features are consistent with the guidelines of RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," June 1978. In response to staff questions, the applicants have added an Appendix A, "Information Related to ALARA for Occupational Exposures," to Chapter 12 of the PSAR.

The applicants have considered means to keep external and internal radiation exposures to personnel, including both individual and collective person-rem doses, as low as is reasonably achievable (ALARA). Shielding has been designed to control radiation exposure so that (1) doses to operating personnel and the general public will satisfy the applicable sections of 10 CFR 20 and 10 CFR 50; (2) doses to plant personnel will be ALARA and the CRBR project will follow the guidance of RG 8.8 and 8.10, as they relate to CRBR, during design and operation of the CRBR project; and (3) the size of the operating crew will be determined by work requirements and not by radiation exposure limitations.

On the basis of its review, the staff has concluded that the radiation protection program will provide reasonable assurance that doses to personnel will be less than the limits established by 10 CFR 20 and will be maintained as low as is reasonably achievable, consistent with the guidelines of RG 8.8. The CRBRP radiation protection program is acceptable. Details are discussed in the following sections.

12.1 Shielding

12.1.1 Design Objectives

The applicants have committed that each system having equipment located in the nuclear island will undergo review to ensure that occupational radiation exposures are as low as is reasonably achievable. This review is to be coordinated

by representatives of the lead reactor manufacturer; information and advice will be obtained from the TVA ALARA committee. In addition, the applicants will conduct two design reviews a year during construction, on the plant radiation protection and shielding program. Health physicists from Commonwealth Edison and/or TVA will participate, and their contributions will be factored into the plant design and procedures, as appropriate.

The shielding thicknesses at CRBR will be provided to ensure compliance with the plant radiation zoning and to minimize plant personnel exposure. Concrete radiation shielding in the facility will be designed in accordance with the recommendations of RG 1.69. Shielding for labyrinths has been designed using a general purpose gamma-ray scattering code. The applicants used the following acceptable computer codes for other shielding designs:

ANISN-W	LSD-2	SCAP	XSRES-WIDX
APPROPOS	ORIGEN	SCAP-BR	
DOTIIIW	QAD	S-4M	
GAMLEGW	RIBD	TRIPLET	

Abstracts of these codes are provided in the PSAR.

The basic objectives of the shielding at CRBR are (1) to reduce exposures to plant personnel and the public to levels that are within the dose limits of 10 CFR 20, and are ALARA; (2) to ensure sufficient personnel access and occupancy time for normal maintenance, operation, and inspection activities; (3) to reduce potential neutron activation of equipment; and (4) to mitigate the possibility of radiation damage to materials. Radioactive components and piping will be located in separate shielded cubicles to minimize exposure during maintenance and inspection activities. Potentially high radiation components will be totally enclosed in shielded compartments with hatch openings or removable block walls. Readouts and controls for radioactive equipment will be remotely located to reduce personnel exposures. Design features to minimize radiation streaming through penetrations include: (1) providing offsets between the radiation source and accessible areas, (2) locating pentrations as far above the floor elevation as possible, and (3) using baffle shield walls or grouting the area around the penetration. Within containment, shielding will be designed to reduce the dose rates in accessible areas to 100 mrems per hour or less. These shielding techniques will be designed to maintain personnel radiation exposures ALARA, in accordance, as appropriate for CRBR, with RGs 8.8 and 8.10, and are acceptable.

12.1.2 Design Description

Shielding will be placed around any plant component or piping if the expected radiation level at any time during plant life would exceed the zoning criteria. For discontinuities, such as shield penetrations, point kernel and Monte Carlo, methods will be used to design shielding equivalent to the surrounding bulk shielding. Penetrations will be designed for minimal effect on whole-body dose rates in accessible areas outside the shielded cells. Valves and sampling stations with high radiation potential will be located in shielded areas or will be designed to be operated remotely.

Radiation protection concepts directed to keeping personnel exposures well below regulatory limits will be used throughout in the design of the plant. Shielding design and radiation zoning will be based on maximum expected radiation conditions during plant life. To the extent practicable, labyrinths will be provided for shielded cubicles containing major sources. Pipes and ducts will be routed through high-zoned, low-access areas when practicable; shielding will be provided for pipe trenches and penetrations.

The applicants have provided five radiation zones as a basis for classifying occupancy and access restrictions on various areas within the plant. On this basis, maximum design dose rates will be established for each zone and used as input for shielding of the respective zones. For example, design radiation levels in operating areas where personnel are expected to be working for a 40-hour week will be less than 0.2 mrem per hour.

In addition to designing plant equipment and components to comply with ALARA guidelines, the applicants' facility layout will be designed to reduce personnel exposures. Valve galleries will have shielded entrances and floor drains to control radioactive leakage. Whenever practicable, pipes carrying radioactive materials will be separated from nonradioactive piping and will be located in shielded pipe chases. Concrete floors and walls of cubicles that contain equipment handling radioactive liquids will be covered with a smooth-surfaced coating to facilitate decontamination. Major components in radioactive systems will be isolated in individual shielded compartments. These cubicles will be sized to permit adequate space and ease of motion for maintenance purposes. Inservice inspection points will be located in properly shielded low-radiation zones to reduce radiation exposure received by inspectors. The applicants will install backup lighting in areas containing high-radiation equipment so that immediate entry into the area to replace a single burned-out light will be unnecessary. These features conform, as appropriate for CRBR, with RG 8.8 and are acceptable.

TMI Item II.B.2 Plant Shielding To Provide Access to Vital Areas and Protect Safety Equipment for Postaccident Operation

Requirement

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids and (2) implement plant design modifications necessary to permit adequate access to vital areas and protect safety equipment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly before the issuance of operating licenses.

This requirement is directly applicable to CRBRP.

Implementation

As part of the normal design process, the applicants have performed radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids. Source terms for equipment have been developed for the components of these systems. In addition, the applicants have conducted, as part of the key system review process, an assessment of the accessibility of equipment following accidents such as sodium spills, radioactive gas leaks, and a hypothesized core-melt accident. Following these key system reviews, the applicants have modified the design to ensure accessability of components and control panels, and acceptability of the accident consequences. The PSAR has been appropriately modified to reflect the changes resulting from the above.

12.1.3 Source Terms

The reactor core is the primary source of radiation in the containment, emitting neutrons and gamma rays during operation. The reactor coolant system is the next highest source of radiation in the containment. The sources of radiation in the reactor coolant system are fission products from fuel-cladding defects and corrosion and activation products. Of these radiation sources, the activation product, Na-24 is the predominant activity in the reactor coolant pumps, steam generators, and reactor coolant piping. The primary sources of personnel exposure, in buildings other than the containment, are fission, activation, and corrosion products from the primary coolant.

In response to a staff question, the applicants have committed to high-temperature/high-flow-rate filtration of the primary sodium before initial fuel loading. Core special assemblies will be inserted into normal positions before fuel loading for filtration of undetected debris. Co-59 content in stainless steel materials in the core, control and blanket assemblies, removable radial shielding, and upper internals is limited by specification to 0.05%. The only stellite to be used in contact with the primary coolant will be in the cold-leg check valve. These features should serve to reduce the buildup of radiation levels from the formation, transport, and deposition of activated corrosion products.

Concentrations of fission and activation product radionuclides in the reactor, and in plant systems, components, and equipment, are calculated for maximum conditions during plant life. Neutron activation rates are developed from the use of evaluated nuclear data files (ENDF/B-III) (Drake, 1984). The model for the release of fission products assumes continuous operation of the reactor with defects in fuel rods generating 1% of the core power. The major source of tritium is boron activation and decay in the control assemblies; the total projected production of tritium over one fuel cycle averages 89 Ci per day. The applicants have committed to use RGs 1.4 and 1.112, as they relate to CRBR, with regard to source terms.

Given the proposed shielding design and the projected radiation source terms, radiation exposure rates in occupied areas will be within the zoning criteria. Projected doses are within 10 CFR 20 limits for workers in restricted areas and are acceptable.

12.1.4 Area Radiation Monitoring

The area and airborne radioactivity monitoring system will be designed to detect, measure, and provide a continuous record of radiation levels at selected building locations. Criteria for selection of monitor locations are anticipated radiation level under normal, shutdown, and abnormal conditions; frequency and duration of occupancy; flow of traffic under normal and accident conditions; proximity of high radiation sources; and the consequence of an undetected increase in radiation level. The applicants have listed the proposed locations and characteristics of area radioactivity monitoring instruments, and as a result of staff questions, the criteria for selection of such instruments.

The applicants' area radiation monitoring system will be designed to alert immediately any personnel entering or working in monitored areas of increasing radiation levels, and to inform control room personnel of the occurrence and approximate location of abnormal area gamma radiation increases. The applicants plan to use portable area monitors where needed. The area radiation monitoring system will be equipped with local and remote audio and visual alarms and a facility for central recording. At 6-month intervals, the applicants will calibrate each area monitor against a radiation check source traceable to the National Bureau of Standards, or from an equally acceptable source. The objectives and location criteria for this monitoring system are in conformance with 10 CFR 20 and 10 CFR 50, and are acceptable.

12.1.5 Estimates of Exposure

The applicants' estimates of person-rem exposures are based on conservatively assumed radiation source design shielding, calculated peak dose rates at specified locations within the plant, and projected manpower levels. Occupancy times are based on type of operation or work assumed to be performed by each individual. Three categories of workers were characterized: plant operators (including shift engineers and unit operators), other nonmaintenance personnel (including health physicists, chemical engineers/analysts, and instrumentation personnel), and maintenance personnel.

The applicants' estimate for average annual occupational radiation exposure, including maintenance, is approximately 400 person-rems. The bases for this estimate are reasonable, consistent with the provisions of RG 8.19, and are acceptable.

12.2 Ventilation

12.2.1 Design Objectives

The CRBR ventilation system will be designed to provide a suitable environment for personnel and equipment during normal operations and anticipated operational occurrences, and to ensure that plant personnel and the general public are not inadvertently exposed to airborne contaminant concentrations exceeding those given in 10 CFR 20 and 10 CFR 50. The applicants intend to maintain personnel exposure ALARA by (1) maintaining airflows from areas of potentially low airborne contamination to areas of progressively higher potential airborne contamination and (2) exhausting a greater volumetric flow from potentially

contaminated areas through high-efficiency particulate air (HEPA) filters and carbon adsorbers. These design criteria are in accordance with those contained in RG 8.8 and are acceptable.

With regard to occupational radiation exposure techniques, the ventilation system for CRBR will be designed in accordance with the guidelines of RG 1.52. The applicants will replace filter elements before the radioactivity level is great enough to create a personnel hazard. The active elements of the atmosphere cleanup systems will be designed to permit ready removal and to be accessible directly from working platforms, and are acceptable.

12.2.2 Design Description

Potential sources of airborne radioactive contamination will be located principally in the reactor containment, reactor service, and steam generator buildings. Those buildings will be separated by heavy structural shielding and fire separation walls. The relative pressure differentials between buildings cannot cause flow of such contamination from one building to another. All occupied areas where airborne radioactive materials may be present will be maintained at a slight negative pressure with respect to surrounding areas. The reactor containment building's air supply and exhaust ducts will be provided with automatic isolation valves. If airborne radioactivity levels exceed preset limits, the building will be isolated.

These features conform, as appropriate for CRBR, with RG 8.8 and are acceptable.

12.2.3 Source Terms

The principal sources of airborne radioactivity orginate from reactor cover gas leakage and tritium diffusion. Design-base releases are based on operation with 1% failed fuel; estimates of releases are based on operation with 0.1% failed fuel (expected case).

The fresh air makeup rate for the plant ventilation systems is conservatively based on expected frequency of occupancy in the various plant areas. The CRBR ventilation system will be designed to maintain airborne concentrations of radioactive material in areas normally occupied within the limits given in 10 CFR 20. The source terms have been estimated conservatively, and are acceptable.

12.2.4 Airborne Radioactivity Monitoring

Fixed and mobile continuous air monitors (CAMs) will provide local audible and visual alarms if airborne concentrations of airborne radioactive materials exceed the monitor setpoints. These instruments will be used to monitor the atmosphere continuously at (1) any specific location where maintenance is being performed, (2) any specific location where a process system failure is suspected of causing an airborne radioactive materials leakage, and (3) inerted cell purging activities.

High- and low-volume portable air samplers will be used for representative sampling of breathing air at locations infrequently occupied. Samples will be

analyzed for gross radioactivity and radioisotopic content as appropriate. Frequency of use will depend on operational need.

Continuous (fixed) monitors will serve the reactor containment building and reactor service building operating areas. Continuous (mobile) monitors will serve in frequently occupied operating areas adjacent to potential radioactive sources, for example, zones I and II.

The CRBR objectives and location criteria for area and airborne radiation monitoring systems are in conformance with 10 CFR 20, 10 CFR 50, and RG 8.8, and are acceptable.

TMI Item III.D.3.3 In-Plant Radiation Monitoring

Requirement

Applicants shall review their designs to ensure that provisions for monitoring in-plant radiation and airborne radioactivity are appropriate for a broad range of routine and emergency conditions. Applicants shall, to the extent possible provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly before the issuance of operating licenses.

Implementation

The in-plant monitoring system incorporates provisions for monitoring radiation and airborne radioactivity over a broad range of routine and emergency condition.

The complement of portable radiation monitors and samplers and the onsite analysis and calibration capability will be the same as at other nuclear power plants operated by TVA and are therefore acceptable.

12.2.5 Inhalation Doses

The CRBRP ventilation design features help to ensure that the airborne radioactivity concentration in all normally accessible areas will be well below the limits specified in 10 CFR 20. The applicants will make use of occupancy time limits and/or respiratory equipment to protect personnel required to enter plant areas where unusually high radionuclide concentrations occur. The applicants provided a tabulation of the expected annual doses to plant personnel from airborne radionuclides for pertinent areas in the plant. The applicants' assumptions and models, upon which inhalation dose estimates are based, meet the information requirements of WASH-1302, and are acceptable.

12.3 Health Physics Program

12.3.1 Program Objectives

Objectives of the health physics program are to (1) apply radiation standards and procedures; (2) review proposed methods of plant operation; (3) participate in development of relevant plant documents; (4) provide specialized training in radiation protection; (5) provide health physics coverage for all plant operations, including maintenance, fuel handling, waste disposal, and decontamination; (6) provide radiation monitoring; (7) maintain records of personnel exposures and plant radiation and contamination levels; and (8) maintain plant personnel exposures as low as is reasonably achievable.

The Plant Health Physicist, under the administrative supervision of the Chief, Radiological Hygiene Branch in the TVA Division of Occupational Health and Safety, is the onsite supervisor of the branch, responsible for direction of the health surveillance program for all plant operations involving potential radiation hazards. The Plant Health Physicist keeps the Plant Superintendent informed at all times of radiation hazards and conditions related to potential exposure or radioactive contamination.

A four-member corporate ALARA committee, with representation from design, operations, and radiation protection groups, is responsible for (1) reviewing and appraising the effectiveness of the ALARA program, from initial planning to decommissioning; (2) reviewing plant design features and operating procedures; (3) auditing the onsite radiation control program at least annually; (4) determining trends in the exposure of station personnel; and (5) taking appropriate actions to correct adverse trends.

These design objectives are appropriate for CRBR, and are designed to maintain personnel radiation exposures ALARA, in accordance with RGs 8.8 and 8.10, and are acceptable.

12.3.2 Facilities and Equipment

The health physics facilities necessary to monitor and control the routine radiological condition of the plant and to maintain positive control over access to high radiation areas will include offices and a laboratory, containing a fume exhaust hood equipped with HEPA filters. Portable and laboratory radiation monitoring instruments, respiratory protection equipment, and other relevant supplies will be kept in the laboratory. Also to be provided are facilities for locker-change areas, clean laundry storage, and personnel decontamination rooms equipped with contamination-monitoring instruments and showers.

Other radiation-protection facilities will include male and female changing areas, a health physics office and work area, a laboratory with shielded counting areas, a "hot" machine shop, and calibration facilities. These facilities are consistent with the provisions of RG 8.8.

Equipment to be used for radiation protection purposes will include fixed radiation protection instrumentation, portable radiation survey instruments,

personnel monitoring instruments, fixed and portable area and airborne radioactivity monitors, air samplers, respiratory equipment, and protective clothing.

Protective clothing and equipment to be provided will include coveralls, lab coats, gloves, head covers, foot covers, plastic suits, and respiratory protection, including half-face and full-face masks with filters, full-face masks with constant air flow supplies, and self-contained breathing apparatus.

Equipment decontamination facilities will consist of a large component cleaning cell in the reactor containment building, and decontamination facilities for small equipment, in the radwaste area of the reactor service building. A personnel decontamination facility in the plant service building will be equipped with showers and sinks, near the exit from the controlled access area portal.

The proposed CRBR facilities and equipment are designed to maintain personnel radiation exposures ALARA, in accordance with RGs 8.8 and 8.10, and are acceptable.

12.3.3 Personnel Dosimetry

A personnel dosimetry service will be provided for workers for measurement of beta, gamma, and neutron radiation. Additionally, a whole-body counter is proposed to be used for determining body content of gamma-emitting radionuclides.

The personnel monitoring capability, as proposed, is consistent with the provisions of 10 CFR 20 and RG 8.8, and is acceptable.

12.3.4 Estimated Occupancy Times

The applicants have provided estimates of anticipated person-hours of occupancy involved in routine maintenance, operations, and refueling, for normally accessible areas of the reactor containment and reactor service buildings. This information, in conjunction with dose rate estimates presented in Section 12.1.5, provides a reasonable basis for estimating person-rem doses in accordance with RG 8.19, and is acceptable.

12.4 Information Related to ALARA for Occupational Radiation Exposures

12.4.1 CRBRP ALARA Commitment

TVA management is committed to maintaining radiation exposures to its employees and the general public and the release of radioactive materials to unrestricted areas as low as is reasonably achievable (ALARA), as defined in 10 CFR 20. For the protection of its employees, TVA also subscribes to the ALARA philosophy set forth in RGs 8.8 and 8.10 in the design and operation of all facilities utilizing radioactive materials or radiation sources. This commmitment to the ALARA provisions appropriate for CRBR is acceptable.

12.4.2 10 CFR 20 Requirements

The applicants specify that 10 CFR 20 applies to the CRBR and addresses particularly those provisions that relate to maintaining doses ALARA, and exposure limits for restricted areas and unrestricted areas.

This firm commitment to the provisions of 10 CFR 20 is acceptable.

12.4.3 CRBRP ALARA Program

The TVA ALARA program consists of (1) full management commitment to the overall objectives of ALARA, (2) issuance of specific relevant documents and procedures, (3) continued appraisal of in-plant radiation and contamination and conditions, and (4) a four-member ALARA committee, including representatives from design, operations, and radiation protection groups, to review and appraise the effectiveness of the ALARA program. The program is acceptable.

The specific responsibility of the ALARA Committee is to determine (1) that an effective ALARA program is established at each TVA nuclear power plant; (2) that the ALARA program is implemented from the initial planning stage through decommissioning of the plant; (3) by review of design features, operating procedures, and maintenance practices, as well as annual audits of the onsite radiation control program, that the objectives of the ALARA program are attained; (4) that information about ALARA programs at other operating plants is taken into account in the design and operation of TVA plants; (5) that all maintenance activities are planned in accordance with the objectives of the ALARA program; and (6) trends in the exposures of plant personnel to permit corrective actions to be taken.

In answer to staff questions, the applicants described design changes resulting from the ALARA design review which have resulted in reduction of projected occupationial exposures for the following operations:

(1) Liquid Radwaste System--Routine operations will be conducted remotely from a zone 1 control center; required valve manipulation will be done in corridors outside the cells; all piping systems will have low point drains to allow complete drainage of flush liquids; all cells will be provided with floor drains for removing spilled liquids; cell floors will be painted with epoxy for ease of decontamination; filter cartridges will be removed remotely and placed in concrete-lined drums; the drums will be handled remotely; spent resins will be slurried and transported to a decanter in the solid waste system, and concreted automatically.

(2) Liquid Radwaste System Maintenance--Sprinklers will be provided to wash down internal walls in the collection and concentrate tanks before maintenance; the wash water will be sent to other subsystems for processing; major components will be skid mounted; access will be provided by separate hatches in the operating floor; small components and tools can be decontaminated in the decontamination facility before repair or use.

(3) Gaseous Radwaste System Maintenance--Components which contain radioactive gases will be housed in shield-wall cells; service lives of expendable components have been considered; redundant components will be provided to ensure that replacement and/or maintenance can be performed during refueling and periods of low radiation levels; filter vapor traps will be each housed in separate cells so that replacement can be carried out with only one cell shut down; piping for the vapor traps will be fitted with fresh argon purge connections for the displacement of radioactive process gas;

the maintenance sequence for the routine lubrication and inspection of the cold box blowers and motors will involve the use of permanently installed iron shadow shield for the maintenance area.

(4) Solid Radwaste System--All drum movements will be carried out remotely; movement and loading into drums of small contaminated parts will be remotely controlled from the radwaste control room; mixture of radioactive waste with cement will be remotely controlled; provision of two-drum storage vaults will permit emptying one for maintenance.

(5) Closure Head Operations--Access to the head access area will be made possible by shielding and component design; impact of radioactive cover gas on personnel exposure will be limited by hermetically sealing penetrations, and by using double buffered seals and sodium dip seals for the rotating plugs; the rotating plug design configuration will provide steps in the annular gaps through the closure head assembly, thereby reducing dose rates from 100 mrems per hour to less than 1 mrem per hour; all major penetrations will provide a secondary seal system to minimize cover gas leakage into the head access area; periodic leak checks will be performed on all riser seals to ensure seal integrity.

(6) Refueling and Fuel-Handling Systems--Double seals will be provided on all reactor refueling system equipment and facilities which contain radioactive materials; pressurized buffer gas will be provided between the seals to minimize the amount of leakage, to detect whether one of the pair of seals has failed, and to permit replacement of the failed seal before an increase in seal leakage; a siphon will be added to the core component pot so that sodium will not overflow from thermal expansion or sloshing; the core component pot has been designed without protuberances to avoid sodium splashing; drop pans will collect sodium drippage from the core component pot.

(7) Inservice Inspections--Remote viewing capabilities are being developed to permit viewing the primary coolant boundary in the primary heat transport system cells and pipeways, and in the annuli between an intermediate heat exchanger or pump and its respective guard vessel; inservice inspections will normally be scheduled to coincide with refueling periods to allow for decay of the sodium activity; inservice inspection of the reactor vessel/guard vessel will be carried out by remotely operated TV camera.

(8) Control Rod Drive Removal Operations--A 7-day decay time at the storage pits was estimated for control rod drive lines manual disconnect operations; gas purges inside the control rod drive mechanisms will be maintained for as long as several days following plant shutdown.

(9) Maintenance Work on Large Equipment--Continuous access will be provided to the primary pump motors because their cells contain no primary sodium; the intermediate heat exchanger will be accessible for inspection and tube plugging by removal of the circular hatch above the intermediate heat exchanger's (IHX's) upper dome; up to 3 in. of lead will be provided around the working areas adjacent to the IHX; the electromagnetic makeup pumps of the overflow system will be located in individual cells, the valves will be in an adjacent gallery; the power conditioning equipment

will be in a third cell; the control panels will be in a cell having routine access; equipment will be arranged in cells to provide straight pipe lengths to expedite cutting and welding; removable modular insulation with heaters will be provided to minimize maintenance time; the large maintenance stand will be designed with a containment bagging cover to control the atmosphere in the work area and to limit the spread of radioactive materials; bags will protect components introduced into the vessel, minimizing exposure and contamination; equipment used for nonroutine maintenance operations will include shielded casks, shielded floor valves and floor adaptors, portable auxiliary shielding, and a portable control console to operate floor valves and casks remotely; each of the three rooms in the decontamination facility could be isolated from the others; bridge cranes will minimize hands-on contact during disassembly before decontamination; viewing windows will be provided to all enclosed spaces to provide for visual inspection from outside the facility.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of the Applicants

13.1.1 Management and Technical Support Organization

The U.S. Department of Energy (DOE), Project Management Corporation (PMC), and the Tennessee Valley Authority (TVA) are coapplicants for the construction permit for the CRBR. However, DOE has the overall responsibility for managing the design, construction, and operation of the CRBR project. This responsibility has been implemented through the establishment of the Clinch River Breeder Reactor Plant Project Office (CRBRP Project Office).

The CRBRP Project Office, serving in the conventional role as the utility, has established prime contracts with the following organizations: Westinghouse Electric Corporation for the nuclear steam supply system (NSSS), Burns and Roe, Inc., as the architect-engineer, and Stone and Webster as the constructor. In addition, TVA has been contracted to serve as the operator. Westinghouse, acting through its Advanced Reactors Division (ARD), will perform the function of lead reactor manufacturer for the design, manufacture, and test support of the NSSS. Westinghouse in its role as reactor manufacturer has subcontracted with General Electric Company, Energy Systems and Technical Division, and Rockwell International (Atomics International (AI) Division) to provide the design and manufacture of certain systems for the NSSS. TVA, as operator, will be responsible for and will have authority for the testing, operation, and maintenance of the CRBRP.

The organization of the CRBRP Project Office is shown in Figure 13.1. The functions of the divisions within the CRBRP Project Office are:

(1) The Engineering Division is responsible for management of the design, engineering and fabrication of systems, processes, equipment, and facilities, including quality, cost estimates, schedule, and research and development activities.

(2) The Construction Division is responsible for management of the construction of the project, including the quality, timeliness, and cost of the constructed work.

(3) The Operations Division is responsible for overseeing TVA's operation of the plant. Before the operating phase, the Operations Division is responsible for development and implementation of an integrated plant operating program and the orderly transition from the design and construction phases to the operating phases.

(4) The Quality Assurance Division is responsible for planning, developing, and ensuring effective execution of the integrated quality assurance program, including the conduct of the owner program and the integration and coordination of all the quality assurance programs of the project participants.

TVA as plant operator has been assigned responsibility for the conduct of preoperational and startup tests. The normal plant staff will be augmented during the preoperational testing phase by the Preoperative Test Section of TVA's Division of Nuclear Power, and during the startup phase by technical personnel from TVA's Division of Nuclear Power.

The technical support for plant operation will come from two sources: (1) TVA's Division of Nuclear Power, which currently provides technical support to all TVA operating nuclear power plants and (2) the CRBRP Project Office.

The staff has reviewed the above described organizational structure and assignment of responsibilities for the CRBRP and finds them acceptable.

Further details on the subject of technical qualifications for the project are contained in the staff's input to Item (f)(3)(vii) of 10 CFR 50.34.

13.1.2 Operating Organization

The applicants have proposed a plant staff of about 340 persons under the direction of the Plant Manager. Reporting to the Plant Manager will be an Assistant Plant Manager who will assist the Plant Manager in his duties; a Quality Assurance Supervisor; an Engineering Supervisor responsible for analyzing the performance of the reactor and turbine cycle and associated equipment and coordinating the plant staff technical training programs; an Operations Supervisor responsible for the safe operation of the plant in accordance with the operating license, Technical Specifications, and approved procedures; a Mechanical Maintenance Supervisor responsible for all mechanical maintenance work at the plant; an Electrical Maintenance Supervisor responsible for all electrical maintenance work at the facility; an Instrument Maintenance Supervisor responsible for all instrument maintenance at the plant; and a Health Physicist responsible for the plant health physics program.

The proposed shift crew for operation of the CRBRP will consist of six operators, two of whom will be licensed senior operators and two of whom will be licensed operators. In addition, each shift will have a Shift Technical Advisor.

The applicants have proposed that the minimum qualification requirements for plant staff personnel will meet the qualifications described in ANSI/ANS 3.1-1978 and have described the specific requirements for plant positions.

The staff finds these commitments are acceptable, since they meet the qualification requirements for comparable positions described in ANSI/ANS 3.1-1978 and Revision 1 to RG 1.8.

The staff concludes that the applicants' proposed operating organization is acceptable and meets the relevant requirements of 10 CFR 50.40b and 50.54j through m. This conclusion is based on the following: The applicants have described the assignment of plant operating responsibilities, the reporting chain up through the Plant Manager, the proposed size of the regular plant staff, the functions and responsibilities of each major plant staff group, the proposed shift crew complement for unit operation, and the proposed qualification requirements for members of their plant staff. The staff has reviewed this information and concludes that the proposed organization is acceptable.

13.2 Training Program

13.2.1 Licensed Operator Training Program

The overall conduct and administration of the plant training program will be the responsibility of the Plant Manager. The Training Coordinator, reporting to the Plant Manager, will be responsible for development, implementation, and documentation of the training program. This ensures that the content and the intent of the training program will provide the necessary training for personnel associated with reactor operations. The program will be designed to train personnel both with and without previous nuclear experience.

The applicants state that a training program will be established to provide plant personnel with sufficient knowledge and operating experience to start up, operate, and maintain the plant in a safe and efficient manner. When it is time to man the CRBRP, TVA should have highly trained nuclear plant operating personnel at the Browns Ferry, Sequoyah, and Watts Bar nuclear plants. These plants will be the primary source of personnel for CRBRP. Training for the personnel to be licensed will meet the guidance of RG 1.8 and will include basic nuclear training; practical training at TVA's Browns Ferry, Sequoyah, and Watts Bar facilities and at an operating sodium-cooled fast reactor to provide the experience required for applicants for cold licenses; a plant system lecture series; simulator training; and practical on-the-job training. Simulators used for training the personnel to be licensed should meet the guidance of RG 1.149.

The information submitted relative to these subjects is satisfactory at the construction permit stage of review only. For the Final Safety Analysis Report, the applicants will have to submit a more detailed training program for staff review. The applicants plan to provide a control room simulator for training purposes that models the control room and plant response. This is consistent with TMI Action Plan Item I.A.4.2.

13.2.2 Training for Nonlicensed Plant Staff

The applicants' plan for the training of the nonlicensed plant staff is included in Section 13.2 of the PSAR.

The program that will be established will meet that described in Revision 1 to RG 1.8. The non-NRC-licensed personnel will undergo training both before and after assignment to the CRBRP. The training program in which these employees will participate before being assigned to the CRBRP will include formal programs at TVA's Power Operations Training Center (POTC), TVA Administrative and Standards of Apprenticeship, or approved equivalents. After assignment to CRBRP the training will be commensurate with the individual's experience and job assignment and will follow the guidelines of Item I.A.2.2 of NUREG-0660 (1980). The individual tasks and the various components of each position will be examined to identify the knowledge, skill, physical ability, duties, and responsibilities required for the individual to accomplish his or her job. The actual training program and schedule for the plant operating staff will be designed in relation to the findings of a job/task analysis.

The applicants have committed to meet the requirements and guidelines of TMI Action Plan Item I.A.2.1, as it relates to training for mitigating core damage, and TMI Action Plan Item I.A.1.1, as it relates to training of Shift Technical Advisors.

A fire protection training program will be developed before fuel loading, will meet the requirements and guidelines of BTP CMEB 9.5-1-R3-1981 as they apply to CRBRP, and will include details such as interaction requirements between the plant operator and the plant constructor. The fire protection program of the constructor will be the basis for fire-protection training for the construction force. When there is interaction between construction and operations personnel in areas under control of operations, the involved construction personnel will be oriented in the applicable portion of the operator's fire protection program.

The staff concludes that the training for nonlicensed plant staff personnel is acceptable and meets the requirements of 10 CFR 19.12, 10 CFR 50.34a and b, and 10 CFR 50.40(b). This conclusion is based on the following:

The applicants have described in the PSAR, in accordance with the requirements of 10 CFR 50.34(a)(6), a preliminary plan for training of nonlicensed plant personnel. All nonlicensed plant personnel will be trained in accordance with the requirements and guidance of ANSI/ANS 3.1 as endorsed by RG 1.8. In addition, specialized training will be given to fire protection personnel, to the shift technical advisors, and to senior plant operating personnel for mitigating core damage.

This preliminary plan and commitment provides an acceptable basis for finding that, insofar as the plans for training of nonlicensed personnel are concerned, the applicants meet the technical qualification requirements of 10 CFR 50.40(b).

13.3 Emergency Planning

The plume exposure pathway emergency planning zone (EPZ) established for the CRBRP site is about 10 mi in radius. This 10-mi radius encompasses portions of five counties: Roane, Anderson, Morgan, Loudon, and Knox. The 10-mi EPZ is shown in Figure 13.2. The ingestion pathway EPZ is an area of about 50 mi in radius and encompasses east-central Tennessee and a small portion of western North Carolina. The location of the CRBRP site in relation to counties and States is shown in Figure 13.3.

13.3.1 Introduction

The proposed CRBRP site is located in Roane County in east-central Tennessee approximately 25 mi west of Knoxville. The site is bounded on the east, south, and west by the Clinch River and on the north by DOE's Oak Ridge Reservation.

10 CFR 50.34(a)(10) requires each applicant for a construction permit (CP) to include in the Preliminary Safety Analysis Report (PSAR) a discussion of preliminary plans for coping with emergencies. The PSAR must contain sufficient information to ensure that the proposed emergency plans for both the site and the EPZs are compatible with the facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, land use, and local jurisdictional boundaries for the EPZs. The emergency planning requirements at the CP stage are set forth in Part II of Appendix E to 10 CFR 50.

In accordance with 10 CFR 50.34(a)(10), the applicants filed with the NRC a description of the preliminary plans for coping with emergencies at the CRBRP.

The preliminary Clinch River Breeder Reactor Plant Radiological Emergency Plan (CRBRP-REP, plan), described in Section 13.3 of Amendment 65 to the PSAR, was reviewed against the requirements of 10 CFR 50, Appendix E, Part II. On the basis of this review, the staff issued comments and questions to the applicants on May 14, 1982 requesting additional information regarding the items required by Part II to Appendix E. On July 28, 1982, the applicants submitted their responses to the staff's request for additional information, and in a letter dated August 20, 1982, the applicants incorporated their responses into Amendment 70 of Section 13.3 of the PSAR, dated August 1982. The staff has reviewed Amendment 70 to Section 13.3 of the PSAR against the requirements of 10 CFR 50, Appendix E, Part II, including the requirement for information to indicate the means by which the standards of 10 CFR 50.47(b) will be met. The individual requirements of Appendix E, Part II, A through H, are discussed below.

13.3.2 Requirements of Appendix E, Part II

13.3.2.1 Requirements A and B

Requirement A

Describe the "onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations."

Requirement B

Describe the "contacts and arrangements made and documented with local, state and Federal government agencies with responsibility for coping with emergencies, including identification of the principal agencies."

Discussion--Requirements A and B

The CRBRP emergency organization will initially consist of the shift operating crew and will be augmented by the site emergency organization, the staffs of four TVA emergency centers--Central Emergency Control Center (CECC), Division of Nuclear Power Emergency Center (DNPEC), Muscle Shoals Emergency Control Center (MSECC), and the Knoxville Emergency Control Center (KECC)--and other TVA organizations as may be required. The site emergency organization will augment the shift operating crew in accordance with Table B-1 of NUREG-0654, Rev. 1.

Section 13.3.3 of the plan commits to the establishment of agreements with Federal agencies and the State of Tennessee, Tennessee Emergency Management Agency (TEMA). However, the plan does not address the agreements to be established with local support agencies other than medical. The plan states that TEMA will notify the State of North Carolina and surrounding States and coordinate assistance from the various State agencies. The plan identifies the principal local, State, and Federal agencies with responsibility for coping with emergencies at CRBRP, with the exception of the principal agency in North Carolina responsible for coping with emergencies. The plan will use the liaisons already established in developing the Browns Ferry, Sequoyah, Watts Bar, and Bellefonte Radiological Emergency Plans with the States of Alabama and Tennessee. TVA will maintain liaison with TEMA, particularly with respect to

the availability of emergency services provided by State agencies. TEMA will inform these agencies of actions to be taken under their respective statutory authority and assist them in developing emergency procedures. Figure 13.4 illustrates the relationship between the TVA emergency control centers and depicts the interface among TVA, Federal, State and local agencies. The local, State, and Federal agencies that will be participating in the CRBRP-REP are:

(1) Tennessee Emergency Management Agency

(2) Tennessee Department of Public Health

(3) Tennessee Department of Agriculture

(4) Tennessee Department of Public Welfare

(5) Tennessee Department of Safety

(6) Tennessee Department of Conservation

(7) Tennessee National Guard

(8) Tennessee Game and Fish Commission

(9) Tennessee Department of Transportation

(10) city and county officials of Roane, Anderson, Loudon, Morgan, and Knox Counties

(11) Sheriff's Department of Roane, Anderson, Doudon, Morgan, and Knox Counties

(12) civil defense coordinators of Roane, Anderson, Loudon, Morgan, and Knox Counties

(13) local police departments

(14) local ambulance service

(15) local fire department

(16) Radiological Emergency Assistance Center Training Site

(17) Oak Ridge Hospital of the United Methodist Church

(18) Department of Energy

(19) National Aeronautics and Space Administration

(20) U.S. Nuclear Regulatory Commission

(21) U.S. Environmental Protection Agency

(22) Federal Emergency Management Agency

Conclusion--Requirements A and B

Additional information is required as follows:

(1) Describe the arrangements to be made with local civil, law enforcement, and fire-fighting agencies identified above.

(2) Identify the principal agency in North Carolina responsible for coping with emergencies.

With the exception of the informational items described above, the plan adequately describes the means for notifying the site emergency organization and offsite support groups.

13.3.2.2 Requirement C

Describe "protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (for example, in the case of an evacuation, who authorizes an evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency."

Discussion

TVA will use the following emergency classifications: notification of unusual event, alert, site emergency, and general emergency. This system of classification will be consistent with the nomenclature used by State and local emergency organizations. A notification of unusual event will provide early and prompt notification of minor events that could develop into or be indicative of more serious conditions that are not yet fully realized. An alert will be declared when events are in progress or have occurred that involve an actual or potential substantial degradation of the level of safety of the plant. A site emergency will be declared when events are in progress or have occurred that involve actual or likely major failures of plant functions needed for protection of the public. A general emergency will be declared when events are in progress or have occurred that involve actual or imminent substantial core failure with a potential for loss of containment integrity. Recognition of the emergency class that may result from many initiating conditions will be primarily a matter of judgment on the part of plant operating personnel. The initiating conditions used for recognizing and declaring the emergency class will be based on specific measured values or observable conditions defined as emergency action levels (EALs). EALs relate to particular in-plant conditions, instrument readings, and onsite and offsite monitoring results that provide the basis for categorizing the event into one of the four emergency classes.

Onsite protective measures will include exposure control, contamination control, and building and/or area and site evacuation. It will be the responsibility of the Site Emergency Director to make the decision concerning the necessity for evacuation.

MSECC will assess the actual or potential offsite environmental conditions (see Requirement H of this section). State and local agencies will be responsible for implementing actions to protect the health and safety of the public. TVA

recommends protective actions to these agencies, but the State and local governments will be responsible for deciding if any actions are needed and what they should be. TVA will assist State and local governments as necessary to implement protective actions for the public. Specific actions required of offsite TVA support groups will be delineated in the final CRBRP-REP. The principal protective measures to be considered within the 10-mi EPZ will be providing shelter and evacuation. The preliminary evacuation analysis is discussed in Section 13.3.2.5.

TVA will provide a prompt notification system for State and local governments to alert the public within the 10-mi EPZ that protective actions may be required.

Conclusion

On the basis of its review of the information supplied by the applicants, the staff concludes that the requirements of Appendix E, Part II, Item C, have been satisfied.

13.3.2.3 Requirements D and E

Requirement D

Describe "features of the facility to be provided for onsite emergency first aid and decontamination, and for emergency transportation of onsite individuals to offsite treatment facilities."

Requirement E

Describe "provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed actions."

Discussion--Requirements D and E

The final CRBRP-REP will include a description of medical facilities at the plant and arrangements made with other facilities to provide additional medical support. Figure 13.3-2 of the plan depicts the layout of the medical complex (first aid) and the personnel decontamination facility in relation to the proposed technical support center and operations support center. Medical consultation will be available from TVA physicians in Chattanooga and other areas. Members of the plant emergency teams will be trained in first aid.

Arrangements will be made with a local private ambulance service to provide emergency service as required to the plant and affected areas in the event that medical transportation is required. Agreements also will be culminated between TVA and a nearby hospital to provide emergency treatment to irradiated or contaminated patients as required. An agreement also will be made with the Radiation Emergency Assistance Center and Training Site Facility operated by the Oak Ridge Associated Universities for emergency treatment of severely contaminated or irradiated personnel.

Conclusion--Requirements D and E

On the basis of its review of the information supplied by the applicants, the staff concludes that the requirements of Appendix E, Part II, Items D and E have been satisfied.

13.3.2.4 Requirement F

Describe the "provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency."

Discussion

TVA will provide general REP training to all plant personnel and specific training to emergency response personnel. Training in specific duties and emergency responsibilities will be provided personnel, such as the Site Emergency Director, as necessary. Training and periodic retraining will be provided to those offsite agencies that may be involved during an emergency. TVA's Division of Occupational Health and Safety will provide training for local offsite response personnel. This training will include notification procedures, basic radiation protection procedures, expected roles of the personnel, and site access procedures.

Conclusion

On the basis of its review of the information supplied by the applicants, the staff concludes that the requirements of Appendix E, Part II, Item F, have been satisfied.

13.3.2.5 Requirement G

Describe "a preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. A nuclear power plant applicant shall perform a preliminary analysis of the time required to evacuate various sectors and distances, noting major impediments to the evacuation or taking of protective actions."

Discussion

TVA will have direct communication links, such as commercial telephone and radio to notify Federal, State and local officials. Primary and backup means of communication are identified in the plan. On the basis of TVA's experience, initial dose projection and notification of offsite authorities can be accomplished within 15 min. The time required to warn all resident and transient persons in any evacuation sector will conform to the requirements of 10 CFR 50, Appendix E. As noted in Section 13.3.2.2, TVA will provide a prompt notification system for State and local governments to alert the public within the 10-mi EPZ that protective actions may be required.

