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Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 30, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to Questions CS760.11, 21, 27, 32, 44, 46, 48, 51, 52, 55, 82, 90, 93, 94, 95, 97, 102, 106, 107, 108, 109, 114, 115, 117, 118, 120, 121, 122, 127, 129, 130, 132, 135, 139, and 140; which will also be incorporated into the PSAR Amendment 69 scheduled for submittal in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

cc: Service List
Standard Distribution
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D001

Question CS760.11

Discuss how the Doppler coefficient is affected by changes made in going from the homogeneous to the heterogeneous core arrangement. Discuss the uncertainties of the Doppler coefficient and how it impacts under cooling and reactivity insertion events.

Response

The following table summarizes typical CRBRP regionwise flooded Doppler constants in the heterogeneous and homogeneous core designs for clean-core conditions at the beginning of cycle one.

	Doppler Constant, $-T\ dk/dT \times 10^4$	
	Heterogeneous Core	Homogeneous Core
fuel	25.8	43.0 (inner core) 12.9 (outer core)
inner blanket	44.0	----
radial blanket	11.8	7.0
axial blankets	2.6	4.4
TOTAL (isothermal)	84.2	67.3

The lower fuel Doppler in the heterogeneous core is a result of the smaller number of fuel assemblies (156 vs 198 in the homogeneous core) and their placement in the heterogeneous core, the higher fuel enrichment and lower fertile-to-fissile ratio in the fuel, and the harder average neutron energy spectrum. The higher total core (isothermal) Doppler constant in the heterogeneous core is a result of the higher total (fuel plus blankets) heavy metal loading. The Doppler uncertainty, in the temperature range around hot-full-power, is ±10% (1σ) as discussed in Section 4.3.2.3.1 of the CRBRP PSAR.

PSAR Sections 15.1.4.1 and 15.1.4.2 provide analyses for the worst case overpower (Section 15.2) and undercooling (Section 15.3) events, respectively. These include the results of a sensitivity study for both types of transients considering uncertainties in Doppler feedback (see Tables 15.1.4-1 and 15.1.4-7). The maximum and minimum values of Doppler coefficient used in these sensitivity studies are summarized below for the models described in Section 15.1.4 and Doppler information in Table 4.3-16:

EVENT	MAXIMUM DOPPLER	MINIMUM DOPPLER
Undercooling	-0.0118	-0.0019
Overpower	-0.0035	-0.0019

As indicated in Section 15.1.4 minimum Doppler is the worst case for the overpower event and maximum Doppler is the worst case for the undercooling event. The Doppler values used in the analyses appropriately reflect 1) a 30% uncertainty (3σ) in its calculated value and 2) time in core life effects. This establishes the worst case conditions to be used in the nuclear kinetic/thermal-hydraulic evaluations for the fuel and blanket hot rod calculations.

Question CS760.21

In Section 15.2.2.2.1 it is stated that gradual core radial motion in response to normal temperature changes is discussed in Section 4.2.2.4.1.8. However, this section is not in the PSAR. Likewise, the last three paragraphs on page 15.2-2a appear to be misplaced.

Response

In PSAR Section 15.2.2.2 all references to Section 4.2.2.4.1.8 have been changed to Section 4.2.2.4.3.

The latter part of this question, dealing with page 15.2-2a, is addressed in the response to Question CS760.16.

15.2.2.2 Sudden Core Radial Movement

15.2.2.2.1 Identification of Causes and Accident Description

The event to be considered here involves core radial motion which occurs rapidly and is difficult to accurately predict. This is in contrast to normal core radial motion which occurs gradually and predictably in response to normal temperature changes and irradiation induced material swelling and creep. The latter type event is discussed in Secton 4.2.2.4.3.

The type of sudden core radial motion to be evaluated has been termed "stick-slip" motion. Stick-slip motion refers to a situation in which the reactor assemblies are restrained from moving radially by interassembly frictional forces at the assembly load planes (stick) and then suddenly move to a new position dictated by current temperature and irradiation environment as the interassembly frictional forces are suddenly removed or reduced (slip). If it is postulated that sticking occurs while the reactor assemblies are bowed away from the core centerline, a sudden positive reactivity insertion can take place as the assemblies slip to an inwardly bowed shape (towards the core centerline). Such an event is unlikely since the buildup of interassembly frictional forces which would be required to cause sticking would occur only when the assemblies are in a compact inwardly bowed state. If the assemblies are bowed outward away from the core centerline, the interassembly gaps would be larger and then the probability of sticking would be minimal. On the other hand, if because of thermal and irradiation effects the assemblies due to manufacturing tolerances and frictional forces.

If the assemblies are prevented from achieving a compact state due to interassembly frictional effects, it is possible that a seismic event could overcome the frictional effects and allow the reactor assemblies to take on a more compact state. This is considered to be the only realistic initiating mechanism for a stick-slip type event.

If the stick-slip event occurred, the reactivity insertion would cause temperature rises of the fuel, cladding, and coolant. The power rise would trigger a primary control system scram if the limits of Section 15.1.3 were exceeded.

15.2.2.2.2 Event Evaluation: Model, Assumptions, and Conservatisms

To determine the maximum possible reactivity insertion, the following analysis steps were followed:

1. Predict the difference in core assembly positions and bowing between refueling and full power.

2. Determine the reactivity worth factors associated with radial motion of each core assembly.

3. From the predictions of maximum possible radial motion and worth factors, determine an upper limit for possible reactivity insertion from stick-slip.

To predict the core assembly positions and bowing at refueling and full power conditions, a finite element model was constructed of a radial row of core assemblies. The reactor environmental conditions were then applied along with material characteristics to give bowing and position curves like those of Figures 4.2-88 through 4.2-92. Refer to Section 4.2.2.4.3 for further details of the core assembly bowing analysis. Comparison of the bowing shapes for Figures 4.2-88 through 4.2-92 shows an inward bowing at full power (100% power to flow ratio). The reactor assemblies were assumed to stick in the refueling position (at 0% power to flow ratio) and to then slip suddenly to the full power position.

Conservative nominal compaction reactivity worth coefficients were determined by using the assumptions that all control rods would be parked above the core at the beginning of an equilibrum cycle. The worth coefficients are shown in Table 4.3-14.

The above procedure results in a prediction of approximately 60 for the maximum value of step reactivity insertion (see Section 4.2.2.4.3).

The above upper limit is considered to be conservative for the following reasons:

1. In the analysis, all the gaps in the core were compressed compleely out whereas core compaction tests (1) indicate that not all gaps will be compressed out in a real core. This is due to manufacturing tolerances as well as frictional effects in the core.

2. The analysis assumptions were that sticking of the core assemblies would occur where the assemblies are in their maximum outwardly bowed configuration. More realistically the sticking would not occur until substantial inwardly directed thermal bowing had already occurred and forces had begun to build-up between assemblies. Thus, part of the bowing reactivity change can be expected to occur gradually which will be compensated for by Doppler and thermal expansion effects. This would reduce the maximum possible step reactivity change.

3. The inherent vibrational motion of the core assemblies when flow is passing through would tend to prevent sticking. This would aid in allowing smooth translation of the core assemblies in response to thermal bowing.

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1. W. C. Kinsel, "FTR Core Compaction and Withdrawal Tests," May 1973, HEDL-TME-73-58, UC-79 e, g, h.

Question CS760.27

In CRBRP-ARD-0308, Feb. 1982, a new model of the upper internals structure (UIS) is described verbally but no details are given. Please provide the applicable equations and estimate the effect of the new model on temperatures.

Response

The DEMO upper internals structure (UIS) model is based upon a lumped nodalization scheme which represents the thermal-hydraulic characteristics of the collector and 29 chimneys comprising the UIS, as well as the collector radial gap. The nodalization scheme consists of 10 axial nodes and 3 radial nodes (2 metal, 1 sodium) for the chimneys and a single node for the collector. The UIS thermal-hydraulic nodeling is briefly described below

- Thermal Nodel

  1) Heat transfer through the chimney wall is assumed to occur by simple conduction.

  2) Convective heat transfer between the sodium in the chimney and the chimney wall is calculated using the Lyon correlation for fully turbulent flow and heat transfer in pipes:

     $$Nu = 7.0 + 0.025(Pe)^{0.8}$$

     where Pe = Peclet number for 3% of rated chimney mass flow.

  3) Convective heat transfer between the sodium outside the chimney (within the shear web) and the chimney wall is calculated using the Maresca and Dwyer correlation for turbulent flow through unbaffled rod bundles:

     $$Nu = 6.66 + 3.126(P/D) + 1.184(P/D)^2 + 0.0155(\psi Pe)^{0.86}$$

     where P = pitch

     D = chimney inside diameter

     $\psi = \varepsilon_H / \varepsilon_M$

o Hydraulics Model

The pressure drop from the collector node to the reactor outlet nozzle through the upper path (chimney) and the lower path (radial gap) must be equal. Thus,

$$f_{CH}W_{CH}|W_{CH}| + H_{CH} + H_{PU} = f_{GAP}W_{GAP}|W_{GAP}| + H_{PL}$$

where W_{CH} = chimney flow

W_{GAP} = radial gap flow

H_{CH} = chimney natural head

H_{PU} = natural head for upper path from chimney to outlet nozzle

H_{PL} = natural head for lower path from collector through radial gap to outlet nozzle

And

$$W_{GAP} = W_{CORE} - W_{CH}$$

Solution of the above two equations provides values for chimney and radial gap flows. The UIS natural heads are calculated using upper plenum and chimney average sodium temperatures.

