CLINCH RIVER 805

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 19

PROJECT MANAGEMENT CORPORATION

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Question 011.1 (10.4.8)

Provide a basic flow diagram for the Steam Generator Blowdown System.

Response:

Refer to revised section 10.4.8 Steam Generator Blowdown System.

Question 011.2 (11.2.3)

You propose to design the liquid radwaste system in accordance with Quality Group D classification. We do not consider this classificaiton adequate because the design guidance should provide reasonable assurance that equipment and components used in the radioactive waste management system are designed, constructed, installed, and tested on a level commensurate with other plant systems and structures to protect the health and safety of the public and plant operating personnel. You should design the systems handling liquid waste, including components in the solid waste system which contain radioactive liquids, to Quality Group D (augmented) classification as described in the attached Branch Technical Position - ETSB No. 11-1, "Design Guidance for Radioactive Waste Management Systems Installed In Light-Water Cooled Nuclear Power Reactor Plants".

Response:

The response to this question is provided in revised PSAR Section 11.2.3.

Since this question was transmitted originally (June 5, 1975), Branch Technical Position ETSB11-1 was updated deleting the classification of Quality Group D (augmented). This deletion is reflected in the updated response. 25

Question 011.3 (11.2.3):

Provide a table listing indoor tanks, except those tanks located in the reactor containment, which contain potentially radioactive materials. For each tank, indicate the provisions incorporated to monitor tank liquid levels, to annunciate potential overflow conditions, and to collect and process liquids in the event of an overflow. Acceptable provisions include dikes around tanks, retention basins, and elevated thresholds to contain liquids in bays containing the tanks.

Response:

The requested information is provided in Table 11.2-5A. 25

Question 011.4 (11.3.3.3)

In Table 11.3-17, you list the operating pressures and temperature of the RAPS and CAPS process vessels. Provide a listing of the design pressure and temperature of each piece of equipment.

Response:

The design pressures and temperatures of the RAPS and CAPS process vessels are provided in revised table 11.3-17.

Question 011.5 (11.3.4)

If releases to the environment are required from RAPS as an alternate operating procedure, indicate the release path and provide a description of the procedure.

Response:

No releases to the environment are required from RAPS as discussed in revised Section 11.3.4.

Question 011.6 (11.5.3)

Provide the seismic and quality group classifications of structures, piping, and equipment for the solid waste system.

Rsponse:

The requested information is provided in revised PSAR Section 11.5.3. 25

Question 011.7 (11.5.3)

Describe the method of processing or handling of sodium-bearing solids from the primary, intermediate, and ex-vessel storage tank cold traps.

Response:

Response is provided in revised PSAR Section 11.5.3.

Question 011.8 (11.5.5 & 11.5.6)

Provide the capacity available for storage of solid wastes and estimate the expected onsite storage period and the decay realized by such storage.

Response:

The response to this question is provided in PSAR Sections 11.5.5 and 11.5.6. 25

Question 011.9 (6.3.1.2 & 9.6.1)

In Subsections 6.3.1.2 and 9.6.1, you describe the control room habitability system. This should be an ESF filter system. Provide an analysis of this system to show it is designed to mitigate the consequences of a DBA with respect to each position in Regulatory Guide 1.52. See North Coast PSAR, Docket No. 50-376, Vol. VIII, Table 9.4-4 for an acceptable format.

Response:

The response to this question is incorporated in new Table 6.3-1, "Conformance of the Control Room Filtration System With Respect to Each Position of USNRC Regulatory Guide 1.52".

Question 011.10 (11.3.3.3)

In Table 11.3-17, you list the operating pressures and temperatures of the RAPS and CAPS process vessels. Provide a listing of the design pressure and temperature for each piece of equipment.

Response:

The information requested is provided in the responses to questions 011.4 and 020.7.

Question 011.11 (6.3.5)

In Table 6.3-1, you present your analysis of the control room habitability system with respect to each position of Regulatory Guide 1.52. Your response to Position 4.d of Regulatory Guide 1.52 is inadequate. You should provide a minimum of three linear feet from mounting frame to mounting frame between banks of components for ease of system maintenance. If components are to be replaced, the dimension to be provided should be the maximum length of the component plus a minimum of three feet.

Response:

The reponse to this question is incorporated in revised Table 6.3-1, "Conformance of the Control Room Filtration System With Respect to Each Position of USNRC Regulatory Guide 1.52," Regulatory Position 4d.

Question 011.12 (10.4.2)

Indicate on the piping and instrumentation diagram for the condenser air removal system (Figure 10.4-1) where the water from the mechanical vacuum pump reservoir is directed. This water should be classified as a radioactive liquid and handled accordingly.

Response:

This question was answered in response to Q011.21 in PSAR Amendment 28. 32

Question 011.13 (10.4.3)

Provide a piping and instrumentation diagram (P&ID) of the turbine gland sealing system.

Response:

A detailed description of the Turbine Gland Sealing System is provided in Section 10.4.3 of the PSAR. The basic flow diagram is shown in Figure 10.4.7. 39

Question 011.14 (11.3.2.1)

In Subsection 11.3.2.1, you discuss the procedure for periodic bottling of Ar-39 and Kr-85 from the RAPS cryogenic still. Discuss the procedure in greater detail. Provide bottle storage pressure. Discuss procedures and the means for monitoring leakage of gas from the storage bottles. Provide the anticipated onsite storage time. Describe the shipping container to be used for transport of the storage bottles to a licensed burial site. Discuss the acceptability of bottled radioactive gases at the licensed burial sites.

Justify your conclusion that bottling, shipping and ultimate storage of the long-lived gaseous radioisotopes (Kr-85 and Ar-39) represents a lower risk to public health and safety than releasing these isotopes under controlled and favorable conditions to the environment. Include your consideration of keeping occupational exposures as low as practicable.

Response:

The cryogenic still bottoms consist of liquid argon in which krypton and xenon isotopes which pass through the cryogenic charcoal beds dissolve. The major radioactive species in this solution are Kr-85, Ar-39, and Ar-41. In addition there will be an accumulation of stable kryptons and xenons in the still bottoms. Kr-85 and Ar-39 are relatively long-lived and their concentrations will continue to increase with time during the operation of the reactor. The concentration of Ar-41 will approach a steady state value as a consequence of its shorter half-life.

The cryostill serves to collect and remove krypton and xenon isotopes - stable as well as radioactive - in the recycle gas stream. This concentrates these isotopes. To provide the capability to minimize the radioactivity release from the plant, the bottling station has been shown in the conceptual system. (Section 11.3.)

During steady state operation with 1% failed fuel, the cryostill bottoms will accumulate 2.0 curies/day of Kr-85 and about 0.01 curies/day of Ar-39. It has been determined that if the still were operated for 10 years, the one cubic foot of liquid argon would contain 5,377 curies of Kr-85, 26 curies of Ar-39 and 0.05 curies of Ar-41. After this mixture is transferred from the cryostill to the noble gas storage vessel, the Ar-41 will rapidly decay (110 minute half-life), so that only Ar-39 and Kr-85 need be considered regarding storage and transportation. This will fill 52 1.5 cu. ft. laboratory-sized gas bottles at 150 psia, a total of 104 curies in each bottle.

This gas, if released, would represent small site-boundary dose effect. If released at the accumulation rate noted above, and under average meteorological conditions, the additional site boundary dose rate would be less than 1 rem/year. (See CRBRP Draft Environmental Statement, NuReg 0024, Section 3.5.2.6.)

Based on the above, either alternative is acceptable. Thus detailed bottling procedures have not been developed.

The Project is currently assessing the benefits of each alternative from an ALARA standpoint. Any change from the present concept as described in PSAR Section 11.3 resulting from the Project's assessments will be included in a future amendment to the PSAR.

Question 011.15 (11.4.2.2)

Justify why you have not provided a monitor on the cell atmosphere processing system (CAPS) discharge line which would initiate automatic termination of the effluent release when radionuclide concentrations exceed a predetermined level.

Response:

Figure 11.3-13, "P&I Diagram, Cell Atmosphere Processing System," shows the part of the CAPS system in which the effluent from the cold box (Figure 11.3-12) is seen to pass through redundant radiation monitors (RISH) with a high-alarm indicator. When the signal is below the set-point, the 3-way valve RY operates the control valve RV to permit the gas flow to proceed to H&V discharge. Upon a "high" signal, RY operates RV to divert the gas stream into the line which returns to the inlet pipe of the Vacuum Vessel.

This system does divert the effluent stream back to the CAPS vacuum tank when that stream's radioactivity exceeds the predetermined set-point level. Further discussion of this procedure is provided in Section 11.3.4.

Question 011.16 (11.5.3)

Although there are a number of processes available which are capable of solidifying liquid wastes under controlled conditions, there is a potential for free[1] liquids to remain in containers following solidification with the widely varying chemical species encountered during power plant operations. Applications should implement measures to reasonably assure complete solidification of liquid wastes.

Two methods which may assure complete solidification of liquid wastes are:

(1) Process Control Program

(a) Solidification agents and potential waste constituents should be tested and a set of process parameters established which provide boundary conditions within which reasonable assurance can be given that solidification will be complete.

(b) The plant operator should provide assurance that the process is run within the parameters established under (a) above. Appropriate system controls and records should be maintained for individual batches showing conformance with the established parameters.

(2) Means to detect the presence of free liquids in solid waste containers.

You should commit to:

(1) Establish process parameters within which solidification systems must be operated to reasonably assure complete solidification of liquids and provide assurance that the systems are operated within these process parameters, or

(2) Have provisions to verify the absence of free liquid within containers prior to shipment offsite.

Response:

Process parameters will be selected to assure complete solidification of liquid wastes Both a process control program and an administrative control program will be used.

The process control program uses pretested formulas for required portions of waste streams and portland cement. The formulations would establish the required mix compositions as well as the amount of any necessary additives to assure that the mix will harden into a solid, immobile, free-standing monolith. Formulations would also include excess cement to make certain that there will be no free water in the crack between the drum wall and the cement block resulting from shrinkage as the cement sets up.

The administrative controls involve back-up procedures to assure that the pretested formulas will be followed by operating personnel. Suitable records will permit verification of compliance.

[1]Free water is defined as uncombined water not bound in the solid matrix.

Question 011.17 (11.5.3)

In either case, you should provide data which will justify the method finally used and will provide reasonable assurance of complete solidification of liquid wastes encountered in your plant.

Response:

While detailed data on the solidification system is not yet available since a vendor has not been selected, programs to assure complete solidification of liquid wastes are described in the response to Question 011.16. The data requested will be provided after the selection of the vendor for the solidification system.

Question 011.18 (11.5.3)

In subsection 11.5.3, you describe several procedures for handling and disposing of radioactive metallic sodium. Justify the acceptability of disposing of radioactive metallic sodium by storing in 55-gallon drums for subsequent offsite transfer to a licensed contractor for processing.

Response:

Radioactive sodium is placed in sealed 55-gallon drums in solid form and is transferred to a shielded vault, where it is stored to allow radioactive decay of Na24 before processing. Generally, sodium is not exposed to air or water since it is in a sealed drum. Even when exposed to air, solidified sodium will not ignite and no water connections are present in the storage vault. Thus the possibility of a sodium fire in the shielded storage vault is highly unlikely. Adequate shielding is provided by the storage vault to minimize radiation exposure to plant operating personnel. Consequently, no hazard to the general public is presented by usage of this method of storing sodium waste.

Processing of the sodium either in the plant or by a licensed contractor will be determined at a later date. When this processing method has been finally determined, justification for its acceptability will be provided.

Question 011.18 (11.5.3)

In subsection 11.5.3, you describe several procedures for handling and disposing of radioactive metallic sodium. Justify the acceptability of disposing of radioactive metallic sodium by storing in 55-gallon drums for subsequent offsite transfer to a licensed contractor for processing.

Response:

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Processing of the sodium either in the plant or by a licensed contractor will be determined at a later date. When this processing method has been finally determined, justification for its acceptability will be provided in the FSAR. 40

Question 011.20

To enable a comparison of the CRBRP application with the numerical standards of 10CFR50, Appendix I, additional information is required. For each building housing systems containing radioactive materials:

a. Provide a description of the provisions incorporated to reduce radioactive releases (iodine and particulates) from ventilation exhaust systems.

b. Provide the release point description, including height above grade, height above and relative location to adjacent structures, relative temperature difference between gaseous effluent and ambient, flow rate, velocity and size and shape of the flow orifice.

c. For the containment building indicate the expected purge and venting frequencies and duration, and the continuous purge rate (if used).

Response:

(a) As described in Section 11.3, one of the design objectives of the gaseous radwaste processing system is a design that will result in gaseous effluents in quantities that are as low as reasonably achievable. Technical specifications provided to implement the ALARA objective are listed in PSAR Sections 16.3 and 16.4.

The design base and the expected values of the annual activity release for each gaseous radinuclide are listed in Tables 11.3-11 and 11.3-12 of the PSAR. Neither table includes radioiodines or particulates.

As stated in revised PSAR Section 11.3.2.1, particulates and elemental iodine are not expected to enter the cover gas. This statement is substantiated by published results in releases from sodium pools (Ref. Q011.20-1) and in-pile experiments (Ref. Q011.20-2). All cover gas is processed through vapor traps in the Radioactive Argon Processing Subsystem (RAPS) which are expected to remove essentially all non-gaseous isotopes including any trace quantities of sodium iodide. After subsequent decontamination by RAPS, the cover gas is recycled to the seals and cover gas spaces. The Cell Atmosphere Processing Subsystem (CAPS) significantly reduces any radioactivity levels in plant effluents. Discussion of the decontamination capabilities and functions of RAPS and CAPS are provided in PSAR Section 11.3.

There is no expected annual activity release of radioiodines and particulates during normal operations. However, any significant leakages of such radioactive species would be detected as follows:

1. Leakage to the RSB and RCB cells would be detected by CAPS process monitors and/or by the CAPS and RSB Radwaste Area exhaust monitors;

2. Diffusion through the reactor head and buffer seals would be detected by the RCB ventilation exhaust monitors and/or head-access-area monitors.

PSAR Section 11.4 provides a discussion of effluent monitoring.

As discussed in revised Section 11.3.1, CRBRP design objectives include conformance with the requirements of 10CFR20 including ALARA releases. Sections 16.3.11.3 and 16.4.4 discuss the technical specifications on airborne release and monitoring, respectively which provide assurance that the ALARA objective is achieved.

(b) A new Table 11.3-20 has been incorporated which provides the release point elevation, flow rate, velocity and size and shape of the discharge orifice for the effluent release points. Figure 11.3-9 has been revised to add roof elevations to the plan showing nuclear island and balance of plant building effluent discharges which will indicate the height above and relative location of the discharges to adjacent structures. The relative temperature difference between gaseous effluent and ambient is dependent upon the seasonal temperature variations and different plant operating modes. Gaseous effluent temperature ranges for the effluent release points are provided in Table 11.3-20. Monthly Historical Temperature Data for the CRBRP area is provided in Table 2.6-4 of the Environmental Report.

For additional information on the effluent release points, see Sections 11.3 and 11.4 of the PSAR.

(c) During normal plant operation, a 14,000 CFM outside air system provides conditioned fresh air to the normal atmospheric areas of the Reactor Containment Building, as described in Section 9.6.2 of the PSAR, entitled "Reactor Containment Building".

<u>References</u>:

Q011.20-1 R.S. Hart and C.T. Nelson, "Introduction of Cesium, Strontium and Iodine into Sodium", in W.P. Kunkèl, "Fission Products Retention in Sodium - A Summary of Analytical and Experimental Studies of Atomics International", NAA-SR-11766, 1966, pp 11-13.

Q011.20-2 W. Kunkel, D. Elliot and A. Gibson, "High Temperature Sodium Studies in KEWB", in W.P. Kunkel, "Fission Product Retention in Sodium - A Summary of Analytical and Experimental Studies at Atomics International", NAA-SR-11766, 1966, pp 45-46.

Question 011.21 (10.4.2)

Indicate on the piping and instrumentation diagram (P&ID) for the condenser air removal system (Figure 10.4-1) where the water from the mechanical vacuum pump reservoir is directed. This water should be classified as a radioactive liquid and handled accordingly.

Response:

Sections 10.4.2.1, 10.4.2.2 and 11.2.6.2 have been amended to indicate that the mechanical vacuum pump reservoir drains will be discharged to the Clinch River in the same manner as other steam cycle related discharges. Concentrations of tritium released from the steam-water cycle to the river through the blowdown system will be as indicated in Section 10.4.7.3. Figure 10.4-1 (flow diagram for the Condenser Air Extraction
44 System) has been revised to show disposition of water from the mechanical vacuum pump reservoir.

Question 011.22 (10.4.3)

Provide a P&ID of the preliminary design of the turbine gland sealing system.

Response:

A detailed description of the Turbine Gland Sealing System is provided in Section 10.4.3. of the PSAR. The basic flow diagram is shown in Figure 10, 4-7. 39

Question 011.23 (11.3.2.1)

In Subsection 11.3.2.1, you discuss the procedure for periodic bottling of Ar-39 and Kr-85 from the RAPS cryogenic still. Discuss the procedure in greater detail; provide bottle storage pressure; discuss procedures and the means for monitoring leakage of gas from the storage bottles; provide the anticipated onsite storage time; describe the shipping container to be used for transport of the storage bottles to a licensed burial site; and discuss the acceptability of bottled radioactive cases at the licensed burial sites.

Justify your conclusion that bottling, shipping and ultimate storage of the long-lived gaseous radioisotopes (Kr-85 and Ar-39) represents a lower risk to public health and safety than releasing these isotopes under controlled and favorable conditions to the environment. Include your consideration of keeping occupational exposures as low as practicable.

Response:

The procedure for disposing of the RAPS cryostill bottoms is discussed in revised PSAR Sections 11.3.2.1 and 11.3.4. The procedure involves controlled gradual release of the noble gases through CAPS during normal operation.

Considerations of keeping occupational exposures as low as reasonably achievable supported the change to the method for disposal of the RAPS cryostill bottoms as described in PSAR Sections 11.3.2.1 and 11.3.4

Question 011.24

To enable a comparison of the CRBRP application with the numerical design objectives of 10 CFR 50, Appendix I, additional information is required. For each building housing systems containing radioactive materials:

(1) Provide a description of the provisions incorporated to reduce radioactive releases (iodine and particulates) from ventilation exhaust systems.

(2) Provide the release point description, including height above grade, height above and relative location to adjacent structures, relative temperature difference between gaseous effluent and ambient, flow rate, velocity, and size and shape of the vent outlet.

(3) For the containment building indicate the expected purge and venting frequencies and duration, and the continuous purge rate (if used).

Response:

27 The requested information is supplied in response to Question 011. 20.

Question 011.25

In the CRBRP Third Level Thermal Margin (TLTM) Report and PSAR Section 9.6.2.4, a conceptual design for a reactor containment building (RCB) cleanup system is provided. Sufficient information in the TLTM report or PSAR has not been provided to permit a detailed review. Provide the following information in the PSAR.

(a) Provide a detailed piping and instrumentation diagram (P&ID) of the RCB cleanup system.

(b) Provide the RCB cleanup system design parameters, e.g., flow rate, temperature, pressure, and materials of construction. As appropriate, demonstrate component material compatibility with concentrated sodium hydroxide solutions.

(c) Provide information of the efficiency of the sodium scrubber as a function of temperature and pH. Indicate any research and development or testing programs which are ongoing or planned to provide the necessary documentation of scrubber efficiency for the expected operating conditions. Justify that a 90% efficient scrubber is adequate to prevent severe plugging or fouling of the downstream HEPA filter. Provide the maximum loading (pounds of sodium) that the closed cycle scrubbing system can tolerate and still function efficiently.

(d) In view of the fact that hydrogen will be a reaction product of the sodium scrubber, justify the lack of hydrogen gas instrumentation in the RCB cleanup system to prevent the buildup of explosive mixtures.

(e) For the RCB cleanup system, provide in tabular form a comparison between the features of the proposed system and each position in Regulatory Guide 1.52. For each design item, discuss any exceptions to Regulatory Guide 1.52.

Response:

(a) Detailed design of the Containment Cleanup System is presently in progress. A Piping and Instrumentation Diagram (P&ID) will be included in CRBRP-3. Volume 2 (Reference 10b, PSAR Section 1.6) as soon as the design details become available.

(b) The Containment Cleanup System design parameters are included in Sections 2.1.2.8, 2.2.9, and A.7 of the CRBRP-3, Volume 2.

(c) The Containment Cleanup System design has been changed to replace the HEPA-Charcoal-HEPA filters with a high efficiency wetted fiber bed sodium scrubber. See Section 2.2.9 of the CRBRP-3, Volume 2 for a discussion of the scrubber system.

The test program to demonstrate the performance of the TMBDB Air Cleaning System is discussed in Appendix A.7 of the CRBRP-3, Volume 2.

(d) See Section 2.2.9 of the CRBRP-3, Volume 2.

(e) Regulatory Guide 1.52 was developed for an atmosphere cleanup system consisting of some or all of the following components: demisters, heaters, pre-filters, high efficiency particulate air (HEPA) filters, adsorption units, fans and associated ductwork, valving, and instrumentation. The TMBDB Containment Cleanup System does not include any of the filter units above, but instead uses a wet scrubber system consisting of an air washer, venturi scrubber, and high efficiency fiber bed scrubber. Therefore, many of the design criteria of Regulatory Guide 1.52 are not applicable. Table Q011.25-1 lists the applicable Regulatory positions of 1.52.

Comparison of CRBRP Containment Cleanup System to Regulatory Guide 1.52, Rev. 2

Table Q011.25-1

(Sheet 1)

Regulatory Position	Applicable to CRBRP Containment Cleanup System: Yes	No	Remarks
1.a	X		
1.b		X	CRBRP-3, Vol. 2 Report
1.c		X	System does not contain adsorber
1.d		X	CRBRP has no Containment Spray System
1.e	X		
2.a		X	Redundancy is provided for active components only
2.b	X		Redundant components will be physically separated
2.c	X		
2.d	X		System will be designed to withstand maximum expected pressure
2.e	X		
2.f		X	System does not contain HEPA Filters
2.g	X		
2.h	X		System manually activated, see Sect. 2.3 of CRBRP-3 Volume 2
2.i	X		
2.j	X		
2.k		X	No outside air intakes are required
2.l	X		
3.a		X	System does not contain a demister
3.b		X	System does not contain a demister
3.c		X	System does not contain a demister

Revised Response

Table Q011.25-1

(Sheet 2)

Regulatory Position	Applicable to CRBRP Containment Cleanup System		Remarks
	Yes	No	
3.d		X	System does not contain HEPA Filters
3.e		X	System does not contain adsorber
3.f		X	System does not contain adsorber or HEPA filters
3.g		X	System is all welded leaktight
3.h		X	System does not contain water drains
3.i		X	System does not contain adsorber
3.j		X	System does not contain adsorber
3.k		X	System does not contain adsorber
3.l	X		
3.m	X		
3.n	X		
3.o	X		
3.p	X		
4.a	X		
4.b		X	System does not have filter banks
4.c		X	System does not have adsorber and HEPA filters
4.d		X	System does not have heaters
4.e		X	System does not have adsorber or HEPA filters
5.a		X	No DOP or activated carbon test are required
5.b		X	System does not have HEPA filter or iodine adsorber
5.c		X	System does not have HEPA filters
5.d		X	System does not have activated carbon adsorber

Revised Response

Table Q011.25-1

(Sheet 3)

Regulatory Position	Applicable to CRBRP Containment Cleanup System		Remarks
	Yes	No	
6.a		X	System does not have activated carbon adsorber
6.b		X	System does not have activated carbon adsorber

Amendment 44

There are no new NRC Questions in Amendment 44.

Question 020.1 (3.4.1)

The PSAR indicates that the lowest floor of several structures will be below the plant grade. Discuss how the safety related equipment located in all areas below grade will be protected from the effects of the maximum expected groundwater level or flooding caused by natural phenomena. For each item discussed, provide the following additional information:

(1) A list of all entrances, including their elevations, and means to be provided to prevent ingress of water.

(2) A list of other below-grade penetrations, including those for pipes; conduits and floor drains, the means to be provided to seal these penetrations, and provisions for periodic examination of these seals. The list should include the elevation of these penetrations.

Response:

The response to this question is contained in Section 3.4.1. 25

Question 020.2 (3.5.1)

Provide a tabulation of all safety related components that will be located outdoors and describe the protection provided for these components against tornado generated missiles. Include in this tabulation all ventilation system air intakes and exhausts, the diesel generator combustion air intake and exhaust, and all vents for safety related tanks.

Response:

The response to this question is contained in Section 3.5.1. 25

Question 020.3 (3.5)

Include in Section 3.5 a description of protection that will be afforded for all safety related components and equipment outside containment from internally generated missiles. List the missiles considered and their associated kinetic energy.

Response:

The general methods of protection that will be afforded for all safety related components and equipment outside containment from internally generated missiles are described in Section 3.5 under the Design Bases 3, 4 and 6.

Since all rotating parts are fully enclosed in metal casings which are designed to contain any potential missiles, there are no known internally generated missiles identified from the failure of rotating components except the turbine failure missiles which are described in Sections 3.5.2.1.1 and 10.2.3. Detailed discussions of rotating component failure missiles are given in Section 3.5.2.1 for each individual system.

Pressurized component failure missiles are discussed in Section 3.5.2.2 The identified missiles and their characteristics for SGAHRS and SGS are listed in Table 3.5-2 and Table 3.5-3 respectively.

Because of the relatively low pressures in the intermediate heat transport system and components, the energy state of the contained fluid is correspondingly low, and therefore, no potential sources of high-energy missiles have been identified.

Question 020.4(3.6.5)

Provide protection of essential systems and components against postulated failures in high or moderate energy fluid systems in accordance with the requirements of enclosure 1, Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

Provide a complete tabulation of all high and moderate energy piping systems as defined in BTP - APCSB 3-1 together with the essential systems necessary to shut the reactor down and to mitigate the consequences of a postulated pipe break located outside the containment. Revise your application as necessary and indicate your intent to comply with this position.

Response:

The information requested is provided in revised PSAR Section 3.6.

Question 020.5 (3.6.5)

Provide preliminary layout drawings of the safety related areas outside containment showing the major systems and the protection afforded to safety-related equipment as recommended in BTP - APCSB 3-1. Emphasis should be placed in using the separation principal to the maximum extent practical.

Response:

The requested drawings are not yet available because system routing design is not complete. However, the effects of pipe rupture are being considered in the design process using separation of rupture sources and safety related equipment as the principal means of protection from piping rupture.

The attached table lists by building all safety related equipment and the piping systems which threaten their function in the event of postulated piping failures during normal plant operation. The nature of the hazard, the method which will be used for protection, and the scope of the analysis of postulated failures that will be required to comply with BTP-APCSB.3-1 is also listed in the table. Only buildings containing safety related equipment are included in the table.

It is anticipated that as the plant design progresses, the table will expand to include more specific detail until upon completion of pipe rupture analysis, safety related equipment subject to damage by pipe rupture, will be identified by equipment numbers for safety related systems and sources will be designated by piping line numbers. Also at that time, the requested drawings will be prepared. The completed table and drawings will be included at a later design stage.

TABLE Q 020.5-1

SUMMARY OF EX-CONTAINMENT PIPE RUPTURE ANALYSIS

SAFETY RELATED EQUIPMENT TO BE PROTECTED	LOCATION	HAZARD	SOURCE OF HAZARD	METHOD OF PROTECTION	SCOPE OF ANALYSIS REMAINING
IHTS Piping	Steam Generator Building (SGB)	Pipe whip Jet impingement	Main steam line Saturated steam line Recirculated feed line	Separation is the principal means of protection. The IHTS is divided into three loops which are separated by concrete walls. Any one loop may be lost without loss of the decay heat removal function although in two loop operation a loss of redundancy may occur. In order to prevent a major sodium water reaction and limit damage propagation within the cells sufficient pipe restraints will be installed to prevent a whipping steam or feedwater pipe from causing cracks on ruptures in sodium piping.	Complete pipe whip analysis to determine the effect of whipping pipe upon sodium piping within the cell.
Intermediate Na pump casings	SGB Cells	Pipe whip Jet impingement	Main steam line Saturated steam line Recircluated feed line	Same as IHTS piping.	Same as IHTS piping.
Evaporators and Superheaters	SGB Cells	Pipe whip Jet impingement	Main steam line Saturated steam line Recirculated feed line	Same as IHTS piping.	Same as IHTS piping.
Steam Drums	SGB Cells 241, 242, 243	Pipe whip Jet Impingement	Main steam line Saturated steam line Recirculated feed line Main feed line Blowdown line Pressurized portion of SGAHRS	Separation is the principal means of protection. The steam generator system is divided into three loops which are separated by concrete walls. Any loop may be lost without a loss of the decay heat removal function.	None required.
Steam Piping Between Evaporators, Steam Drum and PACC	SGB	Pipe whip Jet Impingement	Main steam line Saturated steam line Recirculated feed line Main feed line Blowdown line Pressurized portion of SGAHRS	Same as steam drums.	None required.
Steam Drum recirculation system	SGB	Pipe whip Jet Impingement	Main steam line Saturated steam line Recirculated feed line Main feed line Blowdown line Pressurized portion of SGAHRS	Same as steam drums.	None required.