Preliminary evacuation time estimates for the 10-mi EPZ were prepared for the CRBRP site by the Traffic Management Division of the Tennessee Department of Transportation. The estimated evacuation times found in PSAR Section 13.3 range from 4 hours for the 2-mi EPZ to 7 hours 15 minutes for the 10-mi EPZ. Each estimate contains a preparation time factor of 1 hour 50 minutes. The evacuation time estimates contain the current (1980 Census) permanent and transient population distribution, by sector, within 10 mi and the projected

permanent and transient population distribution within 10 mi for the year 2020. Private automobiles will be the primary means of evacuating the population. Buses may be used to evacuate the Edgewood School. The evacuation of the population will be addressed specifically in the final CRBRP-REP. No constraints have been identified that would prevent meeting the evacuation times described above.

On May 14, 1982, the staff asked the applicants to include in the plans for the evacuation/sheltering of the public, consideration of workers at the K-25 complex, a DOE-Oak Ridge facility within the 10-mi EPZ. The applicants provided a response for the CP review and the response committed to providing this additional information in the Tennessee State Plan at the FSAR stage. Since the Tennessee State Plan will be incorporated into the final CRBRP-REP as an appendix, the staff considers the applicants' response acceptable.

Conclusion

On the basis of its review of the information supplied by the applicants, the staff concludes that the requirements of Appendix E, Part II, Item G, have been satisfied.

13.3.2.6 Requirement H

Describe "a preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the onsite technical support center and the near-site emergency operations facility in assessing information, recommending protective action, and disseminating information to the public."

Discussion

A radiological analysis of the facility design features, site layout, and site location with respect to the consideration of surroundings in compliance with 10 CFR 50, Appendix E, has been conducted. The analysis incorporated meteorology, containment modeling, and radiological parameters to effect a series of dose projections based on exposures resulting from the site suitability source term (SSST). The use of the SSST is conservative, since it envelopes the most serious design-basis accident analyzed in the PSAR. The findings of the analysis are included in Appendix 13.3A of the PSAR. The applicants will establish a standard emergency classification and action level scheme based on specific instrument readings and plant parameters.

Emergency facilities needed to support an emergency response will be provided including a technical support center (TSC) and an operations support center (OSC). The TSC will be provided in accordance with the requirements of 10 CFR 50, Appendix E, and the guidance contained in Supplement 1 to NUREG-0737. The OSC will provide a location where support personnel will be able to assemble and where support will be coordinated. It will provide communication with the control room and the TSC. The proposed locations of the TSC and OSC are shown in Figure 13.3-2 of the PSAR.

Under correspondence dated December 22, 1982, the applicants provided the additional emergency personnel during emergencies, data transmission between the various TVA emergency centers, control room location with respect to the TSC, and a description of the TSC, including habitability features. The applicants committed to incorporate this information into Amendment 75 of the PSAR.

TVA has established its Central Emergency Control Center (CECC) in Chattanooga, Tennessee. The stated purpose of the CECC and its staff is to provide the facilities and manpower for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency at any TVA reactor. The CECC will function as and provide all the services of an emergency operations facility (EOF).

In addition to the CECC, the following TVA emergency response centers are provided:

(1) the Division of Nuclear Power Emergency Center (DNPEC), located immediately adjacent to the CECC, which provides organizational, material, personnel and technical support from nuclear maintenance, reactor engineering, controls and test, and outage boundaries

(2) the Knoxville Emergency Control Center, which provides the technical support for the Division of Engineering personnel and the personnel and equipment support for the Division of Construction

(3) the Muscle Shoals Emergency Control Center, which provides staff and equipment for performing environmental radiological monitoring and dose assessments and for recommending protective actions for the public

The plan has also identified facilities and staff at Oak Ridge, the George C. Marshall Space Flight Center, and Tennessee State and local police organizations--all available for assistance in an emergency.

The NRC has received and approved the TVA centralized emergency management organization concept.

Conclusion

On the basis of its review of the information supplied by the applicants, the staff concludes that the requirements of Appendix E, Part II, Item H, have been satisfied, except that the capability for real time meteorology and dose projection should be clarified by the applicants in the CRBRP-REP.

13.3.3 Plans for Coping With Sodium Fires

The CRBRP is a liquid metal fast breeder reactor using sodium (Na) and sodium-potassium (NaK) and, as such, may present unique fire hazards to the plant, personnel, and surroundings. Although 10 CFR 50 does not specifically include a requirement to describe the plans for coping with nonradiological hazards such as sodium fires, the staff believes that the plans should be designed so that they have some flexibility and can provide some protective measures in the event of this type of fire emergency.

The objective is not to develop new requirements to ensure that plans will function normally during these emergencies. Rather, it is to describe techniques for examining the flexibility of the plans so the staff can evaluate the extent to which plans provide some capability for coping with the toxic, nonradiological aspects of Na/NaK fires.

Current NRC guidance specifies that final emergency plans include the capability to identify and classify fires that potentially affect or compromise safety systems. The final plans should describe observable and reliable indicators in the control room to alert station operators. Fire alarms do not provide sufficient information. A prefire plan is an effective tool that serves the same function as an EAL. Upon an alarm, the control room operator would use the prefire plan to determine the degree of hazard associated with the Na/NaK fire, and which situations require offsite support by the local fire department(s).

The applicants should incorporate this prefire strategy into their final CRBRP-REP or specify an alternate means to provide this capability.

In addition, the staff will request that the Federal Emergency Management Agency (FEMA), in its review and evaluation of offsite emergency plans, take into account the impact off site of the toxic, nonradiological hazards associated with sodium-fire emergencies. This will be accomplished at the FSAR stage.

13.3.4 Federal Emergency Management Agency Evaluation of State and Local Plans

FEMA has completed its review of the Tennessee State Radiological Emergency Plan in support of other nuclear power facilities within the State. The local plans for coping with emergencies at Clinch River are in an incomplete stage at this time. FEMA will be asked to provide its findings on the adequacy of State and local plans at the operating license stage. Formal FEMA findings are not required for this review, and FEMA's participation in hearings at the construction permit stage is not contemplated.

13.3.5 Conclusion

On the basis of its review of the applicants' preliminary plans for coping with emergencies, the staff concludes that, provided the items identified in this evaluation as requiring resolution are corrected and the items for which the applicants have made commitments are accomplished, the preliminary plans are acceptable and meet the requirements of 10 CFR 50, Appendix E, Part II. They provide reasonable assurance that the final emergency plans will be compatible with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population, land use, and local jurisdictional boundaries for the EPZs. The plans are also in conformance with TMI Action Plan Item III.A.1.2.

For an operating license to be granted, the onsite and offsite emergency response plans for nuclear power reactors must meet certain standards set forth in the regulations. 10 CFR 50.47(b) lists 16 planning standards that must be met in the emergency response planning for a nuclear power reactor. Specific criteria for these standards are contained in NUREG-0654, which has been endorsed by RG 1.101 (Rev. 2). The staff has reviewed the information in the CRBRP PSAR on emergency planning and concludes that the information presented is sufficient in depth and scope for the construction permit stage to indicate

that the planning standards will be met in the final emergency plan. Further, no special or unique circumstances have been identified that would preclude the development of adequate preparedness plans at the operating license stage of review.

13.4 Operational Review

NUREG-0800 does not require any information for the PSAR.

13.5 Plant Procedures

13.5.1 Administrative Procedures

NUREG-0800 does not require any information for the PSAR.

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Background

A review has been conducted of the applicants' description of their plan for development and implementation of operating and maintenance procedures. The review was conducted to determine the adequacy of the applicants' plan for ensuring that routine operating, off-normal, and emergency activities will be conducted in a safe manner. The following description and evaluation are based on information contained in the applicants' PSAR.

In determining the acceptability of the applicants' program plan, the criteria used were those of SRP Section 13.5.2, "Operating and Maintenance Procedures" (NUREG-0800) and TMI Action Plan Item I.C.9, addressing the upgrading of procedures, as described in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." The review consists of an evaluation of (1) the applicants' schedule for completion of operating and maintenance procedures; (2) the applicants' plan for conforming with the guidance contained in RG 1.33, Revision 2, February 1978, regarding the minimum procedural requirements for safety-related operations; (3) the plan for compliance with the guidance contained in ANSI N18.7-1976/ANS 3.2; and (4) the applicants' program for conformance with TMI Action Plan Item I.C.9.

The criterion of Item I.C.9 is that the applicants describe, in the PSAR, their program plan, which is to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management, operator training, and coordination with the efforts of the Institute for Nuclear Power Operations (INPO) and other industry groups. The applicants shall submit, before the issuance of a construction permit, a general discussion of how these requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly before an operating license is issued.

13.5.2.2 Operating and Maintenance Procedure Program

The applicants have committed in a letter from John R. Longenecker to Paul S. Check dated June 25, 1982, to amend the PSAR to specify that all operator activities will be conducted in accordance with detailed written and approved

procedures meeting the regulatory positions of RG 1.33, Revision 2, February 1978, "Quality Assurance Program Requirements (Operation)," and ANSI N18.7-1976/ ANS 3.2. The letter also states that the procedures will be completed and approved in time for the operators to be trained on the procedures before fuel load.

13.5.2.3 Long-Term Plan for Upgrading of Procedures

The applicants' response to NUREG-0718 is contained in Appendix H of the PSAR. In a letter dated December 28, 1982, from John R. Longenecker to Paul S. Check, the applicants committed to change the PSAR to include a discussion of how the CRBRP Probabilistic Risk Assessment (PRA) Program discussed in Appendix J of the PSAR will support the long-term plan for upgrading procedures. The PSAR, in Appendix H, states that procedures will be prepared from procedure outlines developed by the designers and walked through the control room by a multidisciplinary review team with human factors engineering as a major contributor to the review and that TVA will maintain its awareness of the state of the art in procedure development and upgrade through its membership in INPO. Appendix H also states that in addition to using the PRA Program results as the basis of emergency operating procedures, the applicants will also evaluate plant safety functions to ensure that adequate procedures are developed for events not addressed by the PRA effort. As described in Appendix J, the PRA effort will also (1) define the operator's role in managing an accident, (2) assist in the validation of emergency procedure guidelines, and (3) provide information on the integrated plant performance for operator training. A continuing risk management program is also described in Appendix J that will keep the PRA current with respect to the plant design and operation, and will allow application of its results throughout the life of the plant.

13.5.2.4 Conclusions

On the basis of commitments discussed above, the staff concludes that the program plan for operating and maintenance procedures meets the criteria of the Standard Review Plan and TMI Action Plan Item I.C.9.

13.6 TMI-Related Items

13.6.1 Item I.C.5, Title 10 CFR Part 50.34(f)(3)(i), Procedures for Feedback of Operating, Design, and Construction Experience

This item corresponds to Item I.C.5 of NUREG-0718, "Licensing Reqirements for Pending Applications for Construction Permits and Manufacturing License."

Requirement

Applicants shall "provide administrative procedures for evaluating operating, design, and construction experience and describe how they will assure that applicable important industry experience originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant."

Position

Applicants shall submit a general discussion of how the requirements will be met. These procedures shall (1) clearly identify organizational responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) identify the administrative and technical review steps necessary in implementing applicable important experience; (3) identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; (4) ensure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) provide suitable checks to ensure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) provide practical interim audits to ensure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license.

Discussion

The CRBRP Project Office has the overall responsibility for establishing, implementing, and executing a program on the CRBRP for the feedback of operating, design, and construction experience. The Assistant Director for Engineering is responsible for ensuring that the experience from the fast flux test facility (FFTF) and industry is fed back into the project. The Assistant Director for Public Safety is responsible for ensuring that information in NRC bulletins, circulars, and information notices is fed back into the project.

The CRBRP Project Office and principal contractors have established a procedure to evaluate sources of industry experience. Sources used for feedback include experience at light-water reactors described in NRC's Office of Inspection and Enforcement bulletins, circulars, and information notices; liquid metal fast breeder reactor experience from the FFTF; and each unusual or unplanned event (unusual occurrence) in the CRBRP project having programmatic significance. The applicants have agreed to establish an interface with the Institute of Nuclear Power Operations to pursue the most effective method of considering licensee event reports in the feedback program. The procedure includes provisions for ensuring that extraneous material does not overwhelm personnel; this is accomplished by (1) initial screening to determine appropriate applicability, (2) further assessment by the responsible line organizations that the item is applicable before assigning it for action, (3) ensuring that conflicting contradictory information is not conveyed to personnel assigned action or (accomplished by multiple levels of screening), and (4) auditing regularly for compliance and effectiveness in accordance with established project requirements.

Conclusion

The staff has reviewed the above described program in regard to the assignment of responsibility; the provisions for the review and feedback of design, con-

struction, and operating experience into the design and construction of the CRBRP; and the provisions ensuring the implementation of the feedback program. The staff finds that this program meets the basis of acceptance described above for the feedback of important design, construction, and operating experience into the CRBRP design and construction and is acceptable.

13.6.2 Item II.J.3.1, Title 10 Part 50.34(f)(3)(vii), Management Plan for Design and Construction Activities

This item corresponds to Item II.J.3.1 of NUREG-0718.

Requirement

Applicants shall provide a description of the management plan for the design and construction activities to include: "A. the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, B. technical resources which are directed by the applicant, C. details of the interaction of design and construction within the applicant's organization and the manner by which the utility will ensure close integration of the architect engineer and the nuclear steam supply system vendor, D. proposed procedures of handling the transition to operation, E. the degree of top level management oversight and technical control to be exercised by the utility applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort."

Discussion

The applicants should establish an organizational structure that will be singularly responsible for the overall management and technical support for the project. Key characteristics of such an organization should be:

(1) All necessary responsibilities should be integrated under a single responsible upper-level executive position.

(2) The individual in charge of all nuclear functions should be a senior-level executive in the corporation.

(3) The lines of authority to the upper-level executive position should be clear, and communications channels should be delineated.

The organization under the upper-level executive should have the following duties and responsibilities for the project:

(1) Review and approve the design criteria of the nuclear steam supply system vendor and architect-engineer.

(2) Review and approve principal plant design features.

(3) Review and approve safety analysis reports.

(4) Coordinate licensing activities with the NRC.

(5) Review and audit job management of the principal contractor.

(6) Develop and implement a quality assurance program covering the utility and contractors, including engineering design review.

(7) Receive and report operation, maintenance, and inspection experience from industry.

The U.S. Department of Energy (DOE), Project Management Corporation (PMC), and the Tennessee Valley Authority (TVA) are coapplicants for the construction permit for CRBR. However, DOE has the overall responsibility for managing the design, construction, and operation of the CRBRP project. The project has been assigned to the Oak Ridge Operations Office (ORO) for implementation. The organizational unit of the ORO that has been established to execute the overall responsibility and authority for the management of the project is the CRBRP Project Office located in Oak Ridge, Tennessee. That office is under the CRBRP Project Director. The CRBRP Project Office organization is shown in Figure 13.1.

Reporting to the Director will be an Assistant Director for Engineering who will be responsible for the design of the plant and for overseeing and directing the performance of the contractors responsible for plant design; an Assistant Director of Construction responsible for overseeing and directing the performance of the contractor responsible for the construction of the plant; an Assistant Director of Operations who will be responsible for overseeing the operation of the plant; and a Chief of Quality Assurance who will be responsible for planning, developing, and ensuring execution of the quality assurance (QA) program. The current staffing of the CRBRP Project Office is about 154 individuals, of which about 34 are employees of DOE. It is expected that at its peak, the CRBRP Project Office will have about 185 employees. These numbers do not include any personnel that TVA will employ to operate the CRBRP.

To implement its overall responsibility for the project, the CRBRP Project Office will

(1) define design criteria
(2) review and approve plant design description
(3) review and approve system design criteria and specifications
(4) review and approve system-to-system interfaces
(5) review and approve the Preliminary Safety Analysis Report
(6) review, approve, and audit principal contractor QA programs
(7) manage and direct the construction contractor

The CRBRP Project Office will be responsible for overseeing the principal contractors (Westinghouse, Burns and Roe, Stone and Webster). Westinghouse, through its Advanced Reactors Division (ARD), has been contracted as the lead reactor manufacturer (LRM) for designing and supplying the entire nuclear island. Westinghouse has subcontracted with General Electric Company Energy Systems and Technology Division and Rockwell International (Atomics International Division) to provide the design and manufacture of certain systems for the nuclear steam supply system. Additionally, Westinghouse has the responsibility for the design and procurement of the primary reactor and has established an organization under the CRBRP Project Manager to implement this responsibility. Therefore, a wide range of technical resources are available for the project.

The lead responsibility in the CRBRP Project Office for the management and direction of the project contractors in the development of the plant design and the

manufacture of the plant components has been assigned to the Assistant Director for Engineering. The Project Office Engineering Division functions according to defined procedures in coordinating the Project Office action on all design decisions. The Engineering Division leads the efforts of the Project Office in defining the design criteria, the design planning requirements, the development requirements, the design control requirements, the interface control activities that are to be accomplished, and the design data that are required (e.g., the construction specifications and the technical requirements for operation, maintenance, and testing). To ensure close integration of all design efforts, the Project Office has required that the plant design be developed in sequential steps of conceptual, preliminary, and final design. At the completion of each step, design organizations are required to review and concur in the interfacing efforts of other design organizations. Multidisciplined design reviews are conducted by the designers and observed or participated in by the Project Office. Proposed changes to the design are described in an engineering change proposal, which is submitted for review and approval by the involved project participants. Major changes are reviewed and concurred in by the architect-engineer, the reactor manufacturers, the lead reactor manufacturer, and if constructibility is involved, by the constructor, and approved by the Project Office. The CRBRP designers have defined the system-to-system interfaces, which may consist of physical constraints, process requirements, service demands, and so forth. These are documented in interface control documents, and any subsequent changes are reviewed and approved by all affected participants. A computerized system is used to track and ensure definition of all system interface requirements.

The CRBRP Project Office has established an Operations Division with responsibility for the overall management, coordination, and direction of the operator (TVA) in the testing, operation, and maintenance of the plant. A contract between DOE and TVA establishes TVA's responsibilities for testing, startup, operation, maintenance, and demonstration activities of the project. To ensure continuity from design/construction to operation and to ensure proficiency of operator personnel, TVA will begin assigning operation personnel to the site about 5 years before criticality. From the time of initial assignment on site until start of preoperational testing 3 years later, the staff size will increase to about 170. Principal activities during this period will consist of observation training at an operating liquid metal fast breeder reactor, CRBR-specific plant and system training, observation of the installation of equipment and systems, and preparation of detailed plant procedures, including administrative, operating, maintenance, surveillance test, and preoperational test procedures. From the time preoperational tests are started, about 2 years before criticality, until fuel loading, the staff size will increase to about 380. Principal activities during this period will include completion of the writing of detailed procedures; acceptance of completed systems from the constructor; preoperational testing; operation, maintenance, and surveillance testing of turned-over systems; continuation of training, including simulator training; responsibility for plant security; and preparation for, and obtaining of, NRC licenses for the operators. From fuel loading until about 2 years later, the staff size will decrease from 380 to about 340. Principal duties during this period will include startup testing, plant operation, maintenance, surveillance testing, and modification.

The CRBRP Project Office will continue to perform its overall project management responsibilities throughout this phase-to-phase transition. As the plant is completed and transfer of custody to the operator is accomplished, the

resources of the Project Office will be available to support the Project Office Operations Division in the management and direction of the operating program. The resources of the project contractors and other participating organizations in the design and construction of the plant will be available for any necessary support of the operator with regard to plant testing, operation, maintenance, and modification.

The staff has reviewed the applicants' organization with respect to its structure, technical resources available to work on the CRBRP, and the qualification of key personnel in the design and construction of the CRBRP, as described in the PSAR. To better understand how the organization functions and to gain a feeling for the responsibilities and attitudes of individuals, several members of the staff met with key individuals of Burns and Roe, CRBRP, and Westinghouse in Bethesda, Maryland, on February 18, 1982, March 4, 1982, and June 3, 1982.

Conclusion

On the basis of NRC review of the applicants' organizational structure, the qualifications of key personnel, their technical resources, and the degree of top-level management supervision and technical control, the staff concludes that the organization and staffing of the CRBRP Project Office meet the requirements of Item II.J.3.1 and that the CRBRP Project Office has the resources and management capability to oversee the design and construction of the facility.

13.7 Radiological Security

13.7.1 Area of Review

The review of the PSAR covered the applicants' plans for implementing security measures for protection against sabotage of the facility, theft of special nuclear material, and storage of classified information. The scope of the review covered both transportation and fixed-site protection in accordance with the provisions of 10 CFR 11; 10 CFR 50.34(c), 50.34(d), and 50.34(e); 10 CFR 70.20(a); 10 CFR 73.20, 73.21, 73.25, 73.45, 73.46, 73.55, 73.71, 73.72, and 73.80 and Appendices B and C of 10 CFR 73; 10 CFR 95; and regulatory guides and equivalent guidance documents.

13.7.2 Acceptance Criteria and Basis

The applicants' PSAR must provide sufficient information to indicate that their preliminary planning for security provides reasonable assurance that conformance to the applicable regulations is expected to be achieved. Commitments to each major portion of the regulation and associated guidance documents are also required in the PSAR. The preliminary planning should include a commitment to a design-phase review for physical security and should show how this responsibility is to be implemented by the applicants.

The bases for acceptance are the commitments made by the applicants to conform to the regulations and guidance documents identified in Section 13.7.1 above.

13.7.3 Review Evaluation

The information submitted by the applicants was reviewed against the acceptance criteria. As a result of the staff's initial evaluation, certain portions of

the applicants' PSAR were identified as requiring changes in order to meet a level of protection consistent with that identified in the acceptance criteria. The applicants' subsequent revision to the PSAR resulted in a document that sets forth design features of the plant and outlines in some detail those main topics to be addressed in the security plan.

The PSAR addresses major portions of the applicable regulations beyond what is necessary at this early stage and provides significant assurance that conformance to the applicable regulations is expected to be achieved. The PSAR contains the necessary provisions to protect the facility against radiological sabotage and to prevent the theft of special nuclear material. In addition, it contains particulars for the control and accounting of special nuclear material and provisions for a design-phase review.

13.7.4 Evaluation Summary

The applicants have provided a general description of the plans for protection of the plant against potential acts of radiological sabotage and for prevention of the theft of special nuclear material. The description addresses security measures relating to the personnel employment program and screening procedures, general plant design layout and equipment arrangements, transportation and accountability of material, and protection of classified information. The staff concludes that, on the basis of the program outline contained in the PSAR, there is reasonable assurance that the final physical security plan will meet the requirements of 10 CFR 11, 25, 50, 70, 73, and 95 and conform to the associated regulatory guides and equivalent guidance documents. At this stage of the licensing process, the depth of preliminary planning for protection of the plant is greater than that required for assurance of an adequate security program.

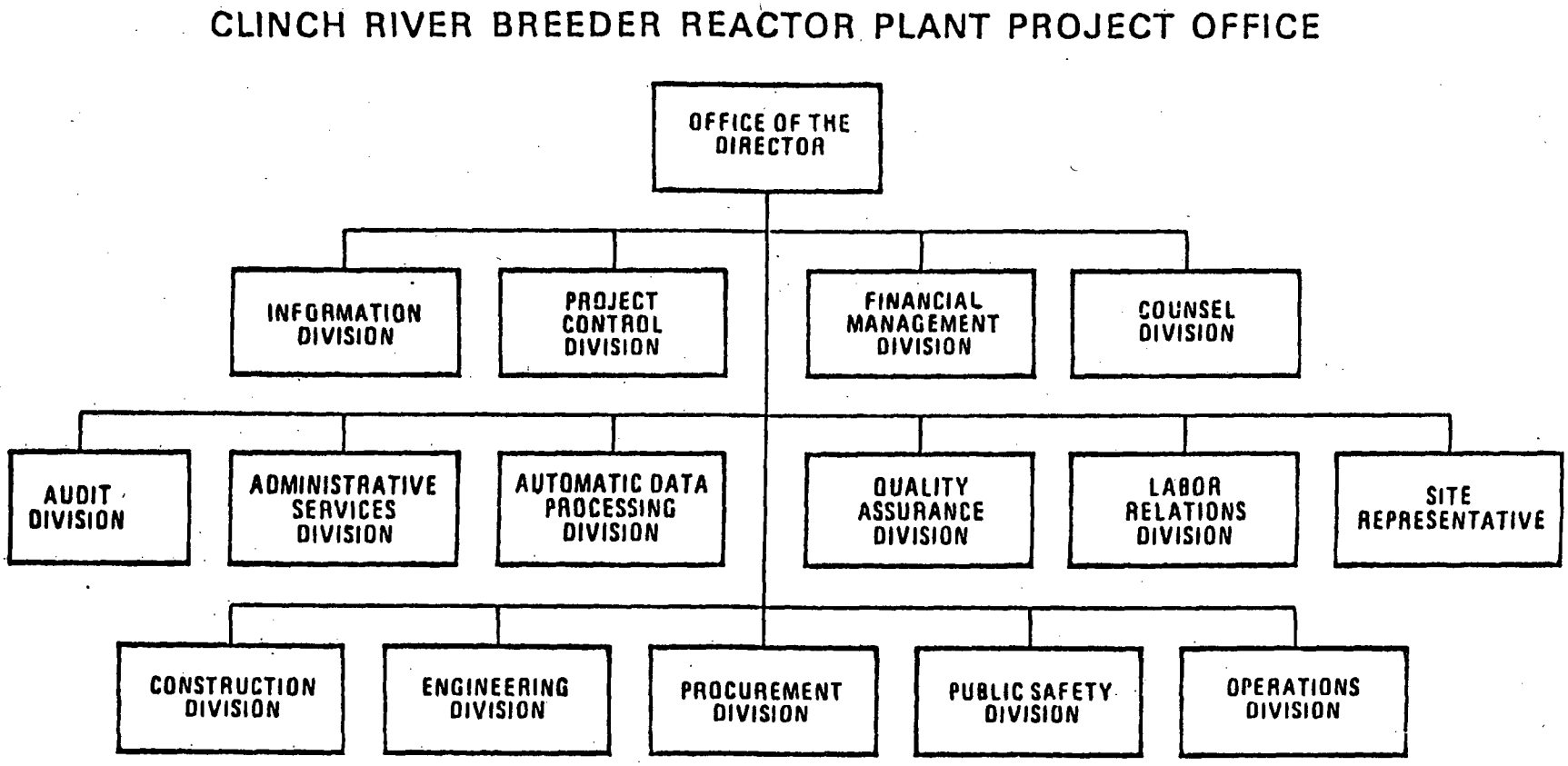


Figure 13.1 Organization of the Clinch River Breeder Reactor Plant Project Office

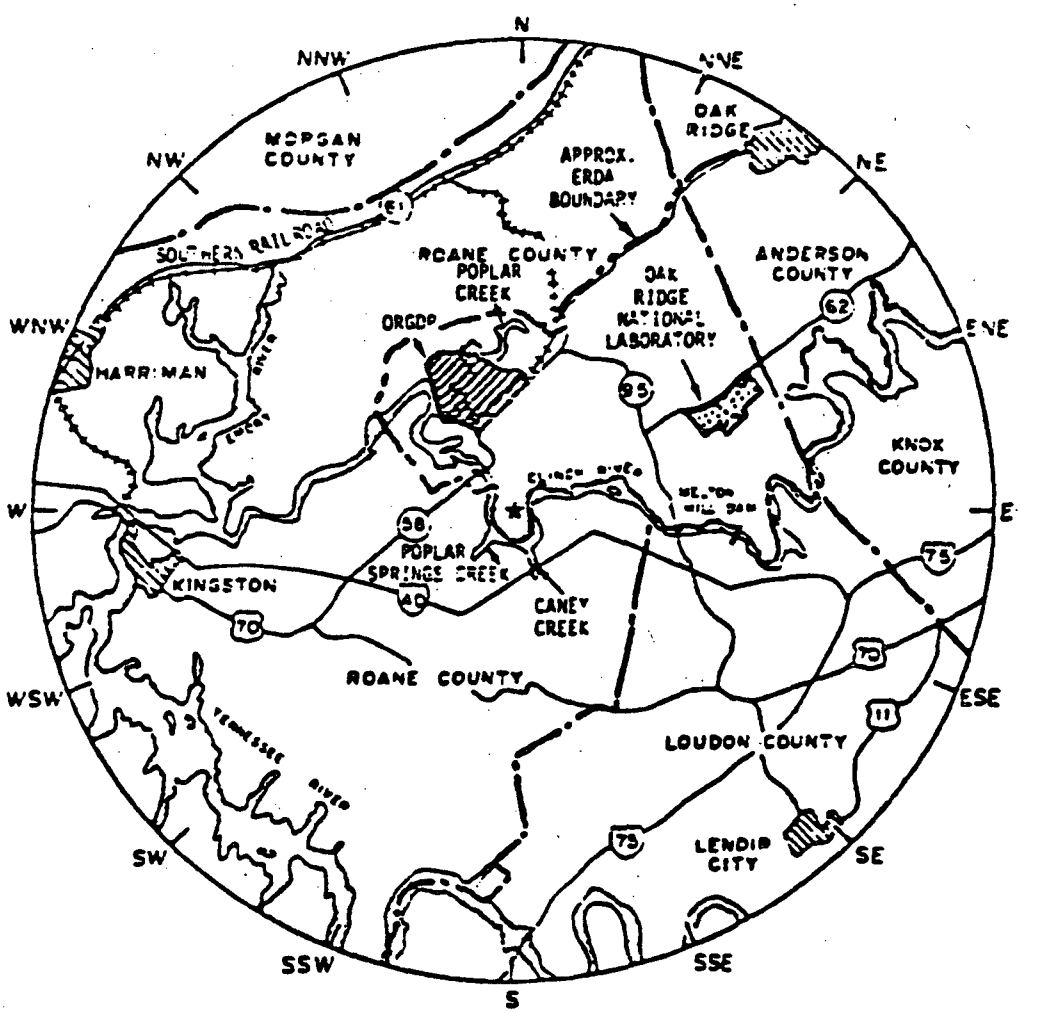


Figure 13.2 Clinch River 10-mi emergency planning zone

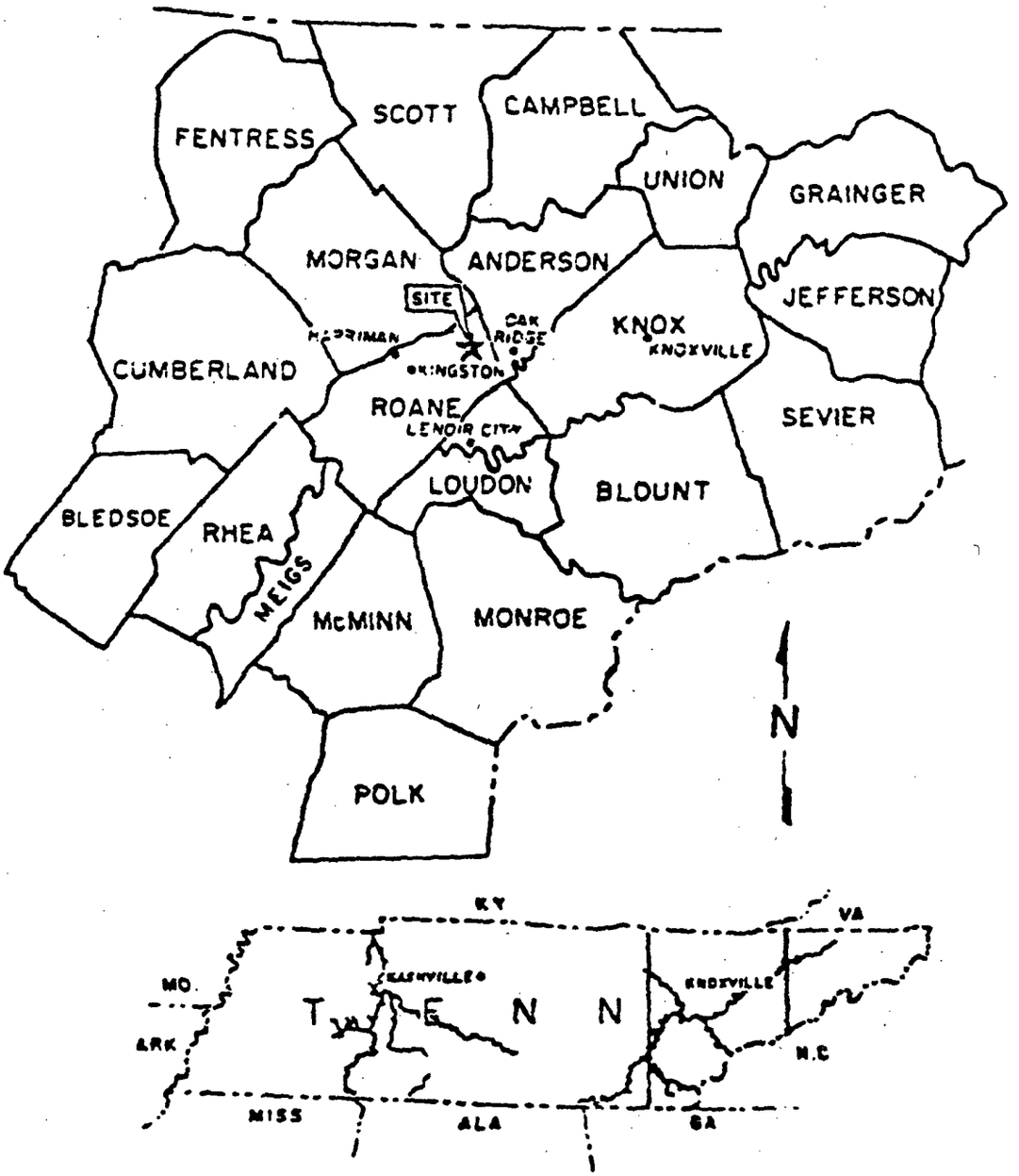


Figure 13.3 Location of Clinch River in relation to counties and States

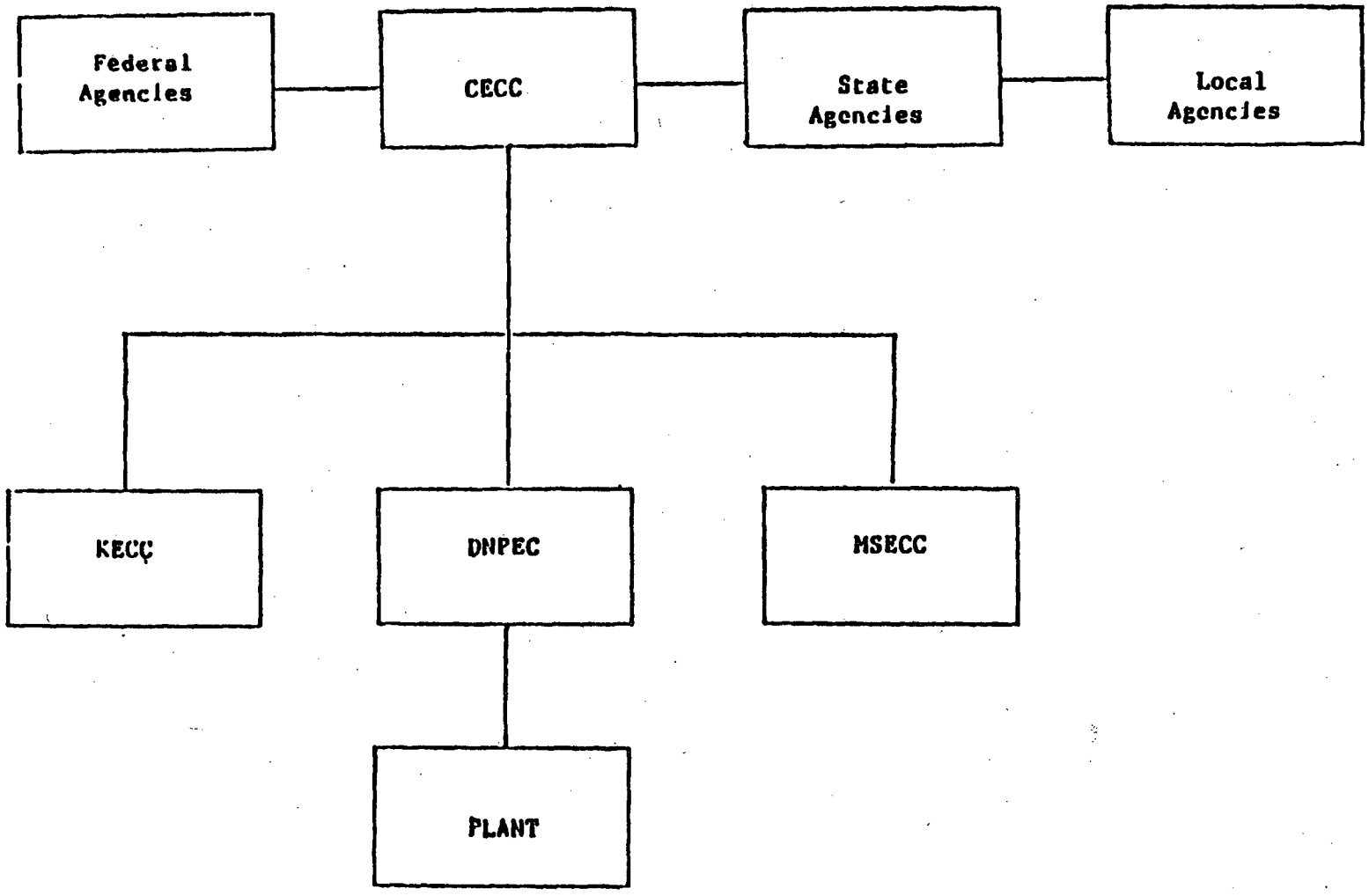


Figure 13.4 Relationship of emergency centers and agencies

14 INITIAL TESTS AND OPERATION

PSAR Acceptance Criteria

The staff has reviewed the initial test program described in the PSAR to determine whether the applicants have established acceptable plans for conducting the initial startup of the plant. The review is based on a comparison of the information presented in the PSAR with the guidance provided in the Standard Review Plan (SRP) (NUREG-75/087) and RG 1.70, Revision 3. Although this guidance has been developed primarily for the licensing of light-water reactors, the staff has determined that it is applicable to the planning for startup of the CRBRP.

In determining the acceptability of the applicants' plans for initial startup, all of the acceptance criteria of the SRP were considered. They are listed below.

(1) Scope of the Test Program

(a) The description of the test program should identify and define the major phases of preoperational testing, fuel loading, precritical testing, low-power testing, and power ascension testing. The definitions provided should be consistent with those included in RG 1.68 or justification should be provided for exceptions.

(b) The applicants' stated test program objectives should provide assurance that the adequacy of construction, as well as design, will be verified for structures, systems, and components important to safety.

(c) The applicants should designate the responsible organizations that will participate in the test program, and there should be reasonable assurance that such designated organizations will be able to collectively provide the necessary level of staffing with suitable skills and experience to develop and conduct the test program.

(d) The applicants should plan to use the plant operating and technical staff in the development and conduct of the test program and in the review of test results.

(e) The applicants should establish plans for the designers of structures, systems, and components, which may include the nuclear steam supply system vendor, architect-engineer, or other design groups, to provide test objectives and acceptance criteria to be used in the development of detailed test procedures.

(f) The applicants' program should provide assurance that construction of structures, systems, and components will be essentially complete before preoperational testing begins.

(g) The applicants' program should provide assurance that construction tests and inspections will be essentially completed before preoperational testing of the item or system begins. Such testing or inspection may include system flushing and cleaning; wiring checks; leak-tightness tests; initial calibration of instrumentation; and subsystem and component functional tests.

(h) The applicants should describe the administrative control that will ensure that preoperational test prerequisites will be satisfied.

(i) The applicants should describe the plans for establishing minimum qualification requirements to ensure that appropriately qualified personnel will be assigned to conduct preoperational and startup tests and to review and approve test procedures and results.

(2) Plant Design Features That Are Special, Unique, or First of a Kind

(a) The applicants should identify all principal plant design features or systems that are special, unique, or first of a kind.

(b) The applicants' description of planned tests should establish that such design features and systems will be functionally tested to demonstrate performance requirements.

(3) Regulatory Guides and Industry Codes and Standards

The applicants should commit to provide, in the FSAR, detailed plans conforming to regulatory positions in regulatory guides that are applicable to testing.

(4) Utilization of Plant Operating and Test Experiences at Other Reactor Facilities

The applicants should describe plans for routine evaluation of the reactor operating experiences of other facilities and describe how they will be factored into the test program. The organizations or designated management positions responsible for this review should be qualified to conduct this activity. The schedule for conducting the review should be consistent with the schedule for the development of the detailed test procedures. The types of reactor facilities to be included in the review should include light-water reactors and test reactors. The time frame included in the applicants' study should be sufficient to achieve a recent survey of plant operating problems. In general, this would involve the previous 2-year period. The sources of information should be adequate to conduct a meaningful and comprehensive review.

(5) Test Program Schedule

(a) The applicants should provide a schedule, relative to the expected fuel-loading date, for the beginning and end of each major phase of the test program. The schedule for conducting preoperational testing should provide for at least 9 months of actual testing.

(b) The preoperational testing phase should be scheduled so that most of the plant operating and technical staff will be able participate in the development, conduct, and review of test results. The hiring and training schedules for these personnel, as given in Chapter 13 of the PSAR, should be compatible with the preoperational testing schedule. In general, the hiring and training of such personnel should be planned for completion at least 1 year before the time of expected fuel loading.

(c) The schedule for the development of plant operating and emergency procedures should establish that they will be available for trial use during the test program.

(6) Trial Use of Plant Operating and Emergency Procedures

The applicants' plans and commitments should establish that plant operating and emergency procedures will be use tested to the extent practicable during the initial test program.

The following sections provide a comparison of the material in the PSAR with these six criteria. The comparison is presented in the order given in the PSAR for the CRBRP.

14.1 Description of Test Programs

Chapter 14 of the PSAR provides information relating to the initial plant startup and operation program to show that the applicants plan to develop and conduct a comprehensive test program for this first-of-a-kind plant, and that necessary early planning has been done for successful achievement of the program goals. The test programs will provide additional assurance that (1) the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, (2) the instructions for operating the plant safely have been evaluated and demonstrated, and (3) the plant operations personnel are knowledgeable in the plant procedures and operating instructions and fully prepared to operate the plant in a safe manner.