Assumptions used in the development of the UIS model and their corresponding justifications are given below:

1) Sodium in the collector mixing chamber is assumed to be completely mixed and at a uniform temperature. Spatial deviations in collector temperature distribution existing at steady state conditions should be sufficiently mitigated during the initial portion of the natural circulation transient to justify a one-node collector representation.

2) The convection heat transfer coefficient between the sodium in the chimney and the chimney wall was assumed to remain constant with changes in sodium flow rate. The addition of a flow-dependent convection coefficient was found to have minimal effects on core flows and temperatures.

3) The twenty-nine chimneys in the UIS are modeled as one chimney. Each chimney can be approximated as two concentric cylinders which divide the flow path into inner and outer regions. Chimney structure and configuration have been accounted for in both flow area and heat transfer area calculations.

4) Sodium on the inside and on the outside of the shear web region of the UIS is assumed to be well mixed and at a uniform temperature.

5) The loss coefficients for the UIS chimney and radial gap pressure drop correlations used in the analysis reported in CRBRP-ARD-0308 were determined from experimental results to be:

$$f_{CM} = 2.047 \times 10^{-8} \frac{\text{psid}}{(\text{lbm/sec})^2}$$

$$f_{GAP} = 6.567 \times 10^{-7} \frac{\text{psid}}{(\text{lbm/sec})^2} \qquad \text{(for a 1 inch gap)}$$

Additional analyses using other gap heights revealed that the total reactor flow was not sensitive to the gap height.

Previous DEMO (Rev. 4) modeling of the upper internals structure lumped the above core structures (upper fuel assemblies and UIS) into one metal-sodium pair of nodes. The revised model described above provides a more detailed representation of the thermal-hydraulic characteristics of the UIS. Comparison of natural circulation transient results using the two models shows that a lower minimum reactor mass flow occurs with the revised UIS model. A preliminary evaluation of the effect of the UIS model was made by adding the model to the simulation used in the analysis reported in CRBRP-ARD-0132 ("A Preliminary Evaluation of the CRBRP Natural Circulation Capability", November, 1977). The peak temperatures were seen to increase by approximately 70°F.

Question CS760.32

Light-water reactor experience indicates that the reliability of auxiliary feed systems which are normally throttled is worse than auxiliary feed systems which initiate at full flow rates. Are there any safety-related reasons which preclude using full auxiliary feed flow at initiation?

Response

The auxiliary feedwater control valves are normally open. With SGAHRS initiation, the AFW isolation valves are opened and the AFW control valves are modulated in position to control the steam drum water level. The AFW flow to each steam drum depends on the water level of that drum.

There is no apparent safety related reason to preclude full flow at initiation. The diversity and redundancy in the CRBRP design results in a highly reliable design and as such the need for full flow auxiliary feed flow at initiation is considered unnecessary.

Question CS760.44

In Section 15.5.2.1, it is noted that if an assembly does not freely drop into an open lattice position and the triple rotating plugs are subsequently operated, "....additional severe damage can be inflicted to the assembly, to the adjacent assemblies, to the IVTM, and to the reactor upper internals." If such an incident does occur, what are the plans to assure that reactor operation does not commence with damages core assemblies or upper internals structures in the vessel?

Response

The accident hypothesized in PSAR Section 15.5.2.1, is prevented from occurring by design. Interlocks are provided to prevent this event. IVTM grapple actuation to release a core assembly is prevented unless the assembly has been lowered to its setdown elevation (reference PSAR Section 9.1.4.4.2). Plug rotation is prevented unless the IVTM grapple is fully withdrawn, and lowering the grapple is prevented if the plugs are rotating (reference PSAR Section 9.1.4.4.1). Furthermore, Control system logic operates without dependency upon these interlocks.

A specific procedure has not been outlined for recovery from this event. The likely approach to responding to this event is: refueling operations would be stopped immediately and the situation reported to plant management, who would assign responsibility for directing recovery operations. Also, the reactor vessel cover gas would be monitored immediately for detection of fission gases to determine if cladding failure of any core assemblies had occurred. (Ref. PSAR Sec. 9.8 for cover gas analysis and Sec. 7.5.4 for failed fuel monitoring). The refueling operations leading up to the event would be reviewed to determine the location and extent of possible contact inside the reactor vessel. Removeable core components might be inspected in the FHC if determined necessary. If inspection of reactor internals were necessary, special inspection equipment would be obtained. The capability is provided for complete unloading of the reactor core and draining of reactor vessel sodium if necessary to facilitate inspection and repairs.

Question CS760.46

In the evaluation of cover gas release during refueling, one cause of this event was identified as separation of the AHM from an open floor valve during a seismic event. During such a seismic event, additional fuel rod failures may occur above the 1% level. Furthermore, the fission gas released by these failed rods would not be processed by the RAPS before release to containment. Given this sequence of events, what would be the effect on the offsite doses as a function of the number of seismically failed fuel rods? Alternatively, demonstrate that the event sequence described above is so improbable as to beyond design basis.

Response

The failure of fuel rods as the result of a seismic event during refueling is beyond the design basis. Refueling preparation and termination operations involving mating the AHM at the IVTM port (when the subject cover gas release event could occur) will be performed with the upper internals structure in its lowered position providing mechanical holddown of the core assemblies.

During operation and the portion of refueling with the AHM mated to the port, the reactor core is a compact unit and there is no mechanical damage to the fuel assemblies due to a seismic event.

Fuel pin failures from a seismic event during operation will be the result of a power transient initiated by a reactivity insertion caused by seismically induced control assembly movement. During refueling, the reactor core will be sufficiently subcritical to prevent a power transient induced fuel pin failure.

Question CS760.48

Section 15.6.1.5 contains analysis of postulated intermediate HTS pipe breaks and resulting sodium fires in the steam generator building. Please provide the detailed design information regarding the high capacity venting which is required to prevent overpressurization.

Response

The design information requested is provided in PSAR Section 6.2.7.

Question CS760.51

Assess the impacts of leak rates beyond the EBL in the intermediate loop. Additionally, here, the consequences of the sodium spray fire will be greater since the atmosphere is deinerted (i.e., 20% O_2 instead of 2-3%). The time dependence of the leak rate itself is an important factor in determining the course and effects of the spray fire. The analyses must substantiate that the consequences of the leak rate and spray fire are conservatively included.

Response

The Project has defined the IHTS Design Basis Leak (DBL) as that equivalent to the flow from a sharp-edged circular orifice whose area is equal to one-half the pipe diameter times one half the pipe wall thickness. This DBL is based on the results of inservice inspection, pipe fabrication and installation quality assurance measures, fracture mechanics analyses and tests and leak detection provisions. These conditions show that a sudden large failure approaching the complete severence of an IHTS pipe is not credible.

As discussed in Section 15.6.1.5 of the PSAR, conservative assumptions have been used to maximize the effects from a IHTS Design Basis Leak spray fire. A leak was postulated to occur in the IHTS hot leg with the IHTS system operating at maximum normal operating temperature and pressure. Sodium discharge from the postulated leak continues at maximum flowrate, ($\sim$1000 GPM) until the IHTS loop and pump tank has been drained ($\sim$8.5 minutes) with subsequent plant shutdown on a plant protection primary-secondary flow mismatch signal. Subsequent to IHTS pump shutdown, sodium discharge continues as a result of static head driving pressure. This scenario results in the maximum challenge to the SGB integrity from a postulated IHTS design basis leak. No action is assumed to be taken by the operator to mitigate the IHTS sodium leakage even though extensive leak detection information would be available in the main control room to confirm the occurrence of significant sodium leakage in the Steam Generator Building (SGB).

In summary, the selected IHTS DBL is conservative and appropriate for assessing the capability of the SGB. No larger leak should be considered in the design and evaluation of the plant.

Question CS760.52

Section 15.7.1.2.1 states that the description of failure consequences of safety-related air supplies will be described only in the FSAR. Is this acceptable? There is essentially no technical information provided in the present report. These are anticipated events. Please provide technical information related to this section.

Response

The air supply system for the CRBRP is not a safety-related system although ample system redundancy and capacity is provided. This air supply system furnishes compressed air to the plant systems. The system is described in Section 9.10. Systems requiring an air supply for their safety-related operations are provided with safety-related accumulators such that the failure of the compressed air system will not result in the loss of any safety function for the duration required. Other safety-related valves are designed to move in a preferred direction with the loss of air supply.

Section 15.7.1.2 has been revised to clarify the failure effects of the compressed air system.