TABLE Q 020.5-1 (Cont'd.)

SAFETY RELATED EQUIPMENT TO BE PROTECTED	LOCATION	HAZARD	SOURCE OF HAZARD	METHOD OF PROTECTION	SCOPE OF ANALYSIS REMAINING
Sodium Water Reaction Products and Relief System	SGB	Flooding	Emergency Chilled water Protected Water Storage tank piping	Building drains/sumps are sized to prevent interruption of safety related function by flooding.	Determination of the maximum flood level that can occur in each cell containing safety related equipment assuming the failure of one active component which would mitigate the effects of a leakage crack. Verification that all safety related equipment is above this level.
Emergency Plant Service Water System	SGB	Water spray	Normal and Emergency Chilled water Normal Plant Service Water; Fire Protection	Routed remotely from high energy piping. Pumps are installed redundantly and separated by a concrete wall.	Determination of the maximum flood level that can occur in each cell containing safety related equipment assuming the failure of one active component which would mitigate the effects of a leakage crack. Verification that all safety related equipment is above this level.
Emergency Plant Chilled Water System	SGB	Flooding Water spray	Emergency Plant Service Water Normal Chilled Water; Fire Protection	Routed remotely from high energy piping. Chillers are installed redundantly and separated by a concrete wall.	Determination of the maximum flood level that can occur in each cell containing safety related equipment assuming the failure of one active component which would mitigate the effects of a leakage crack. Verification that all safety related equipment is above this level.
I&C Panels and Vital Electrical Distribution (IE)	SGB	Flooding Water spray Jet Impingement Pipe whip Environmental	TBD	See PSAR Section 8.3.1.4	Detailed checking of conduit, cable tray and pipe routing to verify sufficient separation of safety related electrical equipment.
Auxiliary Feed Water Pumps	SGB Cell 204	Pipe whip Jet Impingement Environmental Flooding	Aux. feed pump disch. Aux. feed turbine drive steam line. Protected Water Storage Tank piping Emergency Chilled Water	The high energy piping systems located within this cell are not pressurized during normal plant conditions and therefore are not considered as postulated piping failures. Protection from flooding is provided by adequate floor drains/building sumps.	Analysis to prove that floor drains and building sumps are sized to provide adequate protection from flooding assuming a single active failure.
Protected Air Cooling Condenser	SGB Cells 281, 282, 283	Jet Impingement Pipe whip	SGAHRS steam line Steam drum relief and safety valve lines	Separation. The SGAHRS is divided into three loops separated by concrete walls. Only piping associated with a given loop is contained in the cell for that loops PACC. Loss of one loop cannot cause a loss of the decay heat removal function.	None required.

TABLE Q 020.5-1 (Cont'd.)

SAFETY RELATED EQUIPMENT TO BE PROTECTED	LOCATION	HAZARD	SOURCE OF HAZARD	METHOD OF PROTECTION	SCOPE OF ANALYSIS REMAINING
Protected Water Storage Tank	SGB Cell 204	Pipe whip Jet Impingement	Aux. feed pump disch. Aux. feed turbine drive steam line Emergency Chilled Water	Same as Auxiliary Feed Water Pumps	Same as Auxiliary Feed Water Pumps
Connecting Piping between PWST to and including first valve	SGB	Pipe whip Jet Impingement	Aux. feed pump disc. Aux. feed turbine drive steam line Emergency Chilled Water	Same as Auxiliary Feed Water Pumps	Same as Auxiliary Feed Water Pumps
Building Walls	SGB, Cells 241,243,244, 245,246,221, 222,223,281, 282,283,207, 208,209, 224, 225, 226	Gross struct. failure. Punch sheer failure. Hazardous missile from spalling. Fluids leaking thru wall	High energy systems contained within the listed cells.	The preferred method of protection from postulated pipe rupture, separation, is used exclusively for the SGB. Analysis to prove that this method of protection is adequate will be performed on each wall of a cell containing high energy piping. The results of this analysis will be used to size wall thickness or to add piping restraints whichever is more economical.	Detailed pipe rupture analysis to determine the effects of pipe rupture on building walls providing separation of redundantly installed safety related system.
Steam Generator Bulding	S GB/TGB Interface	Flooding	Feedwater and Condensate Secondary Services Cooling Water; Normal Plant Service Water; Fire Protection Circulating Water; Normal Chilled Water	Adequate Floor drains/sumps and curbs are installed to preclude flooding in TGB from entering SGB and affecting SGB safety related components.	NONE
Ventilation System including fan, filters, air conditioning	Control Building	Water spray. Flooding	Normal and Emergency Chilled Water System Fire protection	High energy piping systems are excluded from control building. Floor drains/ sumps adequately sized to prevent flooding in the event of piping failure and single active component failure. Pipe routing is designed to separate chilled water piping and electrical components. Where this is not possible, spray tight panels are used.	Determination of the maximum flood level that can occur in each cell containing chilled water piping and safety related equipment of a piping failure with one active component failure. Verification that all safety related equipment is above this level.
Emergency Batteries	Control Building	Water spray. Flooding	Normal and Emergency Chilled Water Systems Fire Protection	Same as Ventilation System.	Same as Ventilation System.
I&C Panels	Control Building	Water spray. Flooding	Normal and Emergency Chilled Water Systems Fire Protection	Same as Ventilation System.	Same as Ventilation System.

TABLE Q 020.5-1 (Cont'd)

SAFETY RELATED EQUIPMENT TO BE PROTECTED	LOCATION	HAZARD	SOURCE OF HAZARD	METHOD OF PROTECTION	SCOPE OF ANALYSIS REMAINING
Emergency Plant Service Water System	Diesel Generator Bldg. (DGB)	Pipe whip Jet impingement	H.P. Air-Diesel starting air Fire Protection	Separation of the Emergency Plant Service Water supplies is established so that a HP air line rupture of one engine will not affect the service water line of the other. Additionally, the pipe size of the H.P. air lines is smaller than that of the EPSW so that only through wall cracks are to be expected. Moderate energy systems do not pose a hazard to this portion of this system.	TBD
Auxiliary Mech. Systems for Diesel Generator	DGB Cells 511, 512	Pipe whip Jet impingement Water spray Flooding	H.P. Air - Diesel starting air. Emergency Plant Service Water Fire Protection	Diesel engines are installed redundantly. A pipe rupture in either "source" system precludes diesel operation for that unit but does not cause a loss of safety related function.	None required.
Diesel starting air system	DGB Cells 511, 512	None	None	There are no other high energy systems in these cells. Failure of a moderate energy system will not prevent this system from operating.	None required.
DGB Building Walls	DGB Cells 511, 512	Gross structural failure Punch sheer failure Hazardous missile from spalling Fluids leaking through wall.	H.P. Air - Diesel starting air	Reliability of the diesel electric generators is dependent upon the integrity of the boundary between the two systems. The wall will be sized and constructed to prevent all of the hazardous occurrences.	Detailed analysis of the effects of high energy pipe rupture on the walls of cells 511, 512.
Emergency Electrical Switchgear (IE)	DGB	Flooding Water spray	Emergency Plant Service Water	The service water header is separated by a concrete pipe chase. Adequate floor drains/sumps are installed to preclude flooding of safety related components.	Determination of the maximum flood level that can occur in each cell containing safety related equipment assuming the failure of one active component which would mitigate the effects of a leakage crack. Verification that all safety related equipment is above this level.
Safety Related Control and Instrumentation Panels	DGB	Flooding Water spray	Emergency Plant Service Water	Same as Emergency Electrical Switchgear	Same as Emergency Electrical Switchgear

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TABLE Q 020.5-1 (Cont'd.)

SAFETY RELATED EQUIPMENT TO BE PROTECTED	LOCATION	HAZARD	SOURCE OF HAZARD	METHOD OF PROTECTION	SCOPE OF ANALYSIS REMAINING
Air Blast Heat Exchangers	Reactor Services Building (RSB)	None	None	There are no high energy systems in the RSB. Non-sodium moderate energy systems other than ventilation are excluded from these cells.	Final check upon completion of final pipe routing to verify exclusion of non-sodiumppiping other than ducting from these cells.
EVST Na & NaK Cooling System Components	RSB Cells 336A, B,C, 319C,D	None	None	There are no high energy piping systems in the RSB. There are no non-sodium systems other than ventilation in these sealed and inerted cells.	Final check upon completion of final pipe routing to verify exclusion of non-sodium piping other than ducting from these cells.
EVST and EVST Guard Vessel	RSB Cell 327	None	None	There are no high energy piping systems in the RSB. There are no non-sodium systems other than ventilation in these sealed and inerted cells.	Final check upon completion of final pipe routing to verify exclusion of non-sodium piping other than ducting from these cells.
System 82 providing and servicing primary cover gas	RSB	Flooding. Water spray.	TBD	There are no high energy piping systems in the RSB. Adequate floor drains/sumps will be provided to protect safety related components from flooding. Spray tight covers will be installed where required.	Argon pipe routing incomplete. Analysis will determine the effect of water sprays on valve operators and other controls necessary for system operation where it is impossible to separate them from the system.
Emergency Plant Service Water	RSB	None	None	There are no high energy piping systems in the RSB. Moderate energy systems do not threaten this system's function.	None

Question 020.6 (None)

In regard to potential failures or malfunctions caused by freezing, icing, and other adverse environmental conditions, discuss the protective measures to be provided to assure the proper function of those components not housed within temperature controlled areas, and that are essential in attaining and maintaining a safe reactor shutdown.

Response:

The only components not housed within temperature controlled areas and essential in attaining and maintaining a safe reactor shutdown are the Emergency Cooling Towers. Piping to the Emergency Cooling Towers will be routed underground below the frost level to prevent freezing. The piping at the Emergency Cooling Tower is drained by a 3/4" permanent bleed line to the basin to prevent freezing of the pipes when the Emergency Cooling Tower is not operating. Electrical power cables, Emergency Cooling Tower fan motors, instrumentation and control equipment are provided with proper electrical insulation and selected such that the adverse environmental conditions will not affect their operability and safety function.

The water in the Emergency Cooling Tower storage basin is not affected severely by adverse weather conditions, specifically freezing, since the water level is below the ground level and the basin is approximately 40 ft. deep.

Under extreme cold weather conditions the two Emergency Plant Service Water Systems can be alternately operated to maintain the idle reservoir temperature above freezing.

Question 020.7

For all vessels that will contain gases under pressure (such as argon, nitrogen, chlorine, hydrogen, oxygen, air, and CO_2 tanks) provide the following information:

1. The design and operating pressures of the vessels,

2. The maximum pressure of the gas supply

3. The total amount of energy which could be released in the event that the largest pipe connected to the storage vessel should rupture

4. The protective measures that will be taken to prevent the loss of functions of adjacent equipment essential for a safe reactor shutdown,

5. Preliminary drawings that indicate storage locations and arrangements of components within each storage area.

Response:

Revised Section 9.5 provides the requested information.

Question 020.8 (None)

Provide the results of an analysis to demonstrate that failure of any non-seismic Category I auxiliary system or component (including associated turbine systems and components) will not have a detrimental effect (such as flood, spray, leaks) on safety related systems or will not prevent safe shutdown of the plant.

Response:

An analysis to demonstrate that failure of any non-seismic Category I auxiliary system and component will not have a detrimental effect on safety related systems outside containment is incorporated in the reply to Question 020.5 (3.6.5).

In response to Question 020.5, a table is provided which identifies the safety related target, its location, the hazard considered (pipe whip, jet, impurgement, or flooding), the source of the hazard, and the method of protection used to protect the safety related system. Non-seismic Category I auxiliary systems and components have been considered as a source of hazard to safety related systems in this evaluation.

In addition, in evaluating the auxiliary systems and components to be included in the containment building, each system or component having any possible effect on a safety related system or any possible effect on preventing a safe shutdown of the plant is designed in accordance with the requirements for seismic Category I systems. Thus, there are no non-seismic Category I auxiliary systems or components within containment which can have a detrimental effect on safety related systems.

Question 020.9 (5.6.1)

Additional information is required to evaluate the safety aspects of the Steam Generator Auxiliary Heat Removal System (SGAHRS). Provide description and analyses to demonstrate that the protected water storage tank (PWST) is capable of providing makeup to the steam drums until the residual heat load is reduced to a level that is within the capability of the Protected Air Cooled Condensers (PACC).

Response:

Updated PSAR Section 5.6.1.3.9 provides the analysis requested. 17

Question 020.10 (5.6.1)

Provide description and analyses to demonstrate that the Protected Air Cooled Condensers (PACC) are capable of removing the total residual heat upon depletion of the Protected Water Storage Tank (PWST) inventory.

Response:

This question has been answered in revised PSAR Section 5.6.1.3.9 as part of the response to PSAR Question 001.169.

Question 020.11 (RSP) (5.6.1 & 5.1.5)

It is our position that sufficient redundancy and diversity of power source be incorporated into the design of the Auxiliary Feedwater System (AFS) as described in BTP APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants", attached as Enclosure 3.

Response:

The information requested is contained in new PSAR Section 5.6.1.3.12. 17

Question 020.12 (9.4)

Provide the criteria and bases to be used in the design of electric heaters and associated mounting hardware applied to piping and components that are safety related. Provide single failure analysis to demonstrate that the failure of electrical heating system would not impair the ability of the safety related systems and components to perform their safety function.

Response:

See revised section 9.4.3.

Question 020.13 (9.6.1)

Provide additional description, piping and instrumentation diagrams (P&IDs) and single failure analysis for the Control Building Heating, Ventilating and Air Conditioning System. P&IDs should indicate design classification of each component and subsystem, and means for isolating the essential portions of the system from the non-essential portions. The system design should be such that the failure of non-essential portions of the system, or of other systems or structures not designed to seismic Category I requirements, will not prevent the operation of the essential portions of the control room area ventilation system.

Response:

The description, piping and instrumentation diagrams (P&IDs) for the Control Building HVAC System are incorporated in revised Section 9.6.1 and on the revised Figures 9.6-1, 9.6-2 and 9.6-3. The single failure analysis for the HVAC System is incorporated in new Table 9.6-2, "Single Failure Analysis, Control Room HVAC System" and in new Table 9.6-3, "Single Failure Analysis, Control and Diesel Generator Buildings Emergency HVAC System." The design classification of each component is incorporated in new Table 9.6-1, "Control Building HVAC System Equipment List".

Question 020.14 (9.6.2 and 9.6.3)

Provide additional description and design criteria for the Reactor Containment Building and the Reactor Service Building HVAC Systems. The information should include provisions to maintain the atmosphere in these areas suitable for the operating personnel and the equipment.

Response:

The response to this question is incorporated in revised Sections 9.6.2 and
49 9.6.3 and the revised P & ID's are provided on Figures 9.6-4 through 9.6-10.

Question 020.15 (9.6.5)

Provide additional description and design criteria for Diesel Generator Building HVAC System. The information should include a single failure analysis of the systems located in the Diesel Generator Building.

Response:

The description and design criteria for the Diesel Generator Building HVAC System is incorporated in revised Section 9.6.5. The single failure analysis for the Diesel Generator Building HVAC System is incorporated in new Table 9.6-7, "Single Failure Analysis, Diesel Generator Rooms HVAC System".

Question 020.16 (9.6.6)

Provide additional description and design criteria for the Steam Generator Building HVAC System. The information should include a single-failure analysis and the resultant ambient temperatures.

Response:

The description and design criteria for the Steam Generator Building HVAC System is incorporated in revised Section 9.6.6 and revised P&ID's are
49| provided on Figures 9.6-12 through 9.6-15. The single failure analysis for the safety related Steam Generator Building HVAC System is incorporated in new Table 9.6-9, "Single Failure Analysis, Steam Generator Building, Steam Generator Cells HVAC System."

Question 020.17 (9.7)

You state in the PSAR that the safety-related portion of the Auxiliary Coolant Fluid System (ACFS) has sufficient redundancy in equipment and piping to avoid fuel damage. The description and drawings provided in the PSAR are not sufficient in detail to permit an evaluation of this redundancy. Provide additional description, piping and instrument diagrams and a single-failure analysis for the ACFS system.

Response:

The response to this question is incorporated into the response to Question 020.32. 15

Question 020.18 (9.9.2)

Provide design criteria and single-failure analysis for the Emergency Chilled Water System (ECHWS). Provide, in table form, individual cooling requirements of various coolers served by the emergency chilled water system.

Response

Section 9.9.2.1 of the PSAR is revised to include design criteria for Emergency Chilled Water System (ECHWS). Single failure analysis of the system is presented in Table 9.9.2-3. Table 9.9.2-1 is revised to include individual cooling load requirements and locations of the various components served by Emergency Chilled Water System. Table 9.9.2-2 is revised to reflect the changes in cooling water requirements shown in revised Table 9.9.2-1.

Question 020.19 (9.9.4 & 9.9.6)

In order to permit an assessment of the ultimate heat sink, provide the results of an analysis of the thirty-day period following a design basis accident that determines the total heat rejected, the sensible heat rejected, the station auxiliary system heat rejected, and the decay heat release from the reactor. In submitting the results of the analysis requested, include the following information in both tabular and graphical presentations:

(1) The decay heat rate and total integrated decay heat.

(2) The heat rejection rate and integrated heat rejected by the station auxiliary systems, including all operating pumps ventilation equipment, diesels and other heat sources.

(3) The heat rejection rate and integrated heat rejected due to sensible heat removed from containment and the primary system.

(4) The total integrated heat rejected due to the above.

(5) The maximum allowable cooling water inlet temperature taking into account the rate at which the heat energy must be removed, cooling water flow rate, and the capabilities of the respective heat exchangers.

(6) The maximum design ambient air temperature.

The above analysis, including pertinent backup information, should demonstrate the capability of the ultimate heat sink to provide sufficient heat dissipation to limit cooling water operating temperatures within the design ranges of system components, and should be based on the guidelines provided in Regulatory Position C.1.a and C.1.b of the Regulatory Guide 1.27.

Response:

The Steam Generator Auxiliary Heat Removal System (SGAHRS) provides primary heat removal for the reactor decay heat. The Overflow Heat Removal System (OHRS) provides backup for the SGAHRS as a secondary means for the reactor decay heat removal. The 100% redundant Ex-Vessel Storage Tank Cooling System (EVS) provides spent fuel decay heat removal service. All three systems use the atmosphere as an ultimate heat sink. The Emergency Plant Service Water System with the Emergency Cooling Towers provides auxiliary heat removal to support the operation of the above systems, and provides the ultimate heat sink for the Control Room HVAC and Emergency Chilled Water System, Auxiliary Coolant Fluid System, Standby Diesel Generators and other heat sources. The SGAHRS, OHRS and EVS Systems are described in Sections 5.6.1, 5.6.2 and 9.3.3 of the PSAR. The Emergency Plant Service Water System and the Emergency Cooling Tower are described in Sections 9.9.4 and 9.9.6 of the PSAR.

(1) The present selection of the Emergency Cooling Tower Storage Basin is not based on the integrated decay heat to auxiliary heat removal, but it is based on the maximum decay heat rate applied for the required 30 day period. Additionally 10% extra capacity is incorporated into the basin design. This provides a conservative approach in accordance with the presently available information. The decay heat rate and the integrated decay heat rate will be incorporated into the Emergency Cooling Tower design and will be presented in the FSAR.

(2) Table 9.9.4-1 of the PSAR is revised to show the maximum heat removal from the various components served by Emergency Plant Service Water System.

(3) The heat rejection rate removed from the primary system is described in Sections 5.6.1 and 5.6.2 of the PSAR. The integrated heat rejection from the primary system will be presented in the FSAR.

(4) The total heat is rejected from the plant by the above systems. The integrated total heat for each of the systems and the total integrated heat rejected from the plant will be summarized and presented in the FSAR.

(5) The maximum cooling water inlet temperature to each component will be presented in the FSAR.

(6) The cooling load calculations are based on 95^{o}F DB and 77^{o}F WB ambient temperatures simultaneously to establish the Emergency Cooling Tower heat rejection requirements. The above temperatures represent 0.424% and 0.585% duration for all summer hours (2928) at the Oak Ridge Area Station X-10 from 1966 through 1972. The cooling loads and their effect on the Emergency Cooling Tower heat rejection requirements will be analyzed on the basis of the highest historical temperatures. The result of this analysis will be presented in the FSAR.

On the basis of information received from various cooling tower manufacturers having experience with nuclear safety related ultimate heat sinks, it is assumed tha the evaporation rate of the Emergency Cooling Tower will be maintained at a relatively constant rate despite the variations of the loads andmeteorological conditions by controlling the air/water ratio of the tower opration. The anticipated methods for the air/water ratio control are cooling tower fan cycling or discharge dampers modulation. The analysis supporting the above assumption will be presented in the FSAR.

Question 020.20 (9.13)

Description and analyses of the Fire Protection System (FPS) should emphasize protective measures taken to prevent occurrence of fires. These measures should include separation by fire barriers, use of fire-resistant construction material, locating combustible material in separate areas of the plant and providing fire protection system for these facilities.

Description and evaluation of the fire protection system should be in accordance with the requirements of Regulatory Guide 1.70.4.

Response:

The information requested is incorporated into revised Section 9.13.1 for conventional fire protection. The information pertaining to Sodium fire protection is incorporated into revised Section 9.13.2. 2

Question 020.21 (9.14)

Provide sufficient details in Figures 9.14-1, 9.14-2, 9.14-3 and 9.14-4 to permit proper evaluation of the safety aspect of the diesel generator auxiliary systems.

(1) A cross connection with two locked-closed valves should be provided between the two fuel oil pump suction lines from each buried fuel oil storage tank to enhance the diversity of the emergency power generation.

(2) Provide design parameters for the diesel generator auxiliary system components.

(3) Indicate the source of cooling water supply to the diesel-driven and motor-driven air compressors for the diesel-generator starting air system.

Response:

1. Section 9.14.1.1 and Figure 9.14-1 have been revised to reflect the suggested change.

2. More definitive parameters for the diesel generator auxiliary system components will be supplied in the FSAR following the purchase of the diesel generator sets. However, certain design parameters have been added to the revised Sections 9.14.1, 9.14.3 and Figure 9.14-1.

3. The source of cooling water supply to the starting air compressors is identified in revised Figure 9.14-3. Revised Section 9.14.3.2 reflects the deletion of the diesel driven air compressor.

Question 020.22 (10.3)

Provide design criteria and bases to ensure that the main steam isolation valves will be capable of closing against accident flow rates caused by a steam line break downstream of these valves.

Response:

The temperatures, pressures, flow rates and response times which provide specific design bases for the main steam isolation valves are not yet defined. The design features planned to meet the eventual design criteria are discussed in revised Section 10.3.1 of the PSAR.

Question 020.23 (10.2.1)

Provide additional description, design criteria, and bases for the turbine speed control system.

Response:

The description, design criteria, and bases for the turbine speed control system have been provided in the revised PSAR Sections 10.2.1 and 10.2.2. 41

Amend. 41
Oct. 1977

Question 020.24 (10.4.7)

Determine the volume of water that could drain to the turbine building as a result of a failure in the recirculating water piping, and discuss the precautions taken to ensure that the intended function of the safety-related equipment will not be impaired by the flow of this water to the steam generator building via stairways or other openings.

Response:

The maximum amount of feedwater volume that can drain into the Turbine Generator Building (TGB) in case of a break in the feedwater and condensate piping is ~205 cu. ft. If this water spreads over the Ground Floor it would result in a ~0.1 in. water layer spreading uniformly over the floor, assuming no provisions for drain water disposal. However, there are several floor drain fittings provided in the ground floor of the TGB which are piped to floor sumps. The capacity of each sump is ~54 cu. ft. and therefore four of these would be adequate for protection. In addition to these sumps, there are trenches covered with gratings in the ground floor, their total holding capacity is ~980 cu. ft., and so they are more than adequate for preventing water buildup in the TGB. Therefore, no water would flow into the Steam Generator Building (SGB) via the two openings which are presently two 7 ft. doors for fire escape only and are kept closed. Also a 4 in. curb will be provided at these doors.

Question 020.25 (5.6.1)

Provide an explanation of nomenclature used in Figure 5.6.1 and demonstrate that the selected design load of 15 MWt for the PACC is compatible with the data given in Figure 5.6-1.

Response:

The first part of this question has been answered in the response to NRC Question 001.169.

PSAR Section 5.6.1.3.9 provides the requested identification of SGAHRS component sizing criteria. In summary, in order to size critical components of SGAHRS such as the PACC's the transient heat load on the system was considered. If the normal feedwater supply is unavailable upon plant shutdown, Auxiliary Feedwater (AFW) will be supplied to the steam drums thereby allowing sensible and decay heat to be removed by venting steam in addition to PACC operation. Venting continues until the heat load decreases to the level at which the PACC's can reject all incoming heat. The PACC's are sized to remove 15 MWt each (45 MWt total) with forced convection on the air side. The heat load is expected to reach this level within approximately 3/4 hour as shown in Figure 5.6-1. Feedwater consumed by steam venting during the transient will be available from the Protected Water Storage Tank (PWST). The quantity of feedwater used during the venting process is directly proportional to the heat rejected. The PWST has been sized to provide sufficient water for all postulated occurrences including those requiring venting for longer periods of time than shown in Figure 5.6-1 (e.g., loss of loop).

Question 020.26 (9.5.1)

Description and drawings provided for the Argon Distribution System are not adequate to evaluate the system. Provide additional description and P&ID's for the Argon Distribution System. The information should include the type of valves to be used and their ability in containing the radioactive cover gas in the event of packing gland failure. Figure 9.5-2, Sheets 2, 3, and 4, appear to be identical. If they show Argon Distribution System in Steam Generator Building for three separate, but identical, loops, the equipment in each should be so designated. The same comments apply to Sheets 2, 3, and 4 of Figure 9.5-3.

Response:

The P&ID's for the Argon Distribution System are currently being prepared and will be supplied to NRC in July, 1976. The valves to be used for the Argon Distribution System have not been specified. The criteria for specification of the valves will include consideration of the effects of packing gland failures.

Question 020.27 (9.5.1)

Explain the purpose of the arrows with numbers that occur in several flow diagrams. If these arrows and numbers designate system interfaces and system numbers, these systems should be so designated in the description.

Response:

The numbers adjacent to the interface arrows do refer to the interfacing systems. Amended Section 9.5 identifies the systems and their numbers.

Question 020.28:

Provide criteria and bases used in determining the size of the liquid argon and liquid nitrogen storage on site. Fresh argon supply rate should be based on the possibility of failure of the Radioactive Argon Processing Subsystem (RAPS) and, consequently, no purified argon return from RAPS to the Primary Recycle Cover Gas Storage Tank. Provide information to demonstrate that argon can be delivered to the site in the event of extreme natural phenomena, such as rain, snow, and resultant floods before depleting onsite stored gases.

Response:

There are five liquified gas storage complexes in the IGRP system. Two of these are at the RSB pad, two are at the SGB pad, and one is in the RSB.

The RSB argon supply consists of two 1500 gal. dewars arranged to deliver gas in sequence, or in parallel. Any dewar can be charged at will. The size of these vessels is determined by the projected consumption and the desired reserve capacity. The normal usage of argon, once the system has been filled and settled in its operation will be modest, and a single dewar will provide a minimum of 30 days of normal service. About half of dewar will be required to re-inert the Fuel Handling Cell. Therefore, the two dewars provide the necessary back-up when this large cell is being serviced.