The initial test program for the plant is divided into two parts; preoperational testing and startup testing. These two parts are further divided into four phases:

(1) preoperational testing before the introduction of sodium into the heat transport system (HTS)

(2) preoperational testing that requires sodium in the HTS

(3) initial fuel loading, precritical tests, initial criticality, and low-power testing up to 5% of rated power

(4) power ascension testing from 5% through 100% of rated power

The startup test program will end when the CRBRP is proved acceptable on the basis of satisfactory evaluation of test results and the correction or acceptable provision for corrections of all identified deficiencies and incomplete items.

The applicants' definition of the major phases of the initial test program is consistent with the guidance provided in Revision 2 of RG 1.68 and thereby conforms to Criterion (1)(a) above and is acceptable to the staff.

14.1.1 Preoperational Test Program

The objectives of the preoperational test program are (1) to demonstrate the capability of structures, systems, and components to meet performance requirements, (2) to ensure, to the extent possible, that procedures for operating the plant have been used and evaluated, and (3) to demonstrate that the operating organization has acquired sufficient knowledge about the plant features and procedures to operate the plant in a safe manner. The preoperational test program will demonstrate not only that the design of systems, structures, and components meets the objectives, but that construction of the plant has been conducted in a manner that ensures the plant can be operated safely.

The preoperational test program will begin only after a major portion of the plant construction is complete. Before a structure, system, or component is preoperationally tested, activities on it must be essentially complete and the incomplete portions must be clearly documented in the test report.

Before preoperational tests on a system, structure, or component are started, construction tests such as system flushing and cleaning, wiring checks, and leaktightness tests must be completed until meaningful test results are obtained. Each system, subsystem, or component will have successfully passed the construction test and gone through a turnover procedure before preoperational testing begins.

These elements of the preoperational test program demonstrate conformance with Criteria (1)(b), (1)(d), (1)(f), and (1)(g) above and are acceptable to the staff.

14.1.2 Startup Test Program

The objectives of the startup test program are to (1) ensure orderly, safe fuel loading, low-power testing, and an approach to full-power testing; (2) confirm the design bases; and (3) demonstrate, where practical, that the plant is capable of withstanding the anticipated transients and postulated accidents.

The startup test program will begin with fuel loading and will be started only after all preoperational tests that can be performed with no fuel in the reactor are completed.

This program is consistent with the guidance provided in RG 1.68 (Rev. 2) and is acceptable to the staff.

14.1.3 Administration of Test Program

This description of the administration of the test program applies to both the preoperational and startup phases. In planning and carrying out this program, the guidelines of RG 1.68 (Rev. 2) will be used insofar as they apply to the CRBRP. This includes the entire regulatory guide except those portions that are unique to light-water reactors. Other regulatory guides will be reviewed at the time

detailed test instructions are being developed to establish which guides have applicability to the program. This policy is in conformance with Criterion (3) regarding the use of regulatory guides applicable to initial plant testing and is acceptable to the staff.

The Project Office, Department of Energy (DOE), has overall responsibility for the initial test program. TVA is responsible for the conduct of the preoperational and startup testing and plant operation. It also has the responsibility for (1) reviewing and approving test specifications, (2) preparing test instructions from the test specifications, (3) evaluating the adequacy of operating and emergency instructions during the test program, (4) approving test results and (5) recommending plant modifications as a result of deficiencies discovered during testing.

Westinghouse Advanced Reactors Division (W-ARD), as lead reactor manufacturer (W-LRM), is responsible for preparing the technical aspects of the initial test program. In fulfilling this responsibility, it will use the services of the reactor manufacturers (RMs)--W-ARD as lead reactor manufacturer, General Electric (GE-RM), and Atomics International (AI-RM)--and coordinate and use the input of Burns and Roe, Inc., as architect-engineer (A-E). This includes early planning of the scope, schedule, and sequencing of the tests interfacing with the construction schedule. W-LRM is responsible for preparing test specifications and for reviewing the test instructions and test results for the nuclear steam supply (NSS) systems under its cognizance.

Burns and Roe as A-E is responsible for preparing test specifications and for reviewing the technical adequacy of the test instructions and test results of the balance-of-plant and NSS systems under its cognizance.

W-LRM and the A-E will supply onsite staff during the test and startup period to provide technical direction of the initial test and startup program. This technical direction will include technical advice and information to operations personnel to assist them in making decisions. This staff will have the capability to support the TVA plant staff in both operations and testing. Technical direction does not include supervision of operations personnel. The responsibility for safe operation of the plant rests with TVA as the plant operator.

Stone & Webster Engineering Corporation as the constructor is responsible for (1) assisting TVA as the plant operator in ensuring that all prerequisites will be met before tests are started, (2) inserting detailed scheduling of the test program into the plant construction schedule, and (3) assisting, as required, in repair or modification of the plant that results from deficiencies found during the test program.

W-LRM and the A-E will prepare test specifications that describe each test requirement in considerable detail. These specifications will describe prerequisites, test objectives, general test methods, and acceptance criteria. Test specifications will be required and approved by the DOE Project Office.

Before an operating license is issued, the NRC staff will review the test specifications or test abstracts containing the elements of the specifications described above.

Using test specifications as a basis, TVA will prepare detailed test instructions. The format and content of these test instructions will conform with the guidance given in RG 1.68 (Rev. 2). Operating instructions will be available at the time these test instructions are prepared, so that the operating instructions may be referenced in aligning systems for testing. Each test instruction will contain a prerequisites section that will describe in detail all prerequisites, including those that are construction related, that must be satisfied before a test is performed. Signoff sheets will be provided with each instruction to provide verification that the prerequisites have been satisfied. Completion of this signoff sheet will be mandatory before the test is started.

The applicants' description of the administration of the initial test program satisfies Criteria (1)(c), (1)(d), and (1)(e) with regard to adequate participation by the operating staff and equipment suppliers in developing and executing the program and procedures for safe startup of the CRBRP. It also satisfies Criterion (1)(h) by describing the administrative control that will ensure that preoperational test prerequisites will be satisfied. With regard to Criterion (1)(i), the qualifications of personnel assigned to conduct tests and review and approve the test procedures and results are discussed in Section 13.1.1 of this report.

TVA as plant operator is responsible for the conduct of preoperational and startup tests. During the conduct of the tests, plant operating and emergency procedures will be tested wherever possible. This will help ensure that the procedures can be used to safely operate the plant and provides further assurance that the operator will be familiar with the procedures and thoroughly trained to operate the plant.

The applicants' description of the conduct of the tests with regard to the trial use of plant operating and emergency procedures satisfies Criterion 6 and is acceptable.

The applicants' schedule for the initial test program provides for 2 years of preoperational testing before fuel loading. Most of the TVA operating staff and preoperational test crew will be on site approximately 8 months before the start of preoperational testing. Thus, there will be sufficient time for the plant operating and technical staff to participate in the development and conduct of the tests and to review the test results.

Before preoperational testing, these personnel will have completed basic nuclear courses for operators and specialized training for technical personnel as well as assignments to a sodium-cooled fast reactor. Additional discussion on staffing and training is provided in Section 13 of this report.

The staff has concluded that the schedule satisfies Criteria (5)(a) and (5)(b) with regard to the time scheduled for preoperational testing, the planned staff participation, and the compatibility of the test schedule with the hiring and training of operations and test personnel.

The schedule also shows that procedures for normal operation and emergencies will be outlined from 5 to 8 years before fuel loading and that the detailed procedures will be available approximately 2-1/2 years before fuel loading.

This schedule satisfies Criterion (5)(c) and provides adequate time for trial use of operating and emergency procedures during preoperational and startup testing.

The applicants have described administrative procedures for feedback of significant testing and operating experiences at other facilities, including light-water reactors and the fast flux test facility test reactor (see Appendix H of PSAR). These experiences will be used in the development of the initial test program. This satisfies Criterion (4).

14.1.4 Test Objectives of First-of-a-Kind Principal Design Features

The applicants identified six first-of-a-kind plant design features that will require extensive testing and provided brief outlines of the tests that will be performed. These features are (1) the in-vessel transfer machine, (2) the primary/secondary sodium pump, (3) the secondary control rod drive, (4) the upper internals structure and upper internals structure jacking mechanism, (5) CRBRP intermediate heat exchanger, and (6) the steam generator module. The planned tests establish that these design features and systems will be functionally tested to demonstrate performance requirements. This will be accomplished by prototype testing at test facilities and by preoperational and startup tests after installation. The staff also notes the progress in the applicants' natural circulation verification program and expects this to be followed by confirmatory testing during reactor startup.

The applicants' description for testing first-of-a-kind features conforms with Criterion (2) and is acceptable. Additional unique features may be identified for testing. The staff has concluded that ample time has been provided in the test program schedule to adequately test all features that are important to safety.

14.2 Augmentation of Operator's Staff for Initial Tests and Operation

Section 14.2 of the PSAR discusses the augmentation of TVA operations personnel by technical support personnel from TVA and the use of regulatory guides and ANSI standards in developing the required qualifications of the technical support personnel. This aspect of the management of the initial test program is covered in SRP Section 13.

14.3 Conclusion

On the basis of its review of the information provided in Chapter 14 of the CRBRP PSAR and its review of plant design concepts, the staff has concluded that the applicants have demonstrated sufficient advanced planning to ensure that (1) the initial test program will demonstrate functional adequacy of plant structures, systems, and equipment, (2) adequate administrative controls will be established to govern the initial test program, (3) the test program will be used, to the extent practicable, to train and familiarize the plant operating and technical staff in the operation of the facility, and (4) the adequacy of plant operating and emergency procedures will be verified, to the extent practicable, during the period of the initial test program.

The applicants' description of the initial test program conforms to the relevant sections of RG 1.70, Revision 3, and adequately addresses the acceptance criteria of the SRP (NUREG-75/087).

15 DESIGN-BASIS ACCIDENT ANALYSIS

15.1 Scope of Review

The staff's review of the applicants' analyses of design-basis accidents has focused on the following considerations. In Section 15.1.1, the staff considered (1) the adequacy of the selection of design-basis accidents (DBAs), including assessments of whether the proposed DBAs include all credible categories of transient and upset conditions for which engineered safety features and conservative design measures should be provided, and (2) whether within each category the proposed DBAs have been appropriately classified with regard to the expected frequency of occurrence. In Section 15.1.2, the staff considered whether the proposed acceptance criteria, that is, fuel limits and radiological consequences are acceptable and in Section 15.1.3 whether the assumed initial conditions for the accident analyses are appropriate. In Sections 15.2 through 15.7, the staff considered whether the specific postulated accidents constitute sufficiently rigorous and conservative challenges to proposed safety features and whether it will be feasible to eventually meet the acceptance criteria with the safety features currently proposed.

15.1.1 Selection of Design-Basis Accidents

15.1.1.1 Introduction

The applicants have proposed and analyzed a set of DBAs in Chapter 15 of the PSAR. Synopses of those events are provided for the benefit of the reader in an appendix (Section 15.A) to this section of the SER, and the events are listed below. The events include all those accidents in "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants--LMFBR Edition" (WASH-1302) and five additional events. Categories of events proposed include local faults, reactivity insertions, undercooling events, fuel-handling and storage events, sodium spills, and miscellaneous events.

Representative Types of Events To Be Analyzed in Chapter 15 of the SAR

(1) uncontrolled control rod assembly withdrawal from a subcritical condition assuming the most unfavorable reactivity conditions of the core and reactor coolant system

(2) uncontrolled control rod assembly withdrawal at power assuming the most unfavorable reactivity conditions of the core and reactor coolant system which yield the most severe results (hot at zero power, full power, etc.)

(3) control rod misoperation or sequence of misoperations

(4) partial and total loss of reactor coolant flow including trip of pumps and pump seizures

(5) start-up of an inactive reactor coolant loop or recirculating loop at incorrect temperature

(6) loss of normal and/or emergency feedwater flow

(7) loss of all ac power to the station auxiliaries and loss of emergency diesel generators (station blackout)

(8) loss of intermediate coolant flow

(9) heat removal greater than heat generation because of (a) feedwater system malfunctions, (b) a pressure regulator failure, or inadvertent opening of a relief valve or safety valve, and (c) a regulating instrument failure

(10) maloperation of reactor plant controllers

(11) internal and external events such as major and minor fires, flood, storms or earthquakes

(12) loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant system

(13) spectrum of postulated intermediate coolant system piping breaks inside and outside containment

(14) inadvertent loading and operation of a fuel assembly into an improper position

(15) leak from or rupture of waste gas decay tank

(16) failure of cover gas purification system

(17) the spectrum of rod ejection accidents

(18) fuel-handling accident

(19) small spills or leaks of radioactive material outside containment

(20) fuel cladding failure combined with intermediate heat exchanger and steam generator leaks

(21) control room uninhabitability

(22) loss of heat sink

(23) turbine trip with coincident failure of turbine bypass valves to open

(24) loss of one (redundant) dc system

(25) turbine trip with failure of generator breaker to open

(26) loss of instrument air system

(27) local blockages of a few subchannels within a fuel subassembly

(28) leak in control rod drive housing

(29) inadvertent release of oil in pump seal into sodium

(30) failure of a few steam generator tubes

(31) leaks in intermediate heat exchanger

(32) abnormally high or low cover gas pressure

(33) core flow maldistribution because of fuel loading error

(34) gas bubbles passing through core

(35) failure of steam dump system

(36) inadvertent closure of either reactor coolant valves or intermediate coolant valves

(37) spurious reactor trip
(38) inadvertent drop (delatching) of a single control rod
(39) failure of reactor vessel cover seal
(40) plugging of reactor overflow line
(41) failure of core clamping mechanism
(42) accidental opening of valves to a drained isolated loop
(43) large leak (rupture) in steam or feedwater isolated loop
(44) enrichment error in fuel assembly
(45) misloaded fuel assembly
(46) dropped fuel assembly
(47) unknown stuck control rod
(48) loss of normal shutdown cooling system
(49) simultaneous leak of reactor overflow tank
(50) inadvertent closure of floor valve on canister during fuel handling
(51) failure of any single active component in fuel-handling system
(52) loss of site power during fuel handling
(53) fuel-handling machine jams
(54) leak in fuel storage vessel
(55) failure of single active component in fuel storage cooling system
(56) failure to seat fuel assembly properly
(57) inadvertent opening of floor valve with shield plug removed and fuel-handling machine not in place
(58) leak in fuel canister
(59) inadvertent opening of fuel handling machine valve during transfer
(60) attempt to insert a fuel assembly into occupied position
(61) collision of fuel-handling machine with control rods
(62) dropping shipping cask from maximum possible crane height
(63) collision between fuel-handling machine and crane
(64) loss of all power to fuel-handling machine
(65) removal of jammed fuel assembly
(66) blocked coolant flow to control rods
(67) intermediate coolant system fire

Additional Events Considered

(1) seismic reactivity insertion that results from OBE and SSE
(2) core, radial blanket, and control rod movement that results from SSE
(3) sudden core radial movement

(4) inadvertent actuation of the sodium/water reaction pressure relief system
(5) stochastic pin failure

Design-basis accidents are a set of events used to assess the way specific systems respond to abnormal conditions. As such, these events provide analytic tests of the design, selected to determine if installed or proposed safety features can cope adequately with the postulated event. For LWRs, plant response to these DBAs is assessed using the guidance from 10 CFR 50, primarily the general design criteria, and the Standard Review Plan, primarily Chapter 15. It is normal staff practice to require that conservative margins be demonstrated in analyses of the postulated events. Acceptance criteria applied in the tests range from mechanical stress limits to fuel cladding temperature limits. In addition, the postulated events must be acceptably mitigated, that is, meet all specified acceptance criteria, even if single failures are postulated to have also occurred in the safety systems under evaluation. Potential radiological consequences of DBAs are also assessed to determine whether predicted consequences fall within appropriate radiological dose guidelines. Dose guidelines for specific LWR accidents are specified in the Standard Review Plan, typically as fractions of the site suitability guidelines of 10 CFR 100.11. It is emphasized that these are dose guidelines for review rather than strict limits.

The design-basis accidents were selected to represent a reasonable envelope of the credible events which might occur at a nuclear plant and which require mitigation by active systems or passive structures. The choice of the specific events typically depends on the type of reactor with different sets of events selected for BWRs, HTGRs (high-temperature gas reactors), PWRs, and LMFBRs (liquid metal fast breeder reactors). No regulatory criteria have been established for making these choices. Instead, engineering judgment regarding the kinds of faults or phenomena which might occur for a given kind of nuclear reactor is employed. The selected events may range from those which may occur once per year or to those events which may never occur during the life of the plant.

When events have been judged to be so improbable that they are not "credible" as events against which the design should be tested, they have been excluded from the design-basis envelope. For example, accidents involving an initiating event and simultaneous multiple failures of the mitigating safety systems have been judged so improbable that they have not been included as design-basis accidents. Such accidents often have been designated as Class 9 accidents or "beyond the design-basis accidents." The term "Class 9 accident" has no official regulatory standing, but is used here because of its historical familiarity. Because Class 9 accidents typically involve some degradation of the reactor core, the term "core-disruptive accidents" is also used in this SER to describe such severe accidents.

Although probability is a consideration in distinguishing DBAs from Class 9 accidents, there are no specific numerical probability thresholds which are employed. Engineering judgment, based on such deterministic criteria as quality assurance, compliance with regulatory standards, redundancy, independence, and diversity is more often employed to decide that multiple failure of safety systems need not be postulated as part of the design basis for nuclear plants.

In summary, there is no specific regulatory guidance that prescribes which accidents, transients, or upset conditions are to be included in the DBA spectrum for an LMFBR. Also, there is no well-defined verified methodology that can be used to determine what these events should be. To gain assurance that the CRBR design-basis accidents will include all credible, upset, and transient conditions for which mitigating features should be provided, the staff and its consultants have employed several techniques.

First, the staff reviewed the DBAs considered in LWR designs. Although CRBR is different from LWRs in many ways, there are also a number of similarities, particularly with regard to categories of accidents. For example, SRP Section 15 describes DBAs to be considered in the staff's review of LWRs. The staff considered the Standard Review Plan accidents in its review of the completeness of the CRBR DBAs. Staff consultants from Idaho Nuclear Engineering Laboratory (INEL) have assisted in that review.

Second, staff consultants from Brookhaven National Laboratory (BNL) and INEL have reviewed the DBAs considered in foreign and domestic LMFBR designs, respectively. Foreign LMFBRs considered by BNL in its review included PHENIX, SUPER-PHENIX, SNR-300, PFR, and MONJU. Domestic LMFBRs considered by INEL in its review included EBR-II, SEFOR, FERMI-1, FFTF, and the Large Development Plant (LDP).

Third, the staff considered the various reliability- and risk-oriented studies performed for CRBR or other LMFBRs to gain general insight with regard to event sequences which could occur with sufficiently high probability to be considered credible.

The staff and its consultants have also reviewed the CRBRP systems, controls, design features, and characteristics within the context of the physical phenomena, both internal and external, which could credibly occur at the plant. Staff consultants have included BNL, INEL, LANL, and SAI. This review and the other activities outlined above have been instrumental in formulating the staff's conclusions about the completeness of the CRBR DBA spectrum.

No element of this review process was singularly definitive in reaching conclusions; all elements provided useful insights. Specific conclusions regarding the CRBR DBA spectrum are discussed in detail in the following sections. Briefly, the staff reached the following conclusions:

(1) Core-disruptive accidents (CDAs) (including those initiated by loss-of-core-coolant inventory and anticipated transients without scram (ATWS) may be excluded from the DBA spectrum for CRBR.

(2) The set of DBAs proposed by the applicants and described in PSAR Chapter 15, is acceptable with regard to completeness, with the exception that the possible freezing of sodium in the intermediate heat transport system (IHTS) because of credible overcooling transients initiated by failures in the feedwater and steam systems should be analyzed in the FSAR, and under the condition that additional failure modes and effects analyses of control systems (as discussed in Section 7.7 of this SER) and of the cover gas system be performed.

15.1.1.2 Basis for Excluding Core-Disruptive Accidents From the Design-Basis Accident Spectrum for the Clinch River Breeder Reactor

In this section, the nature of CDAs for an LMFBR such as CRBR are described and the conditions which could lead to a CDA are indicated. The basis for excluding such events from the DBA spectrum is discussed.

A core-disruptive accident is an accident so severe that the reactor core or more specifically the fuel geometry is significantly modified over a significant region of the core. Among the variations in the subsequent behavior are (1) successful in-core cooling of the disrupted core, (2) thermal failure of the reactor vessel because of inability to cool the disrupted core, and (3) mechanical failure of the reactor vessel because of power bursts from reactivity excursions or fuel coolant interactions which might occur as a result of fuel and coolant relocation. All these variations could have serious consequences because of the release of radioisotopes into containment, but if containment failure were to also occur because of excessive mechanical or thermal loadings, the radiological consequences could be even more severe.

A core-disruptive accident can occur if there is either (1) a failure to remove heat from the fuel at a sufficient rate to ensure fuel integrity or (2) a local failure in a fuel assembly propagates beyond that assembly to adjacent regions of the core. Failure to remove sufficient heat from the fuel could occur if any, or a combination, of the following should occur:

(1) failure to shut down the nuclear chain reaction when necessary during an overpower or a flow reduction transient

(2) failure to maintain sufficient primary coolant inventory to keep the fuel covered with coolant

(3) failure to maintain sufficient coolant flow to provide a heat removal path from the fuel

(4) failure to extract sufficient heat from the coolant to maintain fuel integrity

Propagation of local fuel faults to large regions of the core can occur if there is a failure to shut down the nuclear chain reaction before the point at which sufficient local damage has occurred to allow propagating mechanisms to develop.

The most significant difference between LWRs and LMFBRs with regard to the nature of the CDAs is that an LMFBR has more possibilities for reactivity excursions following an accident. The reasons for this are that (1) voiding of coolant from an LWR core decreases the reactivity but for an LMFBR voiding of coolant increases reactivity and (2) fuel relocation for an LWR core decreases reactivity whereas it may increase the reactivity for an LMFBR. For either type of reactor fuel coolant interactions, mechanical and thermal vessel failure, release of radioisotopes into containment, and related containment failure are possible.

Because of the potential severity of CDAs for an LMFBR, considerable attention has been devoted to development of design measures to make CDAs very improbable.

Although it is generally believed that CDAs are very improbable, the consequences of such events have nonetheless been studied in detail and measures to mitigate them have also typically been provided in LMFBR designs.

It is the staff's position that CDAs need not be included in the DBA spectrum for CRBR. The basis for this conclusion is the staff's judgment that the safety functions which must be fulfilled to make CDAs very improbable can be implemented for CRBR. This confidence is based on two points. First, those safety functions which must be achieved for an LMFBR such as CRBR are not fundamentally different from the safety functions successfully implemented for LWRs. Second, the special characteristics associated with design and operation of an LMFBR and the ways they could impact these safety functions are well understood because of the general knowledge and experience gained from design and operation of fast-sodium-cooled reactors such as FERMI-1, EBR-I and II, SEFOR, FFTF, and foreign LMFBRs. More specifically, with only a few exceptions, the staff's review procedures, the criteria and standards in 10 CFR 50, 73, and 100, and the NRC Standard Review Plan normally applied to LWRs, are applicable to LMFBRs. These criteria are the basis for excluding core-disruptive accidents from the design-basis accident spectrum for LWRs, and when supplemented with the special criteria necessary to account for the characteristics of LMFBRs, contribute to the staff's confidence that CDAs can be made very improbable for CRBR.

In reaching this conclusion, the staff has taken into account the possibility that human error could initiate accidents or complicate or defeat the mitigation of accidents by safety systems. Because of the use of automatic initiation of safety functions that must be fulfilled rapidly, the adoption of a modern control room design, and the enhanced attention to operator training and human factors in general that followed the TMI-2 accident, the staff believes that human error will not result in unacceptable risk from CDAs at CRBR. At this point in its review, the staff is confident that sufficient attention has been paid to design criteria related to human performance so that plant construction can proceed. The staff's detailed discussion of this subject is provided in Section 18. At the OL review stage, the staff will require a demonstration that all NRC requirements (such as operator training and development of emergency procedures) to minimize human error have been met.

The safety functions which must be achieved to prevent CDAs are itemized:

(1) Shut down the nuclear chain reaction upon initiation of transients.
(2) Maintain sufficient coolant inventory.
(3) Maintain sufficient coolant flow.
(4) Remove sufficient heat from the fuel.
(5) Avoid propagation of local fuel faults beyond an assembly.

The design measures being required of CRBR to achieve these functions with high reliability are discussed throughout this SER. Item (1) is considered in more detail in Sections 4, 7, and the remainder of Section 15; Items (2), (3), and (4) are considered in more detail in Sections 4, 5, 7, 8, 9, and in the remainder of Section 15; Item (5) is considered in Sections 4, 7, and the remainder of Section 15. Furthermore, additional conservative design measures and practices which help ensure that the basic safety functions will be met for CRBR are described in Sections 2, 3, 6, 13, 14, 17, and 18. The discussion of the CRBR principal design criteria in Section 3.1 provides a good overview of all

these considerations, including the special criteria relating to LMFBR characteristics.

The special design features which provide assurance that CDAs will be very improbable for CRBR are:

(1) the dual, independent, and diverse reactivity shutdown systems

(2) the redundancy, diversity, and independence of the shutdown heat transport system

(3) the provisions for inspection and leak detection to ensure high integrity of the heat transport system components

(4) the provisions to ensure that local fuel faults will not propagate into core disruption

(5) the provisions to prevent flow blockages to core assemblies

Although CDAs need not be included as DBAs for CRBR, the possible consequences of CDAs have been considered in detail and special design features have been provided to ensure that the risks from these events will be very low. The staff's evaluation of the consequences of CDAs in the highly unlikely event they should occur is provided in Appendix A of this SER.

15.1.1.3 Basis for Excluding Loss of Reactor Coolant Inventory From the Core as a Design-Basis Accident for the Clinch River Breader Reactor

In the commercial nuclear power industry, the light-water reactor (LWR) power plant predominates. Although there are different types, basically they all utilize water under high pressure and at temperatures near its saturation point as a coolant. If leaks or breaks develop in these systems, the tendency is for the system to depressurize and, in the process, the coolant changes from a liquid to gas or liquid/gas mixture and escapes from the system. This is the undercooling sequence in an LWR, designated as a loss-of-coolant accident (LOCA). In a system under pressure and at saturated conditions, generally there is little that can be done when a LOCA occurs except to replace the escaping two-phase fluid as quickly as possible. Much attention has been focused on LOCAs in LWRs because of (1) their importance as an accident class, (2) the evolutionary nature of the nuclear industry, and (3) the accident at Three Mile Island.

In CRBR, as in LWRs, loss-of-coolant accidents are of importance; however, the emphasis is considerably different. CRBR will be cooled using liquid sodium. The saturation temperature (boiling point) of liquid sodium under atmospheric conditions will be about 1,600°F. The design of CRBR provides for the use of the sodium coolant at low pressures. The pressures for most conditions will be only those pressures necessary to circulate the coolant through the system and any additional hydraulic static pressure that may result.

In addition, the temperature levels of normal operation will be several hundred degrees below the saturation temperature. When a leak will develop in the CRBR system, the low pressures will force sodium to flow or spray from the opening,

but as a liquid. Assuming no other motive force such as pump pressure, a liquid will drain from the hydraulic static head to the level of the leak. A siphoning effect will occur when the liquid in a vessel can flow through a pipe or duct whose inlet will be below the vessel's liquid-free surface, but above the discharge area of the duct. These conditions have been considered in the design of CRBR and discussed and evaluated in Section 15.3. Leakage from the system in CRBR by virtue of the low pressure and low temperature (with respect to saturation nature of the coolant) will necessitate concern primarily for inventory control and siphoning. The concern of losing the entire coolant inventory through a moderate break in the system regardless of location is not present. CRBR will have safety class guard vessels surrounding the major components in the primary heat transport system (PHTS) system. These vessels will be sized so that leaks occurring inside these guard vessels will fill only to equilibrium levels, which will always leave sufficient sodium in the reactor vessel that the unaffected loops can still function. Siphoning has been precluded because the coolant piping outside the guard vessels will be elevated. This philosophy has been utilized throughout the plant design including ex-reactor features such as the ex-vessel storage tank (EVST).

It is necessary to automatically shut down pumps such as the PHTS pumps that might force excessive fluid out of a pipe break and also to ensure that the pony motor cannot pump fluid out of breaks in the elevated sections of piping. These features have been included in the design.

The design features and characteristics described above, in conjunction with the sensitive leak detection system, will provide a level of protection against fuel damage from pipe breaks comparable to the emergency core cooling system (ECCS) in a LWR. These features are preventive and passive in nature rather than reactive (as the ECCS) as might be expected from the differing phenomena. In addition, the leak-before-break piping argument discussed in detail in Sections 3.9.8.3, 3.9.9, and 5.11 leads to the conclusion that pipe breaks will be highly unlikely and adds assurance that CRBR will be adequately protected with respect to the entire spectrum of loss-of-coolant accidents. These issues are also discussed in Section 15.3.2.1.

15.1.1.4 Consideration of Anticipated Transients Without Scram for the Clinch River Breeder Reactor

Operating experience at foreign and domestic LWRs as related to the potential for anticipated transients without scram (ATWS) has been evaluated by the staff and published in NUREG-0460 ("Anticipated Transients Without Scram for Light Water Reactors"). The staff concluded in that report that the frequency of ATWS events for some LWRs may be sufficiently high that measures should be taken to reduce the risks from such events. Rulemaking will be pursued to resolve the ATWS issue for LWRs.

It should be noted that the LWRs considered by the staff in reaching its conclusion had only one fast-acting reactivity shutdown system. As noted in Sections 3, 4, and 7 and in Section 15.1.2 above, CRBR will have dual, independent, and diverse fast-acting shutdown systems. This special feature of the CRBR design and the evaluations performed for it have led the staff to conclude that ATWS events will be significantly less frequent for CRBR than for LWRs in general, and may therefore be excluded from the DBA spectrum. Also, CRBR will have the capability to limit the consequences of ATWS events, should they occur, to acceptably low levels as discussed in Appendix A of this SER.

15.1.1.5 Requirements for Additional Design-Basis Accident Analyses

As a result of its review of the adequacy of the completeness of the CRBR DBA spectrum, the staff concluded that the spectrum is acceptable with the following exceptions.

There is no apparent way that overcooling in the primary heat transport system could lead to a problem such as that of pressurized thermal shock for LWRs because there is no mechanism to inject large quantities of cold sodium at full power or to subject the system to high pressures. However, the staff has noted the possibility of overcooling the IHTS initiated by failures in the feedwater or steam systems (loss of feedwater heating for example). For reasons discussed below, the staff believes that such events will be acceptably mitigated. However, no specific analysis has been provided to show that there is adequate margin to the sodium freezing point in IHTS or to show that the nature of the initiating causes of such events or the effect of additional system failures are fully understood. The staff, therefore, requires that the FSAR will incorporate detailed system analyses of a range of overcooling events including those initiated by failures in the feedwater system, and that these events will cover an acceptably wide variation in initiators, initial conditions, and postulated system failures to clearly demonstrate that all credible accidents of this type have been addressed. Implicit in this is the requirement that failure modes and effect analyses on the feedwater system, the steam generator auxiliary heat removal system, the protected air-cooled condenser, and the EVST cooling system be performed.

It is emphasized that the staff's conclusion that additional analyses should be performed does not mean that there is an inherent inadequacy in the CRBR design. Given even the most pessimistic circumstances, the most that would be required to resolve any new considerations uncovered by the analysis would be an adjustment in trip settings or addition of another trip function in the final design. Such design fixes are well within the state-of-the-art. The basis for the staff's conclusions on additional analysis and the correction of any new problems disclosed by these evaluations is as follows.

Feedwater temperature will be monitored by three resistance temperature detectors (RTDs) in the steam drum inlet line. The RTD signals will be used to provide temperature compensation for the feedwater flow signal. They will also be displayed in the control room. Feedwater mass flow will be sensed by three differential pressure elements across one Venturi in the inlet line to each steam drum. These temperature-corrected feedwater flow signals will be supplied to the reactor shutdown system logic. The reactor shutdown system will provide buffered signals to the plant control system and the plant data handling and display system. Steam drum level will be sensed by three differential pressure elements measuring the differential pressure between a reference column and the water head in the steam drum. This measurement will be density compensated. The signal will be supplied to the reactor shutdown system logic. Buffered signals will be supplied to the plant control system and the plant data handling and display system. The steam-feedwater flow mismatch subsystem will initiate reactor trip to prevent continued operation with large imbalances between the steam and feedwater flow for each heat transport system loop. These subsystems will protect the steam generators and drums against unacceptable thermal transients. Each subsystem will compare the steam and feedwater flow in two individual comparators. If the difference between the two values exceeds the

setpoint in either of the comparators, a trip will be initiated. Increasing steam flow and decreasing feedwater flow fault events will be sensed by the first comparator. The second comparator will sense decreasing steam flow and increasing feedwater flow fault events.

A decrease in feedwater temperature would likely manifest itself by a change in steam drum level and/or a change in the feedwater mass flow rate (since the flowmeter signals will be temperature compensated). Both process variables will supply signals to the reactor shutdown system. An increase in feedwater flow rate would be detected by the steam-feedwater flow mismatch system and/or the steam drum level sensors. Both process variables will supply signals to the reactor shutdown system. The reactor would most probably be scrammed long before the temperature transient was reflected back into the primary heat transport system and no challenge will be presented to the primary system or the fuel.

The staff's second requirement is that failure modes and effects analyses (FMEA) of the heat transport system and the auxiliary system, and the analyses of control system failures and the cover gas system outlined in Section 7.7.2.1 be factored into the Chapter 15 DBA analyses presented in the FSAR to clearly demonstrate that the DBAs will bound all credible transient and upset conditions adequately. The cover gas system FMEA should be performed with consideration of a range of leaks in the PHTS and various operating conditions.

15.1.1.6 Categorization of Design-Basis Accidents

It has become common practice to define the DBAs in terms of frequency of occurrence. The events are placed in three general categories as shown in Table 15.1 which shows the intercomparison of event categories proposed by the applicants, recognized by the staff, and in use by the nuclear industry. Using the criteria of Table 15.1, it is possible to classify the postulated events for CRBR as shown in Table 15.2. Table 15.2 also includes additional information about required operable systems and which primary and secondary plant protection system (PPS) trip function will be involved. The events thus categorized may be analyzed to determine their consequences. These consequences are compared against an acceptance criteria which is discussed in the following section.

The staff has reviewed the procedure and finds it an acceptable way to analyze the design-basis events. In evaluating this procedure, the staff concludes that this designation of events based on engineering judgment and accident delineation studies is comprehensive and that the categorization was reasonable. More detailed discussion of the duty cycles may be found in Section 3.9. The duty cycle is an important factor in design of the fuel, vessel, piping and components.

In analyzing the accidents, the applicants determined the "bounding case" or umbrella event to present in detail. The results of the other events bounded by these cases were then presented in summary. The bounding events were

(1) "15.2, Reactivity Insertion Design Events," safe shutdown earthquake (SSE)--60-cent reactivity step

(2) "15.3, Undercooling Design Events," loss of offsite electrical power

(3) "15.5, Fuel-Handling and Storage Events," cover gas release during refueling

The staff finds no argument with this procedure and concludes that the selection of bounding events for the various categories is acceptable. Partly to verify this selection procedure, staff audit calculations have been performed in selected cases and are discussed in the appropriate subsections.

15.1.2 Acceptance Criteria

15.1.2.1 Introduction

The hierarchy of criteria for acceptance flows from the general design criteria (GDC) for nuclear power plants, 10 CFR 50, Appendix A. In 10 CFR 50, Appendix A, there are 55 distinct criteria enumerated, which apply primarily to LWR power plants. 10 CFR 50, Appendix A, further acknowledges that although the GDC are formulated for LWRs, they are generally applicable to other types of nuclear power units and serve as guidance in establishing the principal design criteria (PDC) for a specific facility as directed in 10 CFR 50.34. CRBR PDC are listed in Section 3.1 of this SER. These criteria address certain general characteristics of liquid metal fast breeder reactor (LMFBR) systems but are specifically intended for application to CRBR.

In the review procedures for LWRs, NUREG-0800 (Standard Review Plan (SRP)) has been developed previously and incorporates acceptance criteria. The SRP is pertinent for many areas of CRBR, but because of the unique aspects of LMFBRs and the accident analyses considered in the SRP, only select portions were used as a guide. Although there has been an LMFBR edition of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (WASH-1302) (USAEC), no corresponding LMFBR edition of the SRP has yet been issued. Therefore, the applicants have proposed acceptance criteria for the design-basis accidents which are tabulated in Table 15.3. These criteria are more thoroughly discussed in Section 4.2 of this SER. Briefly, these criteria are based on the assumption that if appropriate fuel design and coolable geometry limits can be maintained and if radiation releases are restricted to no greater than one-tenth of the 10 CFR 100 guidelines for all design-basis accidents, then the applicable PDC will be satisfied which, in turn, implies a satisfaction of the GDC as well. The applicants have quantified coolable geometry by stipulating that sodium temperatures will not exceed saturation (no boiling) and that the cladding must not exceed the melting point. Therefore, the acceptance criteria in Table 15.3 for extremely unlikely faults are cladding melting and sodium saturation temperature. The events of less severity, but usually higher frequencies, have more restrictive criteria based on minimization of fuel damage or normal operating radiation limits. There are ad hoc acceptance criteria for special accident categories such as sodium fires and spills, fuel handling, and fuel failure propagation.

15.1.2.2 Evaluation

The staff finds the proposed criteria of Table 15.3 based on fuel, cladding, and coolant temperature limits to be acceptable for the construction permit because these limits are based on fuel damage criteria discussed in Section 4.2 of this SER and the maintenance of coolable core geometry. The staff and its

consultants have proposed modifications to these criteria for the operating license which are discussed in Section 4.2 of this SER. The staff noted that the proposed acceptance criteria for extremely unlikely faults put no bounds on fuel temperature. The staff recognizes that the emphasis is on coolable geometry and that there is considerable experimental evidence that some fuel melting may be tolerable. The staff felt, however, that the applicants should establish bounds of acceptability for fuel melting on extremely unlikely faults for the operating license.

Defining the maintenance of coolable core geometry by excluding sodium boiling in the core is an unambiguous criterion. Sodium boiling in the core does not define unequivocably the initiation of uncoolability. This is even more true for transient boiling. Some boiling could very likely be tolerated without deleterious effects. However, the calculation of sodium boiling is not a routine calculation; void formation in the core has the potential for positive reactivity feedback effects and heat transfer becomes more diffcult to predict. Therefore, the staff agrees with the no-boiling criteria.

Where radiation releases are involved, the criteria for acceptance are proposed in the PSAR as the dose limits of 10 CFR 20 for anticipated and unlikely events and one-tenth of 10 CFR 100 for extremely unlikely events. The dose limits are consistent with the Standard Review Plan. For example, the first acceptance criterion for spent fuel cask drop accidents, SRP Section 15.7.5, is quoted below for reference.

> The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.

To preclude the possible occurrence of cladding failure, a limit called the cumulative damage function (CDF) has been imposed on the fuel and blanket rods for anticipated and unlikely events. In addition, a specially defined strain limit also has been imposed on these rods. The CDF is a complicated function of stress- and strain-related parameters and the duty cycle. The CDF and the strain limit are discussed in more detail in Section 4.2 of this SER. The discussion in Section 4.2 also encompasses the concern for fuel-clad mechanical interactions (FCMI) during transients. The applicants note in their response to questions (CS490.10) that FCMI is intrinsically accounted for in the calculational scheme. Therefore, no explicit FCMI model is used, but FCMI effects are included. The reader is referred to Section 4.2 for further discussion.

As mentioned previously in this section, maintaining the cladding below melting temperature and maintaining the coolant below the boiling point may not be sufficient acceptance criteria. As noted in Section 4.2, it is being proposed by the staff and staff consultants that a fuel enthalpy criterion be adopted to ensure maintenance of coolable geometry. This results from tests in which molten fuel was ejected into the surrounding coolant passages. The release of molten fuel does not automatically result in noncoolable geometry, but it is

very difficult to calculate subsequent core assembly behavior once molten fuel is expelled. Although no faulted transients are currently postulated which result in large quantities of molten fuel, future studies into basic LMFBR technology and accident initiators may change this situation.

15.1.3 Initial Conditions and Assumptions

Special assumptions necessary for unique events are discussed in the sections that address the specific events. However, the set of initial conditions and assumptions that are common to most if not all of the transient events are shown in Table 15.4 and are discussed in Section 4.4, in more detail.

To make the events analyzed as conservative as possible, the optimum condition in core life was selected by the applicants for parameters such as prompt doppler feedback coefficient and decay heat. Maximum hot channel factors were imposed as discussed in Section 4.4. In addition, the following conditions and assumptions were used:

(1) The maximum worth control rod was considered "stuck" for events terminated by either the primary or secondary control rod system.

(2) A 200-msec delay existed between trip signal and control rod insertion.

(3) Additional allowances were made for instrument error and control dead band.

15.1.3.1 Evaluation

The staff and its consultants have reviewed the initial conditions and assumptions and find them acceptably conservative and appropriately applied. The assumptions and initial conditions are generally in accord with the Standard Review Plan where appropriate. Specific satisfaction of regulatory guides and branch technical positions have been noted in the respective event sections.

15.2 Reactivity Insertion Design Events

15.2.1 Introduction

Proposed design-basis accidents associated with the insertion of reactivity during operation are listed in Table 15.5. Events that are initiated by identified phenomena and component malfunctions and nonmechanistic events such as simple ramps and step reactivity input have been considered. The identifiable events that can insert positive reactivity in the core are a malfunction of the control systems, gas bubbles through the core (positive sodium coefficient) and fuel movement.

Although gas bubbles passing through the core are analyzed, design features are provided to preclude gas entrainment. These features include continuous bleed of small bubbles from the system, a vortex suppression plate at the reactor free surface, holes in the core support cone, and a pump design which minimizes the generation of small bubbles. Gas bubbles through the core are, therefore, considered an extremely unlikely event.

The bounding event for reactivity insertion events was selected by the applicants to be a step reactivity input of 60 cents as a result of the safe shutdown earthquake (SSE). The event is assumed to result from the radial gaps between fuel assembly channels failing to follow design behavior as the reactor is brought to power. This situation would require a systematic azimuthal rotation or "cocked" position and high coefficient of friction for the assemblies. For this to happen, the coefficient of friction must be high enough to retain the "cocked" position for the channels until full power is attained.