15.7.1.2 Loss of Instrumentation or Valve Air

15.7.1.2.1 Identification of Causes and Accident Description

The system design precludes the loss of air supply to safety-related valves or instruments due to a single credible event. However, multiple failures, or a single failure occurring at the time of a design basis event, could cause loss of instrumentation or valve air. Among such single failures are check valve malfunction caused by valve seal failure. Table 15.7.1 provides a listing of safety-related valves which requires a compressed air supply and their preferred operating directions.

15.7.1.2.2 Analysis of Effects and Consequences

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. The instrument air system is designed to supply clean, dry, and oil-free air for plant instrumentation and control. The air receiver tanks are designed to the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1. Piping is designed to ANSI B31.1.0. Piping which penetrates the reactor containment walls, and the containment isolation valves are ASME Section III, (Sections 3.9.2 and 6.2.4). Intercooler and after-coolers are designed to TEMA Class R.

All active safety-related, air operated valves will be designed to move in a preferred direction with the loss of air supply. Table 15.7.1.2-1 identifies the safety-related valves requiring compressed air and the normal and failed positions and function performed. Valves required to be operable for a safe shutdown are equipped with safety-related accumulators. Each safety-related system is redundant.

There is no compressed air supplied to safety-related instrumentation such that the loss of compressed air would result in a loss of the instrumentation safety-related function.

15.7.1.2.3 Conclusions

Based on the preceeding discussion, the compressed air system will be designed to prevent any adverse effects on the safe operation of the plant due to loss of instrument or valve air.

TABLE 15.7.1.2-1

ACTIVE SAFETY-RELATED VALVES OPERATED BY COMPRESSED AIR

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Primary Sodium Removal and Decontamination System (Nuclear Island General Purpose Maintenance System)	HV001A	Opened	Closed	Containment Isolation
	HV044A	Opened	Closed	Containment Isolation
	HV004B	Opened	Closed	Containment Isolation
	HV085A	Opened	Closed	Containment Isolation
	HV085B	Opened	Closed	Containment Isolation
	HV086B	Opened	Closed	Containment Isolation
Emergency Chilled Water	NV353	Opened	Failed Open	System Isolation
	NV354	Opened	Failed Open	System Isolation
	NV400	Opened	Failed Open	System Isolation
	NV401	Opened	Failed Open	System Isolation
	NV403	Opened	Failed Open	System Isolation
	NV404	Opened	Failed Open	System Isolation
	NV409	Opened	Failed Open	System Isolation
	NV410	Opened	Failed Open	System Isolation
	NV141AC	Opened	Failed Open	System Isolation
	NV141AD	Opened	Failed Open	System Isolation
	NV141BC	Opened	Failed Open	System Isolation
	NV141BD	Opened	Failed Open	System Isolation

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Emergency Chilled Water (cont'd.)				
	AOV165	Opened	Closed	Containment Isolation
	AOV166	Opened	Closed	Containment Isolation
	AOV167	Opened	Closed	Containment Isolation
	AOV168	Opened	Closed	Containment Isolation
	AOV211	Opened	Closed	Containment Isolation
	AOV212	Opened	Closed	Containment Isolation
	AOV79	Opened	Closed	Containment Isolation
	AOV80	Opened	Closed	Containment Isolation
	AOV415	Opened	Closed	Containment Isolation
	AOV418	Opened	Closed	Containment Isolation
Auxiliary Liquid Metal System				
EVST Na Cooler Outlet NaK				
Loop 1	HV359*	Open	Fail as Is	System Isolation
Loop 2	HV420*	Open	Fail as Is	System Isolation
EVST NaK Loop 1 Isolation	HV357*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 1 Isolation	HV358*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 2 Isolation	HV415*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 2 Isolation	HV416*	Closed	Fail as Is	Containment Isolation

*Air stored in an accumulator for emergency operation of the valve.

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Inert Gas Receiving and Processing System	RPHV001(1)	Opened	To RAPS	Process Effluent
	RPHV002(1)	Opened	Closed	Containment Isolation
	RPUV015A(1)	Opened	Closed	System Isolation
	RPUV015B(1)	Opened	Closed	System Isolation
	RPUV018(1)	Opened	Closed	System Isolation
	RPUV019(1)	Opened	Closed	System Isolation
	APHV001(2)	Opened	Closed	Containment Isolation
	APHV002(2)	Opened	Closed	Containment Isolation
	NGHV351A(3)	Opened	Closed	Containment Isolation
	NGHV351B(3)	Opened	Closed	Containment Isolation
	CGHV501(4)	Opened	Closed	Containment Isolation
	CGHV301(4)	Opened	Closed	Containment Isolation

(1) See Figure 11.3-4; (2) See Figure 11.3-6; (3) See Figure 9.5-8; (4) See Figure 9.5-2

Evaporator Water Dump	53WDV001-004	Closed	Closed	System Isolation
Superheater Outlet	53SGV106-108	Closed	Closed	Relief (Power Operation)
Evaporator Outlet	53SGV100-103	Closed	Closed	Relief (Power Operation)
Steam Drum Outlet	53SGV104-105	Closed	Closed	Relief (Power Operation)

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Heating Ventilation and Air Conditioning System	ARAOV046A	Opened	Closed	Containment Isolation
	ARAOV046B	Opened	Closed	Containment Isolation
	ARAOV046C	Opened	Closed	Containment Isolation
	ARAOV047A	Opened	Closed	Containment Isolation
	ARAOV047B	Opened	Closed	Containment Isolation
	ARAOV047C	Opened	Closed	Containment Isolation
	ACAOV064A	Opened	Closed	System Isolation
	ACACOV064B	Opened	Closed	System Isolation
	ACAOV122A	Opened	Closed	System Isolation
	ACAOV122B	Opened	Closed	System Isolation
	ACAOV123A	Closed	Open	System Isolation
	ACAOV123B	Closed	Open	System Isolation
Floor Drain System	AOV34	Opened	Closed	Containment Isolation
	AOV67	Opened	Closed	Containment Isolation

Question CS760.55

In Section 15.7.1.4, Off-Normal Cover Gas Pressure, the relief valve setpoint pressure is 15 psig, while the elastomer seal system is also designed for 15 psig and it is stated that the dip seals of the reactor vessel closure would be "upset" at this pressure.

a. Why not design the seals for a higher pressure or change the relief valve setpoint to a lower value?

b. What type of valves are being used? What is the failure frequency (failure to close following a discharge)? Are they subject to common cause/mode failures?

c. What are the consequences should the seals fail without any increase in the buffer gas flow rate?

Response

a. PSAR Section 15.7.1.4.2 has been amended to clearly demonstrate that the design pressure of the elastomer seals is 300 psid well above the relief valve setpoint.

b. Two cover gas pressure relief valves are provided for overpressure protection of the reactor and overflow vessels' gas spaces. These valves are located in Cell 107B on the equalization piping connecting the two vessels. The pressure relief valves are spring loaded safety type valves designed to ASME Section III Class 2 requirements, and each valve is protected from the sodium vapor environment by a rupture disk located upstream of the valve. Each pressure relief valve/rupture disk assembly is equipped with a normally open blocking valve located upstream of the assembly. The blocking valve is used to isolate each assembly from the reactor cover gas boundary during testing of the relief valve. It can be used to isolate the line in the event the pressure relief valve does not reset following a pressure relief operation. The back pressure on both pressure relief valves is referenced to the Cell 107B atmospheric pressure.

c. A bounding analysis of the consequences associated with seal failure has been provided in the response to Question 001.81 incorporated into the PSAR in Amendment 2.

15.7.1.4 Off-Normal Cover Gas Pressure in the Reactor Coolant Boundary

15.7.1.4.1 Identification of Causes and Accident Description

As described in Section 9.5.1, the cover gas system serving the Reactor and Primary Heat Transport System maintains a pressure in the gas space of the Reactor Coolant Boundary of 6" ± 2" of H_2O. There is a constant sweep flow into the cover gas spaces and through the shaft seals of the primary pumps. This in-leakage is accommodated by two parallel pressure regulators in the line between the RAPS and the primary system overflow tank, which is maintained at the same pressure by a gas pressure equalization line connecting the pumps, reactor vessel, and overflow tank. The makeup regulators and the regulators controlling the bleed from the overflow tank to RAPS are both controlled from the same pressure signal. Failures of the pressure regulators (primary and redundant) or operator error, could cause deviation from the normal operating pressure of 6" ± 2" W.G.

15.7.1.4.2 Analysis of Effects and Consequences

a. Under pressure: If the pressure regulators (including redundant regulators) between the overflow tank and the RAPS system fail open, the pressure in the cover gas spaces within the Reactor Coolant Boundary will go sub-atmospheric since gas from the overflow tank will flow into the RAPS vacuum vessel. Since the vacuum vessel volume is approximately 300 ft^3 (at 8 psia minimum) and the combined gas volume of the reactor vessel, three primary pumps and the overflow tank is about 4500 ft^3, the reduction in pressure is modest: about 1 psi. Such a reduction in pressure would have no adverse affect on the primary system. The change in NPSH available to the pump would not be significant, and the seals in the reactor and pump closures would not be materially affected.

b. Over pressure: If the regulators between the overflow tank and the RAPS should close and the regulators controlling flow to the reactor vessel should, at the same time, fail open, the cover gas pressure in the reactor coolant boundary would increase. Any potential problem is mitigated however by: 1) the time required to establish any significant overpressure, and 2) pressure relief devices on the overflow tank. As mentioned above, the volume of the cover gas space within the reactor coolant boundary is about 4500 ft^3 at normal operating conditions. Since the gas makeup system will be designed to limit the makeup rate to about 50 SCFM, it would take at least an hour to double the cover gas pressure in the reactor coolant boundary. An annunciator in the control room will alert operators to take appropriate action (such as isolation of the makeup gas regulators) long before any appreciable overpressure will be realized. In addition, relief valves set to relief at 15 psig, will limit the pressure even if no operator action is taken prior to reaching this pressure. The discharge of cover gas from the relief device will be modest and CAPS action will preclude any hazard to the public. Even if the pressure does increase to 15 psig, there will be no affect on reactor vessel level or pump tank level performance.