The RSB nitrogen supply consists of two 6000 gal. dewars arranged to deliver gas in sequence, or in parallel. At the design-value use rate required to supply inerting gas to the RSB and RCB cells, each dewar can provide a 6-day supply. When sodium component cleaning operations are in progress, one dewar can provide a 3 day supply. Vessels provide a minimum of six days of service at the maximum use rate.

The 6,000 gal. dewar size was chosen to coincide with the capacity of a standard long haul cryogenic tanker truck. Such a tanker is expected to be used to provide scheduled recharging service. The currently identified source of supply is located in Huntsville, Ala., which is approximately 175 miles from the plant site via primary surface roads. Normal transit times can be projected to be less than 5 hours so that the 3 day reserve provides sufficient time to recover from late deliveries due to natural causes (weather, accidents, etc.) The delivery of argon will be on a similar basis.

The SGB argon supply consists of two 1500 gal. dewars arranged to deliver gas in sequence, or in parallel. Each dewar is expected to provide normal service for at least 30 days. Maintenance and sodium transfer operations in SGB are not expected to require more argon than can be supplied with adequate reserve by the two dewars. These dewars also provide a back-up supply via a tie-line to the RSB dewar system.

The normal SGB nitrogen supply system consists of two 3,000 gal. dewars. For normal operation, the service period of each tank is expected to be about 30 days. However, a sodium cleaning operation has been projected for an SGB location. Its needs would require one tank's capacity in 4 days. Therefore, the two dewars provide an adequate reserve for future needs.

The sodium-water reaction nitrogen supply consists of one 3,000 gal. dewar with a connection to the normal SGB nitrogen supply for emergency use. This supply is provided for use following sodium/water reaction events. The nitrogen is used as the SWRPRS inert cover gas and for holding the pressure on the water side of the steam generators following sodium dump. The steam generator system has only a small normal use-rate, and will be recharged to fill the dewars when the supply tanker arrives for any liquid nitrogen service. The dewar can supply service to one steam generator module for about 36 hours.

Question 020.29 (9.5)

Abbreviations associated with valves shown in table form on page 1.A-6 of the PSAR do not cover all the designations shown on flow diagrams. For instance, Figure 9.5-1 contains valves that are designated by abbreviations of LV, HV, PV and YV, which are not explained in the above mentioned table. Expand the table to include all abbreviations used in the figures.

Response:

The table in Section 1.A has been revised to include all abbreviations used in the figures of Section 9.5.

Question 020.30 (9.6.1)

The non-essential portions of the control room HVAC system should be isolated from the essential portions appropriately (by two automatically isolated dampers). Revise your design to show this capability.

Response:

The response to this question is incorporated in revised PSAR Section 9.6.1.3, "Safety Evaluation" and revised Figure 9.6-1, "Flow Diagram - Control Room HVAC System." Figure 9.6-1 has been revised to show that the toilet exhaust ductwork connects directly to the Control Building exhaust structure instead of connecting with the Control Room exhaust ductwork and that the toilet exhaust system is provided with two automatically operated isolation dampers.

Question 020.31 (9.6)

Revise your design to include redundant monitors that are capable of detecting radiation, smoke, and toxic chemicals in the control room HVAC system air intakes. These monitors should actuate alarms in the control room.

Response:

The response to this questions is incorporated into revised PSAR Sections 9.6.1.2 and 9.6.1.3, and into revised Figures 9.6-1, "Flow Diagram - Control Room HVAC System," 9.6-2, "Flow Diagram - Control and Diesel Generator
49| Buildings Emergency HVAC Systems," and 9.6-14, "Flow Diagram - Steam Generator Cells and Auxiliary Bay HVAC System".

Figure 9.6-1 has been revised to show redundant toxic chemical and smoke detectors and radiation monitors in the Control Room air intake ducts.
49| Figures 9.6-2 and 9.6-14 have been revised to show the relocation of the radiation monitors from the Control Building and Steam Generator Building air intakes to the Control Room air intake ducts.

Question 020.32 (RSP) (9.7)

Your response to question 020.17 is not complete. Figure 9.7-1, Auxiliary Coolant Fluid System (ACFS) Schematic Flow Diagram does not provide sufficient detail to permit evaluation of safety aspects of the system.

It is our position that isolation valves be provided to isolate the non-safety related portions of the ACFS system from safety related portions. Also, describe how ACFS flow to safety related heat loads inside containment will be maintained in the event of a containment isolation signal.

Provide description and P&ID in sufficient detail showing all valves and pertinent instrumentation for the ACFS system.

Further, your response indicates that redundancy is provided only on the Recirculating Gas Cooling System (RGCS) side of the coolers. Since there is no redundancy in cooler units, single failure analysis should consider loss of cooling capability to any single cooler unit. Consequently, residual heat removal system should not take credit for the OHRS system for removing reactor decay heat.

Response.

Since the preparation of PSAR Section 9.7, "Auxiliary Coolant Fluid System", the use of Dowtherm J has been re-evaluated. As a result of this re-evaluation, Dowtherm J auxiliary coolant has been replaced with chilled water, except in situations where sodium is contained in the fluid being cooled. In these cases, a secondary Dowtherm J cooling loop is interjected between the sodium containing fluid and the chilled water. To implement these changes, several sections of the PSAR have been revised to replace references to Dowtherm J with references to chilled water. In addition, the Normal and Emergency Chilled Water portions of Section 9.9 have been relocated to Section 9.7.

The response to this question, modified to reflect the change from Dowtherm J to chilled water has been incorporated into the following revised Sections of the PSAR:

Sections 3.A, 3.2, 6.2.4, 7.6.1, 9.1, 9.3, 9.6, 9.7, 9.9, 9.15, 15.7, and 16.3.7.

In addition to the detailed information provided in the revised PSAR Sections, the following summarizes the approach taken to resolve each point raised by Question 020.32:

a. Figures 9.7-1 and 9.7-2, "Normal and Emergency Chilled Water Systems" provide sufficient detail to permit evaluation of the safety aspects of the Chilled Water Systems including the former Auxiliary Coolant System.

b. Redundant isolation valves are provided between safety and nonsafety related portions of the Chilled Water Systems (i.e. between the normal and emergency portions) to ensure separation capability.

c. In the event of a containment isolation signal, the Normal and Emergency Chilled Water supply lines to containment will not be automatically isolated, because both Chilled Water Systems are closed systems capable of withstanding containment design pressures. If events (e.g. a pipe break in the Normal or one Emergency Chilled Water line) should require the isolation any chilled water cooling loop, remote manual isolation valves will be shut in the affected loop upon a signal from the chilled water and the drainage system leak detectors. Whenever this is done, all safety-related heat loads inside containment will be supplied by either the Normal Chilled Water System, or if the normal system is unavailable by the redundant Emergency Chilled Water System.

d. Figures 9.7-1 and 9.7-2 and the descriptive subsections in Section 9.7 show all system valves and pertinent instrumentation requested.

e. The two-train, redundant Emergency Chilled Water System supplies cooling to safety-related units in the Recirculating Gas Cooling System.

Question 020.33 (RSP) (9.9.4)

Figure 9.9-4 indicates that each Emergency Plant Service Water (EPSW) pump provides flow through a separate and redundant loop.

Present design is not capable to assure cooling water flow to safety related equipment in the event of a moderate energy line crack in one loop and a simultaneous single active failure in the other. Revise your design to provide this capability.

Response:

As shown on Figure 9.9-4, (Basis Flow Diagram - Emergency Plant Service Water System) EPSW system design meets the requirements of a single failure criterion as defined in Section 3.1.2. The assumption of a simultaneous piping failure (moderate energy line crack) in one loop of the EPSW and an active failure in the other loop goes beyond what is required by the single failure criterion.

However, in the event the above assumption is considered, the leaking loop can be isolated from the NPSW system and normal reactor shutdown can be achieved using the NPSW system. Therefore, no design changes are considered necessary.

Question 020.34 (9.13.1)

Provide P&ID's showing the fire protection system yard piping, storage tank and the pumps.

Response:

The Water Supply System P&I Diagram, Figure 9.13-1 shows the schematic arrangement of the fire protection system yard piping and the pumps. The lower portions of the main station cooling tower basins serve exclusively as the water storage reservoirs for the Non-Sodium Fire Protection System. Table 9.13-4 (sheet 1 of 4) describes the capacity of the fire protection reservoirs and the basic design features of the yard piping loop. 19

Question 020.35 (9.13.1)

The PSAR states that Halon 1301 fire suppression system is provided in the Control Room and the Computer Room. Since Halon 1301 and its products of pyrolytic decomposition carry a risk to personnel, describe how the following precautions and safety measures are considered in the design:

(1) Detectors on each line of the Halon system at its storage location to detect small leaks.

(2) Detectors in the Control Room and its ventilation system, capable of sensing small fractions of a volume percent of Halon 1301 in the Control Room, and alarms to alert personnel of the presence of Halon 1301.

(3) A description of the methods to be used to seal the Computer Room.

(4) A description of how safe access to the Computer Room, after a postulated fire, will be made to assure that the fire has been completely extinguished.

(5) A description of other fire protection systems or equipment that are available within or without the Control Room to completely extinguish a fire in the control cabinets or subfloor, if required, in addition to the Halon system.

(6) The design criteria and bases for the Halon system to withstand natural phenomena, failures in the storage system and single active failures in the distribution system.

Response:

Revised PSAR Section 9.13.1.2 responds to this question.

Question 020.36 (RSP) (9.13)

Your response to question 020.20 is not complete. The description provided in the PSAR does not follow the guidelines set forth in Sections 9.5.1.1, 9.5.1.2, 9.5.1.3 and 9.5.1.4 of the Regulatory Guide 1.70.4, "Additional Information, Fire Protection Considerations for Nuclear Power Plants." Provide the additional information.

Response.

The additional information requested is incorporated into the revised Section 9.13.1 for the Non-Sodium Fire Protection System.

Question 020.37 (E.2.1)

Section E.2.1 lists general requirements and key objectives related to the modifications to the reference design to accommodate the consequences of postulated pipe ruptures. Specify whether the plant design will be committed to make these objectives into design requirements. Modifications to the reference design should not compromise the integrity of the systems and components that are essential for the safe shutdown of the plant. Revise Section E.2.1 accordingly to include a design requirement to address this issue.

Response:

For the purposes of Appendix E, a doubled-ended rupture in the primary heat transport system piping will be accepted as a design basis and the general requirements and key objectives given in Section E.2.1 will become design requirements for the plant. Section E.2.1.has been revised accordingly.

Modifications to the reference design will not compromise the integrity of the systems and components that are essential for the safe shutdown of the plant and Section E.2.1 has been revised to include this requirement.

Question 020.38 (E.3):

Table E.3-1 indicates that one of the assumptions made in pipe rupture core transient analysis for three-loop plant operation involves a maximum cover gas makeup flowrate of 100 scfm to the reactor vessel. Show that the effects of this accelerated argon flow is considered in modifications to the argon storage and distribution capability of the reference design.

Response:

The pipe rupture core transient analysis assumptions include a 100 scfm cover gas flowrate into the reactor. This flowrate is described in Table E.3-1 as being chosen so as to estimate the maximum break outflow. Current design effort results indicate that, in order to minimize the number of cycles that the control valve will experience in providing recycle argon gas to accommodate a reactor trip, the flowrate in this line be limited to about 60 scfm. Although this restriction might result in a reduction of the rate of sodium release in the break, the effect is expected to be small.

The gas which flows through this valve originates (in the reference design) in the recycle argon vessel, which normally contains about 3,000 scf of gas at about 50 psig. This vessel is the first source of gas to respond to the pipe rupture event. The vessel is capable of delivering gas for at least 20 minutes at 100 scfm. Thus, no modification to the reference design is necessary. As the vessel becomes depleted and its pressure drops to the selected set point of approximately 10 psig, fresh argon supply makeup gas enters the distribution line. Normally, this supply is capable of delivering gas from one argon Dewar at 33 scfm, but if a condition persists requiring high flow, the other two Dewars with their 33 scfm evaporation rate can be made available by the operator.

Question 020.39 (6.2.6.1.2 Yellow)

The SHAA cooling and ventilating system does not appear to be designed for a single failure since there is only one supply and one exhaust lines. Demonstrate that the design values of the SHAA will not be exceeded assuming a single failure in the cooling, and ventilating system or, revise your design accordingly.

Response:

This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. The CRBRP no longer has a sealed head access area. 60

Question 020.40 (9.5.1 Yellow)

Provide design criteria and bases used to determine the size of emergency argon storage facility inside the Dump Heat Exchanger (DHX) building.

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer relevent as the features upon which the question is based are no longer a part of the design.

Question 020.41 (6.2.7.2 Yellow)

Provide a description, preliminary layout drawings and P&ID showing the heating, ventilating and cooling, and other auxiliary systems required in the DHX building. Discuss the effect of the additional requirements of the DHX building on the capability of the related auxiliary systems included in the reference design.

Response:

In Amendment 24 to the PSAR, the Project withdrew the Parallel Design from further consideration by the NRC staff. This question requests additional design information on a specific feature of the Parallel Design. Accordingly, the question is no longer relevant.

Question 020.42

Identify the means proposed to isolate the DHX in the event of fire and include the measures to be used to inert the atmosphere. (Refer to our previous request 020.20).

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer applicable as the DHX is no longer a part of the design.

Question 020.43 (9.5 Yellow)

It appears that credit is being taken for operation of the inert gas receiving and processing system following the postulated CDA. Specifically, the capability of the argon gas distribution, radioactive argon processing (RAPS) and cell atmosphere processing (CAPS) subsystems appear to be assumed to be available. The subsystems are not designed as engineered safety features (Table 6.1-1) and postulated single failures (e.g., failure of the cryogenic column) apparently are not considered. The need for these subsystems is not clear. Revise your analyses accordingly or propose revised systems' designs to reflect their safety-related function, if any.

Response:

The current TMBDB analyses [see CRBRP-3, Volume 2 (Reference 10b of PSAR Section 1.6)] take no credit for operation of the inert gas systems during the postulated scenario.

Question 020.44

The response to item 020.2 is not complete. Revise the PSAR to indicate the tornado protection provided for the emergency cooling tower fans and motors.

Response:

As described in updated Section 3.8.4.1.4, the Diesel Generator Building houses and provides tornado protection for the airblast heat exchangers, which dissipate the Emergency Plant Service Water heat load. The revised design does not include emergency cooling towers.

Question 020.45 (RSP) (3.6.5)

You have not responded to item 020.4 with respect to committing to meeting Branch Technical Position APCSB 3-1 in regard to protection against postulated piping failures in fluid systems outside containment.

Response:

PSAR Section 3.6 has been revised in Amendment #27 in response to question 020.4.

Question 020.46 (RSP)(5.6.1)

Your response to item 020.9 and 020.10 are not complete. The DAHRS (Demo Auxiliary Heat Removal Simulation) computer model used to determine the steam venting requirements, which affect the sizing of the Protected Water Storage Tank (PWST), assumes that the Protected Air Cooled Condensers (PACC) operation start twenty minutes after shutdown. The PSAR further states that venting will continue until heat load decreases to the level at which the PACCs can reject all incoming heat at about one hour after shutdown. Also, the operation of the PACCs require AC power to drive the fans. The analysis based on this assumption results in a PWST capacity that is non-conservative.

It is our position that if credit is to be taken for PACC operation for short term (within 2 hours after shutdown) shutdown heat removal, PACC design should meet power diversity requirements of Branch Technical Position 10-1, i.e., PACC should be able to operate without an A-C motive power source within this time period. Alternatively, if no credit is to be taken for PACC operating during the short term (2 hours) the size of the PWST should be sufficient to provide the necessary makeup to the steam drum for two hours.

Response:

The two hour loss of all bulk AC power is not a design basis event for CRBRP. However, section 5.6.1.3.9 has been modified to include a two hours loss of AC power event in evaluation of the PWST size. A new case covering this event has been added to Tables 5.6-7 and 5.6-9. The loss of all bulk AC power is assumed and the volume of the PWST is shown to be adequate. PACC heat rejection is assumed to be zero with the PHTS and IHTS naturally circulating.

Question 020.47

The non-sodium fire protection system should, to the extent reasonable and practicable, conform to the guidelines of Branch Technical Position APCSB 9.5-1, a copy of which is enclosed.

Response:

The CRBRP Project, to the extent reasonable and practicable, is committing to meet the intent of Branch Technical Position CMEB9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" as indicated in PSAR Section 9.13.1.

Question 020.48 (10.4.7)

Your response to item 020.24 is not complete. Determine the volume of water that could drain to the turbine building as a result of a failure in the condenser cooling water piping or expansion joints at the condenser, and discuss precautions taken to ensure that the intended function of the safety related equipment will not be impaired by the flow of this water to the steam generator building via stairways or other openings.

Response:

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 (46,200 gal) plus the circulating water contents of the cooling tower basin (about 1,500,000 gal) i.e., a total of 1,546,200 gal.

There is no safety-related equipment located in the TGB. There are two potential pathways for water to enter buidings where safety-related equipment is located i.e., the personnel door in the Steam Generator Building Auxiliary Bay of the Steam Generator Building (SGB) and the personnel access corridor leading to the Diesel Generator Building (DGB), Control Building (CB) and other Nuclear Island buildings. Other openings in the TGB through which water could empty before entering the Nuclear Island (NI) buildings include doorways leading to the Maintenance Shop and Warehouse Building and the yard transformer area, as well as the roll-up door for the TGB railroad access bay.

The intended function of safety related equipment will not be impaired by the flow of this water into the Nuclear Island buildings since the potential pathways leading into the SGB and DGB from the TGB will be provided with watertight doors.

Instrumentation denoting the initiation of the circulating water system flooding incident for operator attention and action, as required, will be provided. Waste Water Treatment System 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 such that the circulating water pumps can be shutdown, eliminating the water at the source.

Question 020.49

In Amendment 29, you indicated that the impact of Appendix A to Branch Technical Position 9.5-1 is presently under evaluation. Appendix A dated August 23, 1976, gives alternatives to the staff to be utilized in the re-evaluation of fire protection provisions. In order to begin our re-evaluation, perform a fire hazards analysis of your facility with the assistance and the technical direction from a qualified fire protection engineer.

This examination should:

(a) Identify the guidelines in Appendix A which are presently met, and discuss how this is done;

(b) Identify the guidelines for which modifications, procedural changes, or enhanced training of personnel are underway or planned, such that the guidelines will be met, and the date you intend to meet Section B of Appendix A, "Administrative Procedures, Controls and Fire Brigade", and

(c) Indicate which of the guidelines you do not now meet or do not intend to meet in the future. For such items, you should provide a basis for your position.

Response:

The detailed response to Appendix A to Branch Technical Position APCSB 9.5-1 is contained in Appendix A to PSAR Section 9.13.

Question 040.1 (6.2.1.3)

Identify the heat sinks used in the containment pressure/temperature analysis giving, for example, surface area, material, thickness, and location.

Response:

The Reactor Containment Building pressure and temperature transient analysis presented in Section 6.2.1.3 is based on the RCB Design Basis Accident (Primary Sodium In-Containment Storage Tank Failure During Maintenance). This postulated accident assumes a large sodium pool fire on the floor of a sub-grade containment cell (Sodium Overflow Vessel and Storage Vessel Cell). For the accident evaluation, the cell is assumed to be de-inerted (air atmosphere) and in communication with the upper containment volume.

The heat sinks used in the pressure and temperature analysis are: 1) The containment vessel steel shell above the operating floor, 2) the urethane foam insulation covering the outer surface of the containment vessel shell, 3) the steel wall and floor liners in the sub-grade containment cell, and 4) the concrete wall and floor of the sub-grade containment cell.

Tables 6.2-2 and 6.2-2A of the PSAR provide a detailed description of the geometry (thickness, area) and heat transfer properties of these heat sinks. New Table 6.2-4A presents a summary description of the heat sinks used for the analysis.

Question 040.2 (6.2.1.3):

For those analyses in which heat rejection to structure and components were considered, justify the heat transfer analysis including (1) heat transfer coefficients, (2) contact resistances, (3) heat transfer through insulation, and (4) modeling or simulation used. Provide the thermal response of typical sections used for design purposes.

Response:

The structures to which heat rejection was considered are 1) the containment vessel steel shell above the operating floor, 2) the urethane foam insulation covering the outer surface of the containment vessel shell, 3) the steel wall and floor liners in the sub-grade containment cell, and 4) the concrete wall and floor of the sub-grade containment shell. The response to Q040.1 provides a summary description of each of these heat sinks and Tables 6.2-2 and 6.2-2A of the PSAR provide a detailed description of the geometry (thickness, area) and heat transfer properties of these heat sinks.

The applicable heat transfer mechanisms in the analysis are 1) radiation heat transfer from the sodium pool to the gas atmosphere above the pool and to the cell walls above the pool, 2) convection heat transfer from the sodium pool to the gas atmosphere, 3) convection heat transfer from the gas to the walls of the cell and the containment vessel, and 4) conduction heat transfer between inter-connected nodes of the heat sink structures.

The analytical methods used for determining radiation and convection heat transfer from the sodium pool have been justified and fully explained in the response to question 001.241.

Convection heat transfer from the containment gas atmosphere to the steel cell wall liner and steel vessel shell is dependent on the temperature difference between the gas and the wall, the surface area of the wall (steel liner or vessel shell) exposed to the gas, and the convective heat transfer coefficient. The exposed wall areas and initial gas and wall temperatures are input parameters to the computer code SOFIRE-II, (See Tables 6.2-2 and 6.2-2A). The convective heat transfer coefficient is dependent on the thermal conductivity, viscosity, and Prandtl Number of the containment gas atmosphere and the temperature difference between the gas atmosphere and the wall. The actual correlation used in SOFIRE-II to compute this coefficient is described in Reference Q040.2-1

Conduction heat transfer through the heat sinks is based on a nodal heat transfer analysis. Each structure is modelled as a series of inter-connected nodes, e.g., the containment vessel shell and urethane foam insulation covering the shell are modelled as four inter-connected nodes. Node thicknesses and cross-sectional areas are input parameters, as are the material properties for each node, such as density, conductivity, and specific heat. The specific input parameters which describe each node (thickness, area, density, etc.) are itemized in Tables 6.2-2 and 6.2-2A. The conduction heat transfer model used in the SOFIRE-II code is described in detail in Reference Q040.2-1.

Heat transfer through the urethane foam insulation covering the outside of the containment vessel shell is calculated by a nodal conduction heat transfer analysis as discussed in the preceding paragraph.

No contact resistances between any two heat nodes were assumed for the analysis.

The basic model used for the analysis was the SOFIRE-II computer code (Reference Q040.2-1). The accident evaluated in Section 6.2.1.3 results in a large pool fire on the floor of a sub-grade containment cell; the cell is assumed to be de-inerted (air atmosphere) and in direct communication, via an open equipment hatch, with the upper containment volume. The Two-Cell version of SOFIRE-II, which was used for the referenced analysis, effectively models the inter-connected cell geometry associated with the accident.

The applicability of the SOFIRE-II code to sodium pool fire evaluations has been discussed in Section 6.2 of the PSAR and in response to question 001.237. The references cited in response to question 001.237 and the references in Appendix A provide the basis for the heat transfer methodology and modeling used in SOFIRE-II.

Typical thermal responses for the heat sinks used in the analysis are provided for the containment vessel shell, and the cell steel wall and floor liners in Figures 6.2-4, 6.2-8, and 6.2-9 respectively.

References

Q040.2-1. AI-AEC-13055, "SOFIRE-II User Report," March 30, 1975

Question 040.3

Provide a list of all nonseismic systems and components within the containment building and inner-cell system. Discuss the effect that the failure of these systems or components will have on the design basis accidents.

Response:

Detailed analysis of the failure of non-Seismic Category I equipment in the reactor containment building have not been performed. However, failure of any non-Seismic Category I equipment is considered in continuing design and evaluation activities.

New Table 3.2-6 provides a listing of non-Seismic Category I systems and components in containment.

Question 040.4 (15.6.1):

Thirteen inner cells have been identified as containing sodium systems and requiring at least partial inerting. Section 15.6 does not provide analyses which consider all compartments. It is indicated in Section 3A that each cell has been analyzed to determine the functional design requirements--cell pressure, temperature and liner temperature--due to postulated accidents. The following information is required for each cell subject to sodium spills and pressure/temperature transients:

a. Sources of sodium spill(s) within each compartment, identifying system, leak flow rate, duration, and temperature of leaking fluid;

b. Methods of evaluating cell response including analytical techniques and modeling;

c. Peak calculated cell gas pressure and temperature;

d. Cell design pressure and design temperature for the hot and cold liners.

Response:

a. Table Q040.4-1 summarizes the source, rate, quantity, duration and temperature of the design basis spill identified for the inner cells of the Reactor Containment Building. Time sequences for leak termination are detailed in Tables Q040.4-2 and Q040.4-3. The reply 38
to Question Q040.9 describes the method for determining a leak size and flow rate for each Primary Heat Transport Piping System. In inerted cells, the consequences of a given spill depend to a large extent upon the sodium temperature. In general, the location of the design basis leak for each cell was chosen as the highest temperature sodium pipe with the largest diameter and with the largest internal pressure. In one case in particular, the reactor cavity cell 101A, the location of the design basis leak in the cell was chosen as the highest internal pressure pipe (the 24" dia cold leg pipe for cell 101A) and coupled with the highest temperature of a pipe in the cell (the 1015°F hot leg pipe in cell 101A) for conservatism.

b. Cell responses to postulated sodium spills were predicted using SPRAY-I, and SOFIRE-II codes. A description of these codes is provided in Appendix A of the PSAR. Their applicability to sodium fire transient analyses, as evidenced by prediction of experimental data, has been presented in PSAR sections 6.2 and 15.6. The codes were described and their applicability to sodium spill events were discussed in the March 5, 1976 meeting with NRC.

SPRAY predicts short term temperature and pressure transients for postulated sodium sprays; SOFIRE predicts longer term effects due to pool fires in the inerted cell atmospheres. Because SPRAY and SOFIRE are not coupled codes, cell response to an event is calculated by using SPRAY AND SOFIRE in sequence, and then separately. The case yielding the highest cell gas temperature and/or pressure is used to obtain a conservative result. SOFIRE has also been used to predict temperatures in the structural concrete surrounding the cell.

The assumptions used in the SPRAY/SOFIRE calculations are outlined in the reply to Question 001.236.

c. The maximum design basis leak rate of 8 gpm at 1015°F was analyzed for the PHTS cell with the smallest free cell volume. This resulted in a pressure rise of less than 2 psi and a temperature increase of the atmosphere to 154°F (assuming a 100°F ambient). The leak rates for the remaining cells are generally much lower than 8 gpm and would result in lower pressure and temperature increases. It is expected that solidification of the sodium pool volumes would occur in all cases without damage to cell liners.

d. The cell design pressure and temperatures are shown in Table Q040.4-1. The maximum long term design temperatures for the cell liners are equal to the cell design temperature.