At full power, the SSE is assumed to be experienced. It is further assumed that the earthquake has caused the loss of offsite power 0.5 sec earlier so that because of loss of power the pumps are 0.5 sec into their flow coastdown before the shock causes sufficient force to release the channels, allowing them to return to a more reactive configuration. This event is assumed to occur at the worst time in assembly life (beginning of equilibrium cycle). The reactivity insertion occurs in essentially a step fashion and is sufficiently bounded, including uncertainties, by a reactivity value of 60 cents. The insertion of the control rods is assumed to be retarded by the SSE as well as the most reactive rod failing to scram.

Another class of reactivity insertion event includes events initiated by either operator error or the failure of a controller on the reactor control system. A single rod of the primary control rod system would be inadvertently withdrawn. The accidents of most interest are very low reactivity ramp rate insertions, because the higher ramp rates cause the overpower trip to be reached before substantial energy is generated in the core and the temperatures are not as high. In the low reactivity ramp rates, considerably more energy is generated before reactor trip occurs. Control rod withdrawal events are unlikely however, because it is necessary to postulate the failure of a number of blocking functions in the reactor control system for these events to progress, such as high flux blocking circuit, flux/flow mismatch blocking circuit, rod bank position limiter circuit, and single rod out-of-alignment blocking circuit. (See Section 7 for reactor control system detail.) In addition, no intervention by the reactor operator was allowed in the analysis.

15.2.2 Evaluation

From examination of Table 15.5, the bounding event with regard to the magnitude of reactivity inserted is the SSE-initiated reactivity event. The uniform radial expansion coefficient from refueling temperature to hot full power is calculated to be -58.4 cents (PSAR, Section 4.3). Although not related precisely to the SSE-induced reactivity inserted, this radial expansion coefficient does give assurance that the 60-cent reactivity value is a reasonable bound, even including uncertainties. In operating practice, a reactivity variance of 60 cents would be quickly noticed as a deviation from expected control rod positions in all operations except the nuclear preoperational testing phase in which the true "as built" coefficients are not precisely known. Even at the preoperational stage, if the predicted value differed so markedly from the measured values, it is likely startup would be delayed until the situation was resolved.

The staff has had its consultants calculate this SSE reactivity insertion event using a totally independent calculational scheme. The staff's consultants calculated the power history for the heterogeneous core using computer code SAS-3A. The power histories and details were used as input to COBRA to calculate the

cladding temperatures. Results of these calculations are compared with the applicants' in Table 15.6. The results of both the applicants' and the staff consultants' calculated clad temperature and sodium temperature satisfy the acceptance criteria for extremely unlikely events as given in Section 15.1.

The staff has evaluated the possibility that the 60-cent SSE event is not actually the most severe reactivity insertion event from the standpoint of challenging the fuel limits. For rapid reactivity insertions, the SSE transient clearly bounds any events that can result in a rapid insertion of reactivity. The staff's consultants have calculated the 2-cents-per-second reactivity ramp insertions using the SAS-3A/COBRA combination as part of the confirmatory calculations. From these calculations, the staff could confirm that the 60-cent SSE event did bound the reactivity ramp insertions of 2 cents per second or more for the heterogeneous core.

There are two concurrent conditions, however, which require careful consideration of the slow reactivity ramp addition events. These conditions are first that small ramp reactivity insertion rates at or near power bring the plant very slowly to the limits before PPS termination and come very close to violating the acceptance criteria on cladding temperature (see Section 15.1), for example, inadvertent control rod withdrawal at power. This in and of itself might not be a condition of concern, but investigators of large breeder reactors that have enrichment and/or core composition variations have found that the neutronic transient behavior can deviate significantly from the normal assumptions made concerning the point kinetics model (Orechwa, 1980, and Su, 1980). Although a full discussion of the phenomenon is inappropriate in this document, a quote from Orechwa (1980) describes the situation most adequately.

> The point-kinetics neutronics model is almost universally employed in transient analysis studies in support of reactor designs. This model assumes that the spacial and spectral distribution of the neutron flux remains fixed in shape but the magnitude of the flux changes in response to system perturbations such as rod motion or temperature changes. In fact, the reactivity change also modifies the spacial and spectral flux distributions which in turn yield altered reactivities and local power densities. Such space-time effects are neglected in the point-kinetics models.

This phenomenon has been given the designation of "flux tilting" by the authors (Orechwa, 1980, and Su, 1980).

It should also be noted that if the phenomenon is significant for CRBR, it also affects the zonal application of doppler feedback and may have implications for flux detection outside the core.

The applicants in a meeting with the staff discussed the concern and stated that the effect on CRBR was not significant. The applicants made a commitment (by letter of December 6, 1982 reporting that meeting) to assess and include the impact of "flux tilting" on the appropriate transients in the FSAR. On the basis of the applicants' commitment to include the analysis in the FSAR, the staff considers this commitment and fallback alternative of reduction of core power level an adequate resolution at the construction permit stage.

The staff and its consultants verified that the initial conditions and assumptions of Section 15.1.3 of this SER were used on all the analyses and were modified where appropriate (i.e., retarded rod insertions for OBE and SSE) and were further verified by the staff's confirmatory calculations. The most conservative time-of-life conditions were also evaluated and selected to give maximum or minimum doppler effect depending on what is more conservative.

Although acceptable, the applicants have evaluated these events in a manner which is not preferable. The evolution of the design from a homogeneous core to the heterogeneous core resulted in an extensive evaluation of the homogeneous core configuration including accident analysis. When the core configuration was changed to the heterogeneous core, the applicants did not reevaluate all events but simply evaluated the differential changes which resulted in a large body of the PSAR concerned with events associated with the homogeneous core and only a small body of information on the heterogeneous core. Refer to Table 15.7. Although this approach is acceptable at the PSAR stage, for the FSAR a full evaluation should be made for the heterogeneous core and the confusing partial comparison of heterogeneous and homogeneous cores should be eliminated. The review was performed on the information presented and in light of the foregoing comments it was possible to make the review and draw comparative conclusions as to the satisfaction of acceptance criteria, and so forth.

The analytical tool used extensively by the applicants is the FORE-2M code, a coupled neutronics point-kinetics model and thermal-hydraulic computer code. FORE-2M has been verified in a variety of comparisons with experiments, closed form analytic solutions, and other codes (Coffield, 1982). The code is complex with many options and is used extensively in the design analysis of CRBR so that additional verification such as comparison with overpower test data as recommended by the staff's consultants will add extra confidence in FORE-2M as a calculational tool. The staff concludes, however, that the FORE-2M code is sufficiently verified for the CP stage.

As can be seen when examining Table 15.5, all of the events are covered by control measures in the PPS, do fall within the umbrella transient as outlined by the applicants, and are confirmed by independent calculations.

In reviewing the CRBR PSAR transient calculations, it was found that the PSAR was ambiguous with respect to fuel clad mechanical interactions (FCMI). The applicants corrected that ambiguity by their response to Question CS 490.10. The applicants noted that the FCMI are included in their calculational scheme. This is discussed in more detail in Section 4.2.

In addition, since insufficient information has been presented to justify operation with less than three loops, the analysis of the accident "inadvertent startup of a shutdown coolant loop," that is, "cold sodium insertion" has been included, but is presented only for completeness because the applicants are not requesting, nor is sufficient information available, to justify less than three-loop operation.

The applicants' proposed design-basis overpower events were reviewed following the Standard Review Plan (SRP) where appropriate. Because the SRP pertains to the review of LWRs, it has limited applicability to CRBR. For these events, the review procedure outlined in the SRP was applicable. The acceptance criteria

adopted in the SRP include a subset of the GDC. The staff used the comparable subset of PDC, discussed in Section 3, in its review. The applicability of determining setpoints and analytical parameters used as input for analyses of the overpower events were suitably conservative and in accord with the recommendations of RG 1.105.

The staff found these analyses to be in accord with the intent of, and form part of the body of information that satisfies the following CRBR principal design criteria: PDC 8, 11, 18, 25, and 57.

15.3 Undercooling Design Events

15.3.1 Introduction

The analysis and discussion of whole core undercooling events for CRBR is presented in PSAR Section 15.3. Local undercooling design events have been considered in Section 15.4, "Local Fuel Failures." As in other sections the events have been divided into the three respective categories: (1) anticipated events (those events of moderate frequency), (2) unlikely events (those events with infrequent incidence), and (3) extremely unlikely events (limiting faults). The cladding limits previously discussed in Section 15.1 of this SER were used by the applicants as criteria to determine whether the fuel performance limits are met in the course of the undercooling event transient. In the transient analysis, the applicants required that one plant protection system meet the limits applicable to the event classification. If the primary system failed to trip, the secondary plant protection system was required to meet the cladding temperature limits applicable to the next event classification which is less probable. In general, the conditions were initiated from the conservative thermal hydraulic conditions of Section 15.1.3. The conditions of Section 15.1.3 were supplemented by:

(1) maximum decay heat depending upon operating cycle including uncertainties
(2) the most rapid flow coastdown
(3) most conservative conditions with respect to doppler coefficient and cycle conditions

In the applicants' PSAR analyses, the following events were analyzed or considered bounded by a loss-of-offsite-power transient:

(1) partial loss of reactor coolant flow including trip of pumps and pump seizures

(2) loss of normal and/or emergency feedwater flow

(3) loss of intermediate coolant flow

(4) loss-of-coolant accidents resulting from the spectrum of postulated pipe breaks within the reactor coolant system

(5) spectrum of postulated intermediate coolant system breaks, inside and outside containment

(6) turbine trip with coincident failure of turbine bypass valves to open

(7) failure of a few steam generator tubes

(8) spurious reactor trip

(9) large leak (rupture) in steam or feedwater piping

(10) loss of normal shutdown cooling system

Table 15.8 shows several accidents delineated in WASH-1302 (USAEC) which are not considered for various reasons. "Station blackout" or the loss of both onsite and offsite power is considered a beyond the design-basis accident, but is accommodated in the design and is discussed in Appendix B.

The loss of offsite power has been used by the applicants as the bounding event for the design-basis, whole-core, undercooling events. Sixteen different whole-core undercooling events were analyzed. They were covered either by the bounding loss of offsite power or the fact that events that occur in the intermediate heat transport system and beyond are thermally removed from impacting the core by the long transit time in the sodium loops. The plant protection system will sense these events and scram the reactor in advance of the thermal impact, so all undercooling events in the secondary, the applicants assert, will be thermally benign.

The results in Table 15.9 address the short-term undercooling transient results. All events, in the long term, require the removal of decay heat (see Table 15.3). The long-term decay heat removal is described in Section 5.5 and Appendix B, "Unresolved Safety Issues," under "Station Blackout."

Before discussing the loss-of-offsite-power bounding event, it should be noted that CRBR will incorporate engineered safety and design features which are related to undercooling events. Although not included in the PSAR Chapter 15, these are evaluated in that context in the next section. These features are related to PHTS sodium inventory control so that the reactor core will remain covered with sodium under all credible conditions. In order to limit the loss of sodium from the PHTS, guard vessels will be placed around the reactor vessel and all the major components in the loop and will be placed and sized so that leaks in the enclosed guarded volume will equilibrate at levels which will still retain sufficient sodium covering the core called minimum safe level. The piping between guard vessels will be elevated to preclude siphoning.

The loss of all offsite power will trip all primary and intermediate sodium pumps, commencing a flow coastdown. The emergency diesel generators will be started automatically to provide power for the pony motors and auxiliary feedwater pumps for decay heat removal. When the flow will begin to decrease, the fuel cladding temperature with some thermal lag will begin to increase, because the reduced sodium flow will result in increased temperature rise across the core. At 0.5 sec after the loss of electrical power, the plant protection system would scram the primary control rod system on a loss of pump electrics. If the highly unlikely situation of primary control rod system failure were to occur, the PPS would trip the secondary control rod system on flux-total flow signal 2 sec after losing electrical power. The reactor would scram and flow would continue to coast down until it reached pony motor speed (~7% flow) at which time the pony motors would engage and provide continued cooling flow for

normal decay heat removal. The core hot spot cladding temperature will peak about 1 sec into the transient for primary control rod system shutdown and about 3 sec into the transient for secondary control rod system termination. The cladding hot spot will continue to decline as the core power generation is reduced, and the stored energy will be removed by the decreasing sodium flow until it approaches long-term decay heat removal conditions.

15.3.2 Evaluation

The loss of offsite power clearly bounds even the seizure of a single primary pump. The event of real interest, therefore, is the loss of offsite power. The staff and its consultants have reviewed the selection of events, the calculational methodology, and the calculational models. The initial conditions for all the event calculations were the acceptably conservative conditions previously discussed in Sections 15.1.3 and 4.4. These conditions were supplemented by using the most rapid flow coastdown and the most severe operating cycle conditions, beginning-of-equilibrium cycle which has the highest power fuel assembly and smallest doppler coefficient. The conditions of Section 15.1.3 were generally applicable to most of the undercooling events analyzed. For some events, these conditions were either augmented by additional conservative assumptions or some conditions were deleted as appropriate.

Noting that the largest fission product inventory could exist in the ex-vessel storage tank (EVST), the staff carefully reviewed the design of the EVST. The staff noted that the EVST will be adequately protected with redundant and diverse cooling paths which will be capable of operating even under "station blackout" conditions by a natural circulation heat rejection path. The features of the design and their separation and isolation should remove the EVSTs as a credible radiological source for accidents. These features are discussed, in addition, in Section 9.1 of this SER.

The applicants analyzed most of the events using the DEMO computer code. The DEMO computer code is a simulation program designed to calculate thermal-hydraulics transients for CRBR. DEMO has been verified by comparing calculations of identical events with other verified thermal-hydraulic codes. The staff finds DEMO to be an acceptably verified computational model for thermal-hydraulic transient calculations. The staff has had confirmatory calculations made by its consultants using a code called Super System Code (SSC) which performs calculations similar to the DEMO code. The SSC results were compared with DEMO results and good comparison was attained. These calculations are discussed in Sections 4.4 and 15.3.2.1. Some special sensitivity studies were made using FORE-2M code discussed in Section 15.2.

Since none of the undercooling events exceeded the acceptance criteria of Section 15.1.2, the staff finds them acceptable. As stated above, because of the transit delay in the heat transport system (HTS), the core will be well isolated from potential undercooling events in the intermediate heat transport system (IHTS), steam generator/superheater, or power conversion portions of the plant. For the loss of offsite power, the maximum cladding temperature of the fuel will be 1,455°F, for the inner blanket assembly it will be 1,471°F, and for the radial blanket assembly it will be 1,478°F. These are within the acceptance criteria of Section 15.1.2. The applicants estimate that the conservatism results in 175F° temperature margin on the hottest fuel assembly.

Special component considerations relative to structural integrity or other factors related to component design are considered in other sections, but warrant some brief mention since the conditions are accident related. For example, large sodium-water reactions in the steam generator system are of special interest. This event, its mitigation, and its inclusion in the system design are discussed in Section 5.6. As an undercooling event noted in Table 15.9, the event to the rest of the plant will appear no more severe than a normal shutdown. The reason for this is that the plant protection system (PPS) will sense the event on steam flow-feed flow mismatch in less than 4 sec and will scram the reactor. The temperature front will arrive at the core inlet in approximately 120 sec, as confirmed by staff consultants' calculations, after the event with no effect on a subcritical reactor. A special mitigative system, sodium water reaction pressure relief system (SWRPRS) discussed in detail in Section 5.5, will be included at the steam generators/superheaters for sodium-water reactions.

Briefly, the following assumption is made. The event will be initiated by a small water-to-sodium leak which has gone undetected or ignored. This small leak will proceed to raise the pressure of the IHTS loop to just under the rupture pressure of the rupture discs protecting the IHTS expansion tank. This small leak also will cause the conditions that will fail a neighboring tube as an equivalent full double-ended guillotine rupture. This, in turn, will fail a second tube 1 sec later, followed by a third tube 2 sec later. This energetic sodium-water reaction will generate a pressure pulse which will be relieved by failing rupture discs in the SWRPRS. Sodium reaction products and hydrogen will be expelled from the IHTS into SWRPRS where hydrogen will be separated from the particulate and liquid matter. The hydrogen will be vented to a flare stack and liquid and particulates will be contained in the reaction products separation tank under an inert atmosphere. Operation of the rupture discs automatically will initiate isolation and depressurize the water side of the steam generators to limit damage to the system. Inadvertent actuation of the SWRPRS and small sodium leak consequences obviously will be bounded by this event. Pressure pulses and their anticipated frequency have been accounted for in the duty cycle for design of the components in the IHTS.

The applicants' proposed design-basis undercooling events were reviewed following the Standard Review Plan where appropriate. Because the SRP pertains to the review of LWRs, it has limited applicability to CRBR. The application of determining instrument setpoints and analytical parameters is appropriate to both systems. The staff found that the parameters used as input for analyses of the undercooling events were suitably conservative and in accord with the recommendations of RG 1.105.

The staff found these analyses and results to be in accord with the intent of, and form part of the body of information that satisfies the following CRBR principal design criteria: PDC 4, 8, 11, 12, 13, 18, 26, 27, 31, 32, 35, 52, and 58.

A special discussion related to HTS leaks and sodium inventory follows in Section 15.3.2.1.

15.3.2.1 Leaks in Primary and Intermediate Heat Transport Systems

Special discussion must be given to leaks in the primary and intermediate heat transport systems. Considerable effort has been devoted to analyzing and considering leaks in the heat transport system both for the fast flux test facility (FFTF) and CRBR. The integrity of the sodium piping will impact three distinct areas: (1) undercooling of the core; (2) sodium fire effects (temperature, pressure, aerosols and released radiation); and (3) design bases for inerted cell liners and sodium fire suppression devices.

The applicants have proposed a piping integrity argument which has been referred to as leak-before-break and is discussed in more detail in Section 5.11. The argument is summarized briefly as follows. The stainless steels used in the heat transport systems are materials which are characterized by high fracture toughness and corresponding large critical flaw size. A critical flaw is defined as the length of axial crack necessary before it changes from a linear parallel-sided crack to an open, nonparallel-sided "burst" (large area) configuration. In addition, a negligible growth rate of postulated defects and the probability of through-wall growth rather than elongation of defects are characteristic. These characteristics and attention to quality assurance detail, the applicants argue, would only (if at all) lead to small leaks occurring which could be detected by the sensitive leak detection system (see Section 5.7). Early detection would allow actions to be taken long before the small leaks developed into substantial leaks. It should be noted that the pipe-integrity argument actually only implies that the so-called instantaneous double-ended guillotine offset shear as a pipe break concept is so improbable for this piping that it may be considered incredible and excluded as a design-basis pipe leak. It does not remove the responsibility of knowing how the core responds to various size leaks and it does not eliminate the need to establish a design-basis leak.

In FFTF (NUREG-0358), this argument was accepted for the primary system piping providing that:

(1) Representative preservice and in service inspection would be performed.

(2) A plant aerosol leak detection system would be installed and verified.

(3) An adequate material surveillance program in a prototypic environment to provide advance warning of unanticipated material degradation processes would be initiated.

For the current evaluation, the leak-before-break pipe-integrity argument for the PHTS was reviewed and found acceptable by the staff and is discussed in Section 5.11.

With this background, the reader may consider piping leaks in the IHTS. The staff concludes that the thermal impacts on the core for the leaks in the intermediate heat transport system (IHTS) will be bounded by the events considered in Section 15.3. The IHTS is weakly coupled to the reactor core in a thermal sense because of the long transit time around the composite of the primary and intermediate heat transport loops, as discussed previously. Most events in the IHTS will manifest themselves in about the same way with respect to the core

whether large IHTS leak, seized IHTS pump shaft, spurious IHTS pump trip, or even loss of normal shutdown cooling system occurs. The leak spectrum in the IHTS undercooling events is, therefore, relatively unimportant and has little consequence with respect to piping integrity in the IHTS system. Care, however, must be used with respect to sodium fires and fire protection systems in the IHTS, and these are addressed in Sections 15.6 and 9.13.2.

In the PHTS, leaks with regard to undercooling of the core have been analyzed by staff consultants. The results of these analyses have shown that instantaneous double-ended guillotine (DEG) pipe rupture between the reactor vessel inlet nozzle and the check valve of the cold leg would result in exceeding the coolant saturation conditions (boiling) in some channels in the reactor core for a few seconds. Moderate-sized pipe leaks (leak areas less than 2 ft^2) would result in no sodium boiling in the core regardless of location in the PHTS. In locations between the core exit nozzle and the cold-leg check valve only very large (DEG pipe rupture) leaks were capable of approaching boiling in some channels in the core (Albright, 1976). Leaks in sodium piping systems have occurred and these leaks are tabulated in NUREG-0358. Both the leak-before-break and leak experience argue that the small leak is more probable than the large leak. The DEG leak is a nonmechanistic concept to obtain an upper bound for postulated pipe leaks to examine the thermal-hydraulic responses of the system. However, no credible mechanism has been identified to produce such a severe pipe break. Therefore, the staff concludes that the DEG pipe rupture is considered to be an extremely low probability event and the DEG pipe rupture can be excluded as a design-basis event.

As discussed above, large leaks are not credible and small leaks have no undercooling impact. Furthermore, small leaks will be detected readily by the sensitive leak-detection system, allowing timely manual reactor shutdown. The remaining concern, therefore, is that there will remain sufficient sodium inventory so that the reactor core will be covered with sodium and decay heat will be removed. Since the sodium coolant temperature will be several hundred degrees below its saturation temperature, the only way sodium can be lost from the system is as a liquid through a leak. Guard vessels are designed that will surround the reactor vessel, intermediate heat exchangers, and pumps. These guard vessels are designed with such volumes and elevations that, if a leak were to develop in these guarded volumes, ultimately the respective levels would equilibrate so that the sodium level in the vessel would be above the exit nozzles and the two unaffected loops would still be functional. It helps in understanding this concept to recall that considerable sodium inventory will be in the vessel above the exit nozzles (33,200 gal, PSAR Section 5.3-10) and available in the overflow tank (~11,000 gal) if needed. Between the guard vessels, the piping will be elevated so that a leak developing in these sections could not establish a siphon action which would siphon the sodium from the reactor vessel. Finally, redundant pump trips will keep from pumping sodium out (see Section 7.2.2.6). The pump head for a pony motor will be insufficient to lift the sodium to a leak in the piping outside the guard vessels. The staff believes these features will protect CRBR from a loss-of-coolant accident both for short- and long-term considerations. The leak-before-break concept does present some complication in establishing a design-basis leak for sodium fires and the inerted cell liners as discussed in Section 15.6.

15.4 Local Fuel Failures

15.4.1 Introduction

The analysis of local fuel failure events which may lead to failure propagation is the subject of Section 15.4 of the CRBR PSAR. Because of the technological similarity between CRBR and FFTF, there is a body of analysis and experimentation which has common applicability. It is noted in passing that in 1978, when the staff had completed its review of FFTF local fuel failure events (NUREG-0358), the staff concluded:

> Fuel pin failures which might occur under various plant operating conditions, including design transients, are unlikely to create conditions under which significant fuel failure propagation within a subassembly would occur.

The basic driver fuel assembly of FFTF has much geometric and physical similarity to the CRBR fuel assembly, so that the conclusions drawn of FFTF local fuel failure reinforce the conclusions drawn about CRBR local fuel failures.

In the design and operation of a nuclear facility, it is necessary to explore hypothetical events to preclude their occurrence or to mitigate the consequences should those events occur. Perhaps more compelling are the lessons learned from events that have occurred at operating facilities. For example, the most notable local failure event was the damage to two subassemblies in the Enrico Fermi reactor when the inlet flow was plugged. The concern does not just exist for the occurrence of local fuel failures (such as concern that the Fermi subassembly would melt) but that these events may initiate propagation of additional failures in neighboring subassemblies, which might then lead to a rapid propagation through a substantial portion of the core. Therefore, not only the local fuel failure event itself is closely examined but a considerable amount of experimental work has been carried out to determine if flow blockage or local failure events will propagate to the surrounding fuel pins. From these experiments, it is possible to infer what time response is necessary for the detection of the local failure in order to shut down the reactor.

The fuel element failure propagation accidents are usually divided into four initiator categories:

(1) Stochastic fuel pin failures resulting from, for example, manufacturing defects

(2) Incorrect loading of fuel enrichments resulting in local overpower--in the heterogeneous core for CRBR this translates into incorrect removal of an inner blanket assembly, since there is only a single enrichment in the fuel assemblies. There are, however, inner blanket assemblies which are removed and replaced with six fuel assemblies at the end of each odd-numbered cycle. A mechanical discriminator post system is provided so that blanket and fuel assemblies cannot be misloaded. The discriminator post design is discussed in Sections 4.4 and 9.1 of this SER. However, even though highly unlikely, an event considered credible is the failure to replace an inner blanket assembly when required.

(3) Small blockages forming in core subassembly coolant channels

(4) Large planar blockages covering a major portion of the subassembly cross-sectional area

The ultimate concern is whether or not these initiators could lead to subassembly failure propagation and then rapidly to whole-core involvement.

15.4.2 Evaluation

For the initial (two cycles) operation of CRBR, operation with failed fuel is not likely; however, the applicants state that the ultimate objective is the acceptance of operation with failed fuel (1.0%). The staff's position on operation with failed fuel is that when failed fuel is detected by the release of fission gas, the leaking fuel assembly will be located and removed at the next scheduled or unscheduled shutdown. If a signal on the delayed neutron detector (DND) system above a predetermined reference is received either in conjunction with fission gas detection or alone, the reactor will be immediately shut down and the failed assembly removed. As discussed in Section 7.5.2.5, given the staff modifications the capability to detect fuel failure for CRBR will be adequate. The potential problems associated with operation with failed fuel or with very slow deterioration of those failures are thus not germane to the present discussion. The emphasis in this section then is on the question of rapid fuel pin failure which would lead to additional fuel pin failure and the potential for propagation to surrounding subassemblies.

Analytical and experimental work has been performed, and is continuing, to verify that the conditions which might arise during plant operations are unlikely to cause pin-to-pin failure propagation. Aspects that have been analyzed in the PSAR include the effects of dimensional changes, wire wrap failure, fission gas release from pin failures, molten fuel release from pin failures, and other similar conditions that could lead to local flow disturbances or mechanical loadings. The results of this work indicate that there should not be a significant potential for failure propagation beyond a few pins at worst under the anticipated operating and faulted conditions.

Experimental and analytical work also has been conducted on the effects of blockages within a subassembly. The results indicate that substantial blockages at the nonfuel inlet or outlet regions do not cause overheating, that inert planar blockages covering a few cooling channels in the fuel region do not cause any significant overheating, and that small heat-producing (fuel material) blockages do not cause significant overheating of adjacent areas. Recent testing on local failure accommodation (Ragland et al., 1982) indicates that failure propagation does occur within a fuel assembly when that area is subjected to a very severe local heat-generating blockage. However, within the assembly, this propagation occurs over a time interval of several minutes. A blockage of the type used in this test is large enough to be detected by the CRBRP delayed neutron detector (DND) system in the time interval over which propagation occurs. The staff and its consultants evaluated the time sequence of this severe disturbance and the evidence that there was no indication of propagation outside the originating subassembly.

The staff evaluated the design approach to preclude blockage of the inlet core region utilizing multilevel ports, strainers, and debris barriers as detailed

on Figures 15.1 and 15.2 and discussed in detail in Section 4.4. The applicants stated that there are design features to prevent any credible identifiable flow blockage mechanism and, therefore, flow blockage of one or more subassemblies is not a design basis for CRBR. Accordingly, the fuel failure monitoring systems associated with CRBR (PSAR Section 7.5.4) were proposed to be classified as a nonsafety-related system.

From the staff's review of the CRBR design, the staff and its consultants have concluded that fuel pin failures that might occur under various plant operating conditions, including design transients, are unlikely to create conditions under which significant fuel failure or fuel failure propagation within a subassembly would occur.

The inlet flow modules will be designed to preclude most foreign materials of any significant size which might be present in the system. The analyses of various local faults show that the assemblies can tolerate substantial blockages without detrimental effects. Even the failure to remove an inner blanket assembly (the only fuel-loading error not precluded by design) will not lead to serious potential for propagation. This is because there is a high likelihood that the reactivity deficit imposed by failure to replace the inner blanket assembly will be detected. The design is such that the inner blanket assemblies will be replaced to obtain sufficient reactivity for the following run. The exit thermocouples would indicate a lower exit temperature or trend. If the loading error were to go undetected for an extended period, which is unlikely, the assembly might fail on overpressure of the fission gas plenum. Therefore, the staff concludes that the potential for fuel failure propagation in CRBR is very small.

In the past, it has been the staff's desire to have the applicants provide detection systems to limit failure propagation even if the potential is small. This was stated in letters of May 6, 1976 and November 11, 1978. In fact, a new principal design criterion (PDC 59) on failure propagation (see SER Section 3.1) was formulated to acknowledge the importance of this consideration for CRBRP. The intent of this design criterion is to require design features and/or detection systems to limit any failure propagation to within the assembly in which the fault was initiated. In this way, local failures are prevented from becoming initiators of whole-core events.

Staff evaluation of the potential for fuel failure propagation is based upon

(1) a review of the data presented by the applicants

(2) consideration of the amount of operating experience with LMFBRs, especially with failed fuel

(3) ability to meet the CRBR design criteria on failure propagation

The staff concludes that, on the basis of LMFBR operating experience to date and the ex-reactor testing and analysis presented, the probability of a local failure leading to propagation is very low and the probability of that propagation being rapid (propagating through an entire assembly in a matter of seconds) is remote. However, at this time, sufficient data do not exist to entirely

exclude the possibility of fuel and blanket assembly failure propagation without additional measures. Domestic LMFBR experience on operation with failed fuel is limited, which leaves some uncertainty that the failure mechanisms and events analyzed by the applicants are, in fact, the bounding cases.

It is the staff's opinion that additional operating experience with FFTF (whose fuel design closely resembles CRBR's), completion of the run beyond the cladding breach program in EBR-II, and data on the behavior of failed fuel under transient conditions are necessary before unrestricted operation with failed fuel or blanket assemblies should be permitted.

Accordingly, the staff concludes that at this stage a means to detect assembly faults which have the potential for leading to propagation should be provided in the design. The staff considers the delayed neutron detection (DND) system as having the best potential for rapid detection of the failures which could lead to propagation. This judgment is based on the following:

To limit failure propagation, it is considered necessary to detect failures rapidly that could lead to an uncoolable geometry situation or that have been caused by an uncoolable geometry situation. When such situations occur, cladding failure and possibly fuel motion may result. To ensure that the resulting failures do not lead to further failures, it is necessary to detect those failures which could involve fuel motion or cladding melting. Cladding failures which release fission gas only through small breaches have been shown by testing and analysis not to lead to further failures (PSAR Section 15.4). Failures involving fuel motion or cladding melting are expected to also involve exposure of fuel to the coolant, thus providing a signal for the DND. Since the DND signal occurs rapidly (30-50 sec) upon fuel exposure to sodium, it has the best capability for rapidly detecting failures that have been caused by or can lead to an uncoolable geometry situation.

Therefore, the staff feels that the DND system should be considered as an instrumentation system important to safety, should at a minimum be designed to survive the operating basis earthquake conditions, and plant operation should be contingent upon at least two of the three PHTS loops having their DND systems in operation. As part of developing the FSAR, appropriate limits on the DND signal should be established which would result in a plant shutdown or scram upon indication of a situation where coolable geometry can no longer be certain. Further discussion may be found in Section 7.5.2.5 of this SER. In addition the system should be designed in accordance with IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," so as to provide flexibility to install this signal into the plant protection system if further experimental work indicates the necessity for more rapid automatic shutdown. Based on available evidence, however, it appears highly unlikely that rapid failure propagation could result from any of the identifiable events for CRBR.

There are other nonsafety-related systems available which will support the detection of local fuel failures should they occur. These systems are redundant exit thermocouples, reactor cover gas radiation monitoring system, and the failed fuel locator system. These systems are discussed and evaluated in Section 7.5. The DND system is the most important system to detect local fuel

failures which may have the potential to propagate. The collection of nonsafety-related systems will provide added assurance, however, that events which might result in fuel failure propagation will be detected.

Quality control in design and fabrication of the fuel assemblies is intrinsic to precluding local faults in CRBR. The design and fabrication procedures may be readily reviewed for acceptability, but it is prudent to obtain all reasonable assurance that the acceptable procedures have been followed. To this end, the staff is requiring a gas flow test of each completed fuel or blanket assembly to verify that no foreign material has been inadvertently left inside the assembly to form a major blockage. This test may also disclose minor anomalies. Further discussion of this test appears in Section 4.4.

The staff finds that, with the appropriate reclassification of the DND, the provisions to reduce the potential for local fuel failure, the events analyzed, and the consequences of these events are in accord with the intent and form part of the body of information that satisfies the following CRBR principal design criteria: PDC 8, 11, 59, and 60.

15.5 Fuel-Handling and Storage Events

15.5.1 Introduction

There is a category of accidents associated with power reactors which involves the handling and storage of irradiated fuel elements. Accidents involving the irradiated fuel have the potential, at least, of releasing significant quantities of radiation to the surrounding environment. Section 15.5 of the CRBR PSAR is devoted to analyzing these events. In general, the acceptance criterion for these events is that site-boundary or low population zone dose rates for the released radioctivity not exceed the allowable dose guidelines prescribed in 10 CFR 20 or one-tenth of those in 10 CFR 100 for the extremely unlikely events. The accidents analyzed are

(1) fuel assembly dropped within reactor vessel during refueling

(2) fuel assembly insertion into an occupied position

(3) single fuel assembly, cladding failure, and subsequent fission-gas release during refueling

(4) cover gas release during refueling

(5) loss of power during refueling (see Section 9.1)

(6) cooling system failure (see Section 9.1).

In addition to the fuel-handling and storage accidents, there are certain other radioactive releases that have been included in this portion of the PSAR such as (1) crane load impacts to the reactor closure head, (2) collision of the ex-vessel transfer machine (EVTM) with the control rod drive mechanisms, and (3) drop of the overhead crane load on the closure head. The results for these various events and their radioactive releases are tabulated in Table 15.10. The applicants believe these events umbrella any other fuel-handling events.

15.5.2 Evaluation

The staff reviewed the analyses of the events and agrees that the bounding event for this category of radioactive releases is the release of cover gas during refueling. Cover gas may be released during refueling by mechanical damage, by improper sequencing of refueling motions, or by separation of the auxiliary handling machine from an open floor valve during a seismic event. The earliest time this could occur is 30 hours after shutdown, during which period the radioactive argon processing system has been in normal operation, cleaning up releases from the assumed 1% failed fuel.

The cover gas is considered released instantaneously to the reactor containment building (RCB) and reactor service building (RSB) and then to the environment as if the railroad door to the RSB was open in violation of procedures. The maximum 2-hour site-boundary dose is estimated at 4.4 mrems whole body. With respect to radiological consequences for all the fuel-handling and storage events analyzed, this was the bounding event. The staff's independent evaluation confirms this bounding event selection (see Appendix 15B). The staff judged the assumptions and analyses to be sufficiently conservative. The radiological consequences of both the staff's independent evaluation and applicants' calculations are within the proposed dose guidelines of 10 CFR 20, which are the acceptance criteria for unlikely events discussed in Section 15.1.2 of this SER. The staff's independent calculation of this event is discussed in Appendix 15B of this SER.

In order to gain understanding of why the cover gas released during refueling bounds the other events, some familiarity with the refueling scheme is helpful.

For a more detailed discussion, see Section 9.1. Briefly, new fuel will be received and, after the proper inspections, will be stored in the ex-vessel storage tank (EVST) by means of the ex-vessel transfer machine (EVTM) for preheating and storage. After the assembly has been preheated inside the EVST, it will be transferred to a sodium-filled core component pot (CCP) and stored. The assembly will remain and be transferred in the CCP until it is inserted into its proper core location upon refueling.

After the reactor has been shut down, cooled to refueling temperature, and the control rod drive lines have been disconnected and repositioned, the equipment hatch between the RCB and the RSB will be opened. After the proper preparation, which involves balancing pressures in the RSB and RCB and attaining the proper flows, two adapters and floor valves will be installed on the reactor head. The fuel transfer and in-vessel transfer machine (IVTM) port plug will be removed using the EVTM and the auxiliary handling machine (AHM), respectively. (AHM will handle equipment needed in fuel handling.) After some further preparation of the IVTM, the reactor will be ready for the refueling operation. The EVTM and the IVTM will work in conjunction during refueling to exchange spent core assemblies in the reactor; new core assemblies will be stored in the EVST. The operations will occur simultaneously--the EVTM will remove a new core assembly in a sodium-filled CCP from the EVST, while the IVTM will remove a spent core assembly from the core and deposit it in a CCP located outside the reactor core, but still within the reactor vessel. The EVTM will transfer the new assembly in its CCP in a second transfer position inside the vessel, will pick up the spent assembly and return to the EVST with the spent assembly

inside its CCP for storage in the EVST. The IVTM then will install the fresh assembly in the core. The spent fuel will be stored in the EVST approximately 100 days before it will be loaded into a spent fuel shipping cask for offsite shipment using the fuel-handling cell.

The fuel will always be inside its CCP unless it is inside either the fuel-handling cell or the reactor vessel. Therefore, the fuel assembly could be dropped either in the reactor vessel or the EVST. It could be possible to drop a single assembly and CCP in the EVST, but the radioactive release of this event is bounded by a single fuel assembly cladding failure and subsequent fission gas release. The assembly cladding failure event would occur as follows: Spent fuel assemblies are in the core, the IVTM, the EVTM, the fuel-handling cell, and the EVST. It is judged that the EVTM is where containment failure is most likely and could lead to the largest release from a single fuel assembly. The analysis assumes release of 100% of the noble gases and halogens in the fuel assembly to the interior of the EVTM, when it is not connected to a location with gas cleanup and subsequent diffusion of the radioactivity through the EVTM seals into the combined space of the reactor containment building and reactor service building. The diffused radioactivity would pass through the RSB filtration system, since the RSB will be maintained at 1/4-in. negative water gage pressure, and would be directed through the filtration system and discharged. The resulting doses are shown in Table 15.10.

The CRBR polar crane will service the head access area with refueling and maintenance equipment. The heaviest load identified to be handled over the reactor head is the auxiliary handling machine (AHM). The crane is of such a proposed design that the lowering speed will be 5 ft per minute. If the AHM load of approximately 100 tons were accidently lowered on the reactor head at the crane velocity limit, an impact on the reactor enclosure head assembly would be imposed. The polar crane lowering speed restriction would mitigate possible damage to head and head-mounted equipment. Any cover gas release would be bounded by the cover gas release during the refueling event discussed previously.

Finally, the radioactive release associated with the EVTM colliding with the control rod drive mechanisms (CRDMs) will be bounded by the cover gas release event previously described. The EVTM is a massive, railway mounted, shielded cask type fuel transfer machine that will be in the proximity of the CRDM only when the reactor will be shut down and the control rod absorbers will be fully inserted. Therefore, reactor criticality concerns will not be present. There are many design features that must be failed and procedural controls violated before the EVTM (see Section 9.1) would be likely to collide with CRDMs. An attempt to insert an assembly into an occupied location would require the violation of a series of procedural steps and interlock failures (see Section 9.1) in order for this event to occur. The radiologic consequences will be bounded by the cover gas release event.

The applicants' proposed design-basis fuel-handling and storage events were reviewed following the Standard Review Plan (SRP) where appropriate. Because the SRP pertains to the review of LWRs, it has limited applicability to CRBR. In the case of fuel-handling and storage events, the SRP was applicable with only minor variations. Independent dose calculations were performed for the

cover gas release event, EVTM fuel-handling (single assembly cladding failure in EVTM) event, liquid tank failure (Section 15.7.2.1.1) and the site suitability source term (NUREG-0786). The dose guidelines proposed for acceptance criteria by the applicants for these events were consistent with and more conservative than the those in the SRP.

The staff finds these events, the analyses of the events, and their consequences acceptable because they cover the mechanical variations possible and the consequences are within the proposed acceptance criteria discussed in Section 15.1.2. The staff found the selection of these fuel storage and handling events and the analyses of their consequences to be in accord with the intent of, and form part of the body of information that satisfies the following CRBR principal design criteria: PDC 52 and 53.

15.6 Sodium Fires

15.6.1 Introduction

In reviewing the accidents that could result in sodium spills and/or leaks, there is considerable overlap with other areas of the plant and a number of issues associated with the design of components and engineered safety features must be considered. Sodium fires can be divided into those associated with the noninerted areas and those in the low-oxygen-inerted cells. The noninerted areas are, for example, sodium loops outside the containment, and certain selected accidents postulated to occur during maintenance operation on items in cells that would normally be inerted but have been deinerted for the maintenance operation. The argument for piping integrity of the heat transport system (HTS), that is, leak-before-break argument, was discussed in other sections (Sections 5.11 and 15.3.2.1) and will not be reproduced here. However, some discussions of the implications of this argument should be understood. The major implication of the leak-before-break piping argument is that the upper end of the spectrum of leaks from the HTS will be virtually precluded because there are no mechanisms postulated credible enough to result in guillotine double-ended breaks. The argument has been successfully articulated for the primary heat transport system (PHTS). In the PHTS, the consequences of sodium leaks are twofold: (1) sodium fires which might exceed the design pressure of the inerted cells and (2) loss of cooling to the core during operation. The leak-before-break argument is accepted on the intermediate heat transport system (IHTS) as it is for the PHTS. Therefore, the consequences of sodium fire are different because of the higher oxygen content, but the impact on core cooling is negligible. (See Section 15.3.) The radiologic consequences of breaks in the IHTS, even under very conservative assumptions, is well within the guidelines of 10 CFR 100. The selection of a design-basis leak to set the design criteria for the fire-suppression system and other aspects of the steam generator building becomes the central issue. The discussion of each of these topics follows in the next section. The sodium spill events presented in the PSAR by the applicants are shown in Table 15.11.