The pressure boundary margin seals will resist pressure in excess of 300 psid without failure. If the pressure in the reactor vessel should increase to 15 psig, these seals would remain intact. Cover gas would bubble through the dip seal and be trapped in the riser annulus between the dip seal and inflatable elastomer seals.

The primary system (and reactor vessel) design pressures have been established on the basis of a 15 psig cover gas pressure, and therefore the system, from a structural standpoint, is unaffected by any overpressure which could occur. If the primary system gas pressure should drift up due to one of the postulated failures, the 10 psi minimum P between the intermediate and primary sodium in the IHX would decrease; however, this loss of ΔP would be monitored and annunciated and appropriate action would be taken.

15.7.1.4.3 Conclusion

Off-normal cover gas pressures in the Reactor Coolant Boundary will not cause a safety problem. Underpressure would be limited to approximately 1 psi below the normal operating pressure of 6 inches W.G. Overpressure conditions would be limited to 15 psig by relief actions and would take about an hour to achieve. Even if such an overpressure condition were to exist, there will be no deleterious effect on the integrity of the Reactor Coolant Boundary. Since radiation dose rate builds up slowly and adequate radiation monitoring is provided, the radiation consequences would be small to the operating staff and are trivial to the public.

Question CS760.82

One page 4.4-57, it is explained that the THDV conditions are more conservative than the PEOC and therefore, represent the "worst bound" of plant conditions. On page 4.4-13, it is stated that a maximum sodium temperature of 1550°F under transient conditions provides an adequate margin to boiling. The limit assumes THDV conditions and a 750°F inlet temperature. For these conditions, Table 4.4-3 shows fuel and radial blanket temperatures of 1571°F* and 1580°F, respectively, which are well in excess of the 1550°F limit.

Explain why those temperatures can exceed the prescribed limit?

* Note: Table 4.4-3 shows the maximum transient temperature in a fuel assembly as 1571°F for assembly number 46. According to Figure 4.4-9, assembly number 46 is an internal blanket assembly.

Response

We emphasize again that the value of 1550°F is not a limit. It is a conservative guideline used to guide the core orificing. The answer to this question is similar to the answer to the preceding Question CS760.81. Maximum temperatures were calculated for three representative and limiting assemblies having initially assumed flows. A maximum steady-state temperature which corresponds to a 1550°F transient temperature was then calculated. In fact, Table 4.4-3 shows that for the fuel and radial blanket assembly, the steady-state temperature calculated with the assumed flow, while for the inner blanket where the maximum transient calculated temperature was 1498°F, the steady-state temperature corresponding to 1550°F is higher. Thus, to satisfy the 1550°F guideline, the assumed assembly flow must be increased for the fuel and radial blanket assembly but decreased in the inner blanket case. Actually, the three assemblies considered were only representative worst cases and the intent was to establish criteria for the steady-state temperature so as not to exceed a transient temperature of 1550°F. These values were used in determining the TELTs for each assembly.

Regarding the note, Table 4.4-3 has been amended to remove this inconsistency.

TABLE 4.4-3

COOLANT LIMITING TEMPERATURES FOR TELT CALCULATIONS

(TEMPERATURES IN °F)

Typical Worst Case for Assembly Type	HETEROGENEOUS CORE MAXIMUM TRANSIENT TEMP. (FORE-2M CALCULATED)	STEADY STATE TEMP. CORRESPONDING TO HETEROGENEOUS CORE MAXIMUM TRANSIENT TEMP. (FORE-2M)	STEADY STATE TEMP. CORRESPONDING TO 1550°F MAXIMUM TRANSIENT TEMP.	T_M
Fuel Assembly	1571	1331	1316	1252 First Core 1261 Second Core
Inner Blanket Assembly	1498	1247	1282	1198 First Core 1207 Second Core
Radial Blanket Assembly	1580	1331	1310	1232

Temperatures at THDV, 3σ, 750°F Inlet

Temperatures for PEOV, 2σ

51

Question CS760.90

The major concern for the PHTS and the IHTS is in maintaining and assuring the integrity of the sodium piping. The main focus is on the definition of the size of the pipe break which must be considered as part of the design basis for the primary loop hot leg and intermediate loop pipes. The applicant has an on-going effort in this area to substantiate the "leak before break" hypothesis.

a. Sodium-to-gas leak detection systems exhibiting adequate sensitivity for operating in this type of environment have been designed. However, please provide us with evidence that their long-term performance has been adequately demonstrated. Provide evidence that they have been tested on large systems such as will be used for CRBR. What data is being used to verify their reliability?

b. Please provide us with the details regarding the pre- and in-service inspection program.

c. Please provide us with the details regarding the material surveillance program.

d. Since considerable effort is being made on improved weld materials and welding techniques, please provide us with all long-term data on the CRBR weld compositions which demonstrate that adequate residual ductility is available during the lifetime of the plant.

Response

a. Long-term performance testing of prototypical CRBRP Aerosol-Type Leak Detectors (sodium ionization & plugging filter aerosol) was initiated in April, 1977 at EBR-II and is currently in progress. These detectors, on tests for over 5 years, have demonstrated the capability of those devices to meet CRBRP performance objectives.

The FFTF has a permanently installed sodium-to-gas leak detection system. The performance of that system demonstrate reliable functioning under actual plant operating conditions.

Verification testing of the aerosol gas sampling leak detection system was performed with a prototypical section of CRBRP large diameter piping in both an air and an inerted environment. This was an insulated pipe 24 inches in diameter with a 8 foot long annulus and a 1 inch gap between the pipe and the insulation. Aerosol sniffer tubes and a collection manifold representative of the CRBRP design were used to transport the aerosols to the detectors during weeping leaks (100g/hr) at typical CRBRP operating temperatures. These leaks were detected well within the time period which could cause significant corrosion damage to the piping.

Data obtained from the development tests (screening, characterization and optimization, mockup, natural circulation, and verification tests conducted at AI-ESG, the long-term performance tests conducted at EBR-II and performance of the FFTF have sodium-to-gas leak detection system have been used in verifying the reliability of this equipment. An analysis of

the sodium-to-gas leak detection system to reliably detect small leaks in the PHTS has been performed. The assessment results are that the design provided a highly reliable system function.

b. The details of the pre- and in-service inspection program are provided in PSAR, Appendix G.

c. The details of the material surveillance program are provided in WARD-D-0185. (Reference 2 of PSAR Section 1.6).

d. Data on the effect of long-term exposure on weld compositions were evaluated and included in the CRBRP "Integrity of Primary and Intermediate Heat Transport System Piping and Containment" report, WARD-D-0185. (Reference 2 of PSAR Section 1.6).

   In this report, it was recognized that additional information would be required, and, consequently, a long-term thermal aging effects program on CRBRP prototypic welds was identified. This program is contained in Appendix C, Volume 2, of the referenced report.

Question CS760.93

Given the presence of a small, undetected leak, the erosion of external surfaces of the piping by leaking sodium and its reaction products may be minimized, but not eliminated by the nitrogen/oxygen inerting atmosphere. However, while such an environment surrounds the primary piping under normal operation, most of the intermediate pipes are in a normal air atmosphere.*

How has this been accounted for in detail for specification of the pipe break/leak sizes?

Response

The Project is currently compiling additional information regarding piping integrity in the intermediate heat transport piping system. The effects of operation in a normal air environment are being addressed and have been considered in the specification of the design basis leak for the intermediate Heat Transport piping. The Project will submit an amended response to this question by October, 1982.

Question CS760.94

Provide analyses which consider a spectrum of postulated pipe breaks of different sizes (up to double-ended); a) consider critical locations in the primary hot leg (e.g., at the pump discharge, and in the intermediate loop piping and b) are run under varying initial and transient conditions to provide sufficient assurance that the entire range of potential thermal/hydraulic consequences to the system have been assessed, c) are based on analytical techniques and computer codes which are verified to conservatively bound the effects of such pipe breaks.

Response

The response to this question is contained in the response to Questions CS760.37 and 760.51.

Question CS760.95

The main steam isolation valves (superheater outlet isolation valves) play an important role in many potential events including station blackout and main steam line break. Under what conditions are these valves closed? Which systems close them? What is the OSIS system referenced in Table 5.5-5 and under what circumstances will it close the valve?