	Cell No.	Fluid	Source of Leak	CELL STRUCTURAL DESIGN PARMETERS: Des.Press.psig: Surge	Des.Press.psig: Long Term	Temp. °F: Design	Temp. °F: Operation	Free Cell Vol. Ft^3	Floor Area Ft^2	SODIUM SPILL DESIGN PARAMETERS: Temp. °F	Time Line Condition (1)	Detection Time Min.(4)	Time to Drain Min.	Potential Spill Vol. Gal.	Max. Leak Flow Rate GPM	Total Spill Volume Ft^3	Pool Depth Ft.
Reactor Cavity	101A	Na	24" Dia. Main Coolant Piping	35	10	100	120	53,700	1257	1015	A_1	20	176	18000	8	69.1	.05
Flowmeter Cell PHTS Loop #1	101C	Na	" " " " "	35	10	180	120	53,700	310	1015	A_1	20	176	18000	8	69.1	.22(5)
Flowmeter Cell PHTS Loop #2	101D	Na	" " " " "	35	10	180	120	53,700	310	1015	A_1	20	176	18000	8	69.1	.22(5)
Flowmeter Cell PHTS Loop #3	101E	Na	" " " " "	35	10	100	120	53,700	362	1015	A_1	20	176	18000	8	69.1	.19(5)
Overflow & Pri. Na Stg. Tank	102A	Na	6" Dia. Sch. 40 Piping	12	10	180	120	68,364	840(3)	830	A_1	975	176	32000	.15	21.1	.03
R.V. Cavity Piping Pent. Area	102B	Na	" " " " "	12	10	180	120	68,364	236	830	A_1	975	243	32000	.15	21.1	.09
Primary Na Makeup Pump	103	Na	4" Dia. Na Makeup Piping	12	10	180	120	3,422	247	830	A_1	300	292	17500	.51	23.4	.09
Future Pri. Na Makeup & Pump	104	Na	4" Dia. Na Makeup Piping	12	10	180	120	4,000	465	830	A_1	300	173	17500	.51	23.4	.05
Pri. Na Makeup Pump Valve Gallery (B)	107A	Na	4" Dia. Na Makeup Piping	12	10	180	120	40,967	365	830	A_1	300	173	17500	.51	23.4	.06
Pipeway & Valve Gallery	107B	Na	6" Dia. DHRS Cooler	12	10	180	120	40,967	450(3)	500	A_1	395	173	17500	.38	22.4	.05
Pri. Na Makeup Pump Valve Gallery (A)	107C	Na	4" Dia. Na Makeup Piping	12	10	180	120	40,967	266	880	A_1	300	173	17500	.51	23.4	.09
PHTS Loop #1	121	Na	24" Dia. Main Coolant Piping	30	10	180	120	116,292	1630(3)	1015	A_1	20	176	18000	8	69.1	.04
PHTS Loop #2	122	Na	" " " " "	30	10	180	120	109,292	1630(3)	1015	A_1	20	176	18000	8	69.1	.04
PHTS Loop #3	123	Na	" " " " "	30	10	180	120	116,292	1630(3)	1015	A_1	20	176	18000	8	69.1	.04
Nak Cooling Equipment	131	Nak	3" Dia. Sch. 40 Piping	12	10	180	120	5,239	253	150	B	3450	11	700	.04	18.5	.07
SSP Cell	132	Na	1" " " " "	12	10	180	120	1,428	116	830	A_2	11030	348	17500	.01	14.8	.13
PTI Cell	141	Na	1" " " " "	12	10	180	120	1,223	94	830	A_2	11030	348	17500	.01	14.8	.16
PTI Cell (Future)	143	Na	1" " " " "	12	10	180	120	1,487	116	830	A_2	11030	348	17500	.01	14.8	.13
Pri. Na Cold Trap (A)	157A	Na	2" " " " "	12	10	180	120	552	202	830	A_2	2500	347	17500	.06	20.6	.10
Pri. Na Cold Trap (B)	157B	Na	2" " " " "	12	10	180	120	4,475	285	830	A_2	2500	347	17500	.06	20.6	.07
Cold Trap Valve Gallery (a)	157D	Na	2" " " " "	12	10	180	120	2,227	136	830	A_2	2500	347	17500	.06	20.6	.15
Cold Trap Valve Gallery (B)	157E	Na	2" " " " "	12	10	180	120	2,075	155	830	A_2	2500	347	17500	.06	20.6	.13

38 | Notes:
1. See Table 040.4-2 or 040.4-3 for termination of leak time sequence.
2. All cells are nitrogen inerted.
3. Floor areas shown represent the areas of the lowest elevation in the cells.
4. Time to detect derived from leak detection requirements in PSAR Section 7.5.5.
5. Cell floors to slope to permit drainage back into PHTS cells.

TABLE Q040.4-1

CELL PARAMETERS FOR REACTOR CONTAINMENT BUILDING

TABLE Q040.4-2

TERMINATION OF LEAK TIME SEQUENCE

TIME LINE CONDITIONS A_1 AND A_2

Amend. 36
March 1977

SEE NOTE 1.

DETECTION - MAX. FLOW RATE

A_1 - BASED ON 100 GPM DRAIN & FREEZE VENT

A_2 - BASED ON 50 GPM DRAIN & FREEZE VENT

10 SHUTDOWN - MAX. FLOW RATE

30 REVIEW OF CAUSE - __ (MAX. FLOW RATE) /10

20 PROCEDURES - __ (MAX. FLOW RATE)/10

120 MIN. VENT (FREEZE PLUG) __ (MAX. FLOW RATE)/10

SEE NOTE 2.

DRAIN (__ (MAX. FLOW RATE)/10)

NOTES: 1. SEE TABLE Q40.4-1 FOR LEAK DETECTION TIME.

2. DRAIN TIME DEPENDS ON VOLUME IN LOCAL SYSTEM, BASED ON 50 AND 100 GPM DRAIN, WITH (MAX. FLOW RATE) 10 LEAKAGE OCCURRING DURING DRAINING, SEE TABLE FOR DRAIN TIME.

TABLE Q040.4-3

TERMINATION OF LEAK TIME SEQUENCE

TIME LINE CONDITION B

SEE NOTE 1.

DETECTION - MAX. FLOW RATE   B. - BASED ON MECHANICAL VENT

10 SHUTDOWN - 8 GPM MAX. FLOW RATE

30 REVIEW OF CAUSE - __ (MAX. FLOW RATE)/10

20 PROCEDURES - __ (MAX. FLOW RATE)/10

15 MECH. VENT __ (MAX. FLOW RATE)/10

SEE NOTE 2.

DRAIN (__ 0.8 GPM)

NOTES:
1. SEE TABLE 040.4-1 FOR LEAK DETECTION TIME.
2. DRAIN TIME DEPENDS ON VALUME IN LOCAL SYSTEM. BASED ON 50 GPM DRAIN WITH (MAX. FLOW RATE)/10 LEAKAGE OCCURRING DURING DRAINING, SEE TABLE FOR DRAIN TIME.

Question 040.5 (15.6.1)

e. Provide the results of an analysis of the pressure buildup behind the hot and cold liners as the result of sodium spills considering air-gap heat up and the possibility of gaseous releases from the concrete or other sources.

Response:

The cell liner design includes venting space behind the steel liner. The vent is in the form of a gap between the liner and the concrete, which serves to collect any gases evolved from the concrete, and vent piping which allows the excess gas to travel to a non-critical area. This venting system will be designed to limit the maximum pressure buildup behind the liners to less than 5 psi.

Question 040.6

Figures 12.-4 through 1.2-9 indicate that hot cell liners are provided for portions of the Reactor Cavity, Primary Heat Transport (PHTS) compartments, and sodium storage tank cell. Justify the partial hot liner coverage provided in these cells and the exclusion of hot liners from other cells containing sodium systems.

Response:

The current cell liner design concept is that of a fixed or "cold liner;" i.e., there will be no differentiation made between "hot" and "cold" liners.

Under the fixed liner concept, the liner is fully restrained and thermal stresses will be accommodated by inelastic strains in the liner material. The design calls for prefabricated cell wall panels consisting of a steel liner on which is placed an insulating layer of concrete. These precast composite panels will serve as wall forms for pouring the structural concrete walls. The liner will be vented by providing an air gap sufficient to allow venting.

Question 040.7 (15.6)

The introduction to Section 15.6, "Sodium spills at potential locations other than those discussed in this section have been examined; however, the results of these spills were considered to be less severe in terms of radiological consequences and cell temperature/pressure transients...". Identify these other spells and provide justification that the cell pressure temperature transient and liner thermal conditions which are "less severe." Consider, for instance, leaks in the Intermediate Heat Transport System (IHTS), or other secondary systems, as well as other locations in the PHTS within the inner-cell system. Provide justification for the exclusion or elimination of these leaks.

Response:

The design basis leak for inerted cells has been defined as described in the response to Q040.9. The project is currently defining a design basis leak rate for non-inerted cells. These design basis leaks will be used to provide the basis for cell design parameters. In addition to defining the design basis for non-inerted cells containing non-radioactive liquid metal, analyses of worst-case sodium spills will be performed to evaluate the design margin.

The criteria for determining the severity of the consequences of those spills are cell pressure, cell structural concrete temperatures and release of radioactivity to containment. Although where liners are used they are to accommodate the maximum sodium temperature of the spill, the cell design conservatively assumes that sodium/concrete reactions may occur. The cell design thus considers additional pressure buildup because of the sodium/concrete reaction and deleterious effects of the sodium/concrete reaction on the cell structure.

Question 040.8 (15.6)

Provide an analysis of the hot sodium jet spray on cold liners within the inner-cell system for any of the sodium spills postulated. Justify the assumption of maintaining liner integrity under these conditions.

Response:

The response to this question is provided in new PSAR Section 3A.8.3.3.

Question 040.9

For each sodium leak analyzed for containment/cell response, provide the method (analytical model, computer code, empirical data, etc.) used to determine the mass release rate and temperature of the leaking fluid, particularly for those accidents in which fluid system characteristics affect the release.

Response:

The sodium leaks analyzed for containment/cell response included in the PSAR were based upon an assumed 30 gpm leak. This leak rate was estimated, based upon the anticipated characteristics of the leak detection system. Since this preliminary assessment was performed, additional effort has been expended to develope a design basis leak rate of 8 gpm for the primary sodium system which has as its basis both the leak detection system, and piping and component structural integrity. Using the design basis leak rate, sodium spill design parameters have been determined for each primary sodium cell in the reactor containment. These spills were developed on the basis of a conservative, postulated design basis crack, which is derived from the following rational.

It has been demonstrated, via extensive experimental and analytical work, (see Ref. 2 of Section 1.6) that the largest credible flaw in the CRBRP primary piping will exhibit negligible growth due to the plant duty cycle. In the event that such flaws are forced to grow, by application of load cycles or load levels substantially in excess of the conservatively formulated plant duty cycle, the growth morphology is such that the crack would be expected to penetrate the pipe wall prior to the accumulation of significant crack extension. Thus, a realistic assessment of the available information leads to the conclusion that a major spill will not occur. In order to obtain design basis spills for the cells, then, it was necessary to postulate an ultra-conservative series of events considered to be incredible. The first such postulate was that a crack significantly longer and slightly deeper than the largest credible initial flaw did indeed exist in the pipe although the construction and quality assurance requirements prescribed for primary piping render such a possibility absolutely incredible. It was then assumed that the growth morphology was such that penetration would not occur until the final event in the plant duty cycle. Extensional growth of the crack was not constrained, however, and such growth was modelled, using conservative material properties throughout the complete plant duty cycle. The extended crack was then assumed to penetrate the pipe wall instantaneously, over a major portion of the extended length; the design basis crack was conservatively estimated, on the basis of the penetrated crack, to have a total length of 4 inches.

Having established the 4.0 inch crack as the source of the design basis leak, the leak rate was calculated as a function of the system temperature, pressure and pipe size using the following equation for laminar flow through the crack.

$$Q = \pi r_i^3 (2C)^4 P^4 / 8 \mu E^3 t^4$$

where: r_i = inside pipe radius

$2C$ = crack length = 4.0 inches

E = modulus of elasticity for the pipe

t = pipe thickness

P = internal pressure

and μ = sodium viscosity.

This expression for laminar flow was applied because calculated leak rates were higher than would be obtained using the turbulent flow expression presented in recent cell liner meetings with NRC and ACRS. For each spill postulated, the temperature of the leaking coolant was taken as the maximum system temperature expected. Leak volumes were calculated on the basis of the leak rate and a leak duration based upon conservative system detection and operator action times. The upper limit design basis leak rate is 8 gal/min which is assumed to continue for 30 minutes (20 minutes for leak detection, 10 minutes for operator action). (For additional information concerning the design basis leak, see "Information in Advance of CRBRP Cell Liner Design Meeting with NRC", dated June 1976).

Question 040.10 (6.2.4)

For the RAPS surge tank cell, provide a leakage test program which is in compliance with Appendix J of 10 CFR 50 or justify the exclusion of this cell from these requirements.

Response:

The RAPS surge tank has been relocated from the RSB to the RCB. The RCB has a leakage specification of 0.1% per day at 10 psid. Thus, the surge vessel cell need not have a leakage specification nor a leakage test program. The testing requirements for the RAPS Cold Box Cell are identified in PSAR Section 16.4.8. 36

Question 040.11

The following tabular information is required for the isolation system:

a. Type of valve used and location (inside/outside containment);

b. Indication of which valves will meet Type C test requirements (and justification for not requiring Type C tests on the others);

c. Line sizes;

d. Quality group (safety class) and seismic classification for all piping, valves, or other components which form the isolation system boundary;

e. Actuation signal(s) for each line;

f. Valve position with loss of actuation power and for accident conditions.

Response:

Revised Table 6.2-5 provides the information requested in items a, c, e and f. As indicated in PSAR Section 16.4.3.2, all containment isolation valves will be tested to demonstrate compliance with Type C test requirements. As indicated in Table 3.2-5, all the isolation valves, and the piping between the valves and the attachment to the containment vessel will be no less than ASME Code Class 2 (Quality Group B).

Question 040.12 (Table 6.2-5A)

It is not acceptable basis to merely state (Item 3) that the IHTS lines meet the requirements of GDC 57 for the CRBRP, and therefore, are not required to include isolation system valves. Criterion 57 specifically requires isolation valves "unless it can be demonstrated that containment isolation provision for a specific class of lines are acceptable on some other defined basis." You have not complied with your criterion as stated in Section 3.1. Provide the qualitative and quantitative information to justify that the IHTS as desinged will (1) provide a comparable solution (or leakage capability) to other systems which to include one or more isolation valves and (2) show that this design achieves greater plant safety without isolation valves.

Response:

PSAR Sections 3.1.3.5 and 6.2.4.1 have been expanded to provide the CRBRP justification for using the IHTS as an isolation boundary. 30

Question 040.16 (Table 6.2-5, 3.1)

The use of remote manual actuation as the primary actuation for the Argon Exhaust to RAPS, Nitrogen Supply Line, and Gas Supply Line (Table 6.2-5) is not in conformance with your GDC 55 (Section 3.2). Revise your design to include automatic actuation of these valves for the primary mode. The comments noted in our Item 040.12 are applicable to this issue.

Response:

NRC Questions 1.308 and 222.75 also relate to Table 6.2-5 and valve actuation. Please note that GDC 55 referenced in the question now corresponds to CRBRP Design Criterion 46.

The assignment of the Nitrogen Supply Line (Item 11 in Table 6.2-5) to 46 is erroneous. CRBRP Design Criterion 46 refers to lines which connect to the reactor coolant boundary. The Nitrogen Supply Line connects only to the containment atmosphere, and is properly classified under CRBRP Design Criteria 47. However, the required containment penetration valve and control requirements for CRBRP Design Criterion 46 and 47 are identical.

Revised Section 3.1 corrects the assignments of these lines to the proper CRBRP Design Criteria. The revised Table 6.2-5a shows the primary actuation for the "Argon Exhaust to RAPS," "Nitrogen Supply Line," and the "Argon Supply Line (70 psig)" as well as for the "Argon Supply (30 psig)," "Nitrogen Exhaust to CAPS," and the "Gas Sampling Line" to be automatic.

Question 040.17 (6.2.1.4)

Exceptions to Appendix J of 10 CFR 50 for Sections III.A.1, III.B.3, III.C.3, and III.D.2 as given in Section 6.2.1.4 which constitutes a deviation from the testing and acceptance criteria of this regulation, are not acceptable. Provide a test program in Section 6.2.1.4 which complies with the requirements of Appendix J. Note that the leakage rate must be given in terms of weight percent per twenty-four hours. The maximum leak rate should be expressed as a percentage of the weight of the original content of containment air at leakage rate test pressure; correct the PSAR as necessary to reflect this type of measurement.

Response:

Section 6.2.1.4 has been revised to:

a) withdraw the exceptions to 10CFR50, Appendix J, Sections III.A.1, III.B.3, and III.C.3.

b) justify the exception to Section III.D.2

c) discuss the test program for the containment vessel.

Question 040.18 (16.4.3)

Section 16.4.3 dealing with Containment Tests, requires additional information:

a. Specify Pa, the calculated peak containment internal pressure and La, the maximum allowable leak rate at pressure Pa;

b. Identify those portions of fluid systems which will be opened or vented to atmosphere and drained of fluids to assure that isolation valves are exposed to containment test pressure. Identify those systems which will not be vented and state the reason;

c. Identify and justify any containment penetration or isolation valve which will not be leak tested.

d. Appendix J specifies acceptance criteria for Type B and C testing (leakage from all Type B&C tests shall be less than 0.6 La). Provide an explanation of your acceptance criteria as stated in sub-section 16.4.3.2.3.

Response:

a) Pa, the peak calculated containment pressure resulting from an extremely unlikely event (sodium spill in the primary sodium storage cell) is 1.8 psig as shown in table 15.6-1 of the PSAR. However, to provide a testable pressure, Pa is specified at 10 psig. (See Section 6.2.1.2).

PSAR Section 16.4.3.2 has been expanded to respond to items b,c, and d of the question.

Question 040.19

Clarify that lines penetrating containment which are connected to the reactor coolant boundary, primary cover gas space, or inerted cell atmospheres will have the capability of periodically testing the operability and leakage of the containment isolation valves (Section 6.2.4.1.).

Response:

Seven inert gas process pipes penetrate the containment building. Two isolation valves are provided at each penetration point as shown in Figure 6.2-10 and discussed in Table 6.2-5. Ten of the isolation valves close automatically when a CIS signal is received. Four valves close on loss of line pressure. Test taps are provided outboard of each isolation valve to enable performance of leak testing. Valve position indicators on each automatic closing valve verifies operation of the valve when each valve is exercised by its respective remote valve switch located on the CIS panel.

The cover gas sampling line isolation valves will have the same leakage and operability testing capabilities as the process line valves. A typical schematic for the valve arrangements is shown in Figure Q040.19-1.

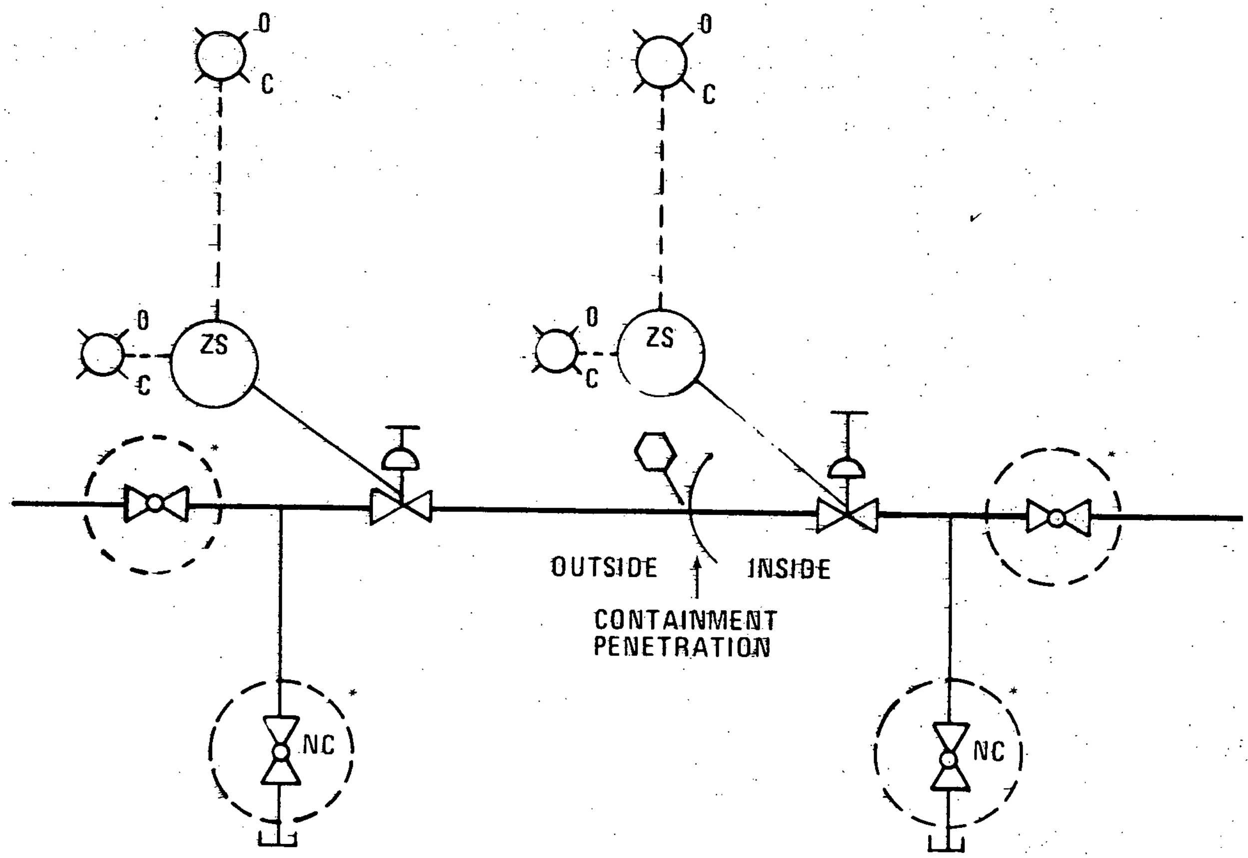

FIGURE Q040.19-1 Typical Containment Penetration Configuration

11-76-P0802-9

Question 040.20

There is no description of the containment/confinement systems. This system should be described in detail, e.g., plant areas served by the system should be described, system diagrams provided, design bases described, fan capacities and actuation times should be stated and justified.

Response:

A detailed description of the containment confinement buildings was provided in response to Question 130.74. 34

The annulus filtration system is described in Section 6.2.5, the Reactor Containment Building HVAC system is described in Section 9.6.2 and the annulus cooling system is described in Section 9.6.2.5. Table 9.6-4 includes fan capacities
49 for these systems. Figures 9.6-4 through 9.6-6 indicate areas served and flow diagrams for the above systems. A discussion of actuation time for the annulus filtration system filter fan units and justification of the capacity of the pressure maintenance fan is provided in revised Section 6.2.5.

Question 040.21

Identify all potential containment bypass leakage paths and provide an evaluation of the allowable leakage limit. Discuss the tests which will be performed to assure that the design bypass leakage limit will not be exceeded. Attached is Branch Technical CSB 6-3, "Determination of Bypass Leakage in Dual Containment Plants", which has been developed for the review of light water reactors. However, many of the positions are applicable to the CRBRP. Describe how the plant desgin meets this position.

Response:

This response addresses compliance with BTP CSB 6-3, "Determination of Bypass Leakage Paths In Dual Containment Plants" in the same order as paragraphs of the Branch Technical Position.

(1) The CRBRP confinement building is concentric to and completely surrounds all portions of the primary containment with the exception of the base mat which is embedded in soil. Therefore leak rates less than the design basis leak rate of containment can be used.

(2) The use of a pressure maintenance fan which maintains the confinement under negative pressure during plant operation precludes a period of direct leakage to the outdoor environment during startup of the fan units for the filtered ventilation system. A pressure response analysis of the secondary containment volume will be included in the FSAR and the system will be designed such that the gradual thermal transient associated with the DBA for containment will not cause a pressure less negative than -.25 in. of H_2O.

(3) The secondary containment depressurization and filtration systems will be designed in accordance with Regulatory Guide 1.52. Although not presently developed, test operation and monitoring programs for these systems will include means for determining the infiltration rate during initial testing and plant operation. The capability of the pressure maintenance fan arrangement to maintain the annulus at less than -.25 in. of H_2O during a containment isolation sequence will be verified by preoperational testing.

(4) The leakage rate of the CRBRP secondary containment has been conservatively calculated to be greater than 100 volume % per day. Fans have been sized using this leakage rate, to maintain a negative pressure of -0.25 inches water in the annulus. Since the annulus will be maintained at this pressure during all plant conditions, no exfiltration analysis is required. The infiltration rate will be verified during preoperational tests.

(5) and (6) A partial listing of bypass leakage paths is identified in PSAR Table 6.2-6. A complete listing of these paths will be provided by 1/7/77.

For each leakage path, a leakage rate has been determined in a realistic manner and the total leakage has thus been calculated. The total leakage from the containment is 0.1% per day and the allowable bypass leakage is 1.0 of the total leakage of 0.001 % per day.

Since at least 40% of the leakage paths are directly to the filtered reactor service building, a bypass leakage value of .0006 wt % per 24 hours will be used in calculating offsite radiological consequences (e.g. in responding to Question 310.43) and setting technical specification limits.

(7) All of the bypass leakage paths identified in PSAR Table 6.2-6 will be subjected to either Type B or Type C tests in accordance with 10CFR50, Appendix J. Therefore the value of bypass leakage will be tabulated from data obtained from regular containment vessel leakage tests.

(8) At the present time no air or water sealing systems or other leakage control systems are provided.

(9) The only systems for which credit is taken for being closed to the containment atmosphere to preclude bypass leakage are the sodium systems which penetrate containment and which are designed to Quality Group C Standards. These systems do not communicate directly with containment, the containment atmosphere inside containment, or the environment outside containment. They are designed to Seismic Category I standards and for temperature and pressure conditions greater than the containment vessel design conditions. In addition, these systems are protected against pipe whip, jet forces, and missiles in a manner similar to that for engineered safety features. The Project is evaluating the qualification of certain systems which are designed to Quality Group C standards as closed systems. These systems will be identified when the complete listing of bypass paths is submitted on 1/7/77.

Question 040.22 (RSP)

In response to 040.17 it is stated that the air locks will not be tested after each opening nor after every 6 months, but rather they will be tested once a year. An on-line monitoring system is proposed as justification for this exception to Part III.D.2 of Appendix J to 10 CFR 50. It is our position that the air locks be tested at 6 month intervals at the highest calculated accident pressure in accordance with Appendix J and that a detailed description of the on-line monitoring system be provided justifying exception to Appendix J's requirement for testing after each opening.

Response:

Consistent with the NRC position, PSAR Section 6.2 has been revised to eliminate the exception to the semi-annual testing of the air locks at the pressure calculated for the RCB design basis accident.

The on-line testing system for the air lock seals have not yet been designed in detail, hence a detailed description is not available at this time. The requirements specified for the design of the seals and testing systems are as follows.

The seal at each door shall be made by means of two seals with a test connection provided to allow pressurizing of the air space between the seals for leak testing. The elastomer portion of each seal will be fully molded in a continuous length to provide a continuous barrier between the door and the door frame completely around the perimeter of the door. The testing system will detect failure or degradation of any seal by leakage from the pressurized air space between it and its companion seal.

Further details of the on line testing system will be provided as they become available.

Question 040.23

The specification of containment leakage should be in terms of weight percent is as pointed out in item 040.17. Therefore, revise those portions of the PSAR which refer to containment leakage to specify weight percent leakage in accordance with the requirements of Appendix J to 10CFR50.

Response:

As can be seen from the following, a weight percentage leak rate has the same numeric value as a volume percentage leak rate.

P_t = Test pressure

$\rho_{AIR@P_t}$ = Density of air at test pressure

$$\text{Weight Percent} = \frac{\text{Weight of air leaked at test pressure}}{\text{Weight of air in entire vessel at test pressure}}$$

$$= \frac{(\text{Volume percent})\ (\text{Containment vessel volume})\ (\rho_{AIR@Pt})}{(\text{Containment vessel volume})\ (\rho_{AIR@Pt})}$$

Weight Percent = Volume Percent

Sections 16.4.3.2 and 6.2.1.5 have been revised to specify the containment leakage rate in weight percent.

Question 040.25

Your response to item 040.12 is unacceptable. The NRC CRBRP Design Criterion 48 does not contian the statement "unless the boundary is protected against accidents, extreme environmental conditions and natural phenomena". Therefore, in accordance with Criterion 48, justify the acceptability and desirability of using the IHTS as an isolation boundary as originally requested in item 040.12.

Response:

PSAR Sections 3.1.3.5 and 6.2.4.1 have been expanded to provide the CRBRP justification for using the IHTS as an isolation boundary. The previous response to Question 040.12 has been modified to reference the sections describing this justification.

Question 040.26

Reference the system design which shows the isolation arrangement for each line penetrating containment or alternatively provide an isolation valve arrangement diagram for each penetration.

Response

Revised Table 6.2-5 and Figure 6.2-10 show the isolation arrangement for each line penetrating containment.

Question 040.27

Provide justification that the entire confinement annulus can be maintained at a minimum of 1/4" W.G. negative pressure considering the effect of the annulus partitions which have been added as part of the Third Level Margins Systems.

Response:

The response to this question is provided in revised PSAR Section 6.2.1.2.