15.6.1.1 Discussion

(1) Sodium Leaks in the Primary Heat Transport System

As stated in other sections of the SER, accepting the leak-before-break argument for the PHTS acknowledges that the likelihood of large sodium

leaks during operation are precluded. The applicants evaluated a spectrum of sodium leaks in the inerted cells and concluded from the standpoint of pressurization and containment of the radioactive products that the consequences of a leak up to a certain point will be limited by the amount of available oxygen within the inerted cells. Leaks larger than this size would produce no additional effect on the spill because the available oxygen would be exhausted. The consequences of fires in the inerted cells will be basically bounded by available oxygen. Detailed discussion of sodium fires are in Section 9.13.

The impact of leak size on the coolability of the core was evaluated for a spectrum of leaks and discussed in Section 15.3. It should be emphasized here that calculations have shown that sodium spills with failure of the cell liners would result in considerably higher pressures than the design pressures for the PHTS cells, so that it is important that the cell liners are designed as engineering safety features. Confirmatory sodium-fire calculations have been made by the staff's consultants and are discussed in Section 9.13.

The staff review included the design bases for the cell liners, catch pans and fire suppression decks for which the available sodium inventory that they may be subject to is more important than the leak size (i.e., the rate at which this quantity of sodium is delivered) (see Section 15.3.2.1).

(2) Sodium Leaks in the Intermediate Heat Transport System

For the IHTS as discussed in Section 5.11, the leak-before-break concept has been accepted as for the PHTS. Consistent with the leak-before-break concept, a design-basis leak for design of the steam generator building as well as the sodium-fire-suppression system was selected. The design-basis leak selected for the IHTS corresponds to a leak recommended by the staff in BTP MEB 3-1, "Postulated Rupture Location in Fluid System Piping Inside and Outside Containment." This position pertains to piping for moderate energy systems and states:

> Section II, Moderate-Energy Fluid System Piping; 3(c)2, Through-Wall Leakage Cracks
>
> > Fluid flow from a crack should be based on a circular opening of area equal to that of a rectangle 1/2 pipe diameter in length and 1/2 pipe wall thickness in width.

Therefore, the applicants' selection of that leak size for the design basis of the IHTS seems to be reasonable. Prudence, however, would dictate that the fire-suppression system be capable of taking and suppressing a sodium fire for the available sodium inventory in any respective IHTS loop, which the staff verified was the approach taken (see Section 9.13.2).

The applicants stated that this leak was postulated to occur in the 24-in. (OD) IHTS piping in cell 226 which was evaluated to be the bounding case. In addition, leakage from the PHTS through the IHX was postulated to bound the radiological releases to the environment and no operator intervention was assumed.

15.6.1.2 Containment Design-Basis Accident

The design-basis accident for the containment is a sequence in which 35,000 gal of sodium are assumed to be instantaneously discharged on to the floor of the overflow tank cell (cell 102a). The sodium would be initially 400°F and the cell would be open to the reactor containment building atmosphere through a 21-ft^2 hatch.

This nonmechanistic hypothetical event is the bounding design-basis sodium fire. Even though hypothetical, this event is to be judged, from parametric analyses, the most severe sodium fire for in-containment sodium volumes. More detailed discussions of the containment evaluation may be found in Section 6.2.1.1.

The event is conservative and was selected to optimize the challenge to the containment. Because the event requires the concurrence of (1) an instantaneous nonmechanistic tank failure, (2) maximum sodium volume in the tank (the contents of one PHTS loop), and (3) the cell deinerted and open to the containment volume, this event is considered a hypothetical event selected to challenge the containment within the design-basis envelope.

15.6.2 Evaluation

In the inerted cells, consequences of a sodium fire have been shown to be bounded by the available oxygen which will be maintained at 2% or less (Q 001.700). The likelihood of exceeding these oxygen limits in the PHTS cells is small because the level will be procedurally controlled and the inerted cells will be operated at only a small negative pressure to control tritium release. To exceed the oxygen limit would require that at least a portion of the inert gas system would fail and procedural controls would be ignored for a fairly lengthy time (the back diffusion of oxygen is relatively slow). The staff's consultants have performed overcheck calculations to verify the acceptability of the applicants' codes. The results of these comparisons are discussed in Section 9.13.2. The design of the cell liners, however, should be bounded by the amount of sodium inventory which can be spilled. Further, the staff has accepted the leak-before-break concept (Section 5.11) for the HTS piping, which excludes very large leaks as design-basis events. Therefore, the staff finds acceptable the selection of the design-basis leak as that stipulated by BTP MEB 3-1. Fracture mechanics arguments support this leak size as well (see Section 5.11) and demonstrate that it is conservative.

In the intermediate heat transport system, the result of a large sodium leak with respect to its radiological consequences would be well within even a fraction of 10 CFR 100, on the basis of conservative assumptions as noted in Table 15.11. Although the leak that has been selected by the applicants is not the maximum leak which is hypothetically possible, consistent with the arguments of the leak-before-break concept, it seems to be reasonable to select this leak based on the aforementioned NRC guidance for moderate-energy pressure systems. The IHTS, although in a more aggressive environment from a corrosion standpoint, is also non-radioactive and subject to routine visual inspection by operating personnel. Therefore, the likelihood of a large leak going undetected would be much smaller. The consequences of large leaks are related to the nature of toxicity associated with the combustion products and the fouling of safety-related

equipment rather than a radiological problem or an effect on core cooling. The staff concurs that the selection of the design-basis leak made by the applicants appears to be reasonable.

No operator intervention was assumed in any of the sodium leaks in the IHTS. In general, IHTS leaks could be mitigated considerably by operator intervention such as shutting off the pump and draining the affected portion of the IHTS loop.

The aerosol relief of the system for the steam generator building is discussed in some detail in Sections 9.13.2, 6.10, and 7.3.1. The design basis for the steam generator building is most critically predicated on the leak size. On the basis of BTP MEB 3-1, the staff finds acceptable in conjunction with the leak-before-break concept the leak size selected. The design of the sodium aerosol mitigation system is based on the resulting accident sequence and is discussed in detail in Section 9.13.2. Given the BTP MEB 3-1 IHTS design-basis leak, the steam generator building will be held within temperatures and pressures acceptable for structures. The release of airborne aerosols that could contaminate safety-grade equipment would be held to limits proposed by the applicants and considered acceptable by the staff for the construction permit. The proposed criteria are evaluated in Section 9.13.2. If the design-basis leak were to be exceeded, the most severe consequence would appear to be a failure of the aerosol mitigation device which would release more aerosols than desirable for fouling or plugging of safety-related devices or increasing site boundary combustion product concentration. If this were to occur, the equipment would be accessible for recovery and the consequences of possibly exceeding the design basis for the steam generator building therefore appear both remote and mild. Offsite concentrations of combustion products would be higher than anticipated, but response to such an event is to be addressed in the plant's emergency plans. The staff, therefore, finds this design acceptable.

In addition, the applicants have committed to a fire test program that will confirm the adequacy of the design (PSAR Section 1.5.2.8).

The containment design-basis accident is acceptable because it is the largest source of sodium available for a sodium fire and is postulated to occur under conservative conditions. A more complete discussion of the containment evaluation appears in Section 6.2.1.1. Even though the event would be so severe, it must be considered hypothetical. A maximum pressure of 0.8 psig would present a mild challenge for the containment designed for 10-psig internal pressure. Although the containment is an engineered safety feature for design-basis events, it also serves as an engineered safety accommodation feature for beyond-design-basis accommodation (see Appendix A.1). Therefore, the containment design tends to have considerable margin for any credible or near-credible design-basis event postulated. The staff's consultants have performed confirmatory sodium-fire calculations discussed in Section 9.13.2.

It should be emphasized that the inerted cells and their liners are to be engineered safety features. This is because the consequences of a sodium fire will be more severe if sodium-concrete interactions are possible. Sodium tests on prototypic liners (Hanford Engineering Development Laboratory, HEDL-TME 79-35) give assurance that liners for both CRBR and FFTF are capable of withstanding their design sodium spills for times considerably in excess of

time-temperature for most postulated sodium accidents. The results of this test and resulting action items are discussed in Section 3.8 of this SER.

The staff found the analyses of the sodium spills and fires and their consequences to be in accord with the intent of, and to form part of the body of information that satisfies the following CRBR principal design criteria: PDC 4, 5, 14, 41, and 52.

15.7 Other Events

15.7.1 Introduction

This section covers the events which do not logically group into one of the other categories: reactivity insertion events, undercooling events, fuel-handling and storage events, or sodium spills and fires. In general, the event severity categorization is similar to that in other sections. The events are categorized as anticipated, unlikely, and extremely unlikely. The events considered are shown with their results in Table 15.12. In the cases where a release of radioactivity would occur, the acceptance criteria have been compared to 10% of 10 CFR 100 dose guideline values. The consideration of these accidents forms a closure set for the potential events of WASH-1302 (USAEC), for example, the closure with accidents resulting from phenomena such as floods, storm, and conventional fires. The natural phenomena (floods and storms) and conventional fires are covered also in Sections 3.3, 3.4 and 9.13.1, respectively.

This section includes the more traditional accidents which are not unique to LMFBRs, such as the failure of one dc electrical system, loss of instrument or valve air supply, liquid radwaste system failure, generator breaker failure to open at turbine trip, and fuel shipping cask drop from maximum possible height.

There are some unique CRBR accidents postulated in this category which warrant elaboration. These are accidents involving the radioactive handling systems--radioactive argon processing system (RAPS) and cell atmosphere processing system (CAPS)--for the cover gas and cell atmospheres. RAPS will be essentially a closed system which will take the reactor cover gas (argon), remove most of the radioactivity, and return it for reuse. It will also provide a source of low-radioactivity gas for use in reactor seals and cover gas pressure control. From time to time there will be releases from RAPS to CAPS. CAPS will take some inputs from RAPS, but will primarily process the cells and pipeways atmospheres (nitrogen/low oxygen content) to reduce the radioactivity of the effluent for subsequent release. (See Section 11 for more detailed system description of RAPS and CAPS.)

The largest source of concentrated radioactivity will be the RAPS cryostill. During normal operation, the argon will be directed through the cryostill. The argon will be condensed to a liquid as it passes through the coiled tubing in the cryostill condenser, which will be surrounded by liquid nitrogen. It is assumed that the cryostill will contain a maximum inventory of radioactivity (operation of 1% failed fuel for 1 year). A rupture is considered to occur releasing the contents to the heating and ventilating system. The RAPS cell exhaust will be isolated, but the incoming nitrogen will continue to feed and pressurize the cell until a high cell pressure signal is reached. It is

assumed that (1) the RAPS cell will not be leak tight, (2) the refueling door will be open at the time of the accident, and (3) the released gas will vent out through the heating and ventilating exhausts. The postulated results of this event are shown in Table 15.12. The rupture in RAPS noble gas storage vessel presents a similar accident with identical consequences. These two events will bound the other postulated accidents in RAPS and CAPS.

The other postulated accident requires leakage through the intermediate heat exchanger (IHX) and a subsequent steam generator (SG) leak. If the SG leak were large enough, a sodium-water reaction would activate the sodium-water reaction protection system (SWRPS) which would provide a pathway to the environment via the hydrogen flaring stack. The postulated consequences of SWRPS action are discussed in Section 15.3 and are shown to have no undercooling impact because of PPS actions and the long thermal transit time. Significant radioactivity leaking into the IHTS would be controlled by keeping the IHTS at a higher static pressure than the PHTS.

15.7.2 Evaluation

The events and analysis of the consequences in Table 15.12 were reviewed. Since this composite of events is, generally, not generically related as in other subsections of Section 15 (undercooling, overpower), they will be discussed in relation to their own subgroup similarity.

The consequences of traditional events--loss of one dc system, loss of instrument or valve air supply, generator breaker failure to open at turbine trip, floods, storms, and conventional fires--were found acceptable and are reviewed in depth in other sections of the SER. For example, see Sections 2.4, 3.3, and 9.13 for floods, storms, and conventional fires, respectively. Section 8.3 covers the dc systems, and Section 10 covers steam and power conversion systems.

The staff independently evaluated liquid radwaste system failure. On the basis of the applicants' and staff's independent evaluations, the consequences of failures of the liquid radwaste systems are judged acceptable (see Section 15.7.2.1).

Events associated with operation of the HTS, such as cover gas pressures, intermediate heat exchanger leakage, leakage by failure of plug seals and annuli, either will be bounded by other events or the consequences are acceptable and the analyses are conservative.

The staff reviewed the event "Off-Normal Cover Gas Pressure in the Reactor Primary Coolant Boundary" (Section 15.7.1.4 of the PSAR) in depth because the German LMFBR, SNR-300, has protection functions for off-normal cover gas pressure, both high and low.

The staff review of the abnormal cover gas pressure event for CRBR indicated that CRBR will be adequately protected for both overpressure and underpressure cover gas events. This conclusion is based on the following considerations.

The reactor cover gas pressure will be maintained at approximately 6 in. of water gage or essentially atmospheric pressure. Argon makeup to the system will be accomplished by two parallel sets of pressure regulators for redundancy. There will be two sets of series pressure control valves that will be

connected in parallel. All these valves will fail open. The bleed from the system will be accomplished by a similar set of series-parallel pressure regulator valves. In this configuration one series string will fail open and one series string will fail closed.

If the bleed valves fail open, the reactor cover gas space, primary pump gas space, and reactor overflow tank space will flow gas uncontrolled to the RAPS vacuum vessel. The pressure in the cover gas spaces will drop to subatmospheric levels by approximately 1-2 psi because of the relative volumes between the cover gas spaces (4,500 ft^3) and the RAPS vacuum vessel (300 ft^3) and their respective pressures. The RAPS vacuum vessel will be limited to a minimum of 8 psia. This mild vacuum would be insufficient to present a problem for the primary pump or seals.

For overpressure to occur, both feed and bleed valves should fail; that is, the redundant complexes must both fail. The feed valves as designed would fail open and the bleed valves must fail closed. The gas makeup system will be designed to provide a makeup rate of about 50 scfm. At that rate, to overpressure the cover gas spaces to 15 psig, the pressure relief setpoint requires at least an hour. There will be redundant pressure relief valves. The pressure relief valves will relieve into the primary overflow tank cell and then into the CAPS, if no operator action is taken before this pressure limit is reached. The primary system will be designed for 15 psig (safety valve setting); therefore, it will have no effect on the system from a structural standpoint. Pressure boundary margin seals have been designed for pressures in excess of 300 psid. The pressure differential between the primary and intermediate HTS of 10 psid, if these postulated failures should occur, would be approached. Because the system flow rate will be limited, as previously discussed, the decrease will be slow. The differential pressure will be monitored and annunciated for operator action.

The staff speculated about setting up some siphoning action, the pressure impact on leaks, and any other possible impairment of system operation and could find no reason to indicate that abnormal cover gas pressure was a safety concern in CRBR. The staff has requested that the applicants perform a failure modes and effects analysis (FMEA) on the cover gas system as noted in Section 15.1 for the FSAR to ensure that the system has been thoroughly explored and completely understood.

Postulated consequences from miscellaneous sodium interactions, leaks, and fires, such as sodium interaction with the chilled water system and leakage from the sodium cold trap, are well within dose guidelines and are conservatively formulated. Therefore, the consequences of these events are acceptable.

Ruptures of RAPS will bound the radioactivity releases possible from accidents in the RAPS and CAPS. However, the calculated dose at the site boundary from such events slightly exceeds the acceptance criteria for such radioactive-release events--one-tenth of the dose guideline of 10 CFR 100, which is 2.5 rems whole-body dose for a 2-hour exposure. The staff finds this unacceptable. The staff notes, however, that the consequences of this event can be mitigated by procedural control. If the inventory in RAP is limited by Technical Specifications, the consequences can be reduced to an acceptable level. It is the staff's opinion that appropriate Technical Specifications to limit the radioactive inventory in the RAPS cryostill and noble gas storage vessel should be

developed for the operating license stage of review. The analysis failed to present the dose for the low population zone (LPZ) for this event. The LPZ dose analysis should be included in the FSAR.

Failure in the EVST NaK system is discussed in Section 15.3.2 and the sodium-fire results are in Table 15.11.

The applicants' proposed design-basis "other" events were reviewed following the Standard Review Plan (SRP) where appropriate. Because the SRP pertains to the review of LWRs, it has limited applicability to CRBR. In this category of events, the dose guidelines in the SRP were consistent with the acceptance criteria for events in which radioactivity was released. The staff found the CRBR dose guidelines acceptance criteria to be more conservative than and consistent with those in the SRP. Therefore, the applicants' proposed dose guidelines were acceptable. The events and the analyses of their consequences are in accord with the intent and form part of the body of information that satisfies the following principal design criteria: PDC 2, 3, 4, 5, 15, 34, 52, 53, and 56.

15.7.2.1 Postulated Radioactive Releases From Liquid Tank Failure

15.7.2.1.1 Introduction

Section 11.2 of the PSAR contains the applicants' provisions for mitigating the consequences of, or the potential for, release of radioactive materials to the environment resulting from the failure of a tank or component containing radioactive liquid wastes.

As part of the review of radioactive liquid waste systems, the staff conducted an independent evaluation of liquid tank failures. This event is evaluated by the applicants in Section 15.7.2.5 of the PSAR.

15.7.2.1.2 Acceptance Criteria

The staff conducted an independent evaluation of liquid tank failures for the CRBRP. The acceptance criteria that were used as the basis for the staff's evaluation are set forth in Section II of SRP Section 15.7.3 (NUREG-0800). These acceptance criteria consist of GDC 60, as it relates to the design of radioactive waste management systems to control releases of radioactive materials to the environment, and 10 CFR 20, as it relates to radioactivity in effluents to unrestricted areas. CRBR PDC 52 corresponds to GDC 60 and satisfaction of either satisfies both criteria. Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides and other documents referenced in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the proposed radioactive liquid waste storage tanks meet the requirements of 10 CFR 20 and 10 CFR 50.

15.7.2.1.3 Evaluation

The review was conducted in accordance with the SRP. The consequences of component failures for components located outside the reactor containment which could result in releases of liquids containing radioactive materials to the

environment were evaluated. Considered in the staff's evaluation were (1) the radionuclide inventory in each component assuming a 1% operating power fission product source term (a conservative, hypothetical bounding value), (2) a component liquid inventory equal to 80% of its design capacity, (3) the mitigating effects of plant design, including overflow lines and the location of storage tanks in curbed areas designed to retain spillage, and (4) the effects of site geology and hydrology.

The applicants have incorporated provisions in the design to retain releases from liquid overflows as discussed in Section 11.2.1 of this SER. In the event of a spill, the staff postulated liquid flow directly to the groundwater beneath the planned location of the tank. The flow was assumed to seep through the ground and flow along the shortest path to the Clinch River, adjacent to the site, mix with the lowest average annual flow of record for the Clinch River, and hence travel 9,000 ft downstream to the nearest user. The staff calculated an overall dilution factor of 7×10^6.

On the basis of the staff's evaluation, the potential tank failure resulting in the greatest quantity of activity released to the environment is failure of one of the intermediate activity level liquid system collection tanks. In its evaluation, the staff has determined the liquid transit time for the leakage to the river to be 30 years. Considering the leakage dilution transit time, the calculated radionuclide concentrations in the Clinch River result in values that are small fractions of the limits of 10 CFR 20 (Appendix B, Table II, Column 2) for unrestricted areas. On the basis of the foregoing evaluation, the staff concludes that the provisions incorporated in the applicants' design to mitigate the effects of component failures involving contaminated liquids are acceptable.

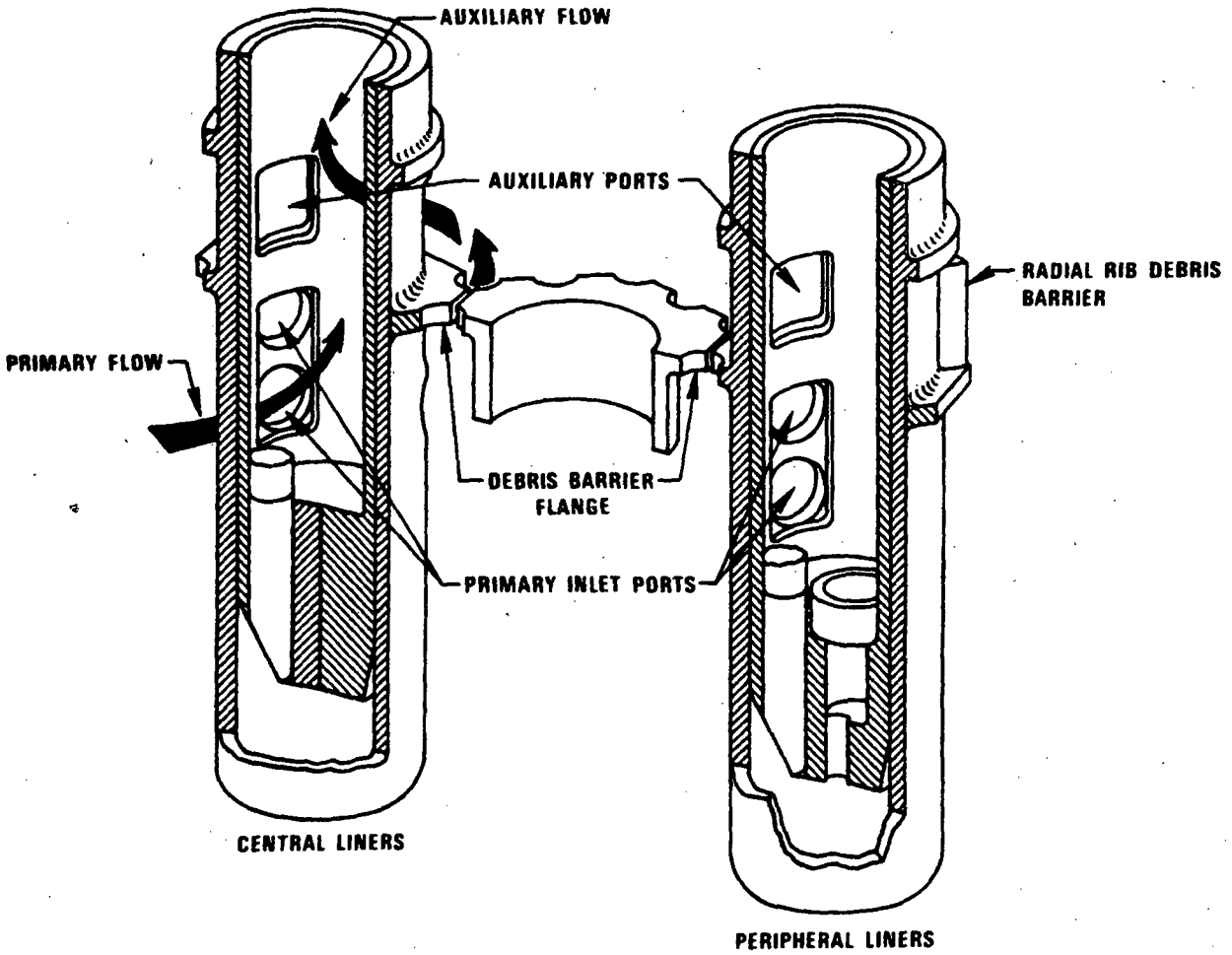


Figure 15.1 Core module liner

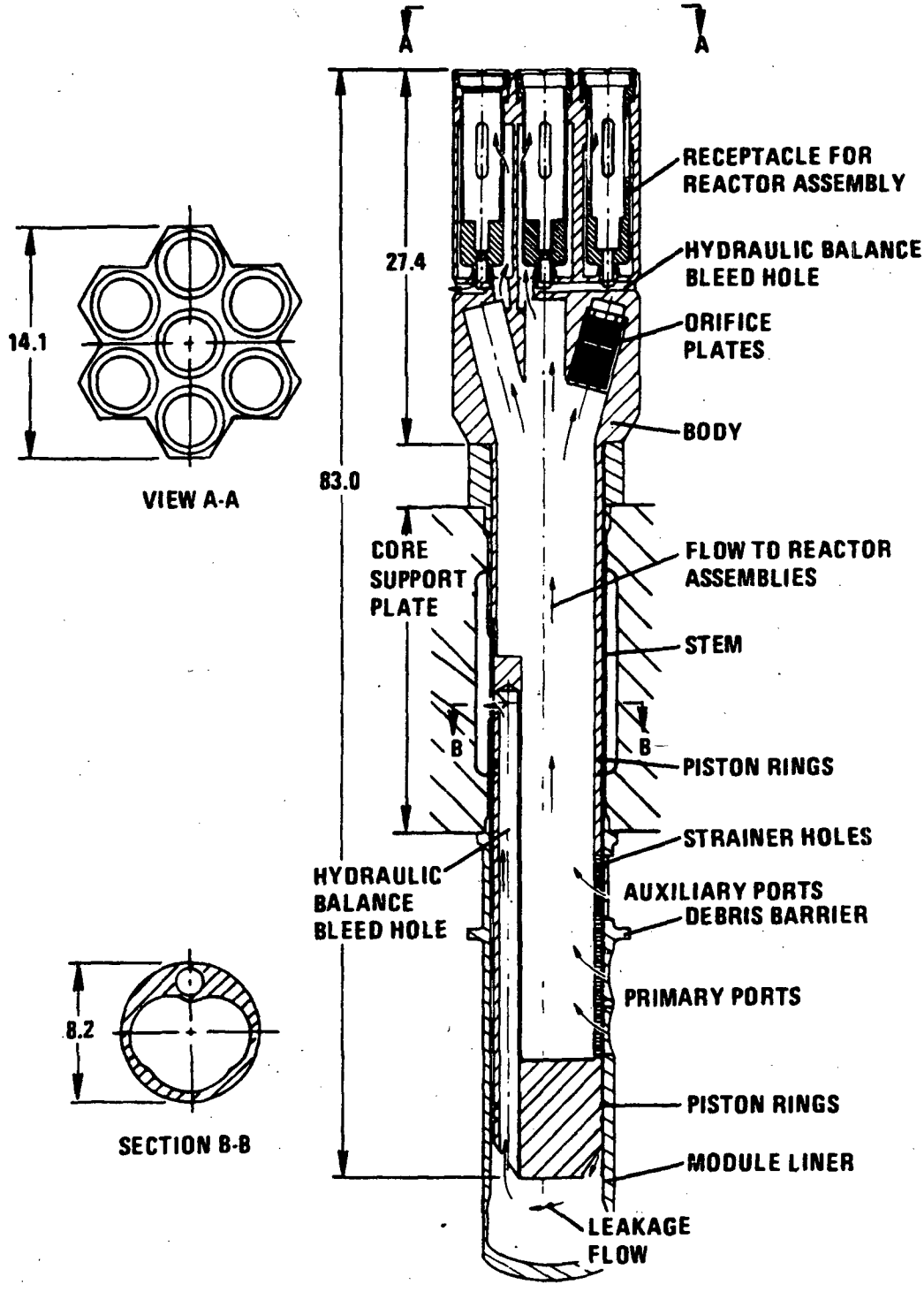


Figure 15.2 Elevation of typical lower inlet module

Table 15.1 Event classification and definitions

Event or plant condition	LWR terminology* and bases/ definition of event class	CRBR terminology** and bases/ definition of event class
Normal operation	Condition I	Normal operation
Incidents of moderate frequency	Condition II: Incidents, any one of which may occur during a calendar year for a particular plant	Anticipated event: An off-normal condition which individually may be expected to occur once or more during the plant's lifetime
Infrequent incidents	Condition III: Incidents, any one of which may occur during the lifetime of a particular plant	Unlikely event: An off-normal condition which individually is not expected to occur during the plant's lifetime; however, when integrated over all plant components, events in this category may be expected to occur a number of times
Limiting faults	Condition IV: Faults that are most drastic and must be designed against, and thus represent the limiting design case	Extremely unlikely event: An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant's lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as design bases

*ANS Std. N18.2 (1973) supplemented by N18.2a (1975).
**DOE Std. RDT C-16-1.

Table 15.2 Systems assumed operable to mitigate the consequences of each accident event

PSAR section	Event	Required operable system	Primary	Secondary
15.2	REACTIVITY INSERTION DESIGN EVENTS			
15.2.1	Anticipated events			
15.2.1.1	Control assembly withdrawal at startup	PPS followed in long term by decay heat removal*	Flux-pressure Flux-delayed flux	Flux-total flow
15.2.1.2	Control assembly withdrawal at power	PPS followed in long term by decay heat removal	High flux Flux-pressure	Flux-total flow
15.2.1.3	Seismic reactivity insertions, OBE	PPS followed in long term by decay heat removal	High flux Flux-pressure	Flux-total flow
15.2.1.4	Small reactivity insertions	PPS followed in long term by decay heat removal	High flux	Flux-total flow
15.2.1.5	Inadvertent drop of a single control rod at full power	PPS followed in long term by decay heat removal	Flux-delayed flux	Modified nuclear rate
15.2.2	Unlikely events			
15.2.2.1	Loss of hydraulic holddown	PPS followed in long term by decay heat removal	High flux Flux-pressure	Flux-total flow
15.2.2.2	Sudden core radial movement	PPS followed in long term by decay heat removal	High flux Flux-pressure	Flux-total flow
15.2.2.3	Maloperation of reactor plant controllers	PPS followed in long term by decay heat removal	High flux Flux-pressure	Flux-total flow
15.2.3	Extremely unlikely events			
15.2.3.1	Cold sodium insertion	PPS followed in long term by decay heat removal	Speed ratio	Flow ratio
15.2.3.2	Gas bubble passage through fuel, radial blanket, and control assemblies	PPS followed in long term by decay heat removal	High flux	Flux-total flow
15.2.3.3	Seismic reactivity insertion, SSE	PPS followed in long term by decay heat removal	High flux Flux-pressure Heat transport system (HTS) pump frequency	Flux-total flow

See footnotes at end of table.

Table 15.2 (Continued)

PSAR section	Event	Required operable system	Primary	Secondary
15.2.3.4	Control assembly withdrawal at startup-maximum mechanical speed	PPS followed in long term by decay heat removal	Flux-pressure Flux-delayed flux	Flux-total flow
15.2.3.5	Control assembly withdrawal at power--maximum mechanical speed	PPS followed in long term by decay heat removal	High flux	Flux-total flow
15.3	UNDERCOOLING DESIGN EVENTS			
15.3.1	Anticipated events			
15.3.1.1	Loss of offsite electrical power	PPS followed in long term by decay heat removal	HTS pump frequency	HTS pump voltage
15.3.1.2	Spurious primary pump trip	PPS followed in long term by decay heat removal	Flux-to-pressure speed ratio	Flow ratio
15.3.1.3	Spurious intermediate pump trip	PPS followed in long term by decay heat removal	Speed ratio	Flow ratio
15.3.1.4	Inadvertent closure of one evaporator or superheater module isolation valve	PPS followed in long term by decay heat removal	Steam-feedwater	Evap. outlet temp.
15.3.1.5	Turbine trip	Long term by decay heat removal**	Steam-feedwater	Loss of condenser vacuum
15.3.1.6	Loss of normal feedwater	PPS followed in long term by decay heat removal	Steam-feedwater	Steam drum level
15.3.1.7	Inadvertent actuation of the sodium-water reaction pressure relief system	PPS followed in long term by decay heat removal	Steam-feedwater	Evap. outlet temp.
15.3.2	Unlikely events			
15.3.2.1	Single primary pump seizure	PPS followed in long term by decay heat removal	Speed ratio	Flow ratio
15.3.2.2	Single intermediate loop pump seizure	PPS followed in long term by decay heat removal	Speed ratio	Flow ratio

See footnotes at end of table.

Table 15.2 (Continued)

PSAR section	Event	Required operable system	Primary	Secondary
15.3.2.3	Small water-to-sodium leaks in steam generator tubes	†		
15.3.2.4	Failure of the steam bypass system	PPS followed in long term by decay heat removal	Steam-feedwater	Steam drum level
15.3.3	Extremely unlikely events			
15.3.3.1	Steam or feedline pipe break	PPS followed in long term by decay heat removal	Steam-feedwater	Evap. outlet temp.
15.3.3.2	Loss of normal shutdown cooling system	PPS followed in long term by decay heat removal	Steam-feedwater	Steam drum level
15.3.3.3	Large sodium-water reaction	Sodium-water reaction pressure relief system rupture discs	Steam-feedwater	Sodium-water reaction
15.3.3.4	Primary heat transport system pipe leak	†		
15.3.3.5	Intermediate heat transport system pipe leak	PPS followed in long term by decay heat removal	Intermediate heat exchanger primary outlet	Flow ratio
15.5	FUEL-HANDLING AND 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	†		
15.5.2.2	Damage of fuel assembly from attempt to insert a fuel assembly into an occupied position	†		

See footnotes at end of table.

Table 15.2 (Continued)

PSAR section	Event	Required operable system	Primary	Secondary
15.5.2.3	Single fuel assembly cladding failure and subsequent fission gas release during refueling	Ex-vessel transfer machine		
15.5.2.4	Cover gas release during refueling	†		
15.5.2.5	Heaviest crane load impacts the reactor closure head	†		
15.5.3	Extremely unlikely events			
15.5.3.1	Collision of ex-vessel transfer machine with control rod drive mechanism	†		
15.6	SODIUM SPILLS AND FIRES			
15.6.1	Anticipated events			
15.6.1.1	Primary sodium in-containment storage tank failure during maintenance	Containment isolation system		
15.6.1.2	Failure of the ex-vessel storage tank sodium cooling system during refueling	†		
15.6.1.3	Failure of an ex-containment primary sodium storage tank	†		
15.6.1.4	Primary heat transport system piping leak	†		
15.6.1.5	Intermediate heat transport pipe leak	†		

See footnotes at end of table.

Table 15.2 (Continued)

PSAR section	Event	Required operable system	Primary	Secondary
15.7	OTHER EVENTS			
15.7.1	Anticipated events			
15.7.1.1	Loss of one dc system	†		
15.7.1.2	Loss of instrument or valve system	†		
15.7.1.3	Intermediate heat exchanger leak	†		
15.7.1.4	Off-normal cover gas pressure in the reactor coolant boundary	†		
15.7.1.5	Off-normal cover gas pressure in the intermediate heat transport system	†		
15.7.2	Unlikely events			
15.7.2.1	Inadvertent release of oil through the pump seal (primary heat transport system)	†		
15.7.2.2	Inadvertent release of oil through the pump seal (intermediate heat transport system)	†		
15.7.2.3	Generator breaker failure to open at turbine trip	†		
15.7.2.4	Rupture of radioactive argon processing system surge vessel	†		
15.7.2.5	Liquid radwaste system failure	†		

See footnotes at end of table.

Table 15.2 (Continued)

PSAR section	Event	Required operable system	Primary	Secondary
15.7.2.6	Failure (leak or rupture) in the ex-vessel storage tank NaK system	†		
15.7.2.7	Leakage from sodium cold traps	†		
15.7.2.8	Rupture in radioactive argon processing system noble gas storage vessel cell	†		
15.7.2.9	Rupture in the cell atmosphere processing system cold box	†		
15.7.3	Extremely unlikely events			
15.7.3.1	Leak in a core component pot	Ex-vessel transfer machine seals		
15.7.3.2	Spent fuel shipping cask drop from maximum possible height	†		
15.7.3.3	Maximum possible conventional fires, flood, storms, or minimum river level	†		
15.7.3.4	Failure of plug seals and annuli	Containment isolation system		
15.7.3.5	Fuel rod leakage combined with intermediate heat exchanger and steam generator leakage	†		
15.7.3.6	Sodium interaction with chilled water	†		
15.7.3.7	Sodium-water reaction in large component cleaning vessel	†		

Table 15.2 (Continued)

*The redundant and diverse decay heat removal capabilities are described in Section 5.6.

**The plant controller will be designed to bring the plant to a steady-state operating condition in the event of a turbine trip. Reactor shutdown may be initiated by the operator if desired. The plant protection system (PPS) would react (primary-steam feedwater, secondary-evaporator outlet temperature) only if the (nonsafety class) plant controller failed.

†These events do not constitute challenges requiring operation of plant protection systems to protect the health and safety of the public.

NOTE: Where more than one trip function is listed, the separate trip functions would each adequately mitigate the event, though the first trip listed is expected to act first to safely shut down the plant.

Table 15.3 Acceptance criteria for preliminary safety evaluation

Event classification*	Fuel temperature	Cladding temperature (°F)	Cooling temperature (°F)	Dose limits
Anticipated fault	Solidus**,***	1,500**	N/A	10 CFR 20
Unlikely fault	Solidus**,***	1,600**	N/A	10 CFR 20
Extremely unlikely fault or postulated accident	---	Solidus† (2,475°F)	Saturation††	1/10 of 10 CFR 100

*Applicable "event class" or "severity level" is based on primary shutdown system action. For secondary system shutdown see Table 15.2. Events terminated by secondary system shutdown will be judged against the acceptance criteria for the next more severe event classification unless the event is already extremely unlikely, then the extremely unlikely limit will control for both the primary and secondary shutdown system.

**For temperatures in excess of these values, transients shall be assessed using mechanical design procedures and design limits of Section 4.2, PSAR.

***No fuel melting at existing conditions.

†This criterion has been considered adequate for the CP stage only. See Section 4.2 of this SER.

††No sodium boiling at existing pressure.

Table 15.4 Thermal-hydraulic initial conditions

Thermal power (MWt)	975*
Primary flow (lb/sec/loop)	3,842
Primary hot-leg temperature (°F)	1,015**
Primary cold-leg temperature (°F)	750*
Intermediate flow (lb/sec/loop)	3,555
Intermediate hot-leg temperature (°F)	956**
Intermediate cold-leg temperature (°F)	671**
Hot spot clad midwall temperature (°F)	1,365

*Power uncertainties are discussed in PSAR Chapter 4, Section 4.4 for three-loop operation.

**These values include an additional 20F° over their normal to allow for instrument error and to control dead band allowance.

Table 15.5 Reactivity insertion design events

PSAR section	Event	Max. cladding temp.* Primary scram	Max. cladding temp.* Secondary** scram	Comments
15.2.1	Anticipated events			
15.2.1.1	Control assembly withdrawal at startup	Not calc.	1,383°F	Temp shown for 1¢/sec. withdrawal. Resultant temp. less than operating condition (full power).
15.2.1.2	Control assembly withdrawal at power	1,510°F	1,610°F	Based on extremely small withdrawal rate. Results are within the guidelines.
15.2.1.3	Seismic reactivity insertion (core, radial blanket, and control rod) OBE	1,440°F	1,440°F	Based on postulated 30¢ step reactivity insertion. Results are within guidelines.
15.2.1.4	Small reactivity insertions	1,500°F	1,560°F	For 2¢/sec insertion case. Results are within guidelines.
15.2.1.5	Inadvertent drop of single control rod at full power	Less than init. cond.	Less than init. cond.	Results are within guidelines.
15.2.2	Unlikely events			
15.2.2.1	Loss of hydraulic holddown	1,400°F	1,420°F	Results are within guidelines.
15.2.2.2	Sudden core radial movement	1,470°F	1,510°F	For nonseismic conditions, results are within guidelines.
15.2.2.3	Maloperation of reactor plant controllers	<1,510°F	<1,610°F	Less than limiting condition.
15.2.3	Extremely unlikely events			
15.2.3.1	Cold sodium insertion	Less than init. cond.	Less than init. cond.	Results are within the guidelines.
15.2.3.2	Gas bubble through core	<1,480°F	<1,480°F	Results are within the guidelines.
15.2.3.3	Seismic reactivity insertion (core, radial blanket and control rod) - SSE	*** 1,491°F	*** 1,544°F	Based on postulated 60¢ step reactivity insertion. Results are within guidelines.
15.2.3.4	Control assembly withdrawal at startup--max. mech. speed	Not calc.	800°F	For 20¢/sec reactivity insertion. Results are within guidelines.
15.2.3.5	Control assembly withdrawal at power--max. mech. speed	1,420°F	1,460°F	For 20¢/sec reactivity insertion. Results are within guidelines.

*Fuel pin inside diameter cladding temperature (under wire wrap).
**Some secondary trip points have been changed.
***Heterogeneous core.

Table 15.6 Clinch River Breeder Reactor audit analysis results--60-cent step/safe shutdown earthquake

Fuel assembly	Fuel temperature (°F)		Cladding temperature (°F)		Coolant temperature (°F)	
	Audit	PSAR	Audit	PSAR	Audit	PSAR
F/A 52						
Nominal	4,002	3,942	1,494	1,529	1,454	1,479
3σ	4,595	4,521	1,687	1,728	1,623	1,654
F/A 101						
Nominal	4,767	4,603	1,464	1,492	1,428	1,431
3σ	5,056*	5,056*	1,653	1,687	1,593	1,597

*Both PSAR and audit calculations predict molten fuel under +3σ conditions.

Table 15.7 Key differences between homogeneous core and heterogeneous core that impact accident analysis

Differences	Impact
One enrichment zone vs. two as in the homogeneous core	Reduced potential for enrichment errors
Inner blanket (IB) assemblies in heterogeneous core	IB assemblies analyzed
9 primary control rod assemblies vs. 15	No impact on event scenarios
6 secondary control rod assemblies vs. 4	Values for scram rate and total rod worth will be different
Lower fission gas plenum pressure	No impact on event scenarios
	Values used in analyses changed from end of cycle 1 to end of cycle 2
Smaller (bulk) positive sodium void worth in fuel assemblies lower doppler	Parameter values affected
	Minor impact on design-basis accidents
Fuel assembly peak linear power rating is slightly higher	Parameter values used in analyses affected (14.1 kW/ft vs. 12.8 kW/ft in fuel assemblies)
Higher Pu enrichment.	Slightly lower melting point

Table 15.8 Events not considered

Events not analyzed	Rationale for omission
Loss of heat sink	Residual heat removal system redundancy and diversity protects against loss of heat sink
Failure of steam dump system	Redundant loops with independent and redundant safety relief valves
Inadvertent closure of either of reactor coolant valves or intermediate coolant valves	No stop valves in either the reactor coolant system or IHTS
Plugging of reactor overflow line	Detected by reactor vessel and overflow tank level probes, line is oversized and heated
Accidental opening of valves to a drained isolated loop	Not applicable because of inability to isolate any of the three loops from the system
Simultaneous leak of reactor overflow tank	No safety consequences involved. Effects from spills are enveloped by the PHTS leaks analyzed.