Response

There are two conditions which result in automatic closing of the superheater outlet isolation valve and superheater bypass valve for any single loop. The first is a large or intermediate-sized sodium water reaction pressure relief system (SWRPRS) trip signal. The conditions that result in a SWRPRS trip are given in PSAR Section 7.5.6. The second condition which results in automatic closure of a superheater outlet isolation valve and superheater bypass valve is a low superheater outlet pressure (< 1100 psig). This function is identified in PSAR Table 5.5-5.

The outlet steam isolation subsystem (OSIS) closes the superheater outlet isolation valves and superheater bypass valves in all three loops as a result of either a high steam-to-feedwater flow ratio signal or a low steam drum level signal. The OSIS is discussed in PSAR Section 7.4.2.

PSAR Table 5.5-5, Section 7.4.2.1.3, and Figure 7.5-6 (Sheets 3, 4, 5, and 6) change pages are provided to clarify the OSIS function.

TABLE 5.5-5

SGS PUMP AND VALVE DESCRIPTION

PUMPS	ACTIVE	INACTIVE	ACTUATING SIGNAL
Recirculation Pump		X	N/A
VALVES			
Pump Suction Isolation		X	Manual (Remote)
Evaporator Inlet Isolation		X	SWRPRS
Evaporator Inlet Water Dump		X	SWRPRS
Evaporator Outlet Relief	X		SWRPRS**, High Pressure Evaporator
Steam Drum Relief	X		High Pressure - Steam Drum
Superheater Inlet Isolation		X	SWRPRS
Superheater Relief	X		SWRPRS**,High Pressure Superheater
Superheater Outlet Isolation	X		SWRPRS**, OSIS/SGAHRS,or Low Super-heater Outlet Pressure
Superheater Bypass Valve	X		SWRPRS**, OSIS/SGAHRS, or Low Super-heater Outlet Pressure
Steam to SGAHRS HX		X	Manual (L.O.)*
Water from SGAHRS HX		X	Manual (L.O.)*
Steam to SGAHRS Auxiliary FW Pump		X	Manual
Feedwater from SGAHRS		X	Manual (L.O.)*
Main Feedwater SGB Isolation	X		SWRPRS**, High Steam Drum Level, Low Steam Drum Pressure, Cell Temp and Humidity
Main Feedwater Drum Isolation		X	High Steam Drum Level
Main Feedwater Check Valve		X	Simple Check
Main Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Startup Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Evaporator Outlet Check Valve		X	Check Valve
Superheater Outlet Check Valve		X	Check Valve
Steam Drum Drain Isolation	X		SWRPRS**, SGAHRS Initiation, Low Steam Drum Pressure

* L.O. - Locked open

** This function is not safety active

7.4.2.1.2 Equipment Design

A high steam flow-to-feedwater flow ratio is indicative of a main steam supply leak down stream from the flow meter or insufficient feedwater flow. The superheater steam outlet valves and superheater bypass valves shall be closed with the appropriate signal supplied by the heat transport instrumentation system (Section 7.5). This action will assure the isolation of any steam system leak common to all three loops and also provide protection against a major steam condenser leak during a steam bypass heat removal operation.

7.4.2.1.3 Initiating Circuits

The OSIS is initiated by the SGAHRS initiation signal. The SGAHRS initiation signal is described in 7.4.1.1.3. This initiation signal closes the superheater outlet isolation valves in all 3 loops when a high steam-to-feedwater flow ratio 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 ratio and the low steam drum level are input to a two of three logic network. If two of three trip signals occur in any of the 3 loops, the OSIS is initiated, and all 3 loops are isolated from the main superheated steam system by closure of the superheater outlet isolation valves and superheater bypass valves.

7.4.2.1.4 Bypasses and Interlocks

Control interlocks and operator overrides associated with the operation of the superheater outlet isolation valves have not been completely defined.

Bypass of OSIS may be required to allow use of the main steam bypass and condenser for reactor heat removal. In case the OSIS is initiated by a leak in the feedwater supply system, the operator may decide to override the closure of certain superheater outlet isolation valves.

7.4.2.1.5 Redundancy and Diversity

Redundancy is provided within the initiating circuits of OSIS. The primary trip function takes place when a high steam-to-feedwater flow ratio is sensed by two of three redundant subsystems on any one SGS loop. The low steam drum

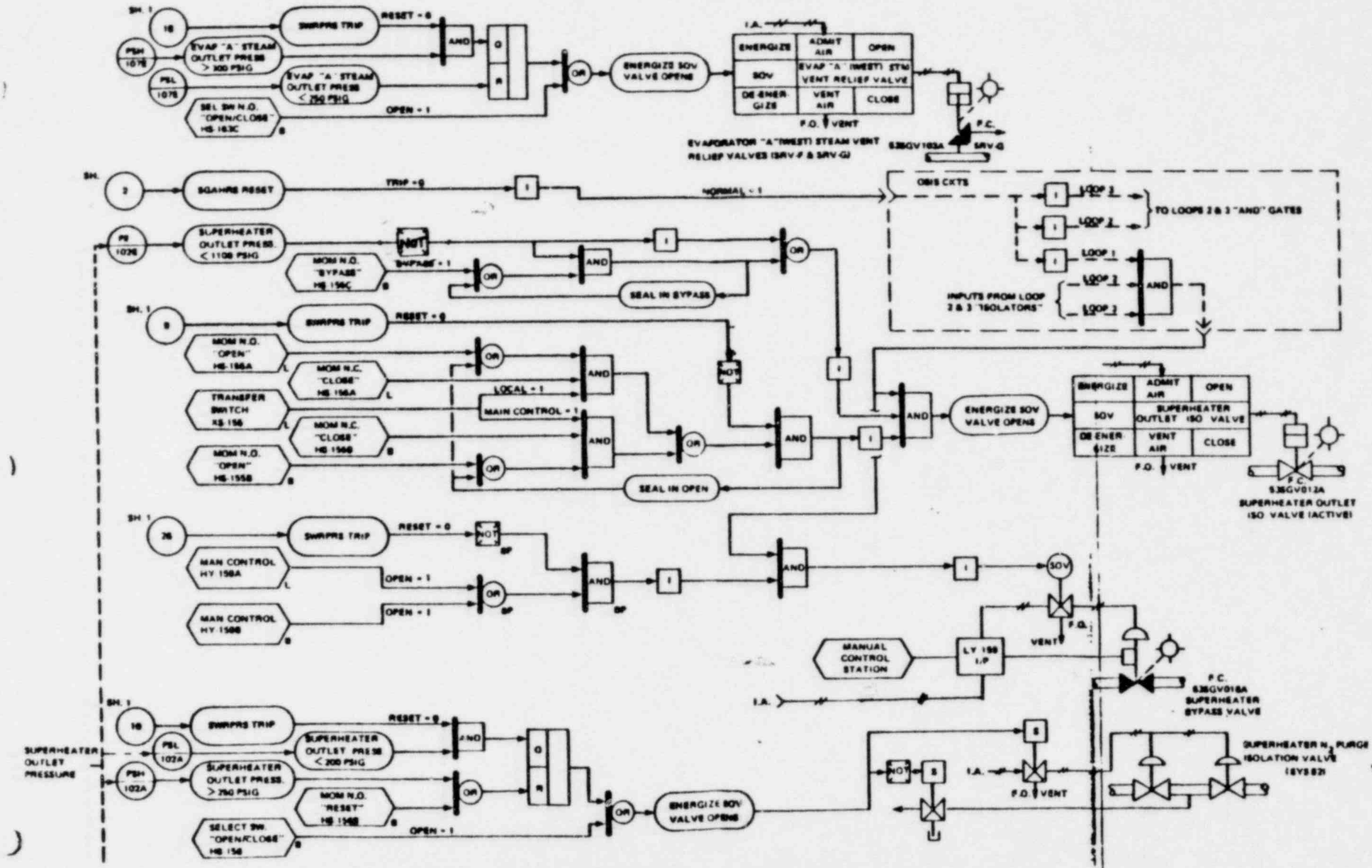

Figure 7.5-6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

SHEET 5 of 6

82-135-02

Amend. 67
Mar. 1982

Question CS760.97

What, if any, protection or control systems are capable of detecting a recirculation pump trip directly?

Response

The Plant Protection System does not detect a recirculation pump trip directly. However, a reactor trip is initiated via PPS when a recirculation pump trips from High Evaporator Outlet Temperature.

The recirculation pump trip can be detected and verified at the following locations:

- Discharge pressure indicator located on local SGS panel.
- Suction pressure indicator located on local SGS and main control panels.
- Differential pressure indicator located on local SGS and main control panels.
- Group alarm for low differential pressure located in the main control room.

Question CS760.102

In Section 5.5.2.3.4 (Steam Generator Module), the presentation on accident analysis takes credit for improved methods of welding the tube to the tubesheet. However, the PSAR indicates that this weld is in a developmental stage.

If the weld method is important to safety (e.g. failure frequency), please provide details of the method and any supporting evidence that indicates its superiority over previous methods.