Question 040.28

Provide the basis which justifies the assumption that the containment isolation system (valves, operators, instrumentation and control components) will be capable of performing its intended function in the containment environment associated with your proposed TLTM scenario. Specifically address the capability of the containment isolation system to maintain the assumed leak tightness under the calculated environment conditions. Include a description of the environmental tests planned to verify the performance of this equipment identifying the key parameters, such as pressure, temperature, chemical composition, and any other which may affect system functionability.

Response:

The assumption that the containment isolation system will be capable of performing its intended function in the containment environment associated with the TMBDB (TLTM) scenario is based on the following:

1. Containment isolation is initiated upon detection of radioactivity in the Head Access Area and/or the exhaust duct of the containment ventilation system (see PSAR Section 7.3). The Reactor Containment Building will be isolated, as for a design basis accident, long before an accident is recognized by the operator as a TMBDB event. Environmental conditions at the time of containment isolation for a TMBDB event are the same or less severe than those encountered for design basis accidents (see Section 2.2.11 of CRBRP-3, Volume 2 (Reference 10b, PSAR Section 1.6)). Environmental qualification requirements of the Containment Isolation System for design basis accident environments are discussed in WARD-D-0165 (Reference 13, PSAR Section 1.6).

2. Valves are commercially available to withstand the temperature and pressure exposure at their physical location during the TMBDB scenario.

No environmental qualification tests are planned under the TMBDB environment for the containment isolation system. This is justified on the basis that the TMBDB environment is not applicable for the initiation of the isolation as described above and that design basis accident qualification for the containment isolation valves will envelope the expected TMBDB conditions.

Question 3

Since the CRBRP fuel will be of relatively low density, the subject of fuel densification must be addressed. Emphasis should be placed on experimental data that relates directly to pellet dimensions rather than just microstructural appearance.

Response:

The information requested has been provided under separate cover in the topical report WARD-D-0168, "Impact of Fuel Densification on CRBRP Fuel Performance."

Question 110.1 (3.6)

In section 3.6 of the PSAR, the statement is made that even though spontaneous ruptures of the sodium piping are not considered to be credible, the intent of Regulatory Guide 1.46 will be met by analyzing selected runs of sodium piping to identify postulated break locations. Provide the basis for these selected locations if the criteria for postulating pipe breaks differs from the intent of Regulatory Guide 1.46 which is applicable to light water reactors.

Response:

The information requested is incorporated in the revised Section 3.6.

Question 110.2 (3.6.2)

The pipe break criteria in Section 3.6.2 may not be completely acceptable. Provide more specific criteria. If the pipe break criteria for the Clinch River Breeder Reactor plant differs from Regulatory Guide 1.46, provide the justification for these differences. In addition, clarify whether the criteria in this section is applicable to sodium piping as well as all ASME Class 1, 2 and 3 high-pressure steam and water piping inside containment.

Response:

The information requested is incorporated in the changed PSAR page 3.6-2.

Question 110.3 (3.6.4):

Section 3.6.4 is not completely acceptable. Provide the basis for the statement that a single degree-of-freedom model is conservative for a pipe whip analysis. Specify your criteria and formulate a dynamic model for both the pipe and pipe restraints. State system conditions, pipe and pipe restraint boundary conditions, forcing function time histories and, if applicable, any impact and rebound. State if the analysis will be elastic or inelastic. If inelastic, verify deformation compatibility. An acceptable guide for the dynamic analysis is given in NRC Standard Review Plan, section 3.6.2.

Response:

Revised Section 3.6.4.1 provides a description of methods and criteria to be used in pipe whip analysis.

Question 110.4 (3.6.5.1)

Clarify the design criteria for the design of pipe whip restraints which is discussed in Section 3.6.5.1 (3) of the PSAR. To be acceptable, the design strain limits for restraints should not exceed .5 of the ultimate uniform strain of the materials of the restraints.

Response:

This information can be found in revised Section 3.6.5.1. 25

Question 110.5 (3.9.1.6)

With respect to Section 3.9.1.6(2), identify and provide the basis for any deviations from Code Case 1592.

Response:

The information requested is incorporated in the changed PSAR page 3.9-3.

Question 110.6 (3.9.2.2)

The design loading combinations for ASME Class 2 and 3 components listed in Table 3.9-2 are not completely acceptable. Acceptable loading combinations are:

Upset Condition - Pressure + Dead Weight + OBE + Transients Associated With the Upset Condition

Faulted Condition - Pressure + Dead Weight + SSE + Transients Associated With the Faulted Condition.

Response:

Revised Section 3.9.2.2. contains the required information. 25

Question 110.7 (3.9.2.4)

Verify that the information presented in Section 5.3.2.1.2 - Design of Active Pumps and Valves, apply to ASME Class 2 and 3 active pumps and valves as well as Class 1.

Response:

The information provided in Section 5.3.2.1.2 applies only to ASME Class 1 active pumps and valves. For ASME Class 2 and 3 active pumps and valves, the information given in Section 3.9.2.4 and Table 3.9-3 will apply.

Question 110.8 (3.10)

IEEE-344-1971, "Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations" which is referenced in Section 3.10 of the PSAR is not completely acceptable. Acceptable criteria is contained in IEEE-344-1975 with the following exceptions:

(a) The use of a factor of 1.5 in Section 6.6.2.1 of IEEE-344-1975 for single frequency test and in Section 5.3 for static coefficient analysis to account for multi-frequency excitation and multi-mode response should not be construed as being acceptable in the absence of justification, since these provisions are inconsistent with the general requirements as stated in Section 6.6.1 requiring that the RRS envelope the TRS.

(b) The use of sine sweep testing in Section 6.6.2.5 of IEEE-344-1975 for equipment seismic qualification should not be construed as being acceptable in the absence of justification, since sections 6.6.2 and 6.6.2.1 do not provide specific guidelines concerning a justifiable methodology to define the TRS for a sweep input motion and to quantify the multi-frequency effects. State your intent to use criteria which is consistent with (a) and (b) above.

Response:

IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations" will be used as described in Section 3.10.1 for seismic qualification by testing of Class 1 electrical equipment.

In response to item (a) and (b) above, Section 3.10.1 has been revised. 25

Question 110.9 (5.3.2.1.2)

The information presented in Section 5.3.2.1.2 of the PSAR is not completely acceptable for the design of active pumps and valves within the Primary Heat Transport System. Acceptable criteria are contained in NRC Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures".

Response:

Additional information in response to the above question has been added to Section 5.3.2.1.2.

Question 110.10 (5.0)

Provide the criteria utilized to ensure that all supports for Category I ASME Class 1, 2 & 3 active components will be designed so that they will not deform to the extent that would impair the required operability of the active component, e.g., the specification of maximum allowable support deformation limit for the most adverse loading conditions in the design specification for each active component.

Response:

The criteria for stress limits for ASME Class 1, 2 & 3 active components and their supports to assure operability are contained in WARD-D-0174 "CRBRP; Active Pump and Valve Operability Verification Plan".

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Question 110.11 (5.0)

Provide a list of loading combinations which will be used in the design of all ASME Class 1 systems, components, equipment and supports in the Heat Transport and connected Systems.

Response:

The information requested is contained in revised Section 5.3.1.1 of the PSAR.

Question 110.12 (3.6)

To be acceptable, the PSAR should present the specific criteria which will be used to postulate pipe break location outside containment and for those portions of piping which penetrate containment. Acceptable criteria are contained in Attachment A.

Response:

The specific criteria which will be used to postulate pipe break location outside containment and for those portions of piping which penetrate containment is provided in revised section 3.6 (see response to question 110.1).

Question 110.13 (3.9.1.1)

To be acceptable, the PSAR should present a description of the pre-operational piping vibrational and dynamic effects tests which will be conducted on all ASME Class 1, 2 and 3 safety-related piping. Acceptable criteria are contained in NRC Standard Review Plan, Section 3.9.2, Dynamic Testing and Analysis of Systems, Components and Equipment.

Response:

Regulatory Guide 1.70 requested that this information be provided in the FSAR, and it is presently planned to supply this information in the FSAR. However, PSAR Section 3.9.1.1 has been expanded to provide information on the preoperational tests. 25

Question 110.14 (3.9.1.2)

To be acceptable, the PSAR should present a description of the seismic qualification program which will be employed to qualify all safety-related ASME Class 1, 2 and 3 mechanical equipment. Acceptable criteria are contained in NRC Standard Review Plan, Section 3.9.2.

Response:

Regulatory Guide 1.70 requested that this information be provided in the FSAR and it is presently planned to supply this information in the FSAR. However, Section 3.9.1.2 has been expanded to provide additional detail on the seismic qualification program. 25

Question 110.15 (3.9.1.3)

To be acceptable, the PSAR should present the dynamic system analysis methods and procedures which will be used to determine the dynamic responses of reactor internals and associated Class 1 components of the Heat Transport and Connected Systems which have an effect on the responses. Acceptable criteria are contained in NRC Standard Review Plan, Section 3.9.2.

Response:

The requested description of analysis methods and procedures are provided in revised Section 3.9.1.3. 25

Question 110.16 (3.9.1.6)

Provide, in the PSAR, the information required in Section 3.9.1.6 of the Standard Format and Content,LMFBR Edition. Specifically, if inelastic design stress analyses and inelastic design stress limits are used in conjunction with an elastic dynamic system analysis, provide the bases upon which these procedures are used. Acceptable criteria are contained in NRC Standard Review Plan, Section 3.9.1, Special Topics for Mechanical Components.

Response:

The information requested is incorporated in the changed PSAR page 3.9-3 and 3.9-3a.

Question 110.17 (3.9.2.5)

Provide in the PSAR, the information requested in Section 3.9.2.5, "Design and Installation Criteria, Pressure-Relieving Devices" of the Standard Format and Content, LMFBR Edition.

Response:

Section 3.9.2.5 has been modified in response to this question.

Question 110.18 (5.2.1.22)

Provide in the PSAR, the information requested in sections 3.9.2.7 and 5.2.1.22, "Field Run Piping" of the Standard Format and Content, LMFBR Edition.

Response:

There is no field run piping in the PHTS nor IHTS as indicated in Sections 5.3.2.3.4 and 5.4.2.3.3, respectively.

Revised Sections 5.5.2.3.3 and 5.6.1.2.3.4 provide the requested information for the SGS and SGAHRS, respectively.

Question 110.19 (3.6, 3.6.1, 3.6.2)

Your intent to use the J.F. O'Leary Letter of July 12, 1973, for pipe break locations, break sizes and orientations for systems inside and outside containment is not completely acceptable. State your intent to comply with the latest criteria as specified in NRC Standard Review Plan (SRP) 3.6.2 "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" for all systems in which breaks are postulated. With respect to SRP 3.6.2, include the following and provide illustrations as applicable:

1) Provide the details to be used for piping penetrations of containment areas. Indicate the use of protective assemblies or guard pipes. State whether such protective assemblies serve to provide an extension of containment, prevent overpressurization, or provide both functions.

2) Indicate the use of moment limiting restraint's at the extremities or within the penetration assembly.

3) Provide the criteria for the design of the process pipe within the penetration assembly. Include type of material (Seamless or welded), allowable stress level, and loading combinations.

4) Provide the design criteria to be used for flued heads and bellows expansion joints.

5) Provide the design criteria applicable to any guard pipe which is utilized with the assembly.

Response:

CRBRP compliance with the J. F. O'Leary letter was discussed with NRC as documented in Reference Q110.19-1. The response to question 020.4 will further document the CRBRP compliance with the intent of the NRC position.

In response to the specific concerns identified, there are no high energy piping penetrations in the RCB; however, the following discussion is provided for the IHTS containment penetrations:

1) Figures Q110.19-1 shows the typical arrangement of the pipe penetration. The boundary of the containment shell is evident from the figure.

2) There are no pipe rupture restraints for the intermediate bay side of the penetration.

3) Design loading for the penetration provides for the maximum forces and moments which could be imposed by the pipe. The intermediate pipe is 316 SS in hot leg and 304 SS in the cold leg. There will be a transition weld between the carbon steel containment penetration and the 316 or 304 SS adaptor. The integrity of the assembly will be demonstrated against sodium spray by analysis demonstrating conformance to the ASME Section III limits.

4) The design criteria for the flued head is ASME Section III, Class 1.

5) There is no guard pipe in the conceptual design (as evident from the attached figure).

The following discussion is provided for the compressed gas, chilled water and drain system piping penetrations:

(1) Figures Q110.19-2 shows the typical arrangement of the pipe penetrations which serve to provide an extension of the containment.

(2) These piping systems are considered moderate energy systems and moment limiting restraints for pipe rupture loadings are not required. However, pipe stops/guides are provided within the penetration assembly to limit moment loadings on the closure due to other normal design conditions.

(3) Process pipe within the penetration assembly will meet the allowable stress levels and loading combinations as required by ASME Section III.

(4) Bellows expansion joints and closure plates (or flued heads) will meet the design criteria of ASME Section III, Subsection NC.

(5) Guard pipes will not be used..

The detailed configurations of other penetrations have not yet been determined, but will be similar in concept to the IHTS penetrations in that (1) a cylinder is welded to the containment vessel; (2) the process piping passes through the cylinder without guard pipes or piping rupture restraints; (3) the design criteria for the penetration will be at least as conservative as the criteria of ASME Section III, Class 2; (4) no expansion joints (bellows) will be used except as necessary for testing purposes.

Reference Q110.19-1:

Letter S:L:653, P. S. Van Nort to R. S. Boyd, "Summary of CRBRP/NRC Meeting on Pipe Breaks Outside of Containment," March 3, 1976.

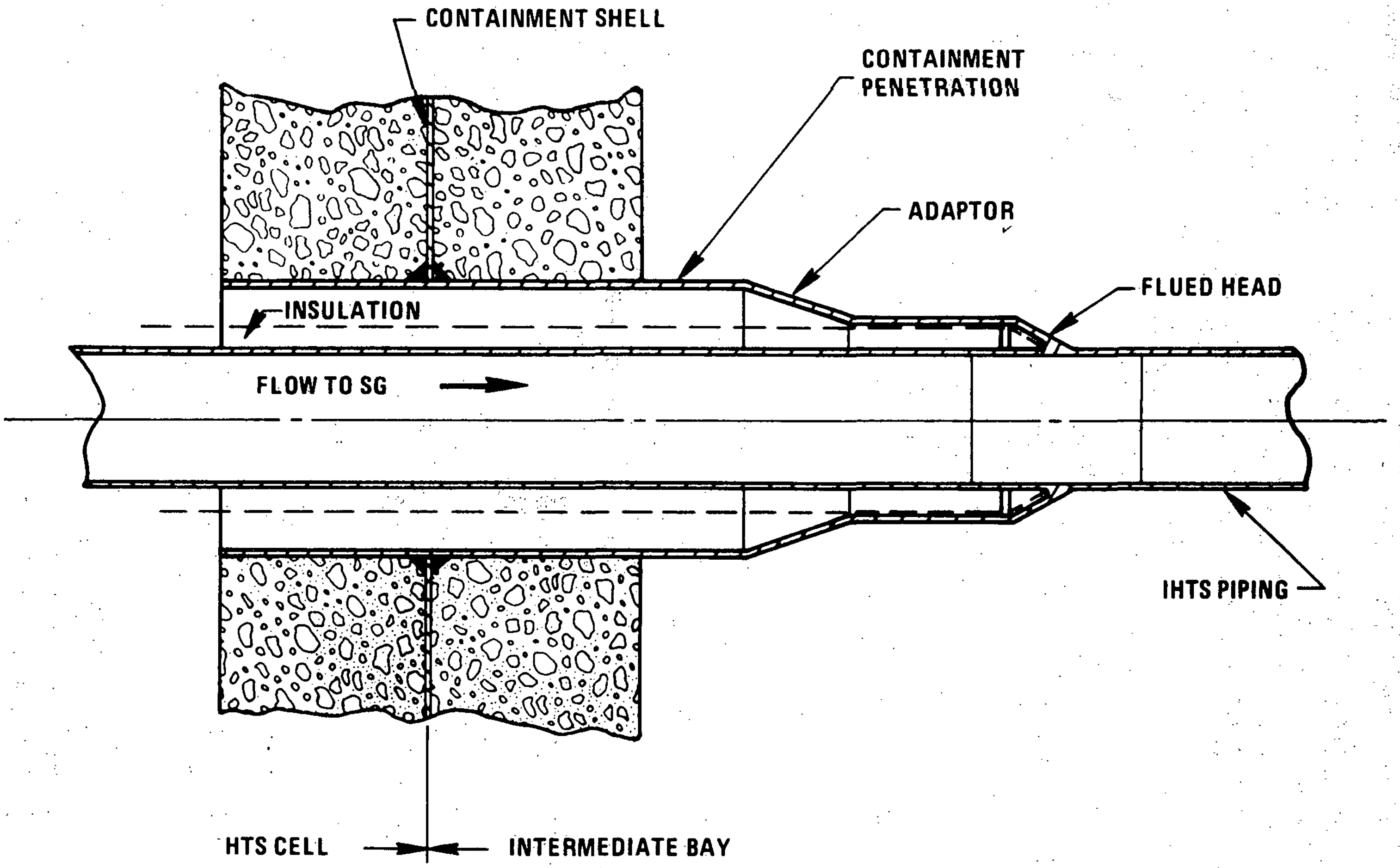
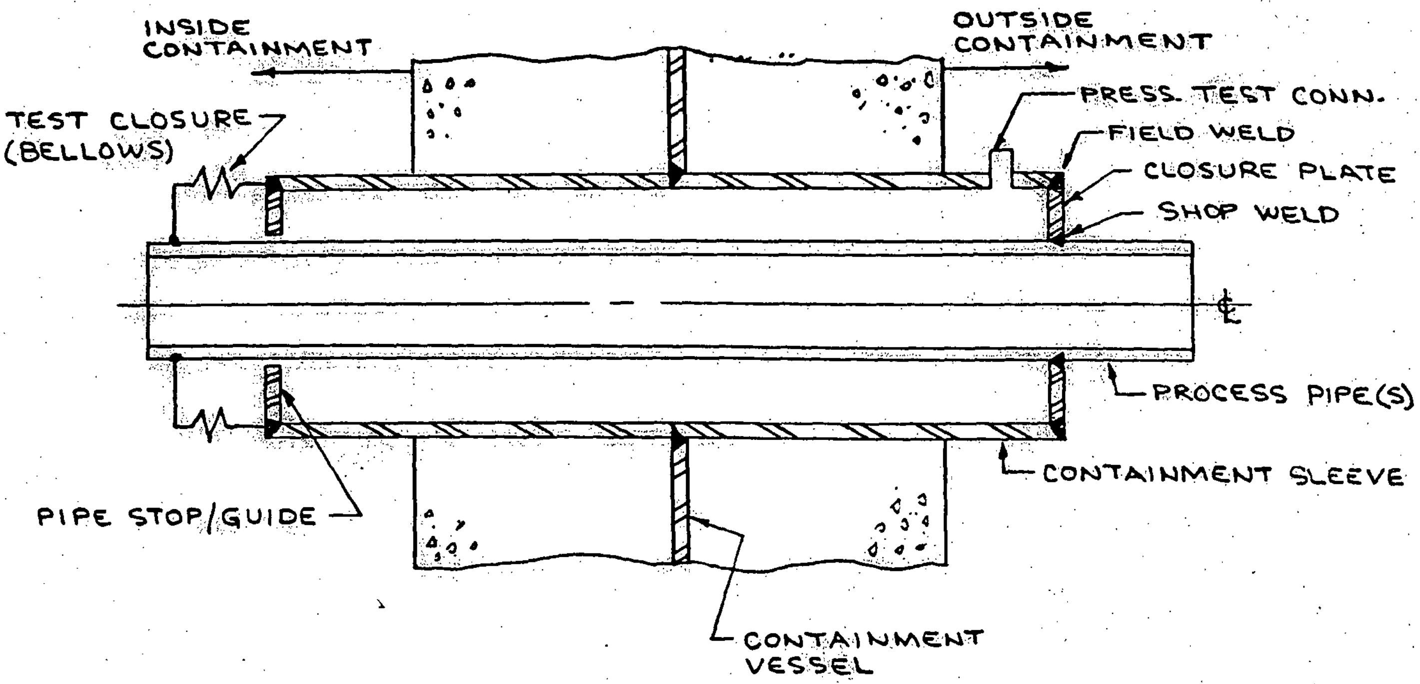
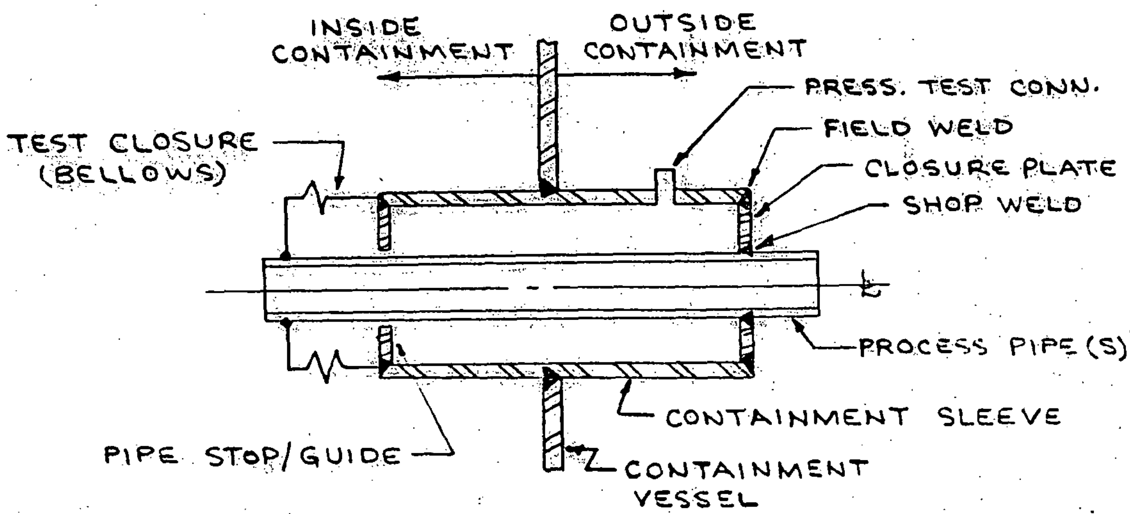


Figure Q110.19-1. Pipe Penetration Arrangement



Typical Below Elev. 816'-0"



Typical Above Elev. 816'-0"

Figure Q110.19-2

Pipe Penetration Arrangement For Pipe Design Temperature $\leq 200^{o}$F.

Question 110.20 (3.6.5.1)

State your assumptions regarding the motion of the free ends of the pipe at a postulated circumferential break. Include in your discussion the possible effects of support reaction forces and moments (even though wave phenomena are involved) and the possible effect of the three-dimensional nature of system flexibilities in addition to the effects of direction and point of application of the driving force(s) and of the inertia properties of the piping system. Provide assurance that all these effects are taken into account where necessary in estimating dynamic loads and directions following a postulated pipe break.

Response:

Section 3.6.4.1 previously revised in response to Question 110.3 has been further revised in response to this question.

Question 110.21 (3.6.3)

Provide assurance that in developing the design loading combinations in Section 3.6.3 of the PSAR you have considered forces and moments resulting from internal and external system asymmetrical pressures + seismic forces + deadweight.

Response:

Asymmetrical pressures, deadweight and seismic forces will be considered in the analyses of events involving postulated pipe leaks and ruptures. Details of the manner in which seismic forces are combined with other loadings are provided in revised PSAR Section 3.9.

Question 110.22 (3.9.12, 3.10)

Expand Tables 3.2-2 and 3.2-3 to show the expected method of seismic qualification (test or analysis) for both the NSSS and BOP supplied Category I mechanical and electrical equipment.

Response:

Tables 3.2-2 and 3.2-3 have been revised to indicate the expected method of seismic qualification of the components.

Question 110.23 (3.9.1.3)

The response to Question 110.15 in the PSAR presents information which is applicable to Section 3.9.1.5 of the PSAR rather than Section 3.9.1.3. Although Regulatory Guide 1.70 does imply that the information in Section 3.9.1.3 be supplied at FSAR, the NRC staff has determined (subsequent to the publication of Regulatory Guide 1.70) that this information is required in the PSAR.

(1) In Section 3.9.1.3, provide a discussion of the program planned for the Clinch River Breeder Reactor Plant (CRBRP) which will verify the structural integrity of the reactor internals due to flow-induced vibrations prior to commercial operation of the plant.

NRC Standard Review Plan, Section 3.9.2, "Dynamic Testing Analysis of Systems, Components and Equipment" outlines acceptable criteria in Paragraphs I.3, II.3 and III.3 of that document.

(2) In Table I of Section 1.1.3 of the PSAR, the statement is made that the intent of NRC Regulatory Guide 1.20 is applicable to the CRBRP, but the testing details given in that document are not appropriate to LMFBR's. Identify and justify the testing details which are not applicable to LMFBR's. Provide a discussion of a testing program consistent with the intent of Regulatory Guide 1.20 and which will be applicable to the CRBRP.

Response:

In response to this question, a discussion of the testing program planned for the CRBRP has been provided in the expanded PSAR Section 3.9.1.3. 25

Question 110.24 (3.9.1.5)

Provide the technical bases to establish the precedent for using the "non-mandatory guidelines" for coupling of mathematical models given in Table 3.9-1.

Response:

See revised PSAR Section **3.9.**

Question 110.25 (3.9.1.5)

Provide a specific and detailed explanation of the manner in which individual loads, in a design loading combination, such as those corresponding to LOCA and SSE are combined. That is, provide the method for combining LOCA and SSE loads as they affect the design of seismic Category I items, including the effect upon the calculated dynamic response of the reactor internals to flow-induced and inertia-load-induced excitation.

Response:

The requested information is provided in revised Section 3.9.1.5.

Question 110.26 (3.9.2.4, 5.3.2.1, 5.4.2.1, 5.5.2.1, 5.6.2.2)

(1) The response to Question 110.9 and the information in Sections 3.9.2.4, 5.3.2.1.2, 5.4.2.1.2, 5.5.2.1.2, and 5.6.2.2.1.2 is unacceptable. Provide additional information on the operability programs of all ASME Class 1, 2, and 3 active pumps and valves. Acceptable criteria are contained in the NRC Standard Review Plan, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures." An acceptable program is contained in the Catawba Nuclear Station (Docket No. 50-413/414) PSAR, Amendment 15.

(2) The design limits discussed in Sections 5.3.1.2, 5.4.1.2, and 5.5.1.2 are not acceptable for ASME Class 1 active pumps and valves. Provide design limits for all Class 1 active pumps and valves subjected to the component normal, upset, emergency, and faulted operating conditions which are at least as conservative as those in Table 3.9-3 for ASME Class 2 and 3 active pumps and valves.

Response:

The design, fabrication and testing requirements imposed on Class 1, 2 and 3 active components are discussed in WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification Plan".

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Question 110.27 (3.9,4.1.7,5.2):

Provide a list of computer programs that will be used in dynamic and static analyses to determine structural and functional integrity of all Seismic Category I systems, components, equipment and supports. Include a brief description of each program, the extent of its application and the design control measures that will be employed to demonstrate the applicability and validity of each program. Guidance is provided in NRC Standard Review Plan, Section 3.9.1.

Response:

The analytical computer programs that will be utilized for the static and dynamic analyses of Seismic Category I structures listed below are discussed in revised Appendix A to the PSAR:

ANSYS
ASHSD2
CHERN
CREEP-PLAST
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DRIPS
DUNHAM's
DYNAPLAS
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ELTEMP
E0984A
E1682A
FBRDSAP
FESAP
GASP
GSAP4
HAPII
HYTRAN
KALNINS
MARC
NASTRAN
NONSAP
SAPIV
SAP4GE
SPECEQ/SPECUQ
SUPERPIPE
WECAN
WESTDYN

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Question 110.28 (3.9.2.2)

Section 3.9.2.2 in the PSAR references the ASME Section III Code and Code Case 1606 for design requirements for piping. These references do not define the loading combinations for the various component operating conditions. Provide the loading combinations for all ASME Class 2 and 3 piping subjected to the normal, upset, emergency and faulted component operating conditions (Reference the Regulatory Position on Code Case 1606 in NRC Regulatory Guide 1.84, "Code Case Acceptability, ASME Section III Design and Fabrication").