Table 15.9 Undercooling design event

PSAR section	Event	Max. cladding temp.* Primary scram	Max. cladding temp.* Secondary scram	Comments
15.3.1	Anticipated events			
15.3.1.1	Loss of offsite electric power	1,410°F (1,455°F)**	1,480°F	Primary shutdown within upset umbrella. Temperature spike associated with secondary is considerably less severe than the umbrella transient (see PSAR Section 15.3.1.1)
15.3.1.2	Spurious primary pump trip	1,390°F	1,445°F	Within the umbrella
15.3.1.3	Spurious intermediate pump trip	<1,365°F	<1,365°F	Core sees only normal trip
15.3.1.4	Inadvertent closure of one evaporator or superheater module isolation valve	<1,365°F	<1,365°F	Core sees only normal trip
15.3.1.5	Turbine trip	<1,365°F	<1,365°F	Temperature decreasing continuously
15.3.1.6	Loss of normal feedwater	<1,365°F	<1,365°F	Core sees only normal trip
15.3.1.7	Inadvertent actuation of the sodium-water reaction pressure relief system	<1,365°F	<1,365°F	Core sees only normal trip
15.3.2	Unlikely events			
15.3.2.1	Single primary pump seizure	1,400°F	1,470°F	Within the umbrella
15.3.2.2	Single intermediate loop pump seizure	<1,365°F	<1,365°F	Core sees only normal trip
15.3.2.3	Small water-to-sodium leaks in steam generator tubes	<1,365°F	<1,365°F	Core sees only normal trip
15.3.2.4	Failure of the steam bypass system	<1,365°F	<1,365°F	Core sees only normal trip

See footnotes at end of table.

Table 15.9 (Continued)

PSAR section	Event	Max. cladding temp.* Primary scram	Secondary scram	Comments
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	<1,365°F	<1,365°F	Core sees only normal trip
15.3.3.3	Large sodium-water reaction	<1,365°F	<1,365°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 (underwire wrap)
**Heterogeneous core

Table 15.10 Fuel-handling and storage events

PSAR section	Event	At site boundary (2-hr)(rem)	At low population zone (30-day) (rem)	Comments
15.5.1	Anticipated events (None)			
15.5.2	Unlikely events			
15.5.2.1	Fuel assembly dropped within reactor vessel during refueling	$<3.0 \times 10^{-4}$	$<2.42 \times 10^{-5}$	Consequences of this event are within the umbrella of Section 15.5.2.4
15.5.2.2	Damage of fuel assembly from attempt to insert a fuel assembly into an occupied position	$<2.77 \times 10^{-3}$	$<1.0 \times 10^{-3}$	This event is well within the suggested guideline dose limits
15.5.2.3	Single fuel assembly cladding failure and subsequent fission gas release during refueling	2.77×10^{-3}	1.0×10^{-3}	This event is well within the suggested guideline dose limits
15.5.2.4	Cover gas release during refueling	4.4×10^{-3}	1.2×10^{-3}	This event is well within the suggested guideline dose limits
15.5.2.5	Heaviest crane load impacts reactor closure head	$<3.0 \times 10^{-4}$	$<2.42 \times 10^{-5}$	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 ex-vessel transfer machine with control rod drive mechanism	$<3.0 \times 10^{-3}$	$<2.42 \times 10^{-5}$	Consequences of this event are within the umbrella of Section 15.5.2.4

Table 15.11 Sodium spill events

PSAR section	Event	Sodium spill Gallons	Sodium spill Temp. (°F)	Atmosphere	Location*	Cell	Max. offsite dose (% of 10 CFR 100)	Max. cell gas press./temp. (psig/°F)	Design press./ liner design temp. (psig/°F)
15.6.1	Extremely unlikely events								
15.6.1.1	Primary sodium in-containment, storage tank failure during maintenance	35,000	400	Normal air	RCB	Overflow tank cell	0.19	0.8/138**	10/1,015
15.6.1.2	Failure of ex-vessel storage tank sodium cooling system during operation	7,500	475	Inerted	RSB	Ex-vessel sodium tank cell	0.48	3.8/254***	12/1,015
15.6.1.3	Failure of ex-containment primary sodium storage tank	45,000	450	Inerted	SGB/IB	Storage tank cell	2.13	3.5/260***	4/1,000 (catch-pan)
15.6.1.4	Primary heat transport system (PHTS) pipe leak	35,000 (PHTS cell)	1,015	Inerted	RCB	PHTS cell	$<10^{-4}$	14.4/680***	30/1,015
		29,200 (reactor cavity)	750	Inerted	RCB	Reactor cavity		10.3/650	35/1,015
15.1.6.5	Intermediate heat transport system pipe leak	39,000	800	Normal air	SGB/IB	IB	3	0.4/630	3/1,000 (catch-pan)

*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.

Table 15.12 Other events

PSAR section	Event	Limiting parameter	Comment
15.7.1	Anticipated events		
15.7.1.1	Loss of one dc system	None	No adverse operating conditions identified
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	Intermediate heat exchanger leak	None	Core sees normal shutdown
15.7.1.4	Off-normal cover gas pressure in the reactor primary coolant boundary	None	No associated adverse operating conditions
15.7.1.5	Off-normal cover gas pressure in the intermediate heat transport system	None	No associated adverse operating conditions
15.7.2	Unlikely events		
15.7.2.1	Inadvertent release of oil through the pump seal (primary heat transport system)	None	No adverse consequence identified
15.7.2.2	Inadvertent release of oil through the pump seal (intermediate heat transport system)	None	No adverse consequence identified
15.7.2.3	Generator breaker failure to open at turbine trip	None	Core sees normal shutdown
15.7.2.4	Rupture of radioactive argon processing system cryostill	<2.5 rem (integrated 2-hour dose at site boundary	Consequences will be within suggested guideline doses
15.7.2.5	Liquid radwaste system failure	3.7×10^{-6} rem at site boundary 3.05×10^{-7} rem at low population zone (LPZ)	Consequences will be within suggested guideline doses
15.7.2.6	Failure in the ex-vessel storage tank NaK system	None	No associated adverse consequences
15.7.2.7	Leakage from sodium cold traps	None	No associated adverse consequences

Table 15.12 (Continued)

PSAR section	Event	Limiting parameter	Comment
15.2.7.8	Rupture in radioactive argon processing system noble gas storage vessel cell	<2.5 rem (integrated 2-hr dose) at site boundary	Consequences will be within suggested guideline doses
15.7.3	Extremely unlikely events		
15.7.3.1	Leak in core component pot	~3200°F at 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	8.89×10^{-7} rem (2-hr whole-body dose) at site boundary 1.13×10^{-6} rem (30-day whole-body dose) at LPZ	Consequences will be within suggested guidelines
15.7.3.3	Maximum possible conventional fires, floods, and storms	None	None
15.7.3.4	Failure of plug seals and annuli	None	No associated adverse consequences
15.7.3.5	Fuel rod leakage combined with intermediate heat exchanger and steam generator leakage	None	No associated adverse consequences
15.7.3.6	Sodium interaction with chilled water	None	None

15A APPENDIX: SYNOPSIS OF CRBR PSAR CHAPTER 15 EVENTS (MARCH 8, 1982, THROUGH AMENDMENT 65)

15A.1 Introduction

This synopsis of events has been included to help the reader understand the accident progressions considered. The discussions are very brief and should not be interpreted as any staff judgment of acceptability. The sequences have been extracted from Chapter 15 of the applicants' PSAR. Because the discussions are so brief and most of the sequences are complicated, the reader is advised that limits or sequences could be misinterpreted. The reader should proceed to the body of this SER, if the events are discussed in detail, or to the PSAR for clarification should uncertainty or confusion arise from these brief synopses.

15A.2 Reactivity Insertion Design Events (PSAR Section 15.2)

15.2.1.1, Control Assembly Withdrawal at Startup

During startup, after criticality, a malfunction of an electronic control causes withdrawal of one primary control rod at maximum speed. The result is maximum cladding temperatures less than temperatures expected at full-power operation.

15.2.1.2, Control Assembly Withdrawal at Power

Withdrawal of one control rod (assembly) at power can lead to transients to 115% of full power if begun at 100%, or to 120% if begun at less than 100%. The results are maximum cladding temperature not more than about 100F° over temperatures at full power or no significant effect, respectively.

15.2.1.3, Seismic Reactivity Insertion, OBE

Operational basis earthquake causes loss of power to the pumps and a step reactivity insertion (by compaction of the core) 0.5 sec thereafter, followed by control rod insertion slowed by seismically induced friction. The result is maximum cladding temperatures less than 50°F over temperatures at full power.

15.2.1.4, Small Reactivity Insertions

This analysis includes several possible small reactivity insertions (e.g., a small shift in position of a core component), either ramp or step reactivity insertions. Assuming worst-case combinations of tolerances, maximum cladding temperatures of 1,500°F and 1,560°F (secondary scram) result; maximum temperatures during normal full-power operation are over 1,400°F.

15.2.1.5, Inadvertent Drop of a Single Control Rod at Full Power

An electrical or mechanical fault causes a control rod drive mechanism to release its rod. The result is that because of action of the plant protection system, the transient is not more significant than that from a conventional plant trip shutdown.

15.2.2.1, Loss of Hydraulic Holddown

Loss of hydraulic holddown (by "inconceivable" blockage of certain conduits) might cause upward motion of some fuel assemblies by as much as 2.5 in.; the possible reactivity insertions lead to transients resulting in maximum cladding temperatures of 1,420°F and 1,430°F, not significant to cladding lifetime.

15.2.2.2, Sudden Core Radial Movement

Stick-slip motion of fuel assemblies toward a more compact core arrangement causes a reactivity insertion. Maximum cladding temperature would not exceed 1,470°F for normal scram conditions, nor 1,510°F under SSE conditions; no significant degradation of cladding results.

15.2.2.3, Maloperation of Reactor Plant Controllers

Assuming failures of the sequencer and its limit circuits and the rod blocks permits seven control rods to move out. Reactivity ramp is terminated by the PPS within the operational incident limits.

15.2.3.1, Cold Sodium Insertion

Assuming the accidental startup of an inactive loop while on two-loop operation and partial failure of the PPS, cold sodium could be pumped into the reactor vessel causing a reactivity transient. The results are cladding temperatures less than those at full power operation, because two-loop operation would be at not more than 67% of full power. Two-loop operation is not being licensed at the present time.

15.2.3.2, Gas Bubble Passage Through Fuel, Radial Blanket, and Control Assemblies

Assuming that a 4-in. bubble could pass simultaneously into a row of eight assemblies (>3 ft^3 of gas), a reactivity insertion lasting less than 0.3 sec occurs, resulting in a maximum cladding temperature increase of 68F°.

15.2.3.3, Seismic Reactivity Insertion, SSE

Safe shutdown earthquake causes loss of power to the pump after 0.5 sec, a step reactivity insertion (by compaction of the core, followed by control rod insertion slowed by seismically induced friction). The results are maximum cladding temperatures of about 1,505°F, not more than 100F° over normal full-power maximum temperatures.

15.2.3.4, Control Assembly Withdrawal at Startup, Maximum Mechanical Speed

Assuming the worst failure of the electronic controller, such that the 9-in.-per-minute limit is failed, when the CRDM reaches 73 in. per minute, its roller nut will disengage the lead screw resulting in a drop of the rod. The reactivity transient causes maximum cladding temperatures of about 800°F, well below normal full-power temperatures.

15A.3 Undercooling Design Events (PSAR Section 15.3)

15.3.1.1, Loss of Offsite Electrical Power

Simultaneous, multiple failures cause loss of all offsite power to the 13.8-kV buses and, because of loss of sodium pump power, reduction in core flow. The result is that cladding midwall hotspot temperature is limited to 1,410°F by primary shutdown and to 1,630°F if shutdown is by the secondary system. Even the latter case is acceptable because the cladding damage does not exceed the acceptable strain limit. Secondary trip setting has been changed (see Section 15.3).

15.3.1.2, Spurious Primary Pump Trip

The trip of one primary sodium pump may be caused by a pump failure or by an ac bus fault, which would also trip the corresponding intermediate loop sodium pump. The result is a maximum hotspot midwall cladding temperature of 1,390°F.

15.3.1.3, Spurious Intermediate Pump Trip

The trip of one intermediate sodium pump may result from a pump failure or an ac bus fault. The PPS shuts the reactor down, with results similar to loss of offsite power with normal trip; the cladding hotspot temperature is not hotter, but is at high temperature longer.

15.3.1.4, Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve

Inadvertent closure of an isolation valve results in rapid drying out of the affected unit followed by reactor trip because of steam/feed flow mismatch or, failing that, by trip on high IHTS cold-leg temperature. Resulting maximum core temperature increases are about 30F°.

15.3.1.5, Turbine Trip

Turbine trip for whatever reason (e.g., loss of lube oil pressure) leads to closing the turbine inlet valve, opening bypass valves, and reduction in reactor power at 3% per minute. No significant effects on core occur. The steam dump to the atmosphere releases whatever tritium has accumulated in the water system, a maximum of 0.25 μCi per gram from which a 2-hour site boundary dose of about 99 mrems (whole-body dose) was calculated (and about 1 mrem, skin beta dose).

15.3.1.6, Loss of Normal Feedwater

Total loss of normal feedwater (e.g., from failure of a feedwater pump) leads to reactor trip by the steam-feedwater flow ratio and activation of the SGAHRS

including opening of the safety relief valves. After pressure declines and relief valves close, the SGAHRS operates with the auxiliary feedwater and protected air-cooled condensers to remove heat. Core experiences normal plant trip temperatures.

15.3.1.7, Inadvertent Actuation of the Sodium-Water Reaction Pressure Relief System

A spurious activation of the water-steam side of the sodium-water reaction pressure relief system, without the sodium dump, isolates the evaporator/superheater module and dumps its water. The PPS trips the plant on steam-feedwater flow mismatch or on high IHTS cold-leg temperature; core temperatures are less than those of normal full-power operation. The IHX and associated equipment experiences some thermal stress.

15.3.2.1, Single Primary Pump Seizure

Seizure of a primary sodium coolant pump, considered with instantaneous stop of an impeller, leads to plant trip by speed ratio of primary and intermediate pumps, or by flow ratio between primary and secondary loops; resulting maximum cladding midwall temperatures are 1,400°F and 1470°F, respectively. The trip times are so short that action of the check valve does not affect the maximum temperatures.

15.3.2.2, Single Intermediate Loop Pump Seizure

Seizure of an intermediate loop pump leads to plant trip by speed ratio of primary and intermediate pumps, or by flow ratio between primary and secondary loops. The core sees a normal plant trip, with temperatures not above its initial steady state.

15.3.2.3, Small Water-to-Sodium Leaks in Steam Generator Tubes

For small leaks, it is assumed that the water-to-sodium leak detection system will alert the operator who will then shut the plant down normally; this would be followed by controlled cooldown and depressurization of the affected steam generator and drainage of the IHTS loop. Plant trip drops the core temperatures so that the hot sodium from the affected loop does not cause temperatures to reach normal full-power operating temperatures. Radiological impact of any associated releases should be insignificant.

15.3.2.4, Failure of the Steam Bypass System

The steam bypass system, through four independent subsystems, maintains steam pressure while releasing steam to the main condenser following turbine trip. Failure of a bypass valve to open will cause a pressure increase leading to relief valve action. Failure of all four bypass valves to open would lead to reactor trip by the three steam-feedwater flow ratio trips, or, later, by low steam drum level trip. The results are like the loss of normal feedwater, that is, the core experiences normal trip temperatures. (Venting from relief valve operation leads to small radiation doses, from tritium.)

15.3.3.1, Steam or Feed Line Pipe Break

The events considered in this group include main steamline rupture, break in a steamline from a superheater to the main steam header, break in a saturated steamline between the steam drum and the superheater, feedline break, and recirculation line break. The reactor is tripped by a steam-to-feedwater flow ratio trip or by a secondary shutdown system (e.g., low steam drum level); the Na coolant flow is maintained by the pony motors at lower velocity so that the temperature transient is experienced by the core only after 150 sec. The plant design must accommodate the temperature transients and the steam venting in cells in the steam generator building.

15.3.3.2, Loss of Normal Shutdown Cooling System

Loss of the main condenser is equivalent to loss of the normal shutdown cooling; consequences are similar to the case of failure of the steam bypass system. The reactor would be scrammed, and the pressure transient would lead to relief valve venting. Core temperatures would be like those following a normal trip.

15.3.3.3, Large Sodium-Water Reaction

A large, sudden leak in a steam generator tube is the most likely source of a large sodium-water reaction. The leak could have a flow equivalent from a double-ended guillotine failure, and it is considered possible that the water/steam jet could propagate the leak to a maximum of two more tubes. The injection of water into sodium creates a high-pressure pulse which is relieved by rupture discs in the IHTS, draining off the sodium and reaction products and automatically isolating and depressurizing the water side of the steam generators in that IHTS loop. The reactor is tripped on steam-feedwater flow mismatch or on the sodium-water reaction; in either case, the trip is long before the temperature transient is transported back to the core. Calculated radiological consequences, from tritium in vented steam and from small amounts of in-leakage primary sodium in fumes vented from the sodium-water reaction, are about 0.012 mrem whole body, 0.04 mrem lung, and 0.02 mrem bone dose.

15.3.3.4, Primary Heat Transport System Pipe Leak

Analysis indicates that a leak of 75,000 gpm would be required for core sodium temperatures to approach saturation; leak detection capability is estimated to be better than 3 gpm. A 30 gpm PHTS pipe leak would not result in a measurable core transient; a normal reactor shutdown could be made. Coastdown of the pumps (less than a minute) would reduce system pressure and the leak rate. Several days would then be available for further reduction of the leakage. Consequences are plant downtime, sodium cleanup, and pipe repair.

15.3.3.5, Intermediate Heat Transport System Pipe Leak

In most break locations, a large leak will trip the reactor on primary-to-intermediate flow ratio trip. Possibly a large leak between the flow meter and the IHX might not trip the reactor on either primary-to-intermediate pump speed ratio trip; in such a case primary cold leg temperature would shut down the reactor. Reactor trip on flow or pump speed ratio would be early enough that the core would experience no temperatures above those of normal full-power

operation. A later trip, occurring 1 sec after the primary cold leg temperature trip setting is reached, could lead to increases in core sodium temperature but it would be expected to remain below 1,500°F.

15A.4 Local Faults (PSAR Section 15.4)

15.4.1.1, Stochastic Core Fuel Pin Failure

A stochastic failure is a random pin failure that is unpredictable; it could result from an undetected random cladding defect. "Stochastic failure of fuel pins is an anticipated occurrence for the CRBRP and such failures can be accommodated easily. Experiments in support of FFTF supplemented by supporting analyses have shown that for any postulated fission gas release mode, there are no serious thermal effects on adjacent pins or structures within the core. A transient jet of gas could not produce cladding overheating sufficient to cause failure of a neighboring pin, and even a steady jet could not cause failures because of the internal flow resistance in the failed pin. It has also been shown that gas blanketing could not cause significant cladding overheating. Furthermore, a volumetrically large gas release from a pin could not stop local coolant flow long enough to cause significant cladding overheating. Based on experience with stochastic failures in other sodium-cooled plants, there should be no adverse mechanical effects from the expected mode of slow gas release through a small hole. For a postulated burst-release mode through a large rip in the gas plenum, no mechanical damage to the neighboring fuel pins or fuel assembly duct would occur. Finally, no adverse long-term effects of fuel pin failure are expected to occur even if some sodium logging of the fuel and some leaching of fission products would occur. Although fuel pin failure is expected to be such that only very small particles of fuel would be released, the potential for fuel release to cause flow blockage has been examined; from experience and analysis, no such blockage is expected. The plant is designed to accommodate operation with 1% failed fuel.

15.4.1.2, Overenriched Fuel Rod Failure

An overpower rod is one which contains pellets of an enrichment higher than intended for its position in the reactor core. Because of QA controls on pellet enrichment at the fuel fabricators, and because of mechanical keying of the fuel pins and fuel assemblies, it is highly unlikely that an overpower pin will be assembled and find its way into the wrong core location. Pins with maximum credible overenrichment placed into a peak power location may not fail immediately under severe conditions but are marginal for long-term operation and are likely to fail during normal expected transients. The CRBR statistical hot pin requires an overpower greater than 25% to reach the fuel incipient melting point; data for low burnup pins indicates even greater margins, by about 20%. Tests operating at overpower sufficient to melt fuel at the pellet middle have shown that operation with 20% to 30% of the pellet molten will not necessarily fail the cladding. If the cladding should fail with injection of molten fuel into the sodium, it does not appear that damage would be propogated further, to other fuel pins or to the duct.

15.4.1.3, Flow Blockage in a Core Assembly

This analysis considers first various potential sources of a blockage, and then the potential effects of blockage. Blockage initiators considered are wire

wrap failure, excessive pin bowing or clad swelling, and foreign matter in the primary coolant. Experience and analysis indicate that wire wrap failure, excessive pin bowing or clad swelling are both improbable and not effective in forming blockages sufficient to be of concern. Inlet design precludes flat plate blockage as occurred in Fermi Unit I, and a combination of multiple orifices and strainers makes plugging by debris unlikely. For fuel damage to be experienced, an appreciable quantity of foreign matter (with diameters less than 0.25 in.) would have to be deposited in a nonrandom fashion in a fuel assembly. Mechanisms to prevent such occurrence are cleanliness during construction, requirements on cleanliness of sodium during filling and operation, preoperational cleanup of the primary system with the core special filter assemblies, and use of the cold trap during normal operation. Evaluation of potential effects of corrosion products, lubricants and other degradable materials, and failed fuel debris indicates little chance for blockage. Potential pin bowing or swelling may result in local hot spots on cladding but will not lead to significant blockage. Experimental data and analysis of blockages indicate extensive blockage, for example, more than 50% of inlet flow area, must occur before occurrence of reduction in flow significant with regard to temperature increases. Even a heat-generating blockage, comprised of a significant amount of fuel material from failed fuel, would need to be relatively nonporous to restrict flow enough to cause significant temperature effects on the fuel.

15.4.1.4, Postulated Module Inlet Blockage

The combination of multiple orifices with a variety of spacing and locations, some with intervening debris barriers, essentially precludes blockage in significant quantity. It would take a large mass of particles 0.5-in. diameter or smaller to plug enough of the strainer or spaces in the fuel assembly to result in significant temperature increases in the assembly.

15.4.1.5, Small Gas Bubble Passage Through Fuel, Radial Blanket and Control Rod Assemblies

Because bubbles passing through the core can cause a positive reactivity insertion and additional heating of the cladding, design to preclude bubbles entering the core includes venting from potential gas pockets at the core support cone, in the IHX and elsewhere, and pump design and a low cover gas pressure to reduce gas entrainment. The lower plenum design also is intended to break up any large bubbles. If a small bubble, for example, 4 in. high and four assembly rows in radial extent, should pass through the core, the power burst of about 0.1 sec at about 30% increase would cause a cladding temperature increase of less than 25F°; repetition of this a million times would not be expected to cause significant cladding damage.

15.4.2.1, Stochastic Absorber Pin Failures

Quality assurance similar to that for fuel makes the likelihood of stochastic absorber pin failure small. Failure of an absorber pin causing a helium gas jet to impinge on an adjacent pin could raise its clad temperature by only 110F° which for maximum initial cladding midwall temperature and linear heat rate would be 1,335°F, well below the 1,600°F guideline limit for short transients. Gas blanketing duration would be only 0.15 sec, much less than the 2.3 sec required to reach 1,600°F. The maximum force impulse from a gas jet is

also insufficient to damage an adjacent pin by mechanical action. Although cladding failures are expected to be of such small sizes that erosion of the boron carbide (B_4C) is unlikely, erosion tests of irradiated boron carbide have resulted in an average particle diameter of 0.0012-in., with a small number of larger particles expected. From this, it is expected that the smaller particles will circulate with the coolant without causing significant blockage or erosion, and that larger particles will settle out before reaching the pump bearings.

15.4.2.2, Overpower Control Rod Assembly

Mechanisms to cause an overpower condition are a wrong core location, an overenriched pin in a lower enrichment assembly, overenriched pellets, and improper orificing. Mechanical discriminators act against mislocation of control assemblies or control pins; quality assurance reduces the likelihood of overenriched pellets and improper orificing. Primary control pins containing 55% boron-10 are used in outer assemblies; the central one uses natural, 20% boron-10; thus, an overenriched case must be in the central position. Overpower leads to higher temperatures, and increased helium release in the cladding and thus to higher gas pressures. Because of the large margin below boron carbide's melting temperature and no expectation of pellet-cladding interaction, the expected pin failure mechanism is by gas overpressure. In the overpower case, at end of equilibrium cycle, the plenum pressure would be almost at the preliminary design guideline maximum of 4,000 psi; therefore, cladding failure may occur. Transients could raise the temperature and initiate failure earlier. It is expected that pinhole failure is more likely than gross rupture, but if rupture occurs, the erosion rate is expected to be small. Because the eroded particles are expected to be no bigger than 0.002-in. diameter and substantial portions dissolve, blockage resulting from their deposition is not likely.

15.4.2.3, Flow Blockage of a Control Assembly

Measures to avoid blockage of a control assembly are the same as those to avoid blockage of a fuel assembly. However, the control assemblies are operated at lower linear power than fuel assemblies and are overcooled. Thus, control assemblies require an appreciably larger percent of their flow area to be blocked before they will start to boil sodium coolant.

15.4.3.1, Stochastic Radial Blanket Pin Failure

As with fuel and control rods, conservative design and quality assurance will keep the numbers of stochastic radial blanket pin failures small. Expected forms of failure are small leaks in the clad releasing gases, and entry of sodium into the fuel at cracks in the cladding. Radial blanket pins have a large fission gas plenum and small fission gas pressure; these, in addition to high gas flow resistance because of small pellet-to-cladding clearance, will make the outleakage of gas suboptimal for failure propagation. Under conservative maximum conditions, a gas jet will not cause adjacent cladding to exceed 1,541°F nor to be blanketed longer than 0.05 sec, which is adequately less than the 0.096 sec calculated as needed for the cladding temperature to reach 1,600°F. Similarly, the impact of the gas jet is not enough to cause mechanical damage. Most of the analysis is extrapolated from that for fuel, and usually favors the blanket resulting from larger diameter cladding (0.253-in. vs. 0.107 in.) and smaller wire wrap spacing (2 in. vs. 6 in.). There is some experimental

basis for concern about sodium-fuel reactions leading to swelling if sodium leaks into cracked pins; experience indicates this is not a serious danger.

15.4.3.2, Overpower Radial Blanket

Cooling requirements for radial blanket assemblies depend on reactor residence time and proximity to the reactor core center. The requirements are met by coolant flow control plus shuffling of the assemblies within the radial blanket zone. Coolant flow control is by fixed orifices in the inlet modules. A pre-programmed shuffling routine including numerous safeguards makes misplacement of an assembly highly unlikely. The event analyzed, judged to be the worst case, is repeated failure to shuffle elsewhere an assembly which remains in the highest flux position; this event requires only a human error in setting up the shuffling program. Within a year after failure to shuffle, some melting of the oxide fuel would be experienced. Cladding failure where cladding temperature is maximum would be expected; this is above the point of molten fuel ejection and below the top of the pellet stack so the release expected would be a less-than-maximum release of gas. The uncertainties are such that molten fuel ejection can not be excluded; a molten-fuel-coolant interaction is possible. Neither it nor a gas jet release are expected to lead to failure propagation from assembly to assembly. The molten fuel jet could create a small penetration on the duct wall and in the duct wall of the adjacent fuel assembly, but no further consequences.

15.4.3.3, Flow Blockage of a Radial Blanket Assembly

As in other parts of the core, flow blockages are considered extremely unlikely because of inlet design and precautions to be taken in cleanliness of the coolant system and sodium. The prevention and detection systems for blockages are the same as for fuel assemblies, except that delayed neutron detectors will be relatively ineffective because blanket pellets contain fissile material fractions appreciably smaller than fuel pellets. As with fuel assemblies, a large blockage fraction is required for a significant reduction in coolant flow and increase in outlet temperature; generally, to cause sodium boiling, a blockage occupying half or more of the flow area of the assembly is required. Smaller blockages might raise cladding temperatures locally but go undetected because of the minor increase in bulk sodium temperatures; such hot spots would reduce the effective lifetime of the blanket pin cladding.

15A.5 Fuel-Handling and Storage Events (PSAR Section 15.5)

15.5.2.1, Fuel Assembly Dropped Within Reactor Vessel During Refueling

The in-vessel transfer machine (IVTM) draws a fuel assembly out of the core into a tubular housing, then drops it into either an open lattice position or onto a location occupied by another assembly. Fission gases and fuel particles might be released through damaged cladding to the primary heat transport system. Severe mechanical damage to the IVTM and other hardware could occur, especially if the triple rotating plugs in the reactor head are operated. In the worst case, release of some cover gas might occur, with results as in Section 15.5.2.4.

15.5.2.2, Damage of Fuel Assembly From Attempt To Insert a Fuel Assembly Into an Occupied Position

A series of procedural errors and interlock failures would be necessary to lead to damage of a fuel assembly by attempting to insert it into an occupied location. Consequences are most likely within the umbrella of Section 15.5.2.4. (below).

15.5.2.3, Single Fuel Assembly Cladding Failure and Subsequent Fission-Gas Release During Refueling

Spent fuel assemblies are in the core, the IVTM, the ex-vessel transfer machine, the fuel handling cell, and the ex-vessel storage tank. It is judged that the EVTM is where containment failure is most likely and could lead to the largest release for a single fuel assembly. The analysis assumes release of 100% of the noble gases and halogens in the fuel assembly to the interior of the EVTM, with subsequent diffusion of the radioactivity through the EVTM seals into the combined space of the reactor containment building and reactor services building. The release passes the RSB filtration system and is calculated for fuel with 36 hours' decay time after reactor shutdown, and for fuel with 87 hours decay time; the maximum 2-hour dose at the site boundary is 1.89 rems to the thyroid.

15.5.2.4, Cover-Gas Release During Refueling

Cover gas may be released during refueling by mechanical damage, by improper sequencing of refueling motions, or by separation of the auxiliary handling machine from an open floor valve during a seismic event. The earliest time this could occur is 30 hours after shutdown, during which period the RAPS has been in normal operation, cleaning up releases from 1% failed fuel. The cover gas is considered released instantaneously to the RCB and RSB and then to the environment, as if the railroad door in the RSB was open in violation of procedures. The maximum 2-hour site boundary is estimated at 4.4 mrems whole body.

15.5.2.5, Heaviest Crane Load Impacts the Reactor Closure Head

The reactor head structure is designed to accommodate the impact of the 100-ton auxiliary handling machine at maximum lowering speed of 5 ft per minute. If contrary to expectations, this should result in release of the cover gas, the results will not exceed those in Section 15.5.2.4 below, 4.4 mrems whole body at the site boundary.

15.5.3.1, Collision of Ex-Vessel Transfer Machine with Control Rod Drive Mechanisms

The EVTM is a railway-mounted cask-type fuel-transfer machine which moves during refueling to within about 1 ft of the CRDMs. This only occurs when the reactor is shut down and all control rods are fully inserted and disconnected from the CRDMs. The rail structure includes anti-liftoff and anti-overturning restraints. In addition to travel limit switches, there are fixed mechanical stops on the rails to limit the EVTM travel. If, in spite of procedural limits and all electrical controls and mechanical limits, the EVTM collides with the CRDMs, it could cause a release of cover gas through the CRDM seals; at the limit, the consequences would not be worse than those in Section 15.5.2.4 below, 4.4 mrems whole body at the site boundary.

15A.6 Sodium Spills and Fires (PSAR Section 15.6)

15.6.1.1, Primary Sodium In-Containment Storage Tank Failure During Maintenance

Analysis of postulated in-containment sodium fires shows that the limiting event with respect to containment building temperature and pressure resistance capability is the postulated failure of a full primary sodium storage tank during maintenance, with the tank cell deinerted and open to the upper containment volume. The tank is in normally inerted cell 102A below the operating floor; the cell has a steel liner and its walls are 6-ft-thick concrete. The event considered assumes that, with the tank cell deinerted and interfacing with the atmosphere of the RCB, the maximum volume of 35,000 gal of sodium is dumped from the tank onto the cell floor with immediate start of pool burning. It is assumed that the sodium coolant contains its maximum concentration of radionuclides. It is calculated that during the 550 hours of combustion, 179,000 lb of sodium is burned. Most of the fire takes place within the first 70 hours, during which the pressure peaks (0.8 psig at 20 hours) and the temperature peaks (wall temperature 122°F at 70 hours). A total of 3.4 kg of aerosol containing 2.5 kg sodium is released during the 70 hours of containment overpressure. The estimated maximum 2-hour, site-boundary dose to an individual is 0.288 rem to the bone.

15.6.1.2, Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation

The EVST has two normal forced-convection-cooling circuits, used alternately, and one backup natural convection circuit. Each normal circuit is in a cell below grade in the reactor service building, as is the EVST. The event considered is the break of a cooling circuit at the low point of the pump suction line, maximizing the spill resulting from siphoning; about 7,500 gal of 475°F sodium would be spilled into the sealed, inerted cell of 14,950 ft^3 volume. A fire lasting about 9 hours ensues, peaking in 3.6 hours at 254°F and 3.8 psig in the cell (within the retention capability of the cell). The analysis assumes maximum radionuclide concentrations in the sodium, the same proportions of radionuclides in the combustion product aerosol, and that all the aerosol is released directly to the environment. The maximum individual dose calculated over 2 hours at the site boundary is 0.713 rem to the bone.

15.6.1.3, Failure of an Ex-Containment Primary Sodium Storage Tank

The two ex-containment primary sodium storage tanks are in a cell on the lowest level of the steam generator building. These tanks will be used to store sodium only if maintenance requires the drainage of more than one PHTS loop or the EVST, and so forth. When a tank is to be used, the cell is sealed and inerted. The event considered is the spill of all the contents of one tank, 45,000 gal of 450°F sodium onto the cell floor, where it is held by a 3/8-in. steel catch pan. The analysis assumes the ensuing fire consumes all the oxygen in the cell, that the proportion of radionuclides in the combustion product aerosol is the same as in the end-of-life sodium, and that all the aerosol is released to the environment without fallout or other depletion. The maximum individual dose calculated over 2 hours at the site boundary is 3.19 rems to the bone. The cell is designed to withstand the temperatures and pressures generated by this event.

15.6.1.4, Primary Heat Transport System Piping Leaks

The PHTS is entirely within the reactor containment building, either in cells or in the reactor cavity. A cell-design-basis leak was chosen, with flow equivalent to the flow from a sharp-edged circular orifice with an area equal to one-half the pipe diameter times one-half the pipe wall thickness. The maximum leak rates for the PHTS cell and the reactor cavity are 947 and 847 gpm, respectively. The design requirement is that there should be no damage to the cell that could impair the safety functions of other equipment. The leak in the PHTS cell is taken as 947 gpm for 4.5 min (until reactor trip on low sodium level) followed by pump coastdown of 1 min and 500 min at pony-motor-induced flow. The spill is taken as 35,000 gal at 1,015°F. The leak in the reactor cavity is similar, beginning with a flow of 847 gpm and spilling 29,000 gal of 750°F sodium. The larger fire is estimated to burn about 423 lb of sodium, and all the aerosol is assumed released to the containment. Automatic containment isolation follows detection of radioactivity in the RCB HVAC exhaust. Leakage for the containment is based on 0.5 psig pressure, applying a square-root relationship to the design leak rate of 0.1% of volume per day at 10 psig; bypass leakage is taken as 1% of the containment leakage. No credit was taken for aerosol depletion in the PHTS cell or outside the RCB; depletion inside containment was calculated with the HAA-3 code; a total of 3.6 g of sodium is released to the atmosphere over a 30-day period. The maximum individual dose calculated over 2 hours at the site boundary is 0.112 mrem to the bone.

15.6.1.5, Intermediate Heat Transport System Pipe Leak

For the IHTS, the design-basis leak was also taken to be equivalent to the flow from a sharp-edged circular orifice with an area equal to one-half the pipe diameter times one half the pipe wall thickness (e.g., for 24-in. pipe, 2.85 $in.^2$. The limiting case is a leak in the 24-in. (OD) main loop hot-leg pipe in cell 226, at the low point in the pipe. It is assumed that the operator takes no action but that the system is tripped by primary-secondary flow mismatch, so that loop flow continues via the pony motor, until the pump tank is emptied. The total spill is 300,000 lb of sodium at an average bulk temperature of 800°F. For consequence evaluation, this design-basis leak was combined with the maximum undetected IHX leak so that primary sodium leaks into the IHTS when its pressure drops; it was assumed that the leak continued at maximum rate for 2 hours and that operator action is taken within 24 hours to break the siphoning of primary sodium to the IHTS. Codes SPCA and SPRAY are used to calculate conditions in the cell, and HAA-3 to calculate the rate of discharge of aerosols to the environment. All aerosol formed is assumed released at ground level. An analysis was made to determine that the released aerosol would not impair the operation of the plant's safety-related equipment. It is assumed in the analysis that when aerosols are detected in the steam generator building HVAC exhausts, all SGB inlet and exhaust dampers are closed except for one 40-ft^2 opening which remains open 5 min. It is further assumed that protected air cooled condenser fan operation is interrupted for 5,000 sec, and that all other HVAC air intake openings throughout the plant, with the exception of the emergency diesel generator rooms, are automatically closed within 1 min. About 630 lb of aerosol are released. The maximum 38-day dose to an individual at the boundary of the low population zone is calculated to be 2.14 rem to the bone.

15A.7 Other Events (PSAR Section 15.7)

15.7.1.1, Loss of One DC System

The plant design includes three independent battery-supported Class 1E dc systems. The loss of one will not prevent the functioning of Class 1E dc loads because the redundant counterpart of the system that is lost will remain operational.

15.7.1.2, Loss of Instrumentation or Valve Air

All safety-related, air-operated valves will be designed to move in a preferred direction (or in certain cases, fail as is) with the loss of air supply; for example, containment isolation valves will fail closed. The systems supplying compressed air to safety-related valves or instruments will be designed such that a single credible failure will not cause interruption of the air supply. Loss of air supply will not have an adverse effect on safe operation of the plant.

15.7.1.3, Intermediate Heat Exchanger Leak

If an IHX leak occurs, in spite of design precautions, the leakage will be into the primary system from the intermediate system which is kept at higher pressure. Temperature-compensated sodium level sensing devices on each system will indicate to the operator any changes that result from leakage.

15.7.1.4, Off-Normal Cover Gas Pressure in the Reactor Coolant Boundary

The subject cover gas pressure is maintained at 6 ± 2 in. of water. There is a constant sweep flow into the cover gas spaces and through the shaft seals of the primary pumps. There is a gas pressure equalization line connecting the pumps, reactor vessel and overflow tank. Off-normal pressure could result from failure of the regulators controlling flow to and from RAPS. Underpressure would be limited to about 1 psia and would have no significant effect. Because the cover gas volume within the primary coolant boundary is about 4,500 ft^3 and the makeup rate is limited to about 50 scfm, an overpressure buildup will be gradual, allowing the operators to take effective action after the condition is annunciated. Relief valves set at 15 psig will limit the pressure, and CAPS action will prevent hazard to the public.

15.7.1.5, Off-Normal Cover Gas Pressure in the Intermediate Heat Transport System

The IHTS argon cover gas is taken from the steam generator building argon supply which is at 175 psig; the pressure in the pump and expansion tank cover gas is controlled to 100 psig. Failure of regulation on the supply could increase the pressure to 175 psig which would have no significant effect because the IHTS is designed for considerably higher pressure. Low cover gas pressure could result in sodium leakage from the primary system into the IHTS, with insignificant radiological effects since operator action would correct the pressure differential in a short time.

15.7.2.1, Inadvertent Release of Oil Through Pump Seal (Primary Heat Transfer System)

The primary sodium pump has oil-lubricated bearings and seals above the sodium-containing pump tank. For oil leakage to enter the sodium, it must get past the system that collects leakage oil in an oversized reservoir, and also past a shaft oil slinger and second collection reservoir. The oil supply is a fixed capacity (about 6 gal), smaller than either of the reservoirs, so that overflowing the collection reservoirs can be accomplished only by manual addition of excess oil. If oil should, however, enter the sodium, analysis indicates that the carbon particles in the sodium will be negligible, that the dissolved hydrogen will cause plugging temperatures low enough to have a good margin of safety, and that reactivity effects will be less than 1¢.

15.7.2.2, Inadvertent Release of Oil Through the Pump Seal into Sodium (Intermediate Heat Transfer System)

The release of oil to the IHTS sodium from the pump oil bearing requires the failure of multiple barriers, as with the PHTS pump oil. An undetected loss of the entire seal oil supply to the IHTS sodium would have consequences no more severe than those of release of pump oil seal to the PHTS, of intermediate loop pipe break, or of intermediate loop pump seizure.

15.7.2.3, Generator Breaker Failure to Open at Turbine Trip

In the event of a turbine trip, the turbine trip logic signals a generator load break switch opening and, independently, a generator field breaker opening. Failure of the generator load break switch to open causes loss of the preferred ac power supply, in which case the normal ac distribution system and the safety related ac distribution system are transferred automatically to one of the reserve transformers and thus to the other offsite power supply. Thus, consequences are negligible.

15.7.2.4, Rupture in Radioactive Argon Processing System Cold Box

The RAPS cold box contains the cryogenic still (cryostill) which separates krypton and xenon from the reactor's argon cover gas stream. It is assumed that the reactor has been operating a long time with 1% failed fuel and that the cryostill has not been unloaded to the noble gas storage vessel for 1 year, and thus the cryostill contains a maximum inventory of radioactivity. Assuming a cryostill rupture, the cold box cell heating and ventilating (H&V) radiation monitor will signal an alarm, close off the cold box influent and effluent lines, and open the bypass line. The liquid nitrogen will be valved off on a cell high-pressure signal. It is assumed that the RCB refueling door is open and that the released gas vents out of the H&V exhausts, without retention in the cold box cell. The maximum individual dose at the site boundary in 2 hours is 3.0 rems whole body.