Response

The weld method employed in the tube-to-tubesheet welds of the CRBRP Steam Generators is an in-bore butt weld. It was selected to avoid the crevices which exist if a front face fillet weld would be used. The weld method as well as the welding equipment has been utilized before and as such were not the subject of the development program. The development program was aimed at the improvement of the weld quality and dependable repeatability of the process. The measures taken to this end are described in Section 5.5.2.3.4 of the PSAR as follows:

For the steam generator tube-to-tubesheet welds, the ASME Code requirements (NB-4000 and NB-5000) were supplemented by requirements of RDT E15-2 and additional requirements. Requirements imposed on the tube-to-tubesheet welds above those of the Code include:

- Vacuum-Art Remelt or Electroslag Remelt - material is specified to reduce impurities and improve properites for tubesheet forgings and tubes.
- Post weld heat treatment range defined to optimize resistance to caustic stress corrosion cracking.
- Helium leak test.
- Penetrant test requirement limiting defect size to much less than that of the Code.
- Weld geometry requirement limiting concavity, convexity and wall thinning.
- Micro-focus radiographic examination - developed to radiograph tube-to-tubesheet welds with improved resolution.

All of the above measure were taken to assure high quality welds. The actual "weld development" is the weld procedure development required to quality the procedure, equipment and personnel as required by the Code.

In addition, the following efforts were undertaken to improve upon available commerical quality standards to achieve the highest quality, dependable welds obtainable:

1. The tube-to-tubesheet preliminary weld development efforts covered work on CRBRP steam generator tube-to-tubesheet welding up to the beginning of weld qualification. This included laboratory weld development, the check-out and verification of the process under manufacturing conditions, a statistical evaluation of the process to establish acceptance criteria, the associated quality assurance procedures and the development and procurement of appropriate welding power supplies.

2. Definition of tight weld geometry acceptance criteria.

3. Post-weld heat treatment thermal stress evaluation.

4. Investigation to determine the likelihood of cracking of the tube-to-tubesheet welds during PWHT.

autogeneous buttwelded, tube-to-tubesheet joints. The shell and tube material is 2-1/4 Cr-1 Mo steel. There are two evaporator modules and one superheater module per loop. The evaporator modules operate with a recirculating ratio of 2:1.

The steam generator design requires that each loop (two evaporators and one superheater) develop 325 MWT at rated full load. The design life of the steam generator module is 30 years with items that cannot be reasonably expected to last 30 years being replaceable during in-service inspection periods. The steam generator is designed to withstand the normal, upset, emergency and faulted operating conditions in accordance with CRBRP Criterion 26. The steam generator module is also designed to withstand the loading combinations indicated in Section 3.9.2.2.

The materials used in design of the steam generator module major components are as follow:

Pressure Boundary (2-1/4 CR-1 Mo-Ref. 3, Vol. 1, Section 2.2):

Shell Forgings	-	SA 336, Class F22A
Shell Plate	-	SA 387, Grade 22, Class 1
Tubesheet Forgings	-	RDT M2-19 with optional provision 2 and Code Case 1557-2
Tubing	-	RDT M3-33 as modified to limit silicon and carbon content for weldability and carburization considerations (Reference 5).

The steam generator module supplier will provide procedures for welding and heat treating in accordance with the requirements specified in the Code as modified by RDT E15-2NB (see Section 5.5.1.2). Welding qualification is controlled by the Code as modified by RDT F6-5 and RDT E15-2NB (see Section 5.5.1.2).

The weld method employed in the tube-to-tubesheet welds of the CRBRP Steam Generators is an in-bore butt weld. It was selected to avoid the crevices which exist if a front face fillet weld would be used. The weld method as well as the welding equipment has been utilized before. A development program was aimed at the improvement of the weld quality and dependable repeatability of the process. The measures taken to this end are described as follows:

For the steam generator tube-to-tubesheet welds, the ASME Code requirements (NB-4000 and NB-5000) were supplemented by requirements of RDT E15-2 and additional requirements. Requirements imposed on the tube-to-tubesheet welds above those of the Code include:

- Vacuum-Arc Remelt or Electroslag Remelt - material is specified to reduce impurities and improve properties for tubesheet forgings and tubes.

- Post weld heat treatment range defined to optimize resistance to caustic stress corrosion cracking.

- Helium leak test.

- Penetrant test requirement limiting defect size to much less than that of the Code.
- Weld geometry requirement limiting concavity, convexity and wall thinning.
- Micro-focus radiographic examination - developed to radiograph tube-to-tubesheet welds with improved resolution.

Material integrity prior to placing the steam generators in service will be assured by complying with the ASME Code Section III which requires weld radiography, tubing ultrasonic testing, plate ultrasonic testing, tubing hydraulic testing, component pressure testing and helium leak testing.

Material considerations are indicated in Sections 5.5.1.4 and 5.5.3.11. Section 5.5.3.1.5 indicates the tests being conducted to support the steam generator design. It is not anticipated that back-up materials will be required.

Question CS760.106

In the event of pipe breaks, what would cause the various isolation valves to close? It is stated in Section 5.6.1.2.1 that in at least one such break it is necessary for the operator to close an isolation valve to save the inventory of the PWST. How many such postulated breaks required operator intervention? How does the operator determine the break location?

Response

In the event of a large pipe break, in a steam generator loop or in an AFW loop downstream of the AFW check valves, the AFW isolation valves to the affected SGS loop will automatically close following the steam drum depressurization to <200 psig. All postulated breaks that do not allow steam drum depressurization will require operator action to isolate the AFWS if the breaks are large enough to initiate SGAHRS.

The information available to the operator to isolate these pipe breaks is described in PSAR Section 5.6.1.2.1.1 "Identification of Active and Passive Components which Inhibit Leaks".

5.6.1.2 Design Description

5.6.1.2.1 Design Methods and Procedures

5.6.1.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

The equipment of the SGAHRS is shown schematically in Figure 5.1-5. Valves and pumps within the SGAHRS are classified as active or inactive, and their operating mode is given in Tables 5.6-5 and 6.

In the event of a pipe break in the auxiliary feedwater portion of SGAHRS, continued heat removal capability will be assured by the multiple loop feature of the SGAHRS and heat transport system.

If a large pipe break occurs in any portion of a steam generator loop, this will result in a reactor shutdown and an AFW initiation. Automatic isolation of the AFW supply to the affected loop will occur within approximately 2 minutes when the steam drum pressure falls below 200 psig. Operator action as a backup is available.

If after an AFW initiating event, a pipe break were to occur in the auxiliary feedwater piping between the steam drum and the isolation valves immediately downstream of the control valves, the flow in the effective loop will increase until limited by the control valve (at approximately 110% of rated flow). A flow limit alarm in the control room will alert the operator to the fact that corrective action is necessary. Following the control valve flow limit alarm, the operator verifies a leak from information provided by the following instrumentation:

a) Safety-related steam drum level and pressure indication are provided on each loop to assist in making a break determination. An inability to recover level or maintain pressure on any steam drum with a corresponding flow limiting alarm on AFW provides a break indication.

b) The Steam Generator Building (SGB) Flooding Protection Subsystem annunciates abnormal SGB temperature, humidity, and sump level in the control room to alert the operator to pipe breaks that could compromise SGAHRS operation (see Section 7.6.5).

c) The plant trip signal: high or low steam to main feedwater flow ratio or low steam drum level. A trip of this type will direct the operator's attention to the steam/water-side of the plant.

Operator action in the control room will close two AFW supply isolation valves to isolate the defective loop. In addition to the above, automatic isolation will occur when the AFW flow remains above 150% for 5 sec. (indicating a flow limiter failure). Due to the flow limiting capaibility of the control valves, the leakage flow will be minimized and proper flow to the two remaining steam drums will continue even though one loop has suffered a pipe break.

Question CS760.107

How soon must the operator intervene and under what conditions to avoid necessary depletion of the PWST inventory?

Response

The conditions under which the operator must intervene in the event of pipe breaks are described in the response to Question CS760.106. The time available for operator intervention to isolate a pipe break is discussed in PSAR Section 5.6.1.2.1.1.

Question CS760.108

Does the operator have any way of knowing the water levels in steam drums, in the PACC loops?

Response

The main control room operator can continuously monitor the steam drum water level in each loop on main control panel indicators. In addition, steam drum level indication is available in the main control room on the PDH&DS. Steam drum level indication also is provided on the three SGAHRS panels at the remote shutdown station in the SGB intermediate bay. The signals to the indicators mentioned above are generated by three, independent, safety-related differential pressure detectors connected to each steam drum. In addition, each steam drum is equipped with a multiport sight glass which can be observed locally.

There are no level detectors on the PACC loops. However, the PACC common condensate return line contains a venturi flow meter the output of which is indicated on the main control panel, PDH&DS and the SGAHRS panels. During normal plant operation, the water level in the PACC condensate return piping essentially is the same as the water level in the steam drum. During PACC operation, water level in the PACC condensate return lines will be higher due to 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, the condensate flowrate back to the recirculation pump suction as described in PSAR Section 5.6.1.3.2. The PACC is being designed to ensure that the condensate flow from the tube bundles will be stable. Therefore, condensate flow measurement is adequate indication of proper PACC operation and no PACC loop level measurement is required.