Response:

Tables 3.9-2A and 3.9-5 have been added and Sections 3.9.2.2 and 3.9.2.3 have been revised in response to this question.

Question 110.29 (3.9.2.3)

The design limits for Code Class 2 & 3 (non-active) valves, given in Table 3.9-3 are not acceptable. Provide stress and pressure rating limits which are consistent with or no less conservative than those given in ASME Code Case 1635.

Response:

The response to Question 110.10 includes a revised Table 3.9-3. Table 3.9-4 has been added and Section 3.9.2.3 has been revised to indicate pressure limits.

Question 110.30 (3.9.2.4)

Provide a sketch and a description of the design concept of the main steam isolation valves, particularly a discussion of the valves' design adequacy to withstand the loading effects of fast closure.

Response:

Section 5.5.2.3.1 and Section 10.3.1 have been modified, and Figure 5.5-2A has been added in response to this question. Note that reference to the main steam isolation valve has been removed from Section 10.3.1 since the Main Steam Supply System covers only components and piping downstream of the steam piping anchor at the steam generator building penetration. Also note that the impact loading at valve seating is limited by controlling the rate of valve closure, as noted in revised PSAR Section 5.5.2.3.1.

Question 110.31 (3.9.2.5 and 5.5.2.4)

The information presented in Sections 3.9.2.5 and 5.5.2.4 concerning design and installation of overpressure protection devices is not complete. Acceptable criteria for open systems is contained in NRC Regulatory Guide 1.67, "Installation of Overpressure Devices". Provide criteria for all ASME Class 1, (if any) 2 and 3 pressure relieving devices which is consistent with Regulatory Guide 1.67.

Response:

Sections 3.9.2.5 and 5.5.2.4 have been modified in response to this question. The steam generation system has no ASME-III, Class 1 pressure relieving valves.

Question 110.32 (4.2.2.2)

The mechanical loads part of Section 4.2.2.2.1 is not acceptable. Operational transient upset, emergency, and faulted mechanical loads acting on the reactor internal structures must be considered in conjunction with seismic loads. Specify the transient loads that will be applicable for these design conditions.

Response:

For the design of the reactor internal structures, seismic loads (Mechanical Loads) are considered in conjunction with any concurrent and appropriate transient loads (Pressure and Temperature Loads) in accordance with the requirements as set forth in the ASME-III Code and the applicable Code Cases. The pressure and temperature loads are discussed in detail in Section 4.2.2.1.3 of the PSAR. Loading combinations of transient loads with seismic loads are discussed in revised PSAR Section 3.9.1.5.

Question 110.33 (4.2.2.2)

Provide a statement that the design criteria for the core support structure under all operating conditions are no less conservative than those in Subsection NG and Appendix F (F-1380) of the Code. Include criteria for items in compression where the instability load may be a significant factor. State the type of analysis from which the component stresses will be derived.

Provide assurance that the deformation limits applied to the design of the reactor internals are sufficiently conservative to prevent interference with the functioning of all related components, e.g., control rods and standby cooling systems under all plant operating conditions, including all faulted conditions. Include a commitment to provide in the FSAR a numerical comparison between calculated or test-determined displacement and the allowable displacements to verify an adequate margin.

Response:

The statement requested is found in revised Section 4.2.2.3. The information requested is located in revised Sections 4.2.2.1.1.1, 4.2.2.1.1.8, 4.2.2.1.2.1, 4.2.2.1.2.7, 4.2.2.1.2.8 and 4.2.2.3.1. Control rod system maximum misalignment sources for the reactor refueling and operating conditions are shown in Figures 4.2-95A and B. |44 |48

Criteria for compressive loads and the type of analysis employed is discussed in PSAR Section 4.2.2.4.1.1.

Final evaluation of each of the reactor internals components will include displacement calculations for all operating conditions in the stress reports, and will be shown to satisfy design limits.

Question 110.35 (5.1.2)

Figure 5.1.3 indicates several bellows are used in the PHTS. State the design criteria that will be used to design PHTS and all other safety-related cooling system bellows. Indicate the analytical procedures to be used and summarize the experimental programs that will be adopted to verify bellows integrity.

Response:

The information requested for the PHTS bellows is provided in the response to Question Q110.75. 31 33

The information requested for the IHX bellows is provided in revised Section 5.3.2.3.2.

Question 110.36 (5.2.4.3)

In Section 5.2.4.3 it is stated that hollow tubes are used to adjust the peak pressure pulse imparted to the closure head. State the design criteria that will be used for these tubes. Indicate how the pressure distribution in these tubes will change the total pressure impulse on the head. Indicate how this change in impulse will influence the safety of the vessel head design. Identify the inelastic structural concept that may be substituted for the crush tubing.

Response:

Revised Section 5.2.4.3 provides the information requested. 25

Question 110.37 (5.3.1.1)

The loading combinations for Class 1 systems and components given in the seismic loads paragraph of Section 5.3.1.1 in response to Question 110.11 are not consistent with current NRC acceptable loading combinations.

Acceptable loading combinations for all ASME Class 1, 2 and 3 systems, components, equipment and supports (including reactor internals) are:

1. For the component upset condition:
   Sustained Loads* + Transient Dynamic Loads associated with the upset condition + OBE.

2. For the component emergency condition:
   Sustained Loads* + Transient Dynamic Loads associated with the Emergency condition.

3. For the component faulted condition:
   Sustained Loads* + Transient Dynamic Loads associated with the Faulted condition + SSE.

*Sustained Loads are loads such as pressure (for the applicable component operating condition), dead-weight, live loads and thermal loads.

Revise Section 5.3.1.1 to be consistent with the above loading combinations. Identify the events included in the transient dynamic loads associated with the applicable component operating condition.

Response:

The requested information has been included in the amended PSAR Section 5.3.1.1 and revised PSAR Section 3.9.

Question 110.38 (5.3.1.1)

In the seismic part of Section 5.3.1.1, it is stated that the OBE will be included in the Design Mechanical Loads for low temperature Section III design but not for elevated temperature design in accordance with the high temperature Code Case 1592. This is unacceptable to the staff. The OBE must be considered for elevated temperature design. Revise Section 5.3.1.1 to be consistent with this criteria.

Response:

The use of OBE loadings in elevated temperature design is discussed in revised PSAR Section 5.3.1.1.

Question 110.39 (5.3.2.3)

It is stated in Section 5.3.2.3.1 that it will be demonstrated analytically that failure of any part of the pump rotating assembly will not affect the integrity of the pressure boundary. Provide details on how this demonstration will be performed. If by analysis, provide the analytical methods to be used.

Response:

The response to this question is provided in revised Section 5.3.2.3.1.

Question 110.40 (5.3.3.1)

It is stated in Section 5.3.3.1.1 of the PSAR that the results of manufacturer's computer programs will be verified to assure their accuracy and applicability. An acceptable method of verifying the adequacy of computer programs is described in the Regulatory Standard Review Plan 3.9.1. Discuss a proposed program to verify your computer program.

Response:

The codes used for analysis of CRBRP components are listed in Appendix A of the PSAR. In response to question 110.27, a discussion of the verification of each code used for structural analysis was provided in Appendix A.

Assurance of verification of each code used in analysis of a component is the responsibility of the vendor or Reactor Manufacturer, whichever performs the code structural analysis of that component. Vendor code verification is subject to the approval of the purchasing organization.

Question 110.41 (5.5.3.3.1)

Section 5.3.3.1.5 is not adequate. Indicate how creep-fatigue will be considered in the pump and valve analysis. Indicate how time dependent distortions of the pump and check valve will affect their operation. If computer programs are to be used for creep-fatigue analysis state their title and indicate how they will be verified.

Response:

The response to this question with respect to the HTS pumps is included in the response to Questions 110.58 and 110.27.

The valve operating temperature is below 800°F, and is not identified as a component requiring creep-fatigue analysis. For those upset, emergency, or faulted events leading to some time of the check valve above 800°F, Code Case 1592 will be applied. The code case is clear in its application, but the detailed analyses depend upon the actually encountered conditions, and cannot be predefined.

Question 110.42 (5.3.3.1) Pg. 5.3-34

In discussing screening analysis in Section 5.3.3.1.1 of the PSAR, it is stated that an approximate inelastic analysis is a one-dimensional approximation of two and three-dimensional geometries. Provide supporting evidence to demonstrate the accuracy of this approximation.

Response:

The PSAR states that simplified inelastic analyses include one-dimension analysis to approximate the inelastic behavior of two and three-dimensional geometries. Other simplified inelastic methods, such as the Bree or O'Donnell-Porowski Techniques (Reference RDT Standard F9-5T), are pseudo-elastic in that elastic analysis results are used to predict inelastic structural behavior.

Simplified inelastic analysis techniques, as described above, may be used to determine conservative bounds on strains. The determination of the applicability of these methods, and the verification of the results, are the responsibility of the component designer or user. Therefore, the use of these methods must be fully justified by the user in his formal stress reports or other appropriate documentation. It is not possible to create general supporting evidence apart from the specific applications.

Question 110.43 (5.3.3.1)

Section 5.3.3.1.2 is not completely satisfactory. Identify the mechanical components and equipment requiring detailed creep analysis.

Response:

Analysis of any component which has an operating temperature above 800°F must account for the time dependent materials properties. Components in the primary heat transport system which operate above 800°F are the hot leg piping, pump, and IHX.

PSAR Section 5.3.3.1 has been revised to so indicate.

Question 110.44 (5.3.3.1)

In the discussion of pumps in Section 5.3.3.1.5 it is stated that some computer programs have inelastic capabilities. Identify the programs having creep (visco-elastic) capabilities which you intend to use.

Response:

More specific information on the codes to be used for pump analysis is provided in revised Section 5.3.3.1.5.

Question 110.45 (5.3.3.6)

In Section 5.3.3.6.1.1, it is stated that pressure surge, vibration, and temperature fluctuation effects will be assessed along with the usual load effects. State under which component operating condition (upset, emergency, or faulted) the pressure surge condition will be included.

Response:

The only identified cause for a pressure surge in the primary system would be a check valve closure, following a pump mechanical failure. For design purposes, the pump mechanical failure event is treated as an emergency event In the plant duty cycle (See Appendix B).

Question 110.46 (5.3.3.6)

In this section, it is stated that stresses due to the specified temperature gradients are defined by

$$\delta_1 = \frac{E\alpha\Delta T_1}{2(1-\nu)}$$

The resultant stresses will exist only in fully restrained elements, and do not account for thermal distortions. Provide the technical basis for using this equation in view of possible thermal distortion.

Response:

Revised Section 5.3.3.6.1.1 provides the information requested. 25

Question 110.47 (5.4.2.3)

In Section 5.4.2.3.1 it is stated that the peak pressure load in conjunction with the possible dynamic response will be treated as a static primary load set. Provide the technical basis for making this statement.

Response:

Section 5.4.2.3.1 has been modified to clarify that the use of a static analysis is for scoping purposes only.

Question 110.48 (5.5.1.1)

In Section 5.5.1.1 it is stated that the SWRPRS design will consider a full guillotine rupture of a heat transfer tube in the most unfortunate location in the unit which causes the guillotine rupture of six additional tubes. Provide the technical basis for selecting six additional tubes. Discuss the probability of more than six tubes experiencing a guillotine rupture.

Response:

The evaluation of steam generator heat transfer tube leaks is provided in previously revised Section 5.5.3.6.

Question 110.49 (5.5.2.4)

Provide assurance that dynamic loads in the discharge lines from the safety/power relief valves in the steam generation system include the effects of possible water slugs arising from the loop seals, where applicable.

Response:

Since water seals are not used in connection with the steam/generator system safety/power relief valves, the discharge lines will not be subjected to water slug flow.

Question 110.50 (5.5.3.6)

In Section 5.5.3.6 it is stated that the sodium/water reaction creates an expanding bubble which begins to eject the sodium from the faulted unit through the burst rupture discs. Provide the technical bases to verify that the reaction bubble will proceed only through the SWRPRS piping and not through the main sodium line.

Response:

The paragraph titled "Results" in Section 5.5.3.6.2 has been revised in response to this question.

Question 110.51 (3.9, 5.3, 5.4, 5.6)

Provide a statement or table which verifies that the design rules including stress limits for all ASME Code Class 1, 2 and 3 component supports are no less conservative than those of Sub-section NF of ASME Code Section III. Include the stress limits for all supports subjected to component operating conditions identified as normal, upset, emergency and faulted.

Response:

PSAR Section 3.9 presents the design criteria for all ASME Code Class 1, 2 and 3 component supports. All ASME component supports constructed to the rules of Section III, Class 1, 2 or 3 will be designed to satisfy the requirements of Subsection NF, Class 1, 2, or 3, respectively. For Class 2 and 3 component supports explicit criteria for all conditions are specified in Table 3.9-3 (revised in response to Question 110.10).

PSAR Section 5.3.3.1.8 has been amended to provide the required statement for PHTS components.

Question 110.52 (3.9, 5.3, 5.4, 5.5, 5.6)

Identify and provide the technical basis for the variations in your program for the inservice testing of Code Class 1, 2 and safety-related pumps and valves from the test program defined by ASME Section XI, Subsections IWP and IWV. A program acceptable to the staff is given in NRC Standard Review Plan, Section 3.9.6, "Inservice Testing of Pumps and Valves".

Response:

Inservice Testing of Pumps - Preoperational tests will be conducted to establish reference values for speed, pressure, flow rate and vibration. The reference values will be used to establish criteria to evaluate pump performance data obtained during future Inservice Tests. For SGAHRS, the inservice testing program for active pumps will be in compliance with the applicable portions of the ASME Code, Section XI, Division 1. The type and frequency of Inservice Tests of Pumps will not be specified in final form until initial pump development test data is available. Operational experience of FFTF pumps will be considered. The ASME Code Section XI, Division 3 (IMP) for Liquid Metal Cooled Plants is presently being prepared and will define test requirements. It is expected that Inservice Testing for CRBRP will reflect the intent of this code.

Inservice Testing of Valves - An Inservice Testing program for valves will be established based on development tests and specifications for these components. For the steam/water portions of the Steam Generator System and SGAHRS, the inservice testing program for active valves will be according to the applicable requirements of the ASME Code, Section XI, Division 1. The IMV part of the ASME Section XI, Division 3 code is expected to be followed for the CRBRP Inservice Testing.

Question 110.53 (5.3.2.3 Yellow)

The last paragraph in Section 5.3.2.3.4 (Amendment 5) needs clarification. Indicate in more detail what is meant by the statement that although the CDA loading requirements will be analyzed in accordance with ASME Section III rules and criteria for faulted conditions, the CDA loadings are not considered as a faulted condition for Code acceptance. Provide the basis for your conclusions that CDA's should not be considered as faulted conditions.

Response:

This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. Since an HCDA is not a Design Basis Accident (Reference Q110.53-1), CRBRP need not meet the Faulted Condition requirements of the ASME Code during the HCDA loading. 60

Reference:

Q110.53-1 U. S. Nuclear Regulatory Commission, "Final Environmental Statement Related to Construction and Operation of the Clinch River Breeder Reactor Plant", NUREG-0139, Appendix I, Docket No. 50-537, February 1977. 60

Question 110.54 (5.2.4.4 yellow)

Section 5.2.4.4 (Amendment 5) is not entirely acceptable. A description and function of a margin seal must be provided.

Response

The additional description, function and details of the margin seal are provided in Amended Section 5.2.4.4 and Figure 5.2-7. 60 32

Question 110.55 (F6.3.3)

Figure F6.3-4 needs to be clarified. The figure does not indicate the locations within the impact experiment where the experimental pressures were recorded nor where the theoretical pressures were determined. Indicate these locations in Figure F6.3-4.

Response:

19 This question requests clarification of information which is no longer a part of the current documentation. The Project has since consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. 60

Question 110.56 (F6.3.3 & 15.1.1.5 Yellow)

In Section F6.3.3 it is indicated that pressure peak attenuation mechanisms are not considered in the TRANSWRAP analysis program. It is also inferred that by not considering this pressure peak attenuation, the resulting PHTS piping and component loads will be conservative. It is true that the peak pressure will be attenuated, however, the time duration of the impulse will generally increase. Verify the adequacy of the TRANSWRAP program to adequately predict the critical impulse profiles for the PHTS and OHRS. In this verification consider the effects of pressure attentuation, impulse duration, and include system components such as pumps and valves.

Response:

The CRBRP Project has consolidated all considerations given Hypothetical Core Disruptive Accidents into report CRBRP-3 (References 10a and 10b, PSAR Section 1.6) and its associated references; consequently, PSAR Appendices D and F have been withdrawn in Amendments 24 and 60 respectively. The response to this question is now found in Section 6.1.2.2 of Reference 10a, PSAR Section 1.6. 60

Question 110.57 (6.2.7.3.2 Yellow)

The External Cooling System (ECS) must be described in greater detail than that described in Appendix F. The description should contain the functional details of the system, loading and operating design criteria, operational testing methods, and the explicit codes and standards to be adapted.

Response:

With the deletion of the Parallel Design in Amendment 24 this question is no longer applicable as the features upon which the question is based are no longer a part of the design.

Question 110.58

Provide in the PSAR, a description of the methods of analysis that will be used in dynamic and static analyses to determine structural and functional integrity of Seismic Category I components and supports. Include the information in Paragraphs 1 through 4 below for the following HTS components:

- Piping Systems
- PHTS Pump
- IHX
- IHTS Pump
- Superheater
- Evaporator
- Control Valves
- Pressure Relief Valves

1 Identify the failure modes which are expected to dominate the component design and the loading conditions associated therewith

2. Indicate the degree to which elastic, simplified inelastic, detailed inelastic, and creep methods of analysis will be used in design iterations. Also, summarize the time-dependent and cyclic structural analysis that will be performed. Describe the basic details of these methods and state the primary assumptions associated with each analysis. Indicate how component degradation over the life of the component will be treated in the analytical methods.

3. Identify those structural tests that will be required in support of the design analysis.

4. Identify and briefly describe the major computer programs which may be used in the various analyses. (Ref. Question 110.27).

Response:

A discussion of In-containment Heat Transport System piping, pumps, and failure modes in addition to IHX failure modes are provided in revised Section 5.3.3.1.5.

Revised Section 5.4.3.1.5 provides information which pertains to Ex-containment piping.

Revised Section 5.5.3.1.5 provides Steam Generator valves and failure mode information.

Computer programs to be used are listed in response to NRC question 110.27 and Appendix A of the PSAR.

Question 110.59 (3.6.1.1)

Section 3.6.1.1 needs to be clarified. Indicate the pipe break criteria to be used in the design of the piping runs from the Auxiliary Feedwater Pump to the AFW isolation valves and from the turbine drive steam supply isolation valve to the turbine drive.

Response:

Section 3.6.1.2.1.1 was modified in Amendment 27 to specify the criteria used for the subject piping runs with regard to pipe breaks. 31

Question 110.60 (3.6.4)

The response to item 110.3 is not complete. Section 3.6.4.1 only describes the methods used to calculate the jet impingement loads. Provide a complete response to item 110.3.

Response:

A revised section 3.6.2 is included to show the latest pipe break criteria for the steam/water piping. Section 3.6.4 has been extensively updated to include a discussion of the Pipe Dynamic Analysis (PDA) computer program including forcing functions, calculational modes, and problem modeling. Jet impingement geometry for guillotine breaks has been revised to follow Moody's expansion model. Section A.69a has been added to Appendix A to include a description of the PDA computer program.

Question 110.61

The responses to items 110.6 and 110.11 are not acceptable. It is the staffs' position that the acceptable loading combinations which were outlined in item 110.6 are applicable to all ASME Class 1, 2 and 3 components in the CRBRP regardless of whether they are high or low temperature. In addition, the term "thermal transients" in the loading combinations presented in the response to these items apparently does not include all transients associated with the particular condition. It should be changed to reflect a more broad definition of loads. Revise the responses to items 110.6 and 110.11 or Table 3.9.2 and Section 5.3.1.1 in the PSAR to be consistent with the criteria which were presented in items 110.6 and 110.37.

Response:

The components in CRBRP will be designed with appropriate combinations of loads for both high and low temperature conditions. Section 3.9 of the PSAR has been revised in response to NRC question 110.25 to reflect the requirements for such combinations.

Question 110.62 (3.9.1.1)

With respect to the response to Question 110.13, the staff will require that the preoperational vibrational and dynamic effects test program be conducted on all high energy piping systems which are not classified as ASME Class 1, 2 and 3 in addition to the commitment in the response. Revise the response to Question 110.13 or Section 3.9.1.1 in the PSAR to be consistent with this requirement.

Response:

The CRBRP has committed to a pre-operational piping vibrational & dynamic effects test program on all safety related piping systems designed as Class 1, 2 or 3 under the ASME Boiler Pressure Vessel code Section III per the Standard Review Plan 3.9.2.

It is the Project's intention not to perform these tests on non-safety related high energy Balance of Plant piping or non-safety related moderate energy seismic Category I piping as loss of this piping will not compromise the safe shutdown of the plant.

Question 110.63

Justify the 1/3 increase in stress limits for emergency conditions as indicated in Note 3 of Table 3.9-3.

ResponseI

The design of class 2 and 3 linear type supports by analysis is governed by sub-paragraphs NF3330, NF3230 and NF3400 of the ASME Code. The rules so defined are essentially identical to the AISC requirements and are detailed in Appendix XVII of subsection NA. In paragraph NF3231.1 for elastic analysis, it is specifically stated that a 1/3 increase over the design, normal and upset stress allowables is permitted for emergency conditions. Table 3.9-3 has been modified to so note.

Question 110.64

The response to item 110.10 is not adequate. The response references the stress criteria specified in Sections 3.9.1.6 and 3.9.2.3. Indicate how these stress criteria can be interpreted to insure that Category 1 ASME Class 1, 2 and 3 active components will be designed so that they will not deform to the extent that would impair the required operability of the active components. Provide a comparison of maximum allowable support deformations for active components with calculated support deformations.

Response:

A discussion of the calculated deformations and deformation criteria for active component supports and the adequacy of the criteria to ensure operability of the components is provided in WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification Plan".

44

Question 110.65

The responses to items 110.7, 110.9, and 110.26 (part 1), are not completely acceptable. The staff does not concur that the information presented in the PSAR meets the intent of NRC Standard Review Plan, Section 3.9.3 with respect to active pump and valve operability. In addition to the information in Sections 3.9, 5.3, 5.4, 5.5 and 5.6 of the PSAR, the staff requires, as a minimum, the following:

1. A program to demonstrate the operability of all ASME Class 1, 2 and 3 active pumps and valves in the CRBRP which is at least equivalent to that presented in the Catawba Nuclear Station PSAR, Amendment 15, pages QP-4A through AP-4FB (Docket No. 50-413/414). Include: (1) a more detailed description of the types of analyses which will be performed, (2) a commitment to perform static tests to simulate faulted condition loads on representative active components, (3) a commitment to include faulted condition nozzle end loads in the aforementioned analyses and tests, (4) a commitment to seismically qualify all appurtenances which are required for operation of the active component. These tests should be conducted in accordance with the requirements of IEEE-344, 1975 with the exceptions stated in Question 110.8, (5) a commitment to demonstrate by test and/or analysis that all active Class 1, 2 and 3 pumps and valves will operate when subjected to the stress limits specified in Table 3.9-3 and in Sections 5.3, 5.4, 5.5 and 5.6 of the PSAR.

2. A commitment to satisfy the criteria outlined in NRC Standard Review Plan, Section 3.9.3, Paragraph II.2.c, "Design Specifications."

Revise Sections 3.9.2.4, 5.3.2.1.2, 5.5.2.1.2 and 5.6.2.2.1.2 to include the above commitments.

Response:

1. The CRBRP program to verify the operability of active components is provided in WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification Plan". 44

2. The intent of Section II.2.c of SRP 3.9.3 will be complied with.

Question 110.66 (3.9.1.3)

It is stated in Section 3.9.1.3.1 that a specific vibration test of an approximate 0.248 scale model will be conducted. Table 3.9-7 compares model parameters with those of the prototype. Justify the choice of model parameters indicated. Some parameters such as the elastic modular, fluid temperature, and kinetic viscosity are quite different between the model and the prototype. Clarify the significance of Table 3.9-6.

Response:

Section 3.9.1.3.1 has been modified in response to this question.

Question 110.69 (3.9.1.6)

The response to Item 110.16 is insufficient. In Section 3.9.1.6, it is stated that sufficiently low limits for normal, upset and emergency conditions are used to assure that the dynamic elastic system analysis is not invalidated. It is also stated that inelastic and limit analysis methods may be used. Clarify this statement. Also, provide the basis for specified deformation limits.

Response:

The information requested is provided in revised Section 3.9.1.6 of the PSAR.

Question 110.70 (3.9.1.6)

Provide a summary table of design limits for all Class 1 active pumps and valves.

Response:

Section 3.9.1.6 has been modified and Table 3.9-9 has been added to provide the design limits for Class I active components.

Question 110.71 (5.3.3.5)

It is indicated in Section 5.3.3.5 that the sodium pressure in the intermediate heat exchanger is always greater in the intermediate loop side than on the primary loop side. Verify that this will always be true through all parts of the heat exchanger considering pressure tolerances and flow characteristics throughout the heat exchanger.

Response:

To assure that the intermediate pressure is greater than the primary pressure at all points in the IHX, the intermediate pressure at the top tube sheet is set at a minimum of 10 psi higher than the primary pressure at that point. This pressure is assured by setting the IHTS cover gas pressure for the worst pressure condition and holding it constant over the range of plant operation (0% to 100% power). Based on a survey of the IHX over the entire operating range, the limiting point in the IHX is at the top tube sheet and at 100% power.

To assure that the cover gas system is designed for the maximum pressure that would be required to assure a positive 10 psi pressure differential, intermediate to primary, a worst case analysis was performed. The worst case is defined by evaluating the PHTS for maximum loop resistance with minimum resistances from the reactor to the IHX and evaluating the IHTS for minimum loop resistances at 100% power. This case defines the cover gas pressure which will assure that a 10 psi pressure differential is maintained in the IHX for all plant operating conditions. The IHTS cover gas pressure requirement which results from preliminary analysis of this condition is 96 psig. The plant cover gas pressure will initially be set at or above the calculated value to account for control and measurement uncertainties. During plant testing the actual plant characteristics will be evaluated and the cover gas pressure requirement can be lowered if the system characteristics warrant it.

The 10 psi differential at the top tube sheet assures that there will be a positive pressure differential at all points in the IHX, taking into account the pressure tolerances and flow characteristics in the IHX. The 10 psi requirement is adequate because primary side velocities are small and the overall pressure drop of the IHX on both the primary and intermediate sides is approximately the same as the 10 psi differential.

Question 110.72 (5.3.3.6)

The response to Item 110.46 is not adequate. It is true that the induced thermal stress in a long cylinder with linear temperature through the wall will be

$$\sigma_1 = \frac{E\ \alpha(\Delta\ T_1)}{2(1-\nu)}$$

However, it is stated in Section 5.3.3.6 that this stress is applicable to elbows. Verify your justification of this thermal stress equation for elbows.

Response:

The above equation provides accurate thermal stress values (within 2%) for a constrained thin-walled, curved shell with a linear, thru-the-wall, temperature gradient where the ratios of shell thickness (t) to the principal radii of plate curvature (r,R) are sufficiently small (less than 0.06) and where the point of determination is not near gross structural or materal discontinuities. For this reason the ASME Code (see equation (10), paragraph NB-3653.1, Section III at tne ASME Code) allows use at the above equation for most piping products, including elbows. All applications at the above equation to the CRBRP primary and intermediate piping, including elbows, are well within the above restrictions.

Question 110.73 (5.5.3.6)

Section 5.5.3.6.2 states that it is expected that the sodium flow in the pump suction line will reverse before the gas bubble reaches the pump. Discuss the consequences and potential induced loading on the pump and motor if the gas bubble does reach the pump.

Response:

Section 5.5.3.6.2 has been updated to discuss the consequences of a gas bubble reaching the pump.

Question 110.74 (5.5.1.1)

The response to item 110.48 is not adequate. Indicate the reasoning why the assumption was made that a guillotined tube will cause the equivalent of six additional guilloting tubes ruptures. The discussion in section 5.5.3.6. can lead one to believe that more than six tubes could fail from overheating in a few seconds.