15.7.2.5, Liquid Radwaste System Failure (Leak or Rupture)

The liquid radwaste collection and processing system is located in a nonhardened building attached to the reactor service building. The cells where the highest activity streams are processed are below grade in concrete cells coated and sealed to prevent leakage and facilitate decontamination; each has a sump and

sump pump to transfer spills. Each cell is closed during operation. One of the two 20,000-gal tanks when full would contain the largest inventory of radionuclides in the system. If failed, the contents would spread over the floor and drain into the sump. The operator, alerted by the low level indicator in the tank and the level indicator of the sump, would pump the sump contents to another tank. The most volatile radionuclide in the liquid waste is tritium; other radionuclides exist primarily in salts formed from cleaning sodium-contaminated components with water. For conservatism, the spill is evaluated giving no credit to the floor sealing, drains, or operator action, that is, the spill is released to groundwater, flows to the Clinch River, and is diluted therein and available for intake at the potable water facility for the K-25 plant. Maximum dose to an individual exposed to the liquid (including drinking) at the site boundary is estimated to be 0.078 rem to the thyroid. Release of 10% of the tritium to the air would give an estimated whole body inhalation dose to an individual at the site boundary of 3.7 μrems in 2 hours.

15.7.2.6, Failure (Leak or Rupture) in the Ex-Vessel Storage Tank NaK System

There are three NaK systems for cooling the EVST sodium, two active and one natural-draft standby. Each involves a Na/NaK heat exchanger outside the EVST and a loop with a NaK/air heat exchanger using outside air for a heat sink. The NaK systems operate at about 100 psig and about 350°F and are all-welded construction. Leaks of NaK in the inerted cells containing ex-vessel storage (EVS) components will have smaller impacts than those involving EVS sodium because the NaK systems have smaller liquid volumes at lower temperatures and are nonradioactive. Leaks in either the air atmosphere cells or the heat exchangers will result in localized fires handled by catch-pans, sealing of the volume, and extinguishing by the fire protection system or by nitrogen flooding. Consequences to the public consist of the release to the air of minor amounts of NaK combustion products.

15.7.2.7, Leakage From Sodium Cold Traps

In the CRBRP auxiliary systems there are the primary sodium cold traps (cooled by liquid NaK in a jacket welded to the outside of the trap), the EVS sodium cold traps (cooled with nitrogen), and the intermediate sodium cold traps (air cooled). All cold traps collect oxide impurities, tritium, and other radioactive materials as precipitates in their crystallizer component. A leak between the NaK and Na in a primary system cold trap will leak NaK into the primary sodium; loss of NaK inventory will signal the failure. A leak to cooling gas or to cell atmosphere will be signaled by leak detectors at the coolant outlet or in the cell; the trap will be isolated and replaced. A leak in any of the three types of cold traps will have negligible effect on either reactor safety or public safety. When a cold trap is to be removed, it is isolated from the processing system and the sodium allowed to freeze. The trap is cut out and the ends are capped and seal-welded. Once removed, all traps remain frozen without supplementary cooling. Calculated consequences of a design-basis cold trap fire (within the RCB with RCB leakage at 0.032% per day) indicate a maximum individual dose for 2 hours at the site boundary of 1.02 mrems to the bone.

15.7.2.8, Rupture in Radioactive Argon Processing System Noble Gas Storage Vessel Cell

The RAPS noble gas storage vessel normally contains radioactive gas which is unloaded annually from the RAPS cryostill. The gas is mainly argon but also includes krypton and xenon; both stable and radioactive isotopes are present. The gas is bled slowly from the vessel into the cell atmosphere processing system so that its pressure normally decreases over the year. The event analyses assumed that a few weeks after cryostill unloading, it is again unloaded so that the storage vessel contains two charges and is at maximum pressure when it ruptures. A radiation monitor initiates a signal closing off the cell vent line to CAPS. It is assumed that the cell leaks the gases quickly into the RCB, that the RCB refueling door is open, and that the released gases vent out through the H&V exhausts. The estimated maximum dose to an individual at the site boundary for 2 hours is 3.0 rems whole body.

15.7.2.9, Rupture in the CAPS Cold Box

The CAPS System cold box contains two charcoal delay beds in series, which absorb xenon and krypton from the process stream before it is discharged to H&V. The beds are cryogenically cooled by injecting liquid nitrogen into the influent stream. The event analysis assumes a rupture of the charcoal delay beds during refueling when the bed inventory of radionuclides is greatest. Normally the cell H&V radiation monitor would initiate an alarm and closing of the H&V vent, the tritium water removal drain, and process gas flow. High cell pressure would trip off the liquid nitrogen flow. The event analysis assumes all the radioactivity is immediately released from the RSB, leading to an estimated maximum 2-hour site boundary individual dose of 0.14 rem whole body.

15.7.3.1, Leak in a Core Component Pot

A core component pot is a long stainless steel thimble used to hold a fuel assembly or control assembly in transferring them from the reactor vessel to the EVST or reverse. A CCP contains about 22 gal of sodium in addition to the core component; the plant may have 600 or more CCPs. A CCP could fail because of defective manufacture, accidental damage, or corrosion. The limiting event analyzed is undetected loss of sodium from the CCP immediately before transporting a 20 kW-decay-heat fuel assembly in the CCP in the ex-vessel transfer machine. Normally, the time between a CCP emerging from the reactor sodium surface and submerging beneath the EVST sodium surface is 56 min. The event analysis assumes a fuel unloading procedure unperturbed by low-load signal from the grapple load cell, or by the monitor signaling high radiation levels in the EVTM. After 10 min, the fuel would heat to cladding temperatures of 1,500°F and random cladding failure could release some fission gas; the EVTM has a 20-kW capacity gas flow cooling system but the fuel would be insulated by the still gas between it and the CCP. After about 17 min, cladding begins to melt; after 30 min it is melting on all rods. After about an hour, the fuel duct reaches the melting point locally. Even after 1.5 hours, the CCP (at 1,900°F) would maintain its integrity (if it had sufficient integrity initially). It is assumed that all the volatile fission products are released to the inside of the EVTM (if they are liquid or vapor at 3,500°F). The release is assumed to be by diffusion through the EVTM seals (calculated to reach 260°F) to the

RSB/RCB and to involve only radionuclides mobile at 260°F. The results are said to be represented by those of Section 15.5.2.3 (above), in which the maximum 2-hour site boundary dose to an individual is estimated at 1.89 rems to the thyroid.

15.7.3.2, Spent Fuel Shipping Cask Drop From Maximum Possible Height

The maximum possible height for a spent fuel shipping cask to drop is 72 ft from the operating floor of the RSB to the bottom of the cask-handling shaft. Because of design features (such as redundant crane rigging) and operational precautions, dropping a cask is considered extremely unlikely. Cask design is such that the drop is not expected to fail the seals. It is assumed that the cask is loaded with fuel cooled only 80 days (vs. 100 minimum) so that it is at its capacity heat load, 26 kW. It is assumed that all fuel rods on six fuel assemblies fail, releasing their volatiles in the cask's inner canister and that they leak out through the cask's inner seals at the maximum rate of 6×10^{-5} cm^3 per second (standard), adjusted for the canister pressure; no credit is taken for the cask's outer seals nor for holdup in the RSB/RCB. The maximum estimated dose to an individual off site is 0.541 mrem to the thyroid, at the LPZ over 30 days.

15.7.3.3, Maximum Possible Conventional Fires, Flood, Storms, or Minimum River Level

Fire: The plant's fire protection system is designed to provide adequate fire protection, detection, and signaling in areas where a hazard may exist; the nearest tree line is kept 100 yd away to minimize forest fire impact.

Flood: Plant grade is at the 815-ft elevation; maximum flood is estimated at 809.2 ft including combined one-half probable maximum flood, OBE with dam failures and wave runup.

Storms: Drainage facilities for safety-related structures are designed for a maximum 1-hour rain of 14 in. and an 8-hour depth of 29.5 in.; maximum recorded rain is 7.75 in. in 24 hours. The roof is designed for 40 in. of snow, compared with a maximum monthly snowfall of 21 in. Wind design for safety-related structures is for 90 mph at 30 ft above grade, compared to a peak recorded gust of 59 mph at Oak Ridge. Tornado design is for a maximum velocity of 360 mph.

Minimum River Level: The minimum river level is taken as el. 735 ft, and the intake structure is 5.5 ft lower than that.

15.7.3.4, Failure of Plug Seals and Annuli

A transient causing excessively low cover gas pressure at the reactor head, followed by a return to normal, could displace the liquid in the plug annulus dip seals enough to allow cover gas under the seal blade and up into the inner buffer annulus between the dip seal and the inflatable seals. This could increase the rate of leakage of radioisotopes into the head access area; factors against this being significant are (1) there are two outer seals with a buffer space between them and these seals have more than adequate pressure resistance, (2) expected leak rates cause a small fraction of the allowable concentrations in the head access area, and (3) a radiation monitor in the head access area can signal any need for evacuation and isolation of containment.

15.7.3.5, Fuel Rod Leakage Combined with Intermediate Heat Exchanger and Steam Generator Leakage

This event assumes undetected IHX leakage in the same loop with a steam generator leak. Normally, pressure differentials cause leakage to flow from the steam generator into the IHTS and from the IHTS into the PHTS, thus preventing outward migration of radioactivity. Plant procedures are to maintain these pressure differentials or to drain loops; only unlikely events would reverse the pressure differentials and only for short time periods, so that risks of radiation exposure from such leakage are minimal.

15.7.3.6, Sodium Interaction With Chilled Water

For an interaction between sodium and water of the chilled water systems to occur, two pipe failures and a third boundary must fail simultaneously. In recirculating gas fan-cooler units and HVAC coolers serving areas containing sodium piping or equipment, water leakage will trigger redundant moisture or leakage detectors, isolating the unit flows and opening drain valves to remove the water. If automatic response fails, the design permits 2 hours for operator action before water enters areas containing sodium piping. In the RCB and RSB, floor drain system leak detectors will signal leakage, allowing isolation of the affected water line. Upon confirmation of a sodium leak in an area served by a recirculating gas cooling system, the water and gas lines to the cooler serving the area will be valved off.

15.7.3.7, Sodium-Water Reaction in Large Component Cleaning Vessel

The large component cleaning vessel (LCCV) is located in the large component cleaning cell in the lower part of the reactor containment building. The event considered is a runaway reaction of sodium with water in the LCCV. Under normal operations, residual sodium or NaK on a component is removed by a controlled reaction with water vapor at 150°F in a nitrogen-purged atmosphere; the reaction is controlled by regulation of the water vapor concentration with the aim of maintaining hydrogen concentrations below 1%. A runaway reaction could occur by flooding the LCCV with water before the sodium reacted with steam. The analysis considers cleaning the intermediate rotating plug (IRP) (a once-in-30-year task), and reacting water with 200 lb of sodium on the bottom plate of the IRP; the resulting pressure in the LCCB (designed for a static rupture pressure of 298 psig) is between 40 and 85 psig. The event assumes release of 100% of the radioactivity in the 200 lb of sodium, decayed for 10 days. Such a release would isolate the RCB and the effluent would pass through the filter system en route to the environment, with decontamination factors of 20 for iodine and 100 for particulates. The estimated maximum individual dose at the site boundary is 50 mrems to the thyroid.

15B APPENDIX: STAFF EVALUATION OF SELECTED FUEL-HANDLING ACCIDENTS

The CRBR spent fuel assemblies will be handled as single entities by the ex-vessel transfer machine (EVTM), the fuel-handling cell (FHC) gas cooling grapples, and the in-vessel transfer machine (IVTM). Based on the design of these systems, the applicants have provided an analysis of the radiological consequences of two fuel-handling accidents: a nonmechanistic cover gas release accident and a nonmechanistic EVTM fuel-handling accident.

The applicants have assumed that at the time either accident occurs, the refueling hatch, connecting the reactor service building (RSB) and the reactor containment building (RCB), will be open. However, during refueling the railroad door in the RSB cannot be open at the same time as the refueling hatch. Therefore, any radioactive release in the RCB or RSB during refueling when the refueling hatch is open will be confined to these two buildings. In addition, the annulus filtration system will be in operation and in a partial recirculation mode during refueling as will the RSB filtration system. The ESF filters of both system will be on line during refueling operations.

The staff has done an independent evaluation of both the EVTM accident and the cover gas release fuel-handling accident using techniques similar to those specified in SRP Section 15.7.4. The assumptions for the staff's evaluation of the EVTM fuel-handling accident are listed in Table 15B.1 and the assumptions for the cover gas release fuel-handling accident are listed in Table 15B.2. The doses are presented in Table 15B.3. As a result of the staff's independent assessment using conservative assumptions, the staff concludes that the proposed design and operation of the fuel-handling system during refueling operations are acceptable, and that the doses resulting from an EVTM or cover gas release fuel-handling accident will be well within 10 CFR 100 dose guideline values.

Table 15B.1 Assumptions used in ex-vessel transfer machine fuel-handling accident

Parameter	Value
Power level (MWt)	1,121
Number of fuel rods in bundle	217
Number of fuel rods in core	33,852
Failed fuel rods (%)	1
Radial peaking factor	1.20
Inventory released from failed rods, iodines, and noble gases (%)	10
Filter efficiency for iodine removal--elemental (%)	95
Atmospheric diffusion factors, χ/Q (sec/m^3)	
2 hours--exclusion area boundary	1.2×10^{-3}
8 hours-low population zone	1.2×10^{-4}

Table 15B.2 Assumptions used in cover gas release fuel-handling accident

Parameter	Value
Power (MWt)	1,121
Number of fuel rods in core	33,852
Failed fuel rods (%)	1
Radial peaking factor	1.20
Inventory released from failed rods, noble gases (%)	10
Atmospheric diffusion factors, χ/Q (sec/m^3)	
2 hours--exclusion area boundary	1.2×10^{-3}
8 hours--low population zone	1.2×10^{-4}

Table 15B.3 Fuel-handling accident doses

	Accident doses* in rems to:			
Accident and location	Thyroid	Whole body	Lung	Bone
Ex-vessel transfer machine fuel-handling accident				
Exclusion area	6	1	1	1
Low population zone	6	1	1	1
Cover gas release fuel-handling accident				
Exclusion area	0	1	1	0
Low population zone	0	1	1	0

*All doses are based on staff practice for light-water reactors (International Commission on Radiological Protection, ICRP 2) discussed in NUREG-0172 and adjusted for CRBRP considerations as discussed in NUREG-0786.

16 TECHNICAL SPECIFICATIONS

16.1 General

PSAR Chapter 16 was reviewed according to the guidelines in SRP Section 16.0 and the Standard Technical Specifications for Westinghouse pressurized-water reactors (NUREG-0452). The purpose of this review was to identify areas not currently being addressed for inclusion in the Technical Specifications so that any impact they may have on the construction of the plant could be factored into the design.

16.2 Evaluation

The Technical Specifications set forth limits, operating conditions, and other requirements imposed on facility operation to ensure that the facility operation remains within the bounds and assumptions of the facility safety analysis. Accordingly, Technical Specifications are required to ensure the protection of health and safety of the public.

For LWRs generic standard technical specifications are available as guidance in preparing plant-specific technical specifications. For CRBR no such generic technical specifications exist; therefore, a more detailed review of these specifications was performed at the PSAR stage to identify those areas not addressed in PSAR Chapter 16 for which the staff considers technical specifications should be provided. The CRBR Technical Specifications contained in the PSAR are in the initial stage of development with many items of a preliminary or "to-be-determined" nature. Therefore, comments on wording, numerical values, or the detailed contents of each specification contained in these Technical Specifications are not provided as part of this evaluation, since it is expected that these specifications will be refined and revised as part of the final design. What was looked for at this stage was whether or not all areas of the plant which the staff considers appropriate for inclusion in the Technical Specifications were in fact included.

The following are items not included in the Technical Specifications submitted with the PSAR, but which in the staff's opinion should be included in the final Technical Specifications unless justification can be provided for their exclusion. The plant design should make provision for monitoring these items.

(1) reactor

   (a) power rate increase limit consistent with that assumed in the fuel design and supported by experimental data

   (b) burnup limits for the inner and radial blankets

   (c) reactivity anomaly limits based on the difference between expected control rod positions and actual control rod positions

(d) limits on core subassembly outlet temperatures, outlet temperature instrumentation operability, and testing

(2) heat transport system (HTS)

(a) primary and intermediate system relief valve operability, setpoints, and testing requirements

(b) maximum allowable temperature difference between primary HTS cold legs based on assumptions used in the reactor vessel stress analysis

(c) recording of plant design thermal transients to ensure the number and type of transient have not exceeded that assumed in the design

(d) minimum and maximum pump coastdown time, consistent with that assumed in the safety analysis and testing requirements

(e) HTS pony motor operability

(f) drain valve configuration (e.g., locked closed and frozen sodium)

(g) intermediate heat exchanger and steam generator sodium-side gas vent line flow (monitor flow to ensure lines are open and able to vent gas)

(h) maximum and minimum sodium inventories in the reactor vessel and intermediate loop expansion tanks

(i) operability of the turbine overspeed protection system

(j) turbine disc inspection requirements

(k) seismic snubber operability and inspection requirements

(l) maximum activity of intermediate heat transport system (IHTS) sodium and IHTS cold traps

(m) maximum activity of primary heat transport system cold trap

(3) containment

(a) containment isolation system operability, setpoints, and testing requirements

(b) containment isolation valve operability (including leak testing and closure time requirements)

(c) containment penetration configuration

(d) containment heat removal system and vent/purge system operability and testing requirements

(e) operability and testing of containment vacuum breaker system

(f) containment temperature, pressure, and relative humidity limits

(g) containment annulus filtration system operability and testing requirements, including negative pressure requirements

(4) reactivity control system

(a) shutdown margin limits

(b) control rod lifetime limit

(c) control rod maximum withdrawal speed and testing requirements

(5) instrumentation

(a) delayed neutron monitor operability, setpoints, and testing requirements

(b) postaccident monitoring instrumentation operability and testing requirements

(c) control room habitability instrumentation operability and testing requirements

(d) sodium leak detection system instrumentation operability and testing requirements

(e) HTS instrumentation operability and testing requirements

(f) seismic monitoring instrumentation operability and testing requirements

(g) fire detection system instrumentation operability and testing requirements

(h) loose parts monitoring system operability and testing requirements

(i) remote shutdown system operability and testing requirements

(j) plant protection system monitor operability and testing requirements

(k) meteorological instrumentation operability and testing requirements

(6) miscellaneous

(a) maintenance of adequate trace heat temperature on

- standby ex-vessel storage tank (EVST) cooling system
- overflow heat exchanger
- reactor head and primary and secondary control rod drive mechanisms

- primary system argon relief lines
- primary and intermediate system cover gas equalization lines and primary system gas lines to relief valves

(b) operability and testing of safety-related interlocks

(c) limits on control room environment and operability and testing of control room habitability and safety-grade heating and ventilating systems

(d) fuel-handling cell decay heat limits (both individual assembly and total cell)

(e) limit on water vapor nitrogen sodium cleaning process temperatures to less than 190°F

(f) limit on radioactive inventory of cell atmosphere processing system

(g) setpoints, operability, and testing requirements for

- reactor service building filtration system
- steam generator building aerosol release mitigation system

(h) restrictions on lifting loads over the reactor and EVST

(i) fire suppression system operability and testing requirements

(j) maximum activity of the sodium in the ex-containment primary sodium storage tank

In addition to the above specific items the format for the final Technical Specifications should be consistent with that used in LWR standard technical specifications.

Final review of the detailed content, wording, and limiting values of the Technical Specifications will be performed at the operating license review stage. It is suggested that before the FSAR is submitted, further iteration and review of these specifications by the staff and applicants take place so that a complete detailed set of specifications is developed in time to be factored into the final design.

17 QUALITY ASSURANCE

17.1 General

The quality assurance (QA) program for the design and construction phase of the CRBRP is described in Chapter 17 of the Preliminary Safety Analysis Report (PSAR). Section 17.1 of the PSAR describes the organizational relationships and provides an overview of the QA program. Appendices to Section 17.1 describe the QA programs of the owner, the fuel supplier, the nuclear steam supply system supplier, the architect-engineer, the constructor, and the reactor manufacturers. Section 17.2 of the Final Safety Analysis Report will describe the QA program for the operations phase.

The staff's evaluation of this QA program is based on its review of PSAR Chapter 17 and discussions with representatives of organizations with QA programs described in PSAR Chapter 17. The staff assessed the QA program for the design and construction phase to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and if it meets the acceptance criteria of Section 17.1 of the Standard Review Plan (NUREG-0800).

17.2 Organization

17.2.1 Owner

The Department of Energy (DOE), as the owner of the CRBRP, is responsible for the total QA program for the plant and is organized to control and verify the QA programs of the other principal organizations. The relationship of the owner and the other principal organizations is shown in Figure 17.1.

The CRBRP Project Office executes the responsibilities of the owner for the design and construction of the plant. The organizational arrangement for the overall QA program for the CRBRP is shown in Figure 17.2. The owner's organization for the design and construction phase of the CRBRP is shown in Figure 17.3. The Project Director is responsible for the design, procurement, manufacturing, and construction of the plant. He directs all activities of the CRBRP project, and he has overall responsibility for the QA program for these activities. He has approved the CRBRP Project Office's QA Manual and has made the overall QA program mandatory for all activities conducted for safety-related plant structures, systems, and components by all involved organizations. The purpose of the QA program is to ensure the quality of the CRBRP.

The Chief, QA, reports to the Project Director and manages the Quality Assurance Division. He is responsible for establishing, administering, and coordinating the owner's QA program. He is free of prime responsibility for cost and schedule and is organizationally located at the same level as those whose work he verifies. This arrangement should ensure that the owner's QA organization has adequate independence and reports at a sufficiently high management level to permit it to accomplish its objectives effectively.

The Chief, QA, directs implementation of the QA program described in the PSAR. He is responsible for developing the QA program and monitoring its implementation and effectiveness. The QA program is described in greater detail in the QA Manual, which has been approved by the Project Director. The program is implemented through the QA Manual, which provides the details of how each of the Appendix B criteria is met. The staff finds that the PSAR has clearly defined the responsibilities and authorities of the owner's QA organization.

The owner implements its QA functions by means of the QA organization shown in Figure 17.4. This organization is primarily located in Oak Ridge, near the construction site, with some personnel located at contractors' offices. QA personnel have the authority and organizational freedom to (1) identify quality problems; (2) initiate, recommend, or provide solutions; and (3) verify implementation of solutions. The Chief, QA, and his staff, consisting of approximately 15 QA personnel, have the authority to halt (i.e., stop work authority) the continuation of activities performed by the owner or by contractors, including procurement and construction site activities, that could adversely affect the quality of the safety-related structures, systems, and components. The Chief, QA, is responsible for establishing and implementing a system of audits and surveillance of construction, installation, and construction testing activities at the site.

The staff has evaluated the independence, authority, and responsibility of the Owner's QA organization and concludes that the organization complies with 10 CFR 50, Appendix B, and is acceptable.

17.2.2 Fuel Supplier

The U.S. Department of Energy's Richland Operations Office (RL), as the fuel supplier, is responsible for ensuring that the fuel and blanket assemblies are fabricated, assembled, and tested so that they will conform to the design requirements. Westinghouse Electric Corporation, as the supplier nuclear steam system (see Section 17.2.3), is responsible for the fuel and blanket assembly design and design quality assurance.

Figure 17.5 shows the RL QA organization. RL operates under the RL Manager who has mandated the implementation of the QA program. He has assigned responsibility for establishing and managing the fuel supplier QA program to the Director, Safety and Quality Assurance Division, who reports to the RL Assistant Manager for Safety, Safeguards, and Quality Assurance.

RL's Director, Safety and Quality Assurance Division, is free of prime responsibility for schedule or cost, has the authority to stop unsatisfactory work pending resolution of quality matters, and is authorized to (1) identify quality problems; (2) initiate, recommend, or provide solutions through designated channels; (3) verify implementation of solutions; and (4) control further processing, delivery, or installation of nonconforming material. The Director, Safety and Quality Assurance Division, is responsible for establishing and implementating a system of audits and surveillance of activities affecting the fuel and blanket assemblies.

The staff has evaluated the independence, authority, and responsibility of RL's QA organization and concludes that the organization complies with 10 CFR 50, Appendix B, and is acceptable.

17.2.3 Nuclear Steam Supply System Supplier

The Westinghouse Electric Corporation's Advanced Reactor Division (ARD), as the supplier of the nuclear steam supply system (NSSS), is responsible for the QA program applied to the design, procurement, and manufacture of safety-related structures, systems, and components that make up the NSSS. ARD is also responsible for the design and design QA of the fuel and blanket assemblies. Responsibility for the design, procurement, and manufacture of certain portions of the NSSS is assigned to two reactor manufacturers as described in Section 17.2.6.

Figure 17.6 shows the ARD QA organization and principal QA functions for the CRBRP. ARD operates under a General Manager who has overall responsibility for ARD QA programs and the establishment of organizational structures for the preparation, implementation, and monitoring of the ARD QA program. He has delegated the authority and the execution responsibility for the ARD QA program for CRBRP to the ARD CRBRP Project Manager.

The ARD CRBRP Project Manager is responsible for the NSSS technical integration and program management, for all technical and program planning, contract and project administration, customer liaison, and direction of all NSSS development, design, procurement, component fabrication, testing efforts, and the CRBRP QA program. The execution responsibility within ARD is divided between the Oak Ridge and Waltz Mill sites according to system and component responsibility under the direction of the CRBRP Project Manager.

The ARD CRBRP QA Manager reports to the CRBRP Project Manager and has been delegated the authority and execution responsibility by the CRBRP Project Manager for establishing, maintaining, directing, and managing the QA program for CRBRP NSSS and for ensuring that the QA program is effectively executed within ARD between ARD and the reactor manufacturers, and between ARD and ARD suppliers. The CRBRP QA Manager's responsibility includes monitoring ARD, the reactor manufacturers, and ARD suppliers to verify that work is performed in accordance with CRBRP QA program requirements and that products meet applicable specifications, codes, and standards. In fulfilling this responsibility he maintains interface with engineering, procurement, and safety and licensing on quality-related matters such as interface coordination, nonconformance control, and corrective action. His responsibility also includes review and approval of engineering and procurement documents to ensure that quality requirements are properly applied. He is also responsible for ensuring that ARD-Oak Ridge suppliers are performing work in accordance with approved supplier QA programs during component surveillance and audits of supplier operations that relate to design, procurement, fabrication, testing, and shipping.

The Product Assurance Manager of ARD reports to the ARD General Manager and is responsible for the QA functions within ARD-Waltz Mill an the implementation of the CRBRP QA program delegated to Waltz Mill by the CRBRP Project Manager. He is responsible for establishing, maintaining, directing, and managing all QA programs at the Waltz Mill, Pennsylvania, site to meet applicable contractual requirements, and for ensuring that QA programs are effectively implemented throughout ARD. He is also responsible, by review and approval actions and by surveillance and audits, for ensuring that ARD-Waltz Mill suppliers have established and are effectively implementing QA programs to meet contractual requirements. His responsibility also includes assurance that products meet applicable specifications, codes, and standards.

Both ARD's Product Assurance Manager and CRBRP QA Manager are free of prime responsibility for schedule or cost, have the authority to stop unsatisfactory work pending resolution of quality matters, and are authorized to (1) identify quality problems; (2) initiate, recommend, or provide solutions through designated channels; (3) verify implementation of solutions; and (4) control further processing, delivery, or installation of noncomforming material.

The staff has evaluated the independence, authority, and responsibility of ARD's QA organizations and concludes that these organizations comply with 10 CFR 50, Appendix B, and are acceptable.

17.2.4 Architect-Engineer

The Burns and Roe, Inc. Breeder Reactor Division (BRD), as the architect-engineer for the CRBRP, is responsible for the QA program applied to the design of the safety-related structures, systems, and components that make up the balance of plant (BOP). In addition, BRD is also responsible for the surveillance of BOP suppliers.

Figure 17.7 shows the BRD QA organization. The Senior Corporate Vice-President of BRD is responsible for the architect-engineer's QA program. The CRBRP QA Manager, who reports to the Senior Corporate Vice-President of BRD, is responsible for ensuring that the QA program is established and implemented. BRD's QA Manager is free of prime responsibility for schedule or cost, has the authority to stop unsatisfactory work pending resolution of quality matters, and is authorized to (1) identify quality problems; (2) initiate recommend, or provide solutions through designated channels; (3) verify implementation of solutions; and (4) control further processing, delivery, or installation of nonconforming material.

The staff has evaluated the independence, authority, and responsibility of BRD's QA organization and concludes that the organization complies with 10 CFR 50, Appendix B, and is acceptable.

17.2.5 Constructor

The Stone & Webster Engineering Corporation (S&W), as the constructor of the CRBRP, is responsible for the QA program applied to the construction, installation, and construction testing of safety-related structures, systems, and components at the reactor site.

The S&W organization responsible for construction activities is shown in Figure 17.8. The S&W President has established QA policies for the corporation. He has delegated the authority for the development of the QA program to the Vice President of QA who is responsible for the overall S&W QA program. The Vice President of QA is independent of, and has organizational authority equal to, the other S&W Vice Presidents.

The QA Department Manager, who reports to the Vice President of QA, is responsible for administering and managing the QA program for construction activities. For the CRBRP, these responsibilities have been delegated by the QA Department Manager to the CRBRP Project Manager.

Major QA activities that are carried out by S&W's QA organization are

(1) the review and approval of construction and inspection documents

(2) inspections and audits within S&W and at the construction site

S&W QA personnel are free of prime responsibility for schedule and cost, have the authority to stop unsatisfactory work pending resolution of quality matters, and are authorized to (1) identify quality problems; (2) initiate, recommend, or provide solutions through designated channels; (3) verify implementation of solutions; and (4) control further processing, delivery, or installation of nonconforming material.

The staff has evaluated the independence, authority, and responsibility of S&W's QA organization and concludes that the organization complies with 10 CFR 50, Appendix B, and is acceptable.

17.2.6 Reactor Manufacturers

Two organizations have been designated as reactor manufacturers for the CRBRP. These organizations and their responsibilities are

(1) General Electric Company, Advanced Reactor Systems Department--responsible for the design, procurement, and manufacture of the intermediate heat transport system, steam generator system, and related components and controls (Figure 17.9)

(2) Rockwell International Corporation, Atomics International Division--responsible for the design, procurement, and manufacture of fuel-handling systems, auxiliary sodium systems, reactor plant maintenance, and related components and controls (Figure 17.10)

Each reactor manufacturer is responsible for the QA program applied to the design, procurement, and manufacture of the safety-related structure, systems, and components it supplies. Each reactor manufacturer has established a QA organization for the CRBRP. Personnel in these organizations are free of prime responsibility for schedule or cost, have the authority to stop unsatisfactory work pending resolution of quality matters, and are authorized to (1) identify quality problems; (2) initiate, recommend, or provide solutions through designated channels; (3) verify implementation of solutions, and (4) control further processing, delivery, or installation of nonconforming material.

The staff has evaluated the independence, authority, and responsibility of each reactor manufacturer's QA organization and concludes that each of these organizations complies with 10 CFR 50, Appendix B, and is acceptable.

17.3 Program

Each of the organizations noted above has provided in Chapter 17 of the PSAR a description of the QA program for the conduct of assigned responsibilities. Each QA program includes a cross reference that relates its QA program documents to the applicable QA criteria of 10 CFR 50, Appendix B. These documents coordinate the QA activities of the various organizations that are responsible

for design, procurement, construction, testing, and services. On the basis of its review of these cross-reference tabulations, the staff concludes that the appropriate criteria of Appendix B to 10 CFR 50 have been addressed within each organization's documented QA procedures. Further, the management of each organization has made compliance with the QA procedures mandatory.

Each organization's QA program will be applied to all safety-related structures, systems, and components within its scope of supply; a list of the specific plant items subject to the QA program has been incorporated into the PSAR. However, NRC review of this list is incomplete, and this area is an open item in the staff's review of the QA programs. Each QA program includes provisions, as applicable, for design control, document control, documents, purchased item control, special process control, inspections, test control, measuring and test equipment control, nonconformance control, corrective action, control of QA records, and audits.

Except for the constructor's QA program, the QA programs described in Section 17.1 of the CRBRP PSAR include a commitment to comply with the Department of Energy's standards RDT F2-2, "Quality Assurance Program Requirements"; RDT F1-2, "Preparation of System Design Descriptions"; RDT F1-3, "Preparation of Unusual Occurrence Reports"; and RDT F3-2, "Calibration Program Requirements." These standards provide requirements that are generally at least as stringent as QA-related regulatory guides addressing the same subject matter for light-water reactors. Where the requirements are not as stringent (in the area of inspector certification, for example), the QA programs commit to meeting the more stringent requirements. This is acceptable to the staff. The constructor, by referencing its NRC-accepted topical report on QA (SWSQAP 1-74A), has committed to meet the requirements of the QA-related regulatory guides for light-water reactors. This is also acceptable to the staff.

The QA program for each organization that has safety-related design responsibility includes provisions for the control of design information. QA controls for design include document review and approval, design review, and design change review and approval to ensure conformance with design criteria.

Each organization described previously will ensure that its contractors and suppliers have adequate QA programs, that inspections of safety-related items and activities will be performed to documented inspection instructions by qualified personnel, and that results will be recorded. They will ensure by surveillance and audits that personnel performing such inspections are free from undue pressures of cost and schedule pertaining to the project. Quality engineers review purchase requisitions, purchase orders, and subsequent change notices to ensure conformance to requirements. Audits and feedback of discrepancy data are used to measure supplier performance.

Each organization has established program requirements for itself and for its contractors and suppliers that ensure that there will be a documented system of records attesting to the quality of the safety-related plant items. As-built documentation will be verified.

Each organization has developed an indoctrination and training program to ensure that personnel performing quality-related activities are trained and qualified in the principles and techniques of the assigned activities and are instructed

as to the purpose, scope, and implementation of quality-related manuals and procedures. Retraining is provided to ensure these personnel maintain their proficiency.

A comprehensive system of planned and documented internal and external audits, described in the PSAR, will be used by each organization described above and its contractors to verify compliance with all aspects of the QA program and to assess the effectiveness of the program. The audits will encompass the review and evaluation of all quality-related activities subject to the provisions of the QA program and will involve procedures, work areas, hardware, activities, and records. The program requires that the audits be conducted in accordance with preestablished procedures by qualified personnel not having direct responsibilities in the area being audited. Audit results will be documented and reported to appropriate levels of management for corrective action. Responses to audit findings will be verified for implementation and effectiveness by followup audits. The staff finds that each organization has described a satisfactory QA audit program.

In its review, the staff has evaluated each organization's QA program for compliance with the Commission's regulations and applicable regulatory guides and industry standards. On the basis of this review, the staff concludes that each organization's QA program contains the necessary QA provisions, requirements, and controls for compliance with Appendix B to 10 CFR 50 and applicable guides and standards, and is acceptable for the CRBRP.

TMI Action Plan Item

Item I.F of the TMI Action Plan (NUREG-0660) addresses quality assurance and has two subtasks with licensee actions as discussed below.

(1) Expand QA list.

(a) Description: Develop improved QA list.
(b) Schedule: Licensee action is yet to be determined.
(c) Resources:....

The NRC has not published guidance for licensees to expand their QA lists to cover equipment important to safety and rank the equipment in order of its importance to safety as called for in NUREG-0660. Until this occurs, licensees are not required by the NRC to expand their QA list beyond items which are safety related. However, the CRBRP PSAR includes the commitment that the applicants' 10 CFR 50, Appendix B, QA program will not be limited in application to safety-related structures, systems, and components alone but that it "is or will be appropriately applied by the Owner to the CRBRP in its entirety including all structures, systems, and components where satisfactory performance is required for the plant to operate reliably, safely and with minimum environmental effects."

This satisfies the intent of Item 1.F.1 and CRBR PDC 1.

(2) Develop more detailed QA criteria. This is a Decision Group D item, and details of implementation by the licensees are not included in the Action Plan.

The NRC has not completed publication of more detailed criteria for various aspects of quality assurance for design, construction, and operations as called for in NUREG-0660. However, a number of the items are reflected in SRP Section 17.1 (NUREG-0800). The staff reviewed the CRBRP QA program description and concluded that it meets the acceptance criteria given in SRP Section 17.1. This satisfies Item 1.F.2.

17.4 Conclusion

On the basis of its detailed review and evaluation of the QA programs of the owner (DOE's CRBRP Project Office), the fuel supplier (DOE's Richland Operations Office), the NSSS supplier (Westinghouse Electric Corporation, Advanced Reactors Division), the architect-engineer (Burns and Roe, Inc., Breeder Reactor Division), the constructor (Stone & Webster Engineering Corporation), and the two reactor manufacturers (General Electric Company, Fast Breeder Reactor Department and Rockwell International Corporation, Energy Systems Group), the staff concludes that the QA program for design, procurement, and construction of the CRBRP meets the requirements of 10 CFR 50, Appendix B, and is acceptable, except that certain additional information and clarifications are necessary regarding the items that are under the control of the QA program. These items have been identified in the PSAR and are being reviewed by the staff. At the completion of this review, the staff will require additional information from the applicant. This item is considered open until satisfactory resolution is obtained.

17.5 QA Program Implementation

17.5.1 Office of Inspection and Enforcement

The application for a construction permit for the CRBRP was docketed for a licensing review in April 1975. After the licensing evaluation was started, the Office of Inspection and Enforcement began inspections of the implemented QA program for those activities relating to the design, procurement, and fabrication of the facility's major components.

Inspections started in April 1975 were conducted at the site Project Office, the architect-engineer's (Burns and Roe, Inc.) corporate offices, and various manufacturing facilities where CRBRP equipment was being fabricated. Areas examined included (1) program organization, (2) QA program for design and procurement, (3) implementation of QA program for design and procurement, (4) audit reports, (5) manufacture process control, and (6) manufacture of selected equipment. Major components inspected during manufacture included, but were not limited to, the reactor vessel, vessel closure head, core support structures, sodium pumps, and the core restraint systems. These inspection activities were reviewed against the design guidelines that comply with the DOE standard RDT F2-2, "Quality Assurance Program Requirements," and the 10 CFR 50, Appendix B, criteria.

The inspection findings revealed some deviations and unresolved items with respect to the CRBRP PSAR commitments and standard RDT F2-2. However, upon termination of licensing and inspection efforts in 1977, the CRBRP Project Office had completed all corrective actions and resolved all open items identified by the inspectors. The CRBRP Project Office actions were documented in an amendment to the PSAR that was issued in October 1977.

During the period 1977 through 1983, design changes took place (e.g., core design and decay heat removal system) and the Project Office continued a QA audit and surveillance of reactor lead manufacturers and their contractors in the fabrication of components. The last inspection was performed at the Project Office in December 1978. Since the summer of 1977, the Office of Inspection and Enforcement has not performed any additional equipment inspections.

17.5.2 Region II

Since licensing review of CRBRP has begun again, Region II has reinitiated the inspection program. A site inspection was conducted in October 1982 to observe site-clearing activities and to examine stored equipment. An overall inspection program is being developed that will include extensive examination of stored equipment and fabrication records. Review of updated QA programs will begin in the spring of 1983.