Question CS760.109

More information is required concerning the turbine drive for the turbine driven auxiliary feedwater pump. How will pump performance depend on the pressure and temperature of the steam driving the turbine?

Response

The SGAHRS turbine drive is designed to produce a rated 2000 hp at 4000 rpm when supplied with a steam at 1000 psig, 546°F and 96% quality at turbine inlet. The required turbine inlet conditions are regulated and maintained by a pressure control valve which reduces the steam pressure from the steam drum operating pressure (1475 psig during superheater venting, 1550 psig during steam drum venting) to 1000 psig at turbine inlet.

Normally, following SGAHRS venting, the SGS pressure will be 1400 psig as controlled by the PACC heat rejection rate. When the PACC assumes the total heat load (~1 hour following SGAHRS initiation) the need for feedwater is limited to makeup leakages. Under the refueling conditions the pressure will drop as the SGS temperature is reduced to 400°F. At refueling conditions the makeup of water leakage is provided by the main feedwater system. In the event the main feedwater system becomes unavailable, the motor driven auxiliary feedwater (AFW) pumps provide a redundant alternate source of makeup water. Each of the two motor driven AFW pumps has full capability of providing makeup water for these conditions. There is no need to operate the turbine driven auxiliary feedwater pump at steam drum presures below 1000 psig.

Question CS760.114

Please provide direct heat removal service design details including:

a. Air blast heat exchanger and overflow heat exchanger design elevations.

b. Length of the piping

c. Design temperatures at inlet and outlet locations of the heat exchangers.

Response

a) Design elevations:

1) Air Blast Heat Exchanger (ABHX) centerline of the Nak outlet line is elevation 777'-0 5/8".

2) Overflow Heat Exchanger (OHX) centerline of the sodium outlet nozzle is elevation 757' - 10 1/4".

b) The length of the Direct Heat Removal System piping can be determined per Ref. CS760.114-1.

c) Design temperatures:

1) Structural design temperature for the ABHX is 650°F at 100 psig internal pressure.

2) Structural design temperature for the OHX is 650°F at 100 psig internal pressure.

Ref. 760.114-1:
Letter HQ:S:82:29, J. R. Longenecker to P. S. Check, dated June 25, 1982.

Question CS760.115

The auxiliary feedwater system is an integral part of the decay heat removal system, yet for some transients (e.g., steam generator tube leaks) it is necessary to shut down the auxiliary feedwater supply.

a. Does the requirement to be able to isolate the auxiliary feedwater system impair the availability when needed?

b. Can a single failure or spurious isolation signal lead to a simultaneous isolation of all auxiliary feedwater trains?

Response

a. The design of the AFW subsystem of the SGAHRS is such that isolation of AFW to one loop of the Steam Generator System will not impair the availability of the AFW to the other two loops. The AFW subsystem is capable of removing plant decay and sensible heat under all conditions in which at least one heat transfer loop remains intact. Isolation of one AFW loop will, therefore, not impair safe shutdown of the plant.

b. The AFW subsystem design provides separate piping,valving,and controls for each steam generator loop. This arrangement ensures separation of each loop in performing its function. As such a single failure or spurious isolation signal will not lead to simultaneous isolation of all AFW trains.

Question CS760.117

In the PSAR the presentation of the Direct Heat Removal Service (DHRS) does not contain enough detailed information for analysis at this time. The DHRS decay heat removal capabilities cannot be adequately assessed but on the basis of limited information available some very preliminary simplified analysis has been conducted. The DHRS design is such that the sodium intake nozzle only penetrates the reactor vessel wall (not the thermal liner). The sodium flows into the intake nozzle partly from tha hot upper plenum through the thermal liner ports (2.625 inches below normal operating sodium level) and partly from the bypass flow from the cold lower plenum. The DHRS return nozzle penetrates into the hot upper plenum where the returning cold sodium mixes with the hot sodium and depends on the primary pony motors for force the sodium flow through the primary loop.

Our concerns regarding DHRS to loss of heat sink (LOHS) events include:

a. The DHRS intake nozzle doesn't penetrate into the upper plenum. The sodium level falls below the thermal liner ports for an extended period of time, how would level be recovered and if it cannot be recovered what are the consequences?

b. From our DHRS preliminary studies, the temperature of the sodium returning to the upper plenum from the DHRS is about 600°K, and during loss of normal heat sink events, the sodium temperature exiting from the core is about 900°K. What is the thermal mixing in the upper plenum? Is there any thermal stress concentration? Is there any flow reversal?

c. Please provide the details of the 1/21 scale plenum tests which are mentioned in Q001.580.

Response

a. There is no credible combination of events which would result in sodium level falling below the thermal liner ports for an extended period while DHRS is operating to remove decay heat; however, DHRS is designed with the following makeup capability in the event that overflow has been interrupted for any reason: Prior to reactor scram, the overflow vessel is one-half full (approximately 17,000 gallons). The makeup pumps continue to transfer sodium back to the reactor. Pump flow is increased to 560 gpm total (280 gpm per pump) upon initiation of DHRS. The 17,000 gallons of sodium in the overflow tank is equivalent to about 89 inches of available makeup level in the reactor.

b. The fluid mixing in the upper plenum during DHRS operation has been determined experimentally in the 0.248 scale Integral Reactor Flow Model (IRFM). Richardson number modeling was used to simulate CRBRP shut-down conditions for both three-loop and one-loop flow. The test results indicate that the plenum is well mixed with only 6% of the DHRS flow short-circuiting the primary loops during three-loop

operation and 10-11% short-circuiting during one-loop operation. The DHRS design allowable is 20%. A second indication of the fluid mixing is from the small magnitude of the thermal striping measurements.

The highest peak-to-peak fluctuation, measured during three-loop testing, was 18% of the difference between the core exit temperature and the DHRS return nozzle (sodium makeup nozzle) temperature.

Because of the fluid mixing in the plenum the only area with a potential thermal stress concentration is at the sodium makeup nozzle.

There will not be any flow reversal through the core with any primary loop pony motors operating. If one or two of the loops are inoperative there could be reverse flow, without safety consequence, through those loops.

c. In the response to Q001.580 it was stated that the adequacy of the geometric locations of the DHRS makeup and overflow nozzles was demonstrated in a 1/21 scale Outlet Plenum Feature Model Test, and that this adequacy would be confirmed in future testing in the 0.248 scale IRFM. The IRFM tests have been completed. Documentation of test results will be available in August 1982.

Question CS760.118

Provide a Failure Modes and Effects Analysis (FMEA) for the PHTS, IHTS, the DHRS, the PACC's, or other systems involving shutdown heat removal and natural circulation?

Response

Failure Modes and Effects Analysis (FMEA's) for the PHTS, IHTS, DHRS, PACCs, and other systems involving shutdown heat removal and natural circulation are currently being conducted. Current Project FMEAs are not sufficiently finalized to provide adequate or appropriate information at this time. This information will be available at the OL state.

Question CS760.120

Given a complete loss of the PHTS and trip, what time is available to either recover the system or to put the DHRS into operation before (A) unacceptable loss of level in the vessel, (B) dry out of the Steam Generator's or (C) boiling in the hot channel or the core?

Response

The Project interprets the NRC question to be one which questions the operator response time to initiate the DHRS assuming all heat transfer is lost at the IHX at the time of scram.

This information can be found in the response to NRC Question 760.38.

Question CS760.121

Has a reliability analysis on the PACC system been initiated? What methods and models are presently used to determine:

- o corrosion impact
- o heat transfer deterioration
- o monitoring
  - parameters during transients
  - frequency
  - testing
  - maintenance
- o frequency of demand
- o analysis of operation
  - nominal
  - off-normal
    possibilities of overcooling varying steam supply loss of power to fan changing steam conditions

Response

Yes; both quantitative and qualitative reliability analyses on the PACC system have been initiated. The analyses utilizes failure state modeling and Failure Mode and Effects Analysis (FMEA).

o Corrosion Impact

The impact of corrosion on the reliability of the PACC system is being considered. In the quantitative reliability evaluation of the PACC, corrosion is considered as one of several causative factors for PACC leakage and is included in the PACC system leakage failure rate estimate. Corrosion is considered as a possible failure cause. The impact of corrosion would be minimized by inservice inspection and by stringent water chemistry requirments.

o Heat Transfer Deterioration

The impact of heat transfer deterioration on the reliability of the PACC is being considered. Degraded heat transfer capability has been considered as a PACC failure mode. The results how that the fouling factor is not significant.

o Monitoring

The PACC reliability analyses have not taken credit for the benefit of monitoring specific PACC parameters during transients.

In the quantitative reliability evalution of the PACC, consideration of monitoring frequency, testing, and maintenance is being given. Monitoring frequency and testing impact on the PACC reliability is considered by evaluating the PACC model using a one-week inspection and testing interval. The impact of maintenance on the PACC is considered in terms of repair times used to estimate PACC unavailability.

In the FMEA, monitoring frequency, testing and maintenance are being considered (and in most cases eliminated) as possible failure causes.