Response:

The discussion is Section 5.5.3.6. was intended to provide support for the contention that under the worst plausible sequence, the total leak is not expected to exceed that of double-ended guillotine failures. Data from tests and actual operating experience support this contention. The point was also made that in order to generate significant pressure pulses, secondary failures must occur prior to the mitigating effects of SWRPRS actuation and blowdown. Failures which occur later than approximately 1 second following SWRPRS actuation will not significantly increase the system pressure since they will be vented to the atmosphere through the SWRPRS.

Section 5.5.3.6. has been expanded to clarify this position.

Question 110.75

The response to Item 110.35 indicates the analytical procedures, design criteria and experimental programs related to PHTS bellows will be provided in the near future. It is the staff's position that bellows design criteria must be provided and reviewed to support the PHTS integrity argument. Provide this criteria in the PSAR.

Response:

Except for the bellows separating the PHTS sodium from the IHTS sodium in the IHX (discussed in PSAR Section 5.3.2.3.2), there are no bellows in the PHTS coolant boundary. There are no bellows in the PHTS piping runs, thus no bellows impact the piping integrity argument.

The bellows (other than the IHX bellows) referred to in response 110.35 are those between the piping and cell liner at the PHTS cell penetration. This bellows is not a sodium boundary component and is not an ASME-coded item. However, as indicated in revised Section 5.3.2.3.4, the design criteria for the bellows assembly shall be ASME Code Section III, Subsection NF. There are no experimental programs associated with this bellows.

Question 110.77 (5.5.3, 10.3)

With respect to the main steam isolation and stop valves discussed in Section 5.5.3.1.6 of the PSAR, demonstrate the design adequacy of these valves to perform their function during a postulated pipe rupture. Discuss the analytical methods and procedures used to calculate the impact energy and the resulting stresses and strains in the disc and any other parts of the valves which are subject to closure impact following a postulated rupture or a spurious closure of the isolation valve.

Response:

The method of determining the design adequacy of isolation valves, which must close under normal or pipe rupture fluid conditions, has been added to Section 5.5.3.1.5. No response is provided for Section 10.3.1, since reference to the main steam isolation valves was removed from the main steam supply system by Amendment 14.

Question 110.78

The response to Question 110.58 is unsatisfactory. The staff must be assured that the methods to be used in dynamic and static analysis to determine the structural and functional integrity of Seismic Category I components and supports are adequate. Of principal concern is the assurance that the analytical models will adequately represent the physical situations of concern, i.e., proper failure modes (static and creep-fatigue rupture, vibrational distortions, stability and deformations), adequate constituative relationships (elastic, inelastic, viscoelastic, degradation with time), compatible boundary conditions, realistic component material properties.

Also of concern to the staff is the adequacy of the computer programs to produce sound quantitative results. Some recognized computer programs are capable of producing excellent results for one class of component models and possibly not provide accurate results for others. Convincing evidence must be presented to demonstrate that the computer programs used really are adequate.

Supporting tests are generally required to directly verify the structural capabilities of components, and to confirm the analytical methods used. Such tests can be either component model or prototype tests. The staff must be assured that such tests indeed accomplish their purpose.

Include the information in items 1 through 5 below for the following HTS components:

Reactor Vessel

Piping Systems

PHTS Pump

IHX

IHTS Pump

Superheater

Evaporator

Line Valves

Pressure Relief Valves

1. Identify the specific failure modes which are expected to dominate the component design and the loading conditions associated therewith. For larger and more complicated components several critical areas will arise. Identify the failure modes for each of these areas.

2. Indicate the degree to which elastic, simplified inelastic, detailed inelastic, and visco elastic (creep) methods of analysis will be used in design interactions. Also, summarize the time dependent and cyclic structural analysis that will be performed, i.e., vibration, stability, and creep-fatigue analysis. Describe the basic details of these methods. State the primary assumptions associated with each analysis. Indicate how component degradation over life of the component will be treated in the analytical methods.

3. Identify and summarize those structural tests that will be performed in support of the analysis methods or the actual verification of the component structure in lieu of analysis.

4. Indicate how analysis or testing of components from other programs, such as the FFTF, will be adapted into the overall procedure to verify the adequacy of CRBR component design. Summarize or reference the methods used, assumptions made, and the results of such analysis or tests. If the analysis or test has not been performed, summarized the approach to be used in these programs.

5. Identify and briefly describe the major computer programs which will be used in the various analyses (Ref. Question 110.27). The adequacy of these programs for specific application must be provided.

Response:

The following material in this response deals with satisfaction of ASME Code Requirements. These components also have been analyzed for Structural Margin Beyond the Design Base (SMBDB) conditions as discussed in Reference 10a of PSAR Section 1.6.

Response for Reactor Vessel:

The methods used in the static and dynamic analyses of the Reactor Vessel (RV) to determine the adequacy of the structural and functional integrity are summarized in this response.

The RV is a top ring-supported cylindrical structure with a torispherical bottom head. It is roughly 57 feet long with a diameter of about 20 feet. The sodium-containing portion is all stainless steel designed for 900°F in the outlet plenum region and 775°F in the inlet plenum region. The top flange of the vessel and the vessel support ring are fabricated of SA 508 Class 2 low-alloy forgings. There is an Inconel 600 transition section between the low-alloy forgings at the top and the stainless steel in the remainder of the vessel.

The vessel walls and outlet, makeup, and overflow nozzle penetrations are cooled by primary sodium coolant bypass flow to keep the steady-state metal temperature below or equal to 900°F during normal operation and to reduce the rate of vessel wall temperature change during operation transients.

The RV is supported from its upper end. The vessel support system accommodates dead weight, seismic loads, and forces hypothesized under margin loading conditions from the assembled reactor vessel and closure head to the reactor cavity wall through the support ledge.

The RV is designed and analyzed to the Class 1 requirements of the ASME Code, Section III, and Code Case 1592. In addition, simplified inelastic and detailed inelastic methods that are used conform to the requirements of RDT Standard F9-4T and the guidelines of RDT Standard F9-5T.

FAILURE MODES

Analyses of the RV reflect both time-independent and time-dependent materials properties and structural behavior (elastic and inelastic) by considering the following failure modes:

- Ductile rupture from short-time loadings
- Creep-rupture from long-term loadings
- Creep-fatigue failure
- Gross distortion due to incremental collapse and ratchetting
- Buckling due to short-term loading

Specific failure modes critical to the various regions of the vessel are addressed later in the description of the corresponding analyses.

LOADING CONDITIONS

The loading conditions which mostly control the design of the RV are the seismic loadings and the thermal loadings (transient and steady-state conditions). The seismic loads primarily affect the sizing of the upper assembly and the core support cone. The thermal loading is critical for the elevated temperature parts; the vessel thermal liner, makeup nozzle bridge liner, and the outlet nozzle assembly. These parts will experience through-thickness and axial temperature gradients both during steady-state and transient conditions.

ANALYSES

Analysis of the RV has been subdivided into overall system analyses and analyses of several different regions or components of the RV. In addition, analyses have been sequenced as sizing or conceptual design verification, preliminary detailed, and final analyses. The principal features and anticipated critical failure modes associated with each of these analyses is discussed in detail below.

Design Conditions

This analysis covers the basic sizing for all the parts of the RV. The loading conditions to be considered for this analysis are the design and test conditions of Subsection NB AND Code Case 1592 of the ASME Code. Paragraph NB-3112 defines the design conditions for the low temperature parts of the assembly as design pressure, design temperature, and design mechanical loads (e.g., design pipe loads). The design conditions for the elevated temperature parts are defined in Paragraph 1592-3113.1 as the design parameters for normal conditions. The effects of earthquake are not considered as a design condition load for elevated temperature parts. Test conditions are defined in NB-3114 and 1592-3113.7.

The analysis consists of dividing the vessel into simple shell, plate, and beam segments and calculating the primary membrane and bending stresses using conventional, elastic, hand techniques. Nozzle reinforcing calculations are included. The effects of pipe loads on the nozzle and shell are considered. The stress limits are the allowable stresses defined in NB-3221 and 1592-3222.1 of the Code. Appropriate environmental effects are considered.

Highest stressed areas are the nozzles (pipe attachment area) and the nozzle-to-shell junctures. Design piping loads are the significant stress contributors.

Seismic Analysis

This analysis considers the detailed seismic and stress analysis for the total RV assembly. It is analyzed with the response spectra developed from the response motion at the location of the support ring. The reactions imposed upon the RV nozzles by the piping are determined by using the stiffness matrices due to the piping system. A mathematical model of the lower reactor is used to generate loads for all seismic conditions. The design criteria are established in accordance with that stated in Appendix A to Section 3.7 of the PSAR.

The RV assembly is analyzed by a detailed dynamic analysis using the response spectra loading. The 3-D finite element method is used to establish a mathematical model of the RV assembly. The structure is divided into a finite number of appropriate elements, such as beam and plate elements, which are interconnected at a finite number of joints or nodal points. These individual finite elements are then assembled into a simplified mathematical model under the variational principle preserving the shock energy absorption capacity of the total system.

The seismic analysis is performed considering the seismic motion to be acting in the vertical direction and in two orthogonal horizontal axes. The analysis is performed independently in each of the two horizontal directions and vertical directions. Finally, the combined modal responses obtained for each of the vertical and horizontal siesmic loads are combined individually by the square root of the sum of the squares.

Fluid-induced vibration is investigated at this stage for the liner, outlet nozzle sleeve, makeup nozzle sleeve, and inlet nozzle flow deflector to determine the potential for vibration due to fluid flow.

The adequacy of the design is determined in accordance with Section III of the Code. The RV assembly is designed to insure a safe shutdown during and after an SSE. To meet this condition, the RV shall not exceed the limits of Section III of the Code for faulted conditions.

Highest seismic stresses are found in the upper stainless steel shell courses and the Inconel transition section.

Thermal Analysis

This analysis consists of a 2-D axisymmetric finite element thermal analysis to determine the basic thermal gradients in the various aras where potential high thermal stresses can exist.

The analysis includes the evaluation of all the normal, upset, and steady state conditions identified in the equipment specification. The transients are conservatively combined considering coolant flow, maximum temperature ranges, temperature ramp rates, and number. From the resulting bounding transients, film coefficients are calculated and temperature profiles obtained using finite element techniques. This data is subsequently used to determine the time and location of significant thermal stresses. This analysis also determines the metal temperatures from which the allowable stress can be determined.

Thermal-Mechanical Analysis fo Cover Gas Nozzles

This analysis considers the cover gas inlet and outlet nozzles.

The loading conditions considered are the normal, upset, emergency and faulted operating condition thermal and flow transients to determine the maximum primary plus secondary stress range. In addition, loads due to deadweight seismic and thermal expansion are evaluated. Pressure stresses were found to be small.

A 2-D finite element method (FEM) analysis using constant strain triangular ring element is performed. A single mesh is generated for both the thermal and stress analyses. The thermal analysis yields temperature distributions in the nozzles based on gas flow and associated shell temperatures. A special purpose program is used to evaluate the asymmetric pipe loads. The stresses are calculated on a completely elastic basis.

The analysis is performed per NB-3000, para. 3228.3, of the Code. The critical areas are the nozzle-to-shell juncture and the nozzle-to-pipe juncture.

Thermal-Mechanical Analysis of the Lower Head, Shell and Core Support Assembly

This analysis of the lower head, shell, and core support assembly considers the thermal-mechanical loads for operating conditions, normal, upset, emergency, and faulted. The analysis includes evaluation of transient and steady state temperature distributions for all significant transients. Also included are the mechanical stresses due to pressure, seismic, deadweight (weight of the core) and core support plate thermal motions.

The analysis presents an evaluation of transients for the inlet plenum and determines how the transients are combined for analytical consideration. The combining of transients into a lumped transient is based on coolant flow, maximum temperature changes, rate of change and number of cycles. Based on the lumped transient, fluid temperatures and film coefficients are determined. Metal temperature distributions are then determined by FEM. Stresses are calculated for the above thermal and mechanical loads using a 2-D FEM. For perturbed loop transients, the thermal response is approximated by composite solutions. Asymmetric stress conditions are determined using a 2-D asymmetric FEM with Fourier series thermal distributions about circumference.

Buckling is investigated in the areas of the core support cone and the lower torispherical head using analytical procedures based on conservative current state-of-the-art practice.

High stressed areas are the juncture of the core support cone to the shell, knuckle region in the torispherical head and the torispherical head-to-shell juncture.

Thermal-Mechanical Analysis of Upper Assembly

This analysis covers the upper assembly, including the stainless, Inconel shells, the ferritic flange, the radiological shield, and the support ring and the dip seal access ports. The loadings of normal, upset, emergency, and faulted conditions including seismic are considered.

The heat transfer analyses include the determination of transient and steady state thermal distributions. Thermal stresses considered include those due to radial and longitudinal gradients and thermal discontinuity at the juncture of the flange and the tapered shell. Superimposed on these thermal loads are pressure loads, deadweight acting on the flange, seismic loads, and all other externally applied loads.

The evaluation of the design is made in accordance with Section III of the Code and Code Case 1592. Environmental effects are considered.

The highest stressed areas are the stainless to Inconel shell juncture and the upper shell (stainless) courses which are subjected to large longitudinal thermal gradients. The temperatures in the carbon steel parts are below 800^{o}F, therefore the acceptance criteria is NB-3000. In the lower part of this assembly (stainless shells) temperatures are above 800^{o}F, hence Code Case 1592 is used for evaluation of the stresses. Inelastic analysis is performed to demonstrate adequacy in the region where the axial gradient in the shell begins.

Buckling due to seismic loading was investigated in the upper shell courses. The analysis was based on conservative analytical methods.

The analysis considered the basic sizing and design of the dip seal maintenance port. The loading conditions considered include interal and external pressure, deadweight, seismic and temperature effects.

The analysis determines or validates tube wall thickness, gap size between the tube and bore, weld size and type of weld at the vessel flange, flange size and the design of any additional supporting system. Hand calculations are used to perform the analysis. The highest stressed area is the weld at the vessel flange-to-pipe juncture.

The analysis considered the design of the radiological shield, its attachment and its interaction with the vessel flange. The loading conditions are deadweight, seismic, thermal conditions and natural frequency calculations. The seismic analysis is based on a static seismic loading of 1.5 times the maximum OBE response. The analysis uses hand calculations for seismic analysis and a combination of computerized interaction analysis and hand calculations for the thermal evaluation.

Thermal-Mechanical Analysis of the Inlet Nozzle Assembly

This analysis considers the inlet nozzle assembly. The loading conditions considered in this analysis are all the thermal and flow transients for normal, upset, and emergency conditions, and the thermal and siesmic pipe loads.

A 2-D FEM analysis is performed. The grid from the inlet nozzle analysis previously described is utilized in this analysis. The thermal analysis yields temperature distributions throughout the assembly for steady-state and transient conditions. The stresses are determined for most areas using an elastic or inelastic material model. In addition to the thermal stresses, stresses due to pressure are included in the axisymmetric analysis. Pipe loads and the flow deflector loads are also analyzed by FEM using a special program for asymmetric loads. Supporting this analysis is a limited 3-D elastic FEM analysis of the assembly to validate the 2-D assumptions used. This model is also utilized in evaluating the effects of the lower head on the nozzles.

In using a 2-D technique to analyze the juncture between a cylinder and a nozzle, the cylinder is assumed to act as either a sphere or a flat plate, both of which can be modeled as symmetric about the nozzle centerline while a cylinder cannot be so represented. When a small diameter nozzle is inserted into a relatively large diameter cylinder, the approximations used in 2-D analysis are obviously close to the true geometry; however, when the nozzle diameter is large relative to the cylinder a check on the accuracy of stress distribution at their juncture is appropriate. For the inlet nozzle, the ratio of cylinder diameter to nozzle diameter is only 9.44; this low ratio makes a check on the assumptions necessary. Also, the vessel lower head is less than $2.5(Rt)^{1/2}$ from the inlet nozzle; this effect is not considered in the axisymmetric analysis, but is checked by the 3-D test case.

The critical area occurs at the nozzle-to-shell juncture and the pipe-to-nozzle juncture. The adequacy of the assembly is determined with respect to Subsection NB. The environmental effects are considered.

Thermal-Mechanical Analysis of the Outlet Nozzle and Outlet Nozzle Liner

This analysis considers the outlet nozzle assembly. The loading conditions considered in this analysis are the thermal and flow transients for normal, upset, and emergency conditions, and the thermal and seismic pipe loads. The majority of this nozzle will be at temperatures near 800°F; however, outboard of the thermal sleeve and in the sleeve itself, temperatures will exceed 800°F. Therefore, time-dependent effects are considered.

A 2-D axisymmetric FEM analysis using constant strain triangular ring elements is performed. A single mesh is generated for both the thermal and stress analyses for each of the major subassemblies. The thermal analysis yields detailed temperature distributions through the nozzle and sleeve, based on fluid temperature and flow through the outlet and behind the liner and sleeve. Both two and three loop flow conditions are evaluated thermally and, based on results, the decision is made whether to make stress evaluations for one or both conditions.

Thermal distributions are obtained for the transients and stresses are determined using an elastic and/or inelastic material model of the assembly. In addition to the thermal stresses, pipe loads and the effects of the relative motion of the vessel thermal liner are factored into the analysis. Stresses due to these loads are determined using a FEM program for asymmetric loading. Supporting this analysis is a limited 3-D elastic FEM analysis of the assembly to validate the 2-D assumptions used.

The critical stress areas are in the nozzle liner, the nozzle-to-shell juncture, sleeve-to-vessel liner juncture, and the sealing discs. The stresses are compared to the limits of Subsection NB and Code Case 1592. The environmental effects are considered.

Analysis of the Makeup and Overflow Nozzles for Operating Conditions

This analysis considers the sodium makeup and overflow nozzles. The loading conditions considered are the thermal and flow transients for operating conditions transient events, flow rates, and outlet plenum conditions, and thermal and seismic pipe loads. The overflow nozzle is exposed only to bypass flow. The makeup nozzle has a thermal liner to consider and is affected by outlet plenum temperatures. Analysis of the makeup nozzle liner includes the local region of the vessel liner.

The geometries for both these nozzles are modeled axisymmetrically for FEM analysis. For both nozzles, thermal distributions as well as stress distributions are considered axisymmetrically. Asymmetric pipe loads are evaluated and superimposed at critical locations. The stresses are computed on a completely elastic basis.

The acceptance criteria is Code Case 1592 for those areas where significant temperatures above 800°F occur. Environmental effects are considered.

The critical stress areas are the nozzle-to-piping junctures and the shell-to-nozzle junctures.

Thermal-Mechanical Analysis of the Makeup Nozzle Liner

This analysis considers the makeup nozzle liner. The loading condition is the same as described for the makeup nozzle. The 2-D finite element model for the bridge liner is evaluated for convergency capability for the inelastic/creep behavior.

The acceptance criteria is ASME Code Case 1592. Environmental effects are considered.

The critical area is the juncture of the makeup nozzle liner to vessel thermal liner.

A sleeve of 718 material is installed on the makeup nozzle to enable the assembly to withstand the striping caused by incoming cooler sodium.

Thermal-Mechanical Analysis of the Vessel Thermal Liner Attachment Region

This analysis considers the thermal mechanical loading on the thermal liner forging including the adjacent shell and the baffle support ledge. The loading conditions considered in this analysis are the thermal and flow transients for normal, upset, and emergency conditions, and the seismic loads on the attachment pins.

The analysis includes a heat transfer evaluation of both steady state and transient conditions. Due to the complexity of the design, thermal stresses are considered for both radial and longitudinal gradients as well as the thermal discontinuity which exists at the thermal liner forging. Superimposed on these stresses are the mechanical load stresses due to pressure and seismic events.

The attachment region adjacent to the shell complies with the NB-3000 rules for material below 800°F. The vessel thermal liner portion adjacent to the vessel thermal liner support forging reaches a temperature above 800°F and is analyzed to the criteria of Code Case 1592. The stresses and strains in this area are not critical.

Thermal-Mechanical Analysis of the Vessel Thermal Liner

This analysis considers the thermal mechanical loading of the vessel thermal liner. The first phase includes a total evaluation of the radial gradient through the liner to identify operating temperature limits. No discontinuities are considered in this phase. The second phase is an evaluation of the liner stiffening ring, the liner and the bypass flow penetrations. The loading conditions considered are the normal, upset, and emergency conditions. Seismic stresses are also considered.

The analysis includes a heat transfer evaluation of steady state and transient conditions. The maximum thermal stresses are caused by the radial gradient which will exist in the thermal liner. The effect of longitudinal gradient is also evaluated. This location also requires stresses calculated for thermal discontinuity.

This assembly operates normally at elevated temperatures. Therefore, it requires a time-dependent analysis. The major potential failure mechanism is creep-fatigue interaction. The method of analysis is 2-D FEM. The structural program has inelastic and creep capability. The adequacy of the design is evaluated according to Section III and Code Case 1592. The environmental effects are considered.

The critical areas are the vessel thermal liner shell in the vicinity of the sodium level and the striping potential in the bypass flow penetration area.

MATERIAL PROPERTIES

The environmental conditions of the RV also influence the design. Sodium exposure is the only effect of significance and is applicable for the high temperature stainless steel regions, specifically, the vessel liner, outlet nozzle liners, nozzle stub ends, and makeup nozzler liner. There are no environmental effects on material properties for the carbon steel and Inconel 600 low temperature portions of the reactor vessel.

The elastic material properties used in the structural evaluation of the RV are specified in the ASME Code documents. The Nuclear Systems Materials (NSM) Handbook TID-26666 is used as the authoritative source for material properties not specified in the applicable Code documents. All material properties used in the design and analyses of the RV are specified in the Code documents or the NSM Handbook.

A collection of computer files containing material property data, routines for interpolation and routines for material models, which are based on the material data given in the NSM Handbook and the ASME Code, were used for the analysis of the RV.

Where material properties are significantly uncertain, consideration is given to the use of minimum, average or maximum properties as appropriate to obtain a conservative result. This selection of appropriate properties is guided by RDT Standard F9-5T.

Material Degradation

Most of the data used to define the allowable design stresses in the ASME Code were obtained from tests conducted in air. No attempt is made in the Code to account for the effects of other service environment. The LMFBR development program has focused attention on the mechanical behavior of reactor materials when exposed to high-temperature liquid sodium, in addition to fast neutron irradiation and to long time aging at elevated temperatures. A brief discussion of the environmental effects is given in the following paragraphs.

Thermal Aging Effects on Mechanical Properties

Types 304 and 316 stainless steels are non-age hardenable alloys. Thus, no significant changes in strength or hardness of annealed material would accrue from long-term aging at temperatures up to 1200°F, unlike the precipitation-hardened stainless steels. Some slight increases in strength and decreases in ductility may occur due to carbide formation, together with a reduction in the room temperature impact strength. Of more significance, is the fact that these alloys will sensitize during long-term service in the temperature range from 800°F to 1500°F.

In this phenomenon, carbide precipitation occurs at the grain boundaries, the adjacent matrix becomes depleted in chromium and the grain boundary regions become susceptible to attack by corrosive media. Such attack is not likely to occur in sodium, which, if pure, is a relatively inert environment. However, cracking may initiate during fabrication and the other pre-operation periods when the component is not exposed to sodium due to the environmental conditions (presence of water and halides). Because of this, precautions must be taken during such periods to ensure that contact between sensitized material and potentially corrosive media is minimized, if not entirely avoided. Hence, no allowances have been made for the effects of thermal aging on the properties to types 304 and 316 stainless steels. This did, however, demand that control be specified and exercised during the fabrication process to prevent stress corrosion and intergranular attack.

<u>Neutron Irradiation Effects on Mechanical Properties</u>

The effect of neutron irradiation on the mechanical properties of a material are generally to increase the tensile and yield strengths, and to decrease the ductility. The actual magnitude of the effect is dependent on several parameters, such as the temperature of irradiation, the test temperature, the neutron energy spectra and the neutron fluence.

Two alternate procedures have been used to account for the effects of neutron irradiation on the structural integrity of components. The comprehensive approach characterizes the effects of neutron irradiation upon each material response and failure mode considered by the ASME Code. When necessary, additional failure modes are considered. The alternate approach involves finding the threshold where irradiation effects first become measurable (in terms of structural response integrity). The irradiation levels are then held below these threshold levels by shielding.

For austenitic stainless steels (Types 304 and 316), measurable loss of ductility (total elongation) can first be detected at about 10^{21} nvt total fluence for temperatures in the range of 600°F to 1100°F. The reactor vessel end-of-life fluence is less than 6×10^{20} nvt and hence no fluence effects are expected.

<u>Effects of (Nitrogen + 2% Oxygen) Atmosphere on Mechanical Properties</u>

The selection of nitrogen gas as the atmosphere for the reactor cavity was based on the desire to prevent chemical reactions should molten sodium leak into the cavity from any source. However, the exposure of austenitic stainless steel to pure nitrogen for extended periods of time at elevated temperatures may lead to the formation of a thin nitrided layer. This is considered undesirable because of the brittleness of such layers. To minimize the formation of such a layer, a small percentage of oxygen (2%) will be introduced into the nitrogen.

Effects of (Argon-Plus-Sodium Vapor) Atmosphere on Mechanical Properties

Very little is known of the effects of exposure to an argon-plus sodium-vapor atmosphere on the mechanical properties of a material. It is possible that, if the sodium vapor is continually condensing on the material surface and rejoining the main reactor coolant, there could be some interstitial transfer. However, because of the scarcity of data, it is not possible to provide quantitative assessments of such effects at this time. Practically, the potential for significant mass transfer via condensation is insignificant. It is judged that exposure to the cover gas should be considered the same as exposure to liquid sodium without loss of interstitials.

Surface Effects of Liquid Sodium on Mechanical Properties

Compared with air testing, liquid sodium may cause certain metallic elements to be transferred from the hotter to the cooler regions of the system. In addition, surface oxidation in liquid sodium is greatly reduced when compared to air testing. It is believed that these surface effects are insignificant in their influence on short-term tensile properties.

For time-dependent deformation, such as stress-rupture and fatigue, the effects of a liquid sodium environment are complex and need to be considered in detail. In the case of stress-rupture, it has been shown that for a given temperature and stress, rupture times in air are longer than those in liquid sodium. A sodium environment correction factor is applied to the rupture strength data specified in ASME Code Case 1592 for Types 304 and 316 austenitic stainless steel. This effect is used in all evaluations where stress to rupture is involved.

Fatigue properties of materials can be greatly affected by the environment in which the properties are measured. The avoidance of excessive surface oxidation by testing in sodium (or inert gas) instead of in air increases the cycles-to-failure for a given strain range. No increase in the design fatigue limits due to exclusion of oxygen effects is permitted.

Interstitial Transfer Effects on Material Properties

In the reactor system, interstitial carbon and nitrogen are transferred from the hotter to the cooler regions. This leads to weakening in the decarburized and denitrided regions and to strengthening in the carburized and nitrided areas. In the case of fatigue behavior, however, the effects of interstitial absorption at the surface are complicated because of two concurrent mechanisms. On the one hand carburization can lead to enhanced crack nucleation at carbide particles and, on the other, surface strengthening during strain-controlled fatigue will increase the proportion of elastic straining which is less damaging than plastic deformation. Studies indicate that, in general, the austenitic materials will be carburized and the ferritic materials will lose interstitials. However, the crossover from carburization to decarburization is system dependent, and it is likely that in certain systems at least some of the austenitic material will be decarburized. Procedures have been established by which the extent of interstitial transfer for Types 304 and 316 stainless steel can be determined and from this the effects on mechanical behavior is calculated. The procedures include calculations of surface and average interstitial concentrations and interstitial gradients under decarburizing and denitriding conditions. Because of the shortage of data on nitrogen diffusion, the rates of nitrogen transfer are estimated from available carbon transfer data.

TESTS

No structural tests, other than those required by the ASME Code, have been performed in support of the RV analysis methods.

COMPUTER PROGRAMS

The following computer programs were used in the analysis of the RV. These codes are all proprietary to Babcock and Wilcox.