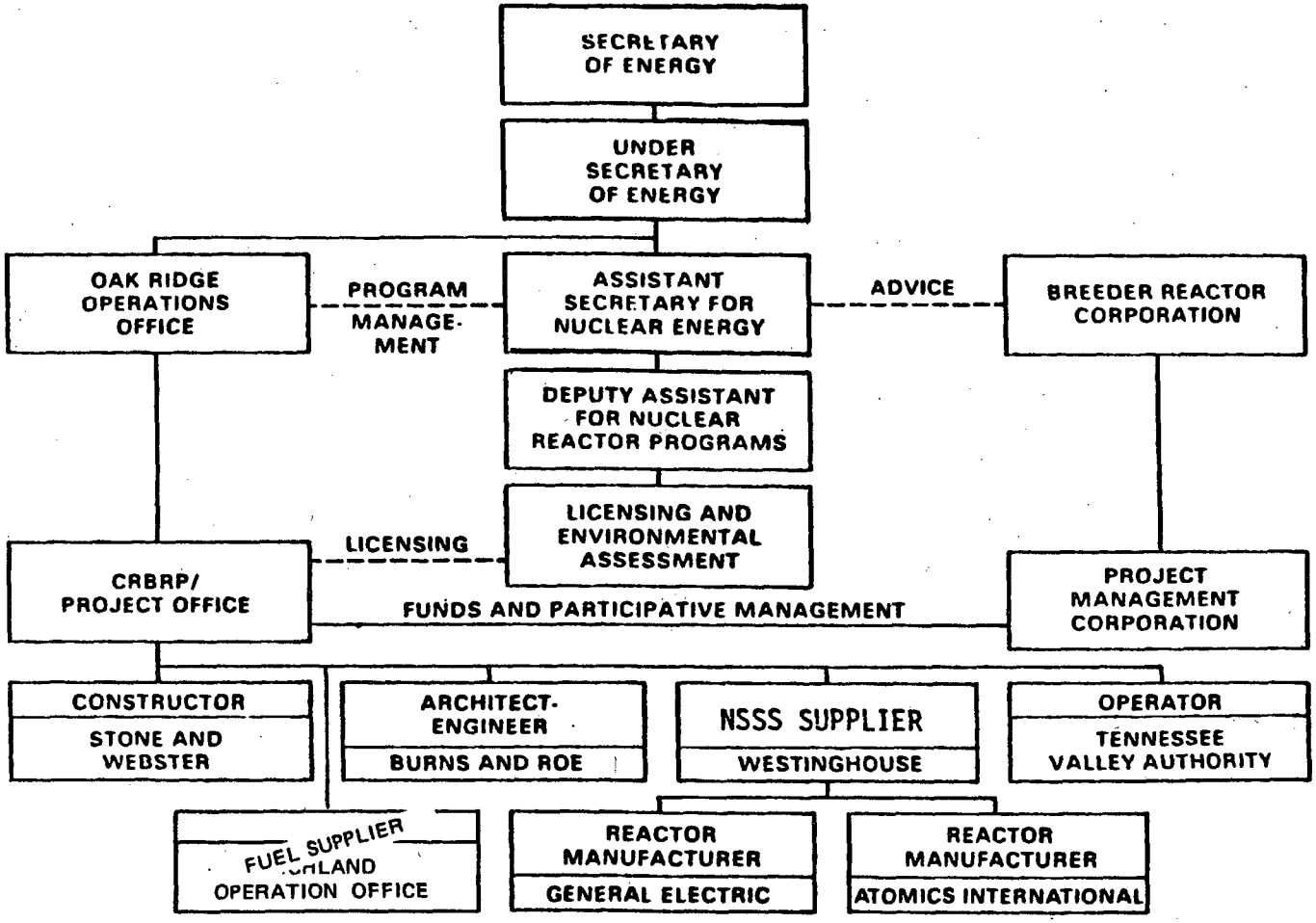


Figure 17.1 Organization of the CRBRP project

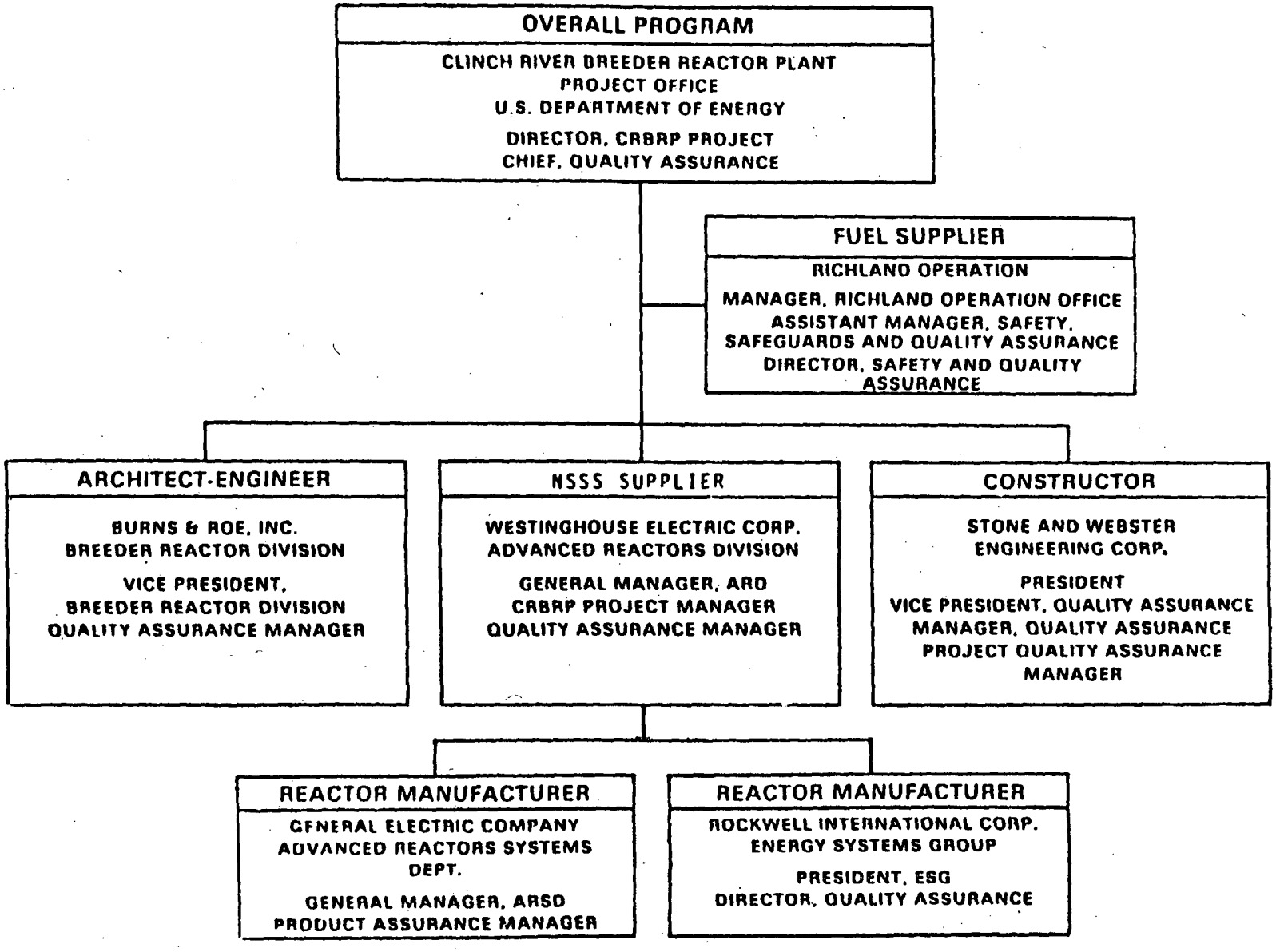


Figure 17.2 Organization of the CRBRP overall quality assurance program

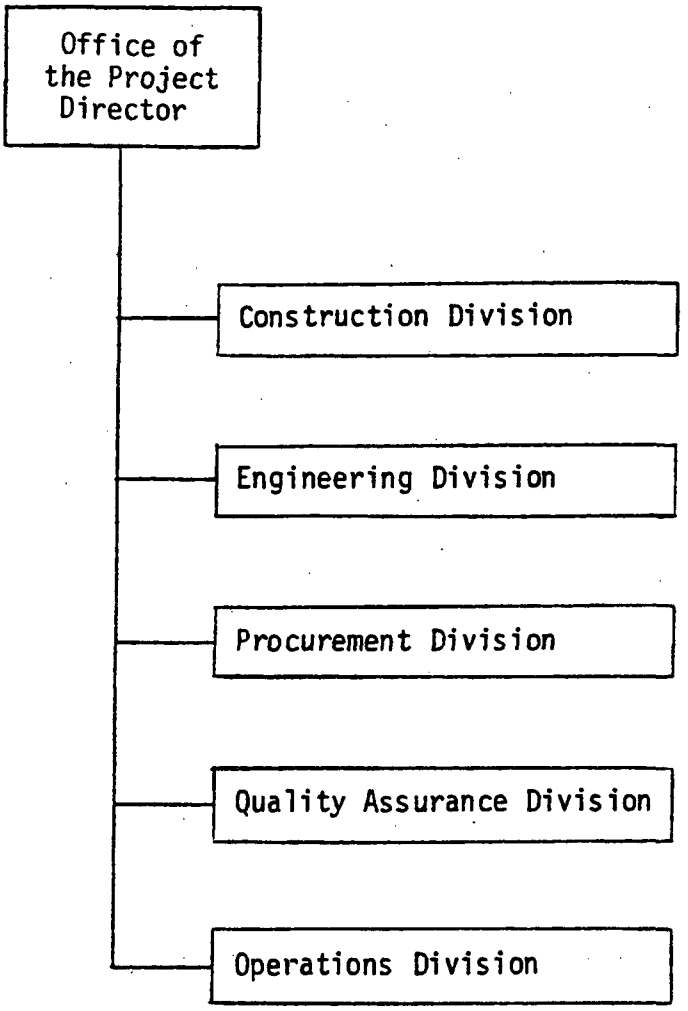


Figure 17.3 CRBRP Project Office organization

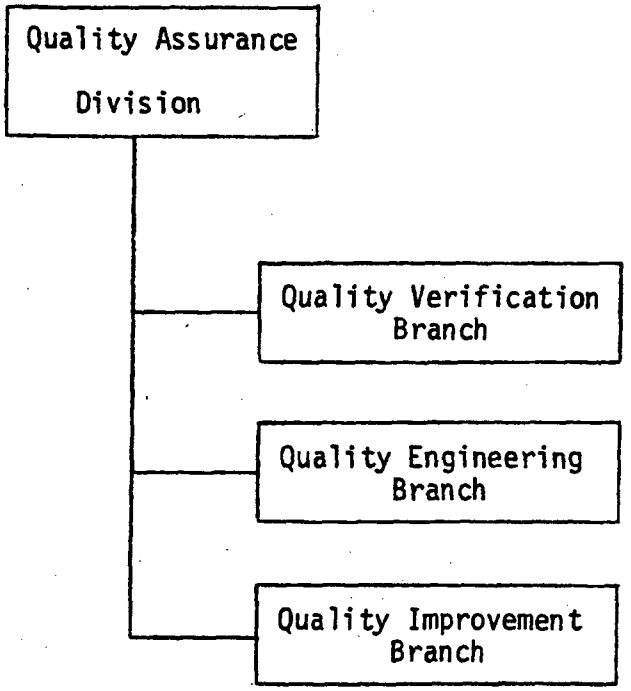


Figure 17.4 CRBRP Quality Assurance Division organization

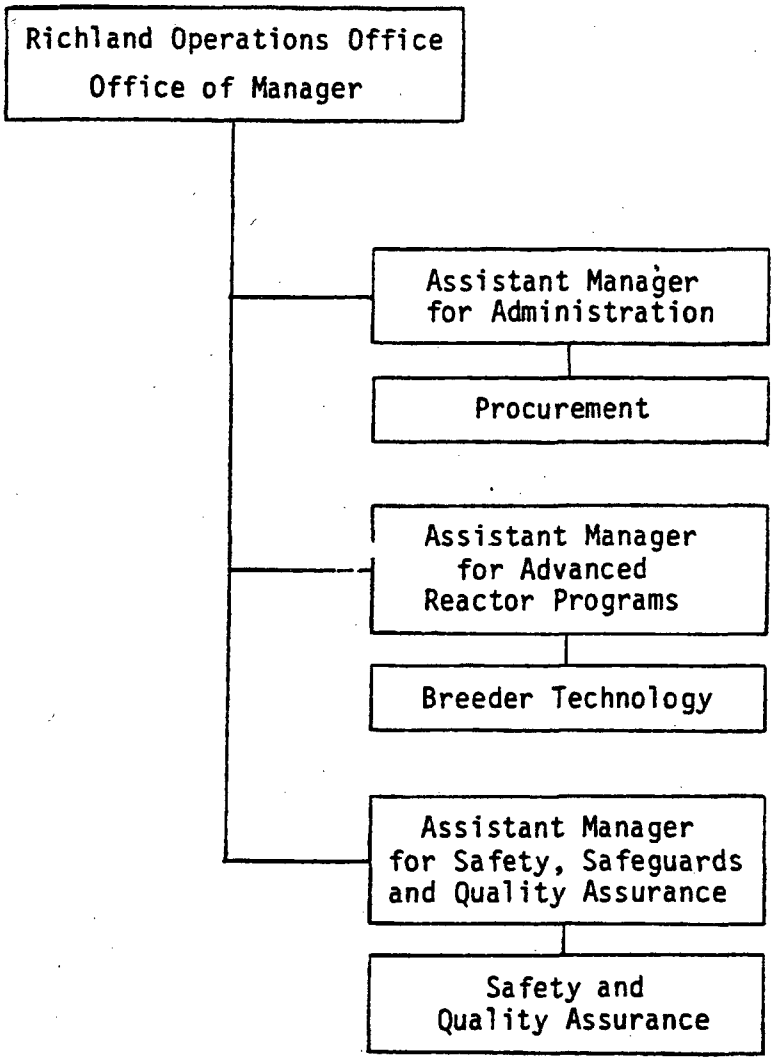


Figure 17.5 Fuel supplier Quality Assurance organization

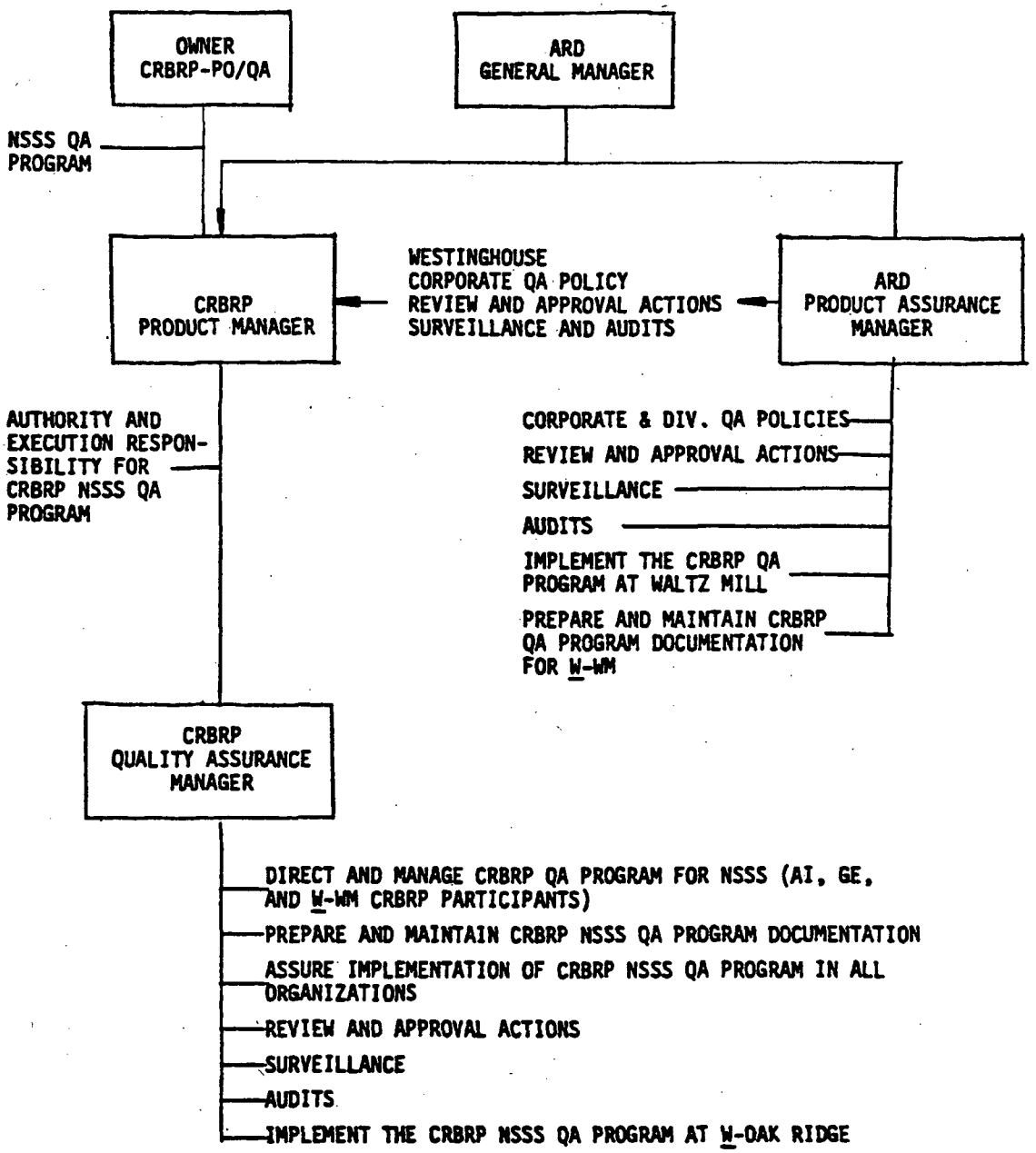


Figure 17.6 ARD organization for CRBRP with principal functions

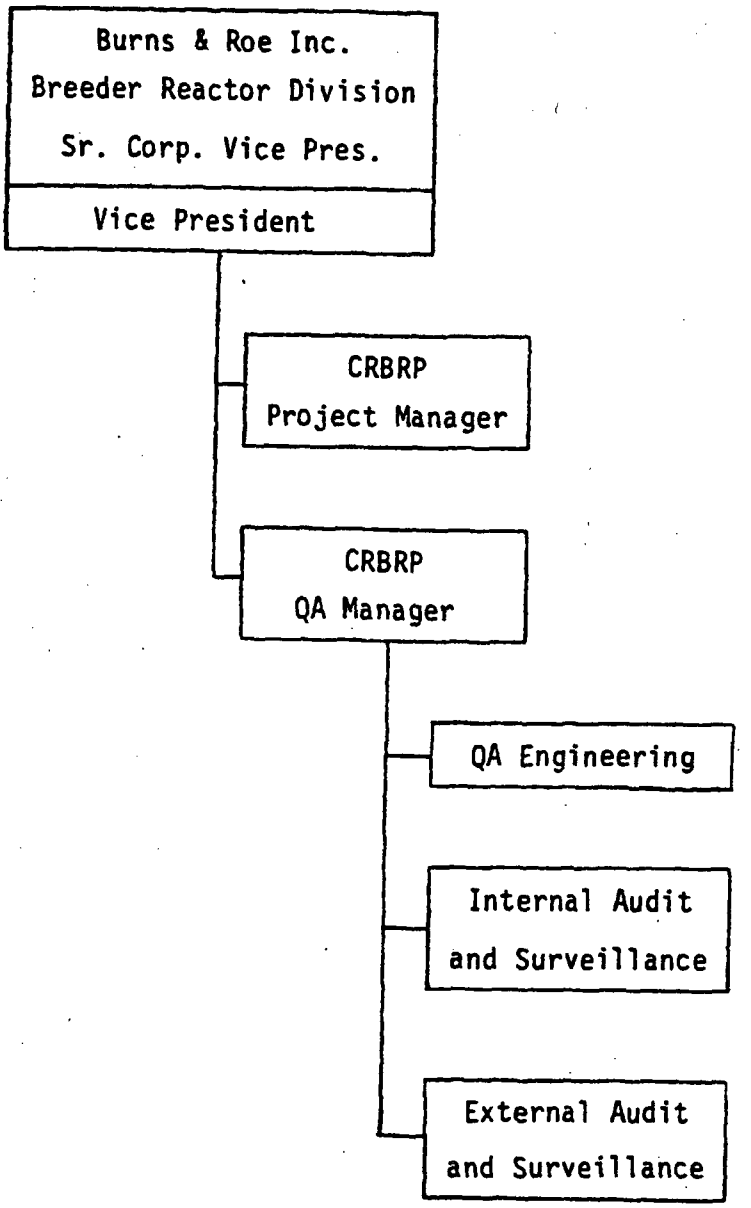


Figure 17.7 Architect-engineer Quality Assurance organization

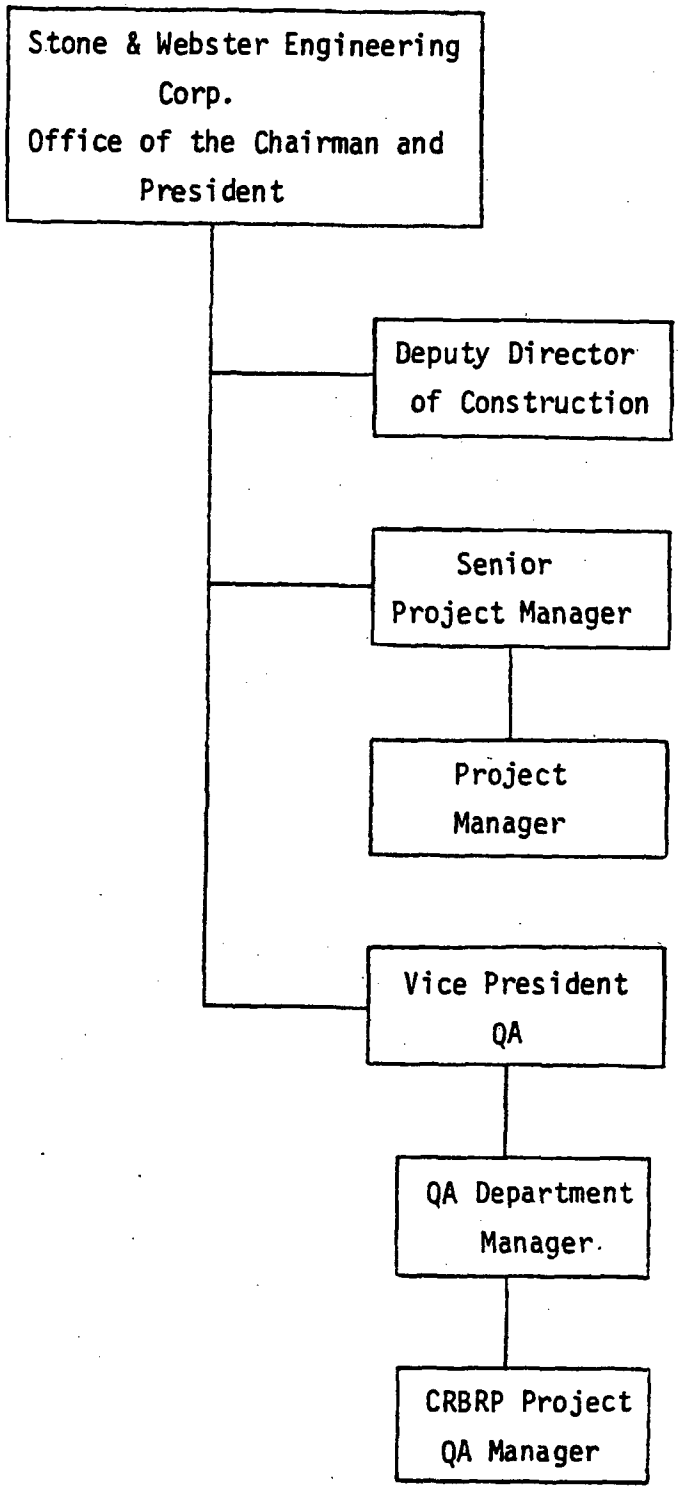
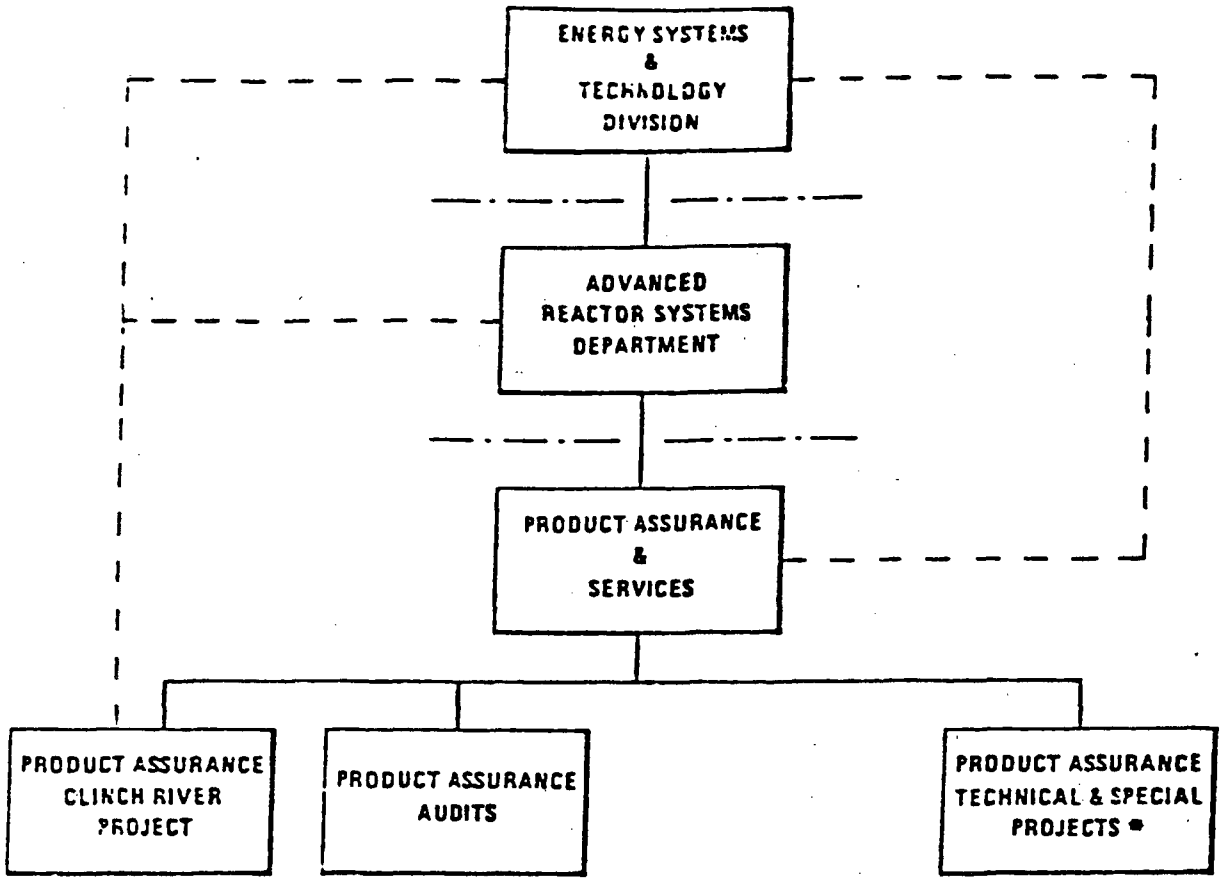


Figure 17.8 Constructor Quality Assurance organization for CRBRP



* DOES NOT MAKE DIRECT CONTRIBUTION TO CLINCH RIVER ACTIVITIES (SUPPORTS BASE PROGRAMS AND OTHER WORK)

Solid lines represent administrative hierarchy.
Dashed lines represent organizational access for functional reporting.

Figure 17.9 Reactor manufacturer Quality Assurance organization for CRBRP (General Electric Company, Advanced Reactor Systems Department)

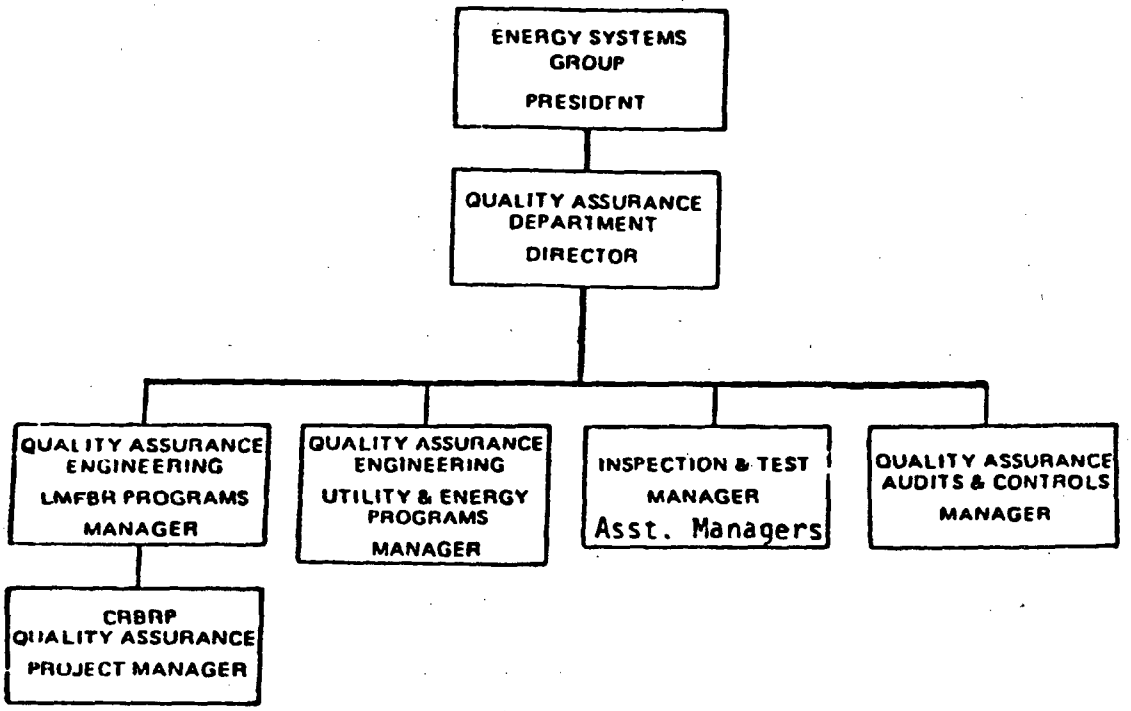


Figure 17.10 Reactor manufacturer Quality Assurance organization for CRBRP (Rockwell International Corporation, Energy Systems Group)

18 HUMAN FACTORS REVIEW

Regarding additional TMI-related requirements (the CP rule), 10 CFR 50.34(f) refers to specific applicants for light-water-reactor construction permits. Although the CRBR is not specifically referred to and is not of the reactor type referred to, the NRC takes the position that the CRBR must comply with 10 CFR 50.34(f). 10 CFR 50.34(f) states in part that the applicant shall (1) provide, for Commission review, a control room design that reflects state-of-the-art human factor principles before committing to fabrication or revision of fabricated control room panels and layouts (Item I.D.1); (2) provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded (Item I.D.2); and (3) provide for automatic indication of the bypassed and inoperable status of safety systems (Item I.D.3). Additional guidance on the implementation of these requirements is provided in NUREG-0718 and NUREG-0737.

18.1 Background and Discussion

The applicants have indicated willingness to comply with the CP rule and submitted supporting information in PSAR Amendment 69, pages H-11 to H-16, dated July 1982 and by letter (J. R. Longenecker to P. S. Check) submitted in December 1982. Two design information meetings were held in Bethesda, Maryland, one in December 1981, and one in October 1982. A design information meeting was also held in Oak Ridge, Tennessee, in December 1982. At these meetings the applicants described the methods used to apply human factors principles to the redesign of the CRBR control room. At the October meeting, the CRBR presentation included information about the following topics:

(1) The approach to alarms. In the human factors review and control room redesign, the number of alarms was considerably reduced (to about one-third the number in a typical LWR control room) by eliminating unnecessary alarms and providing multiple input to single alarms where different inputs are symptomatic of the same condition. The applicants stated that they had considered alarm suppression and other concepts of alarm management. They decided against alarm suppression because they were reluctant to press the state of the technology in too many areas. Their concern is that design errors could result in suppression of important alarms. Although the staff shares the concern about erroneous suppression of important alarms, it is also a matter of concern that the information overload resulting from multiple alarms could represent a risk. The staff agrees with the applicants that it may be too early to "freeze" design in this somewhat experimental area. However, at the staff's suggestion, the applicants committed to continue to study alarm suppression. If the study reveals that alarm suppression would offer a significant advantage, it is likely that a system could be incorporated into the computer/cathode ray tube (CRT) complex without significant disruption.

(2) The dark board concept. The applicants stated that they are utilizing the dark board concept for annunciators and safety panels. No safety panel lights will be illuminated when the plant is in the normal condition.

(3) Operator response under accident conditions. In the CRBR design all prompt safety functions are automatic and redundant. The operators' function during an accident will be to verify that the safety functions are being performed. This is in contrast to LWR practice in which prompt safety actions are automatic and followup safety actions are manual.

In the December 1982 meeting at Oak Ridge, staff members observed a demonstration of a typical walkthrough of a design-basis event on the control room mockup. The applicants and the staff discussed the importance of demonstrating equivalence to NUREG-0700 (Appendix B), the design methodology, information overload resulting from multiple alarms, the computer-based CRT display system, and the safety parameter display system (SPDS). The staff agreed that the SPDS function could be incorporated in a computer/CRT display if the design meets the functional requirements resulting from an acceptable CRBRP analysis. At the conclusion of the meeting, the applicants made the following commitments:

(1) The applicants will perform a study of techniques to reduce alarm overload and assess their applicability to the CRBR alarm system. The report will be submitted in conjunction with the FSAR.

(2) The main control panel layout, a completed human factors check list for the main control panels (addressing applicable portions of Section 6 of NUREG-0700 as guidelines), and a summary report of the control room design process showing equivalence to NUREG-0700 (Appendix B) will be submitted 3 months before initiation of main control room hardware fabrication.

(3) The functional basis of computer graphics and examples of typical top level graphics will be made available for audit before fabrication of the main control room panel.

(4) The PSAR was modified by Amendment 75, docketed February 8, 1983 to clarify how equivalence to NUREG-0700, Appendix B, will be demonstrated.

18.2 Safety Evaluation

The applicants, in docketed information, state that the CRBR control room design was developed in accordance with state-of-the-art human factors principles, and that documentation will be submitted to demonstrate that design methods used are equivalent to those described in NUREG-0700, "Guidelines for Control Room Design Review," including Appendix B which describes methods of systems analysis, functional analysis, task analysis, and sequence analysis. The scope of the systems analysis includes operation of each system, use of the main control board mockup, and the consideration of the human-machine interface. The applicants also state that the control room will incorporate an advanced design information system including computer-based CRT displays. The applicants have committed to supply design information for NRC review before committing to fabrication of main control panels and layouts. The information supplied by the applicants and the stated commitments provide reasonable assurance that Item I.D.1 will be accomplished in an acceptable manner.

The applicants, in docketed information, state that the CRBR design will include a safety parameter display system that will display to operating personnel a minimum set of parameters or derived variables which are representative of the safety status of the plant. The system will have the capability of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits. The SPDS will be designed consistent with the guidance of NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated March 1981. The information provided by the applicants and the applicants' commitments provide reasonable assurance that Item I.D.2 will be accomplished in an acceptable manner.

The applicants, in docketed information, state that the CRBR design includes automatic indication of the bypassed and inoperable status of safety systems. In addition, to the extent practical, inputs to the inoperable status monitoring system will be direct measurements of the desired variable. The information provided by the applicants and the applicants' commitments provide reasonable assurance that Item I.D.3 will be accomplished in an acceptable manner.

18.3 Conclusion

The staff has reviewed the documents submitted by the applicants constituting a commitment to demonstrate a state-of-the-art control room, and to furnish a safety parameter display system and an automatic safety status monitoring system. Design methodology described by the applicants utilizes state-of-the-art analytical techniques, and design concepts indicate the use of computer-based CRT display systems. These facts, combined with the stated intention to apply accepted human factor principles and the commitment to submit the design for NRC review before committing to fabrication, provide bases for approval.

Based on its review, the staff has concluded that the applicants have provided sufficient information to demonstrate that Items I.D.1, I.D.2, and I.D.3 will be acceptably accomplished.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Clinch River Breeder Reactor Plant is being reviewed by the Advisory Committee on Reactor Safeguards. The staff intends to issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission, relative to its review, is available. The supplement will append a copy of the Committee's report, will address comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

20 COMMON DEFENSE

The review of the PSAR covered the applicants' plans for implementing security measures relating to protection against sabotage of the facility, theft of special nuclear material, and storage of classified information. The scope of the review covered both transportation and fixed-site protection in accordance with the provisions of 10 CFR 11; 10 CFR 50.34(c), 50.34(d), and 50.34(e); 10 CFR 73.20, 73.21, 73.25, 73.45, 73.46, 73.55, 73.71, 73.72, and 73.80 and Appendices B and C to 10 CFR 73; 10 CFR 95; and regulatory guides and equivalent guidance documents.

20.1 Acceptance Criteria and Basis

The applicants' PSAR must provide sufficient information to indicate that their preliminary planning for security provides reasonable assurance that conformance to the applicable regulations will be achieved. Commitments to each major portion of the regulation and associated guidance documents are also required in the PSAR. The preliminary planning should include a commitment to a design-phase review for physical security and should show how this responsibility is to be implemented by the applicants.

The bases for acceptance are the commitments made by the applicants to conform to the regulations and guidance documents identified in the first paragraph on this page.

20.2 Review Evaulation

The information submitted by the applicants was reviewed against the acceptance criteria. As a result of its initial evaluation, certain portions of the applicants' PSAR were identified as requiring changes in order to meet a level of protection consistent with that identified in the acceptance criteria. The applicants' subsequent revision to the PSAR resulted in a document that sets forth design features of the plant and outlines in some detail those main topics to be addressed in the security plan.

The PSAR addresses major portions of the applicable regulations to a greater degree than is necessary at this early stage and provides significant assurance that conformance to the applicable regulations will be achieved. The PSAR contains the necessary provisions to protect the facility against radiological sabotage and to prevent the theft of special nuclear material. In addition, it contains particulars for the control and accounting of special nuclear material and provisions for a design-phase review.

20.3 Evaluation Summary

The applicants have provided a general description of the plans for protection of the plant against potential acts of radiological sabotage and for prevention of the theft of special nuclear material. The description addressed security measures relating to the personnel employment program and screening procedures,

general plant design layout and equipment arrangements, transportation and accountability of material, and protection of classified information. The staff concludes that based upon the program outline contained in the PSAR there is reasonable assurance that the final physical security plan will meet the requirements of 10 CFR 11, 25, 50, 70, 73, and 95 and conform to the associated regulatory guides and equivalent guidance documents. At this stage of the licensing process, the depth of preliminary planning for protection of the plant is above that required for assurance of an adequate security program.

21 FINANCIAL QUALIFICATIONS

On March 11, 1982 the Commission approved SECY-82-21, a final rule eliminating entirely the financial qualifications review and findings for "electric utility" applicants, and providing that the financial qualifications of an electric utility applicant are not among the issues to be considered by atomic safety and licensing boards in construction permit or operating licensing proceedings. This aspect of the rule became effective immediately upon publication in the Federal Register and applies to pending licensing proceedings and the issues or contentions raised therein.

Pursuant to final regulations 10 CFR 50.2(x) and 50.33(f), 47 FR 13750 (March 31, 1982), electric utility applicants will no longer be required to submit information on their financial qualifications and the staff "shall" not conduct any financial qualifications reviews of such applicants. "Electric utility" includes investor-owned utilities, public utility districts, municipalities, rural electric cooperatives, and State or Federal agencies, and associations of these entities.

22 FINANCIAL PROTECTION

Under the Price-Anderson Act, a system of private insurance and government indemnity currently totaling $560 million exists to pay public liability claims for personal injury and property damage resulting from a nuclear incident. Under this act, owners of commercial nuclear power plants having a rated capacity of 100 MWe or more must provide proof to the NRC that they have private nuclear liability insurance, or another form of financial protection approved by the NRC, in an amount equal to the maximum amount of insurance available from private sources. That insurance, currently $545 million, consists of primary private nuclear liability insurance of $160 million from two nuclear liability insurance pools, American Nuclear Insurers (ANI) and Mutual Atomic Energy Liability Underwriters (MAELU), and a secondary retrospective premium insurance layer of up to $5 million per reactor per incident but not in excess of $10 million for a single reactor in any year. With 77 commercial reactors operating under the system, the secondary layer totals $385 million. As a licensed facility, CRBRP would be assessed this premium in the event of a nuclear incident resulting in damages exceeding the amount of the current $160 million primary insurance layer. In addition, a licensee is required to execute an indemnity agreement with the NRC that provides up to, but not more than, $500 million in Government idemnity to satisfy public liability claims in excess of the amount of insurance or other financial protection required of the licensee. As a licensed facility, the operators of CRBRP will be expected to meet the above requirements of the Price-Anderson Act, including the execution of an indemnity agreement with the Commission. The provisions of the Price-Anderson Act would, therefore, apply to any nuclear incident occurring at CRBRP.

Public Law 94-197 also provides that the present $560-million limit on liability for a single nuclear incident be retained until the combined primary and retrospective insurance layers reach the $560-million level. After that point the limit on liability would rise, corresponding to increases in the primary and retrospective insurance layers. The present government indemnity level is $15 million, the difference between the financial protection layer of $545 million and the liability limit of $560 million. Although no ultimate dollar limit on liability will be set, there is also no liability on the part of the licensees above the limit of liability, whatever it may be.

Concerning the liability of the other participants in the CRBRP project, the basic financial protection and Government indemnity would extend not only to the licensee, but to "any other person who may be liable," through an "omnibus coverage" feature. For example, should offsite damage be caused by failure of a component, the public currently would have up to $560 million for the payment of claims even though the vendor of the component might otherwise be without substantial coverage.

23 CONCLUSIONS

NRC technical review and evaluation of the information submitted by the applicants considered the principal matters summarized below:

(1) The population density and land-use characteristics of the site environs and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology) have been identified to establish that these characteristics have been determined adequately and have been given appropriate consideration in the plant design, and that the site characteristics are in accordance with NRC's siting criteria of 10 CFR 100, taking into consideration the design of the facility, including engineered safety features provided.

(2) The design, fabrication, construction, and testing criteria, and the expected performance characteristics of the plant structures, systems, and components important to safety have been identified. The NRC review also determines that the principal design criteria (PDC), quality assurance criteria (QAC), regulatory guides (RGs), and other appropriate rules, codes, and standards have been identified and are justified.

(3) The expected response of the Clinch River Breeder Reactor Plant to various anticipated operating transients and to a broad spectrum of postulated accidents has been evaluated. On the basis of this evaluation, NRC determined that the postulated consequences of a few highly unlikely postulated accidents (design-basis accidents (DBAs)) would exceed those of all other accidents considered. The staff performed conservative analyses of these DBAs to determine that the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence, would not exceed the Commission's guidelines for site acceptability given in 10 CFR 100. In addition, the staff reviewed those events that are classified as being beyond the design basis to ensure that (a) their probabilities of occurrence are sufficiently low and (b) the reactor vessel and containment would limit the radiological consequences of such an event, should it occur, to an acceptable level.

(4) The applicants' engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the plans for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, have been reviewed to determine the applicants' technical qualifications to operate the facility safely.

(5) The design criteria of the systems provided for control of the radiological effluents from the CRBRP have been reviewed to determine the capability of these systems to control the release of radioactive wastes from the facility within the limits of the Commission's regulation in 10 CFR 20 and the operation of equipment provided in such a manner as to reduce radioactive

release to levels that are as low as reasonably achievable (ALARA) within the context of the Commission's regulations in 10 CFR 50 and to meet the dose design objectives of Appendix I to 10 CFR 50.

(6) The applicants' quality assurance program for the construction of the facility has been reviewed to ensure that the program complies with the Commission's regulations in 10 CFR 50, and that the applicants will have proper controls over the CRBR construction so that there is reasonable assurance that the facility can be constructed safely and reliably.

NRC FORM 335 (7 77) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG-0968
Volume 1

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant
Main Report-Volume 1 Docket No. 50-537

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
Clinch River Breeder Reactor Program Office

5. DATE REPORT COMPLETED
MONTH: March YEAR: 1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Clinch River Breeder Reactor Program Office
Office of Nuclear Reactor Regulation
Washington, DC 20555

DATE REPORT ISSUED
MONTH: March YEAR: 1983

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Safety Evaluation Report for a CP

PERIOD COVERED *(Inclusive dates)*
April 11, 1975 through March 1983

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

The Safety Evaluation Report for the application by the United States Department of Energy, Tennessee Valley Authority, and the Project Management Corporation, as applicants and owners, for a license to construct the Clinch River Breeder Reactor Plant (Docket No. 50-537) has been prepared by the Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. The facility will be located on the Clinch River approximately 12 miles southwest of downtown Oak Ridge and 25 miles west of Knoxville, Tennessee. Subject to resolution of the items discussed in this report, the staff concludes that the construction permit requested by the applicants should be issued.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Clinch River Breeder Reactor Plant
(CRBRP)
Safety Evaluation Report

17b. IDENTIFIERS/OPEN-ENDED TERMS

AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS *(This page)*

22. PRICE
$

NRC FORM 335 (7-77)