- Frequency of Demand

The impact of frequency of demand on PACC reliability is being considered through the use of Shutdown Heat Removal System (SHRS) shutdown initiators. This is, the PACC failure state model is evaluated using shutdown initiators which place a demand on the PACC.

- Analysis of Operation

In quantitative reliability analysis, nominal operating conditions are being considered. The loss of power to the fans is included in the PACC evaluations. Items: (a) Possibilities of overcooling, (b) varying steam supply, and (c) changing steam conditions, are not appropriate for the failure state model evaluation.

Question CS760.122

What data base was used to determine the design characteristics of the PACC and its operational abilities?

Response

The data base being used for the design of the PACC is described in the response to NRC Question 760.035.

Question CS760.127

The cooldown limit of 150°F/hr is quite high compared to LWR limits. What is the limiting component at this cooldown rate? Are the associated thermal stresses based on perfect mixing in the upper plenum?

Response

No component has been identified as being limiting for the normal cooldown from 40% power. In this regime near perfect mixing in the outlet plenum exists and the thermal stresses are calculated on this basis.

Question CS760.129

Does the estimated frequency of reactor scram (around 10 per year) take into account spurious trips introduced by the severely reduced secondary trip settings?

Response

The following definition of a reactor trip from the PSAR* is provided to define the items included for determining the frequency of reactor scrams:

U-1 Reactor Trip

This transient includes real scrams due to malfunctions (including rapid reactivity transients) which cause a PPS trip level to be exceeded, and spurious scrams covering those situations in which a PPS trip level is not actually exceeded but a scram occurs due to a fault in the PPS, control system, or plant instrumentation.

The effect on plant availability is considered when determining the setpoints for the Primary and Secondary RSS trips. The settings for the secondary trip set points will not lead to excessive spurious trips.

*PSAR Appendix B, Section B.1.2.1

Question CS760.130

What is the basis for the 3% per minute rate of change of power limit during load follow operations? Can you provide data that demonstrate that fuel cladding mechanical interaction will not lead to excessive cladding failure during load follow? Have control system interactions been considered in determining time delays to scram? Please provide a discussion with the details of the analysis.

Response

The CRBRP plant is designed to provide the capability to load follow. The reactor vessel, piping and other systems are being designed to accommodate thermal stresses and heat up rates consistent with a maximum 3% per minute power rate change. However, the applicant has no plans to operate CRBRP in the load follow mode during the first 328 full power days (first core fuel load) because the capability to sustain cyclic rapid power increases in the fuel has not been demonstrated. The 3% per minute is also an upper limit and slower ramp rates (slower heat up rates) can be accommodated by the system. The automatic control system precludes reactivity ramps exceeding ±3% of full power during steady state operation.

Later core loads of CRBRP may be cycled in a load follow type of operational mode. Initial tests to demonstrate this capability have been performed in EBR-II (run 112 with 1%/sec power change). Additional power cycle tests are planned as part of the operational reliability transient testing program in EBR-II. At this time, only calculations with the LIFE computer code can be performed to indicate the adequacy of the fuel to sustain load follow operation. However, load cycle tests in EBR-II and load follow tests in foreign reactors (see References, QCS760.130-1 and 2) indicate no obvious rod damage due to reactor load follow operation.

Load follow power rate changes can be firmly addressed after the irradiation testing has been performed. The rate of power change increases may depend on the power swing and the duration of operation at the lower power.

Power increase rates after extended low power operations will be determined based on the planned Operational Reliability Testing (ORT) program transient tests in EBR-II and FFTF experience. The steady state and transient program

plans will be provided to NRC via a summary description document before the end of FY82 which will include the ORT program transient tests.

Control System interactions have been considered in the Chapter 15 Safety Analysis. For fast transients, PPS action occurs prior to Control System response. For slow transients, Control System response tends to mitigate the event consequences. In general, a conservative accident scenario is postulated in PSAR Chapter 15 analyses by assuming no Control System action.

During power operation, the time response of the Reactor Control System is approximately 20 seconds; the time responses of the Sodium Flow Control Systems are also approximately 20 seconds. For most transients, the time it takes the PPS to recognize a scram condition and initiate a scram is on the order of 0-5 seconds; thus, the control system interactions do not cause appreciable delays in the time to scram. Additionally, control system interactions generally act to decrease reactor power or increase sodium flow during transients.

References

QCS760.130-1 R. Lallement, "French Experience Concerning the Reactor Behavior of Breeder Fuel Elements", 1981 ANS/ENS Meeting on Reactor Safety Aspects of Fuel Behavior.

QCS760.130-2 T. Rousseau, et. al., "Fast Neutron Reactor Fuel Elements and Power Grid Duty Cycling", International Conference on Fast Breeder Reactor Fuel Performance, Monterey, California, 1979.

Question CS760.132

No discussion of pressurization transients is given in 5.7.3.

Please indicate the margin in the design for overpressurization resulting from either postulated accidents in Section 5 or steam generator tube failures, whichever is applicable.

Response

The Sodium-Water Reaction (SWR) Design Basis Leak (DBL) as discussed in Section 5.5.3.6 is the transient which imposes highest pressures in the Intermediate Heat Transport System (IHTS). The Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS) as described in Sections 5.5.2.4 and 5.5.2.6 is an overpressure protection system designed to ASME Code Section III, Division I, Class 3. In the event of a large SWR, the SWRPRS functions automatically to limit IHTS pressures to below emergency (Level C) limits.* The SWRPRS rupture disks are nominally rated at 325 PSID so that sustained overpressure is limited to about 325 PSIG. At major components, transient pressures as high as 395 PSIA (Table 5.5-11) are associated with the SWR DBL. The portion of the IHTS that will be exposed to the above described event is designed to ASME Code, Section III, Division I, Class I. The margine relative to design limits will be determined upon completion of component Final Stress Reports.

*The sodium/water reaction event is classified as faulted for the affected steam generator unit, for the interconnecting piping to the other steam generator units in the loop, for the injected reaction products separator tank(s), affected rupture discs and interconnecting pressure relief piping. For the rest of the plant, the event is classified as an Emergency Event.

Question CS760.135

The saturated steam line rupture is identified as the most severe event in terms of the temperature transient seen by the evaporator and IHX but there does not appear to be any evidence that the designs can accommodate such a severe thermal shock.

Please provide evidence that these components can accept a 500°F change over a 10 minute time span. Please provide a list of critical components and the acceptable thermal transients associated with each of them.

Response

The IHX design specification identifies temperature and flow transients that the IHX must withstand. The saturated steam line rupture event is identified as one of the events that the IHX shall be designed for. The thermal and stress analysis of the IHX has been completed and the vendor has certified that the IHX can accommodate all specified thermal transients including the saturated steam line rupture event.

Question CS760.139

Please indicate the functional requirements for the Battery Backup System for the following areas:

- o Total power requirements
  - valves
  - turbine control
  - other
- o design life
- o maximum length of time under battery operation

Has any reliability study been done for this system?

Response

The CRBRP DC power system, which consists of Class 1E and non-Class 1E power supplies, is described in the PSAR Section 8.3.2. The batteries are sized in accordance with IEEE Standard 485-1978 and include temperature correction and aging factors. Each battery is sized for its maximum expected duty cycle including a design margin for load growth. The battery chargers are sized to provide DC power to all continuous loads and also to charge the batteries from a totally discharged state to full charge within twelve (12) hours.

A list of Class 1E (Division 1, 2 and 3) DC loads is provided in Tables 8.3-2A, 8.3-2B and 8.3-2C of the PSAR. These tables include the power requirement and duration of operation for each load. The Class 1E loads which require uninterruptible AC power supply are powered by the corresponding DC battery through an inverter (DC to AC).

The DC system design for CRBRP is presently under development. The complete list of Class 1E and non-Class 1E loads (DC and AC), which are supported by the DC batteries upon loss of the Plant AC power, will be included in the FSAR Chapter 8. As a minimum this list will provide load description, power requirement and maximum duration of operation for each load, as well as total power requirements for each DC bus.

Ratings of the Class 1E and non-Class 1E DC batteries and DC to AC inverters are shown in the PSAR Chapter 8, Figure 8.3-2.

The qualification life of the station batteries will be in excess of 15 years. It is expected that each battery will be replaced once during the 30 year life of the plant.

Normally, the batteries will be on float with the continuous DC load supplied by the battery chargers. On loss of AC power, the batteries will supply power to the connected DC loads for a period of at least two (2) hours without recharging, except for the security battery which will require recharging after 15 minutes.

A reliability study for the DC system will be performed and the results will be included in the FSAR.

Question CS760.140

Some portions of the hot leg piping require further inelastic calculations (WARD-D-0185, Pg. 4.1-18). Have these calculations been performed? Please provide the results of these calculations.

Response

Inelastic analyses of the CRBRP primary heat transport system hot leg between the pump and intermediate heat exchanger have been performed. These results confirm the integrity of the piping that was forecast based upon elastic analysis results. The results of this analysis is given in Reference QCS760.140-1.

References

QCS760.140-1 A.K. Dhalla, "A Procedure to Evaluate Structural Adequacy of A Piping System in Creep Range", ASME Publication PVP-63, American Society of Mechanical Engineers, New York, 1982.