Code	Used For
ABSA	Axisymmetric Body Stress Analysis
ABTA	Axisymmetric Body Thermal Analysis
ALAS	Axisymmetric Load, Axisymmetric Body Stress Analysis
FESAP	See PSAR Appendix A
FETAP	General Configuration Thermal Analysis
CREEPABSA	Elastic Plastic Creep, Axisymmetric Body Stress Analysis
BIJLARRD	Bijlarrd Shell Stress Analysis
INTERACTION	General Interaction Analysis for Shells of Revolution with Axisymmetric Loading

Response for HTS Piping:

The methods used in the static and dynamic analyses of the primary and intermediate HTS piping to determine structural and functional integrity are summarized in this response.

The Heat Transport System (HTS) consists of piping and components required to transport reactor heat to the steam generators. The system is comprised of three approximately identical cooling circuits, each of which includes a Primary Heat Transport System (PHTS) loop and an Intermediate Heat Transport System (IHTS) loop thermally coupled by an Intermediate Heat Exchanger (IHX). The PHTS and IHTS piping within containment are located within shielded and inerted cells (nitrogen atmosphere with a maximum of 2 percent oxygen). A detailed description of the PHTS and IHTS piping is provided in Chapter 5 of the PSAR.

The HTS Piping shall be designed, constructed and stamped in accordance with the rules for Class 1 (ANS Safety Class 1) Nuclear Components in the ASME Boiler and Pressure Vessel Code, Section 111, 1974 Edition with Addenda through Summer 1975 and Code Case Interpretations 1592-7, 1593-1, 1594-1, 1595-1 and 1596-1 supplemented by RDT Standards F9-4T (dated January 1976) and E15-2NB-T (dated November 1974, Amendments 1, 2 and 3). The piping will be designed to assure that the stresses, strains and deformations are within the applicable Code criteria, and to meet the system functional requirements. In addition, simplified inelastic and detailed inelastic methods that are to be used will conform to the requirements of RDT Standard F9-4T and the guidelines of RDT Standard F9-5T (dated September 1974).

FAILURE MODES

Analyses will be performed on the piping to reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all the modes of failure listed below:

1. Ductile rupture from short-term loadings

2. Creep-rupture from long-term loadings

3. Creep-fatigue failure

4. Gross distortion due to incremental collapse and ratchetting

5. Loss of function due to excessive deformation

6. Buckling due to short-term loadings

7. Creep buckling due to long-term loadings

LOADS

It is convenient in the context of the structural analysis and stress evaluation of the HTS piping to separate the loadings into two categories; System Loads and Piping Component (Local) Loads. Requirements regarding the combination and application of loadings are specified in the applicable ASME Code, RDT Standard documents, NRC Regulatory Guides and CRBRP criteria documents. (See PSAR Section 3.9)

System Loads

System design loads are comprised of internal pressure, deadweight, earthquake loads, thermal expansion, Sodium/Water Reaction (SWR) loads and system thermal transients. The combination or treatment of the system loadings in the analysis process for the HTS piping and support systems is shown on Figure Q110.78-P-1. These loads are described in detail in the following:

1. Internal Pressure

   System pressures include the piping internal pressures for Design, Normal, Upset, Emergency and Faulted Conditions. Local membrane and bending stresses resulting from system pressures in the piping are determined by standard practices and are combined with other calculated stresses.

2. Deadweight

   The deadweight loading imposed by the piping on itself and on the supports consists of the dry weight of the HTS piping and the weight of the sodium contained in piping during the operating conditions. The total weight of the insulation and trace heaters around the piping provides an additional deadweight loading as do the weight of valves, clamps and portions of the restraining devices such as snubbers.

3. Earthquake Loads

   The intensity and character of the earthquake motion which produces forced vibration of the equipment mounted within the containment building are specified in terms of the floor response spectrum curves or time-histories at various elevations within the containment building. These response spectra or time-histories are developed from a three-dimensional multi-mass elastic dynamic model of the reactor containment and steam generator buildings. The forcing function applied to this model is the site seismic ground motion.

The subsequent motion throughout the buildings at various elevations is the basis for the Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) floor response spectrum curves.

4. Thermal Expansion

The vertical and lateral growth of piping and the main HTS components as temperature rises above the ambient temperature impose loads in the piping.

5. Sodium/Water Reaction Loads (IHTS Piping)

A possible event considered is the Design Basis Leak (DBL) within the Steam Generator. Large pressure peaks reverberate through the IHTS pipin when sodium and water react in the steam generator under the postulated rupture of steam/water tubes. As pressure increases the rupture disks fail and sections of the IHTS piping are rapidly evacuated. Both the pressure transients and inertial loading of evacuation produce responses in the IHTS piping.

6. System Thermal Transient Loads

System operating transients such as plant heatup, cooldown, reactor scram, etc. cause changes in thermal expansion loading as described above and in addition may cause large through wall thermal gradients which must be considered in the evaluation. From heat conduction analyses, system thermal transients are analyzed to determine local thermal stresses in the piping system which in turn are combined with the other local calculated stresses.

Piping Component (Local Loads)

The pipeline flexibility analyses under system loadings generate data on displacements, forces and moments at selected points along the piping resulting from deadweight, thermal expansion, seismic conditions and other dynamic conditions such as for sodium/water reaction (SWR) loadings. Local membrane and bending stresses resulting from system pressures in the piping are determined by standard practices. From heat conduction analyses, system thermal transients are analyzed to determine local thermal stresses in the piping system.

The flexibility loads are combined in an appropriate manner and applied in the stress analysis of a local region of the piping system to determine the induced stresses and strains at the piping component level. These are added to the pressure and thermal stresses to obtain the total stresses for comparison with criteria.

Maximum allowable interface loads between the HTS piping and certain attached components such as the reactor vessel and the IHX are specified at the component nozzles. The nozzle loads determined from the flexibility analyses in terms of weight, thermal expansion, seismic, etc. (or combinations thereof) must be within these maximum allowables or redesign of the piping becomes necessary.

In the design of the piping, the interface interaction between components and piping is considered in one of two ways. Either the displacements of the component nozzles are imposed upon the piping or the component is included in the analysis model of the piping. When components are included in flexibility models it is necessary to consider the loads introduced by the relative motion of component support locations. For example, under seismic conditions it may be found that supporting floors have movement relative to each other. Loadings caused by such interface conditions shall be identified and considered in the structural evaluation.

ANALYSES

The evaluation of the HTS piping design is made in accordance with the methods outlined in recognized nuclear industry codes and standards, namely the ASME B&PV Code and RDT Standards. The governing Code and Standards for the piping are identified in the introductory paragraphs of this response.

The evaluation of the HTS piping includes flexibility analyses, heat transfer analyses, and stress analyses; these types of analyses are described in the following subparagraphs.

Flexibility Analyses

The objectives of the flexibility analyses are to determine moments, forces and deformations induced in a piping system due to the types of loadings discussed previously. To a large extent the flexibility analyses consider elastic formulations; the piping is designed, wherever practical, such that the stresses are sufficiently low to ensure elastic behavior. In regions where inelastic behavior is expected, non-linear flexibility analyses are made.

Procedures for constructing elastic flexibility models are based on finite element techniques, using matrix displacement methods. The specific computer finite element model or flexibility model for each pipiing system is composed of a series of pipe elements of the appropriate flexibility character with an appropriate lumping of mass at the intersection of each element (node point). The nodes are selected at changes of sections, at locations of equipment support, at equipment centers of gravity, at points of restraint, at special locations where response is desired and at intermediate locations to limit the length of the elements so that the model will adequately represent the actual system. The number of lumped masses or degrees of freedom shall be such as to insure compliance with requirements of PSAR Section 3.7.2.3. Other assumptions common to this type of analysis include:

- o Deflections are small in proportion to the size of the configuration so that changes in position and shape of a member are ignored in their effect on flexibility of the whole, and
- o effects of direct axial compression or extension, or of shear deflection, are negligible in comparison with bending and torsional effects.

The influence of localized effects on deflections and rotations is provided by the inclusion of flexibility factors in the formulations. Guidelines for the calculation of the flexibility factors are given in Subparagraph NB-3687 of the ASME B&PV Code, Section III. It is noted that the formulations tend to overestimate the stiffness, and therefore are conservative. This same guideline is employed in constructing the flexibility indices for items such as nozzles and anchors for which no ASME Code procedures are specified.

To the maximum practical extent, the HTS piping flexibility analysis models are defined so as to include the connecting component, pipe or auxiliary equipment. This minimizes the number of points at which seismic inputs must be determined. This also avoids over conservatism at interfaces between items when the stiffness of one is not negligible relative to the other. Further, this approach minimizes the number of flexibility analysis models and leads to a consistency among the models used for the several types of static and dynamic loadings.

Representations which are included in the piping system models to represent the connected components (e.g., Reactor Vessel) are checked against more detailed component models to assure correct dynamic response prediction. Piping-type stick elements and lumped masses are used for modeling components just as for piping sections. Component support stiffnesses are included in such models and where appropriately defined, equivalent support of floor masses are included. Models also include modeling of shell nozzle flexibility.

Elastic flexibility analysis for the PHTS and IHTS in-containment piping are made with the WECAN or WESTDYN computer programs. For deadweight and thermal expansion analyses, linear elastic models of the piping from the component nozzle anchors are used. For seismic analyses, extensive use is made of the response spectrum method in accord with the Appendix A to PSAR Section 3.7. Time-history analyses are used wherever the response spectrum method is judged to be overly conservative. For sodium/water reaction (SWR) and SMBDB conditions, the response of the IHTS and PHTS piping, respectively, is determined by integrated time-history analysis, using forcing functions that are prescribed as force-time histories at change-in-direction and flow restriction locations in the piping system.

The piping seismic flexibility analysis for the PHTS 36" Hot Leg includes the reactor vessel, the outlet downcomer and the pump. The PHTS 24" Hot Leg model includes the pump and IHX. The PHTS 24" Cold Leg model includes the IHX, check valve, reactor vessel and inlet downcomer. The in-containment IHTS 24" Hot and Cold Leg models include the IHX and models of the RCB penetration seals. For the SWR analysis of the in-containment IHTS piping, a portion of the ex-containment piping is also added to models. The seismic models for the small-diameter PHTS piping (IHX vent, pump bubbler, and pump drain lines) include the connecting component models (IHX and Pump) as well as the guard vessels when the piping is supported off a guard vessel.

The elastic flexibility analyses for the IHTS ex-containment piping are made with the SAP computer program. Post processing programs are used to made the stress calculations in accordance with the ASME Code. The models for the flexibility analyses of the IHTS ex-containment piping are briefly described as follows:

Hot Leg

The Hot Leg model includes the superheater inlet and outlet, the evaporator inlet and an anchor at the penetration of the HTS cell of the Reactor Containment Building. Normally 3 to 5 thermal expansion cases are performed. These expansion cases will envelope all of the thermal operating conditions. The displacement of the equipment nozzle depends on the particular thermal expansion case being analyzed.

The superheater inlet nozzle is treated as an equivalent pipe that is carried to the superheater shell; at the shell the equivalent pipe is rotationally fixed.

At the superheater outlets, the superheater nozzles (two) and the superheater shell are treated as equivalent pipes. The shell equivalent pipes (one for each side of the shell) are carried to the vertical centerline of the superheater. At this point, the equivalent pipes are rotationally fixed.

The nozzle equivalent pipe is rotationally fixed at the shell.

At the evaporator inlet the evaporator nozzle and the evaporator shell are treated as equivalent pipes. The shell equivalent pipe is carried to the vertical centerline of the evaporator. At this point the same equivalent pipe is rotationally fixed. The nozzle equivalent pipe is rotationally fixed at the shell.

At the anchor of the RCB penetration, the piping is geometrically fixed both with respect to rotational and linear displacements.

Cold Leg

The cold leg model includes the evaporators (outlets), the pump (inlet and outlet) and an anchor at the penetration of the HTS cell of the Reactor Containment Building.

At the outlets of the evaporators, the evaporator nozzle and shell are treated as equivalent pipes. The shell equivalent pipe is carried to the vertical centerline of the evaporator. At this point the same equivalent pipe is rotationally fixed and the thermal displacement imposed. Also, the nozzle equivalent pipe is rotationally fixed at the shell.

The pump inlet is treated in a similar manner. At the inlet, the pump nozzle and shell are treated as equivalent pipe. The shell equivalent pipe is carried to the vertical centerline of the pump. At this point the equivalent pipe is rotationally fixed and the thermal displacement imposed.

At the pump outlet, the pump nozzle is treated as an equivalent pipe. This equivalent pipe is rotationally fixed at the pump shell.

At the anchor at the penetration of the Reactor Containment Building, the cold leg is geometrically fixed - both with respect to rotation and displacement.

Nonlinear flexibility analysis is required for fill and drain loading conditions for the PHTS large piping. This need arises from the use of constant load hangers to support the piping. The hanger load values are set as appropriate for the filled condition. When empty these forces lead to excessive stress and deformation. Devices to limit the travel of the hangers are required and the determination of appropriate limiting values involves a flexibility analysis which is nonlinear due to the changing free/fixed conditions of the hangers during fill and drain.

An inelastic flexibility analysis is required for the PHTS 24-inch hot leg where the calculation of induced forces on an elastic basis is excessively conservative because stress relaxation is not accounted for. The use of an inelastic flexibility analysis to calculate the forces applied to local regions is not considered to invalidate the use of elastic analysis rules in evaluation of the local region for compliance with applicable structural integrity requirements. The inelastic flexibility analysis is performed with the MARC computer program using the curved pipe finite element model specifically developed for such analyses.

No significant design analyses for short-term primary or long-term creep buckling of the in-containment piping is required. Load-controlled forces that can lead to buckling due to short-term loadings are kept small by the piping support arrangements. The predominant operating stresses on the piping are due to thermal expansion and thermal transients (or deformation-controlled loading). Buckling in straight pipe sections under deformation-controlled loadings does not pose a problem because of the low axial load levels and the limited deformations that could result. Also, the buckling or plastic collapse of the elbows in the HTS large-diameter piping is not a practical mode of failure because rotations of the elbows are limited by the piping support system.

Heat Transfer (Thermal Transient) Analysis

Thermal transients are the source of some of the largest variations of stress in the HTS piping. Thermal analysis of piping temperature distributions during such occurrences are therefore an important part of the structural integrity assessment for the piping. In this section, the procedure and principles employed for HTS piping temperature distribution analyses are described.

The transient events used as the basis for piping design/analysis are specified in the piping design specification for each section of piping loop in the piping system. Thermal hydraulic data, in the form of temperature, flow and pressure time plots, are given for each thermal transient in the specification.

The heat transfer analysis of the piping components is carried out with finite element programs because these analyses can become complex. Some examples of when a detailed analysis are necessary include:

(1) When axial heat flow as well as through-the-wall flow is significant such as in a branch connection.

(2) when the specified sodium transient is complex.

(3) When the radiation mode of heat transfer is significant along with convection and conduction.

ANSYS, WECAN and TFEATS are the basic computer programs used to solve temperature distribution problems for the HTS in-containment piping system. Geometry generators are internal to these programs which are used to prepare models and input for both one-dimensional and two-dimensional heat transfer problems.

The computer programs have the capacity to determine through-wall temperature gradients in the piping as a function of time for time-dependent input functions of mass flow rate and bulk fluid temperature. They have the capability of decomposing the through-wall temperature distribution into three components as described in sub-paragraph NB-3653 of the ASME B&PV Code, Section III. These three components are the wall average temperature (T_a), the moment generating equivalent linear distribution (T_1) and the nonlinear portion with zero average value and zero first moment with respect to the mid-thickness (T_2). These quantities are used with ASME code formulas to determine secondary and peak stresses for use in the ratchetting and fatigue evaluation of the piping components.

Several locations in the piping system required special thermal analysis such as at nozzle-to-pipe joints, flued heads, tapers, branch connections, etc. For these general thermal analyses, the large finite elements programs are readily applicable. For nozzle-to-pipe joint discontinuities, the thermal response of the structure on each side of the interface is determined by calculating the radial temperature distribution at various time periods during the various thermal transients and then calculating the average temperature (T_a and T_b as defined in the ASME Code) as a function of time for each region. The maximum temperature difference (T_a - T_b) between these two quantities during each transient is used to determine peak thermal discontinuity stresses at the interface in accord with the ASME Code formulas.

Stress (ASME Code Evaluation) Analysis

The technical approach taken in analyzing and evaluating piping components for compliance with structural integrity requirements follows the procedure outlined in Figure Q110.78-P-2 (Blocks 1 through 9). The process starts off with the through-wall temperature results (T_1 T_2 and T_a-T_b) obtained from the heat transfer analysis as discussed previously and as shown in Block 1. elastic flexibility analysis loads (forces, moments and displacements) are then obtained for all the piping system load conditions as indicated by Blocks 2 and 3.

In all cases the analysis and evaluation of the piping components proceeds on an elastic basis as shown in Block 4. The ELTEMP computer code is used to analyze and evaluate piping components on an elastic basis in accord with Code Case 1592 and RDT Standard F9-4T.

The ELTEMP computer program is operational and the calculations performed by ELTEMP have been verified. This computer program is considered to be satisfactory for use in elastic evaluation of HTS piping.

The ELTEMP computer program will be documented, revised and maintained as a part of the structural evaluation program for the HTS piping and will be used in the preparation of final stress reports for CRBRP Class 1 piping. At present, the applicable ASME Codes and RDT Standards do not provide specific rules for piping at elevated temperature; only general rules in accord with NB-3200 of the code are provided. To assure consistency, the preparation of the ELTEMP computer program is coordinated with other efforts to prepare special rules for elevated temperature piping for inclusion in the applicable Codes and Standards.

A number of fallback approaches are used to assess piping components which are not shown to be satisfactory on an elastic basis using Blocks 3 and 4. If the reason for noncompliance is judged to result from high thermal transient stresses, the procedure is modified to use an elastic flexibility analysis and an inelastic analysis of the piping component (Blocks 3 and 6). If the reasons for noncompliance is judged to result from excessive flexibility analysis forces, the procedure is modified to use an inelastic flexibility analysis and an elastic piping component analysis (Blocks 5 and 4). In some cases both the flexibility and component analyses need to be inelastic (Blocks 5 and 6).

It is anticipated that the foregoing four alternatives will be adequate for the design/analysis of most of the HTS piping components. However, the applicability of the CHERN computer program must be verified for use under nonaxisymmetric conditions.

Block 7 identifies an inelastic analysis approach of last resort. This type of analysis is required for piping systems that are highly loaded by thermal expansion and thermal transient stresses such as the PHTS 24-inch hot leg between the pump and IHX. Block 8 is included in Figure Q110.78-P-2 to indicate that some individual discontinuity regions may require detailed multi-dimensional inelastic analysis. In addition, limited indirect use of these types of analyses is anticipated in the verification of simpler design analysis procedures.

The scope of analysis as described above does not include consideration of vibration. The potential for excitation of vibration is considered in the structural evaluation program. The effects of vibration induced by the pump impellor are considered for the PHTS hot leg piping connected to the pump.

An analysis checklist for the piping is included in Figure Q110.78-P-3. This checklist will be expanded in size and detail as the analyses progress.

MATERIAL PROPERTIES

The elastic material properties used in the structural evaluation of the HTS piping are specified in the ASME Code Documents. The Nuclear Systems Materials (NSM) Handbook (TID-26666) is used as the authoritative source for material properties not specified in the applicable Code Documents. Where material properties are required which are not available in the Nuclear Systems Materials Handbook or the ASME Code, action will be taken using procedures approved by the CRBRP Project Office to obtain the required material properties.

A collection of computer files containing material property data, routines for interpolation and routines for material deformation models, which are based on the materials data given in the Nuclear Systems Materials Handbook and the ASME Code, are used for analysis of the HTS piping (and other CRBRP components).

Thermal and mechanical properties are considered in the selection of materials for use in the HTS piping. Further, consideration is given to material properties in connection with fabrication procedures. For example, the need and procedure for accounting for the effects of cold work in the design analysis is examined. A thickness allowance is provided, in the manner described in the ASME Code Section III Subsection NB-3600, to account for the effects of corrosion and erosion.

For material properties critical to the design/analysis process, consideration is given to the use of minimum, average or maximum properties as appropriate to obtain a reasonably conservative result. For example, in certain critical situations the evaluation of deformation limits is based upon minimum stress-strain curves. The selection of appropriate properties is guided by RDT F9-5T.

Material Degradation

Most of the data used to define the allowable design stresses in the ASME Code were obtained from tests conducted in air. No attempt is made in the Code to account for the effects of other service environment. The LMFBR development program has focused attention on the mechanical behavior of reactor materials when exposed to high-temperature liquid sodium, in addition to fast neutron irradiation and to long time aging at elevated temperatures. Guidance for establishing the effects of the service environment upon the response and failure characteristics of the structural materials is summarized in Table Q110.78-P-1. A brief discussion of the environmental effects is given in the following paragraphs.

Thermal Aging Effects on Mechanical Properties:

Types 304 and 316 stainless steels are non-age hardenable alloys. Thus, no significant changes in strength or hardness of annealed material would accrue from long term aging at temperatures up to 1200°F, unlike the precipitation-hardened stainless steels. Some slight increases in strength and decreases in ductility may occur due to carbide formation, together with a reduction in the room temperature impact strength. Of more significance, is the fact that these alloys will sensitize during long term service in the temperature range from 800° to 1500°F. In this phenomenon, carbide precipitation occurs at the grain boundaries, the adjacent matrix becomes depleted in chromium and the grain boundary regions become susceptible to attack by corrosive media. Such attack is not likely to occur in sodium, which, if pure, is a relatively inert environment. However, cracking may initiate during fabrication and the other pre-operation periods when the component is not exposed to sodium, due to environmental conditions (presence of water and halides). Because of this, precautions must be taken during such periods to ensure that contact between sensitized material and potentially corrosive media is avoided. Hence, no allowances have been made for the effects of thermal aging on the properties of types 304 and 316 stainless steels used in the HTS piping. This did, however, demand that control be specified and exercised during the fabrication process to prevent stress corrosion and intergranular attack.

Neutron Irradiation Effects on Mechanical Properties

The effect of neutron irradiation on the mechanical properties of a material are generally to increase the tensile and yield strengths, and to decrease the ductility. The actual magnitude of the effect is dependent on several parameters, such as the temperature of irradiation, the test temperature, the neutron energy spectra and the neutron fluence.

Two alternate procedures have been used to account for the effects of neutron irradiation on the structural integrity of components. The comprehensive approach characterizes the effects of the neutron irradiation upon each material response and failure mode considered by the ASME Code.

When necessary, additional failure modes are considered. The alternate approach involves finding the threshold where irradiation effects first become measurable (in terms of structural response and integrity). If irradiation levels are held below these threshold levels, no fluence effects will take place.

For austenitic stainless steels (Types 304 and 316), measurable loss of ductility (total elongation) can first be detected at about 10^{21} nvt total fluence for temperatures in the range of 600°F to 1100°F. The pipe wall end-of-life fluence for the HTS piping within the reactor cavity is less than 1×10^{20} nvt and hence no fluence effects are expected.

Effects of (Nitrogen + 2% Oxygen) Atmosphere on Mechanical Properties)

The selection of nitrogen gas as the atmosphere for the reactor cavity and HTS cell was based on the desire to prevent chemical reactions should molten sodium leak into the cavity and HTS cell from any source. However, the exposure of austentic stainless steel to pure nitrogen for extended periods of time at elevated temperatures may lead to the formation of a thin nitrided layer. This is considered undesirable because of the brittleness of such layers. To minimize the formation of such a layer, a small percentage of oxygen (<2%) is introduced into the nitrogen.

Effects of (Argon-Plus-Sodium-Vapor) Atmosphere on Mechanical Properties

Very little is known of the effects of exposure to an argon-plus sodium-vapor atmosphere on the mechanical properties of austenitic stainless steel. It is possible that, if the sodium vapor is continually condensing on the material surface and rejoining the main reactor coolant, there could be some interstitial transfer. However, because of the scarcity of data, it is not possible to provide quantitative assessments of such effects at this time. Practically, the potential for significant mass transfer via condensation is insignificant. It is judged that exposure to the cover gas should be considered the same as exposure to liquid sodium without loss of interstitials.

Surface Effects of Liquid Sodium on Mechanical Properties

The interactions of the sodium environment with the material, excluding interstitial transfer effects, may be defined as surface effects. Compared with air testing, liquid sodium may cause certain metallic elements to be transferred from the hotter to the cooler regions of HTS systems. In addition, surface oxidation in liquid sodium is greatly reduced when compared to air testing. It is believed that these surface effects are insignificant in their influence on short-term tensile properties.

For time-dependent deformation, such as stress-rupture and fatigue, the effects of a liquid sodium environment are complex and need to be considered in detail. In the case of stress-rupture, it has been shown that for a given temperature and stress, rupture times in air are longer than those in liquid sodium. Figure Q110.78-P-4 gives a sodium-environment correction factor which applied to the rupture strength data specified in ASME Code Case 1592 for Types 304 and 316 austenitic stainless steel. This effect is used in all evaluations where stress-to-rupture is involved.

Fatigue properties of materials can be greatly affected by the environment in which the properties are measured. The avoidance of excessive surface oxidation by testing in sodium (or inert gas) instead of in air increases the cycles-to-failure for a given strain range. These increased cycles-to-failure values observed when testing in sodium are being independently verified. No increase in the design fatigue limits due to exclusion of oxygen effects is permitted.

Interstitial Transfer Effects on Material Properties

In the HTS piping, interstitial carbon and nitrogen are transferred from the hotter to the cooler regions. This leads to weakening in the decarburized and denitrided regions and to strengthening in the carburized and nitrided areas. In the case of fatigue behavior, however, the effects of interstitial absorption at the surface are complicated because of two concurrent mechanisms. On the one hand, carburization can lead to enhanced crack nucleation at carbide particles and, on the other, surface strengthening during strain-controlled fatigue will increase the proportion of elastic straining which is less damaging than plastic deformation. Studies indicate that, in general, the austenitic materials will be carburized. However, the crossover from carburization to decarburization is system dependent and it is likely that in certain systems at least some of the austentitic material will be decarburized. Procedures have been established for the CRBRP piping by which the extent of interstitial transfer for Types 304 and 316 stainless steel can be determined and from this the effects of mechanical behavior can be calculated.

ASME Code Case 1592 requires a minimum carbon content of 0.04 percent for austenitic stainless steels. To compensate for any interstitial transfers, the material is being ordered with a minimum carbon content of 0.055 percent for the primary hot-leg piping. This additional carbon percentage is considered sufficient to account for carbon depletion in the high temperature regions of the primary loop.

Although some carbon depletion is expected in the intermediate HTS piping, the specified minimum carbon content of 0.04 percent is considered adequate.

STRUCTURAL VERIFICATION TESTS

The planned technical approach for the design/analysis of the HTS piping includes the use of some methods, procedures, designs, etc. that are not fully developed, substantiated or verified. To account for this, appropriate testing and verification will be carried out as part of the HTS piping design/analysis effort and/or LMFBR base technology programs which are considered relevant to the structural evaluation of the HTS piping. Table Q110.78-P-2 provides a list of completed and ongoing base programs which will contribute to verification of design methods and, hence verification of the adequacy of the HTS piping.

Of specific importance is the qualification testing of the CRBRP HTS piping support system identified as items 3 and 12 on Table Q110.78-P-2. These programs will qualify the load carrying capability of the vertical and horizontal pipe clamp designs. They will also evaluate and assess the mechanical snubbers and constant load hangers when used in combination with the pipe clamps as a complete pipe support and restraint system. Under these programs, models also will be developed and verified for piping restraints for use in the design/analysis of LMFBR piping systems.

To determine the performance capabilities of the pipe clamps, testing has been done at various temperatures up to 1015°F and under various static and dynamic loadings. The test articles were instrumented with thermocouples, strain gages, accelerometers, load transducers and displacement transducers to gather data from which stresses and strains were calculated. The clamps were inspected during testing to ensure proper fit during thermal expansion movements of the pipe test section and to assure load carrying capability during seismic-type shocks and vibrations.

Testing was performed on 24-inch pipe clamps. Shock loadings at forty different frequencies from 5.64 Hz to 62.98 Hz and with corresponding forces between 2 lbs. and 19635 lbs. were used.

Results from all the direct CRBRP test programs and appropriate base technology programs for piping will be used to provide assurance that designs are structurally adequate and analysis calculations reasonable in the certified ASME stress report and FSAR.

COMPUTER PROGRAMS

Responses to the NRC questions 110.27 and 110.58 provide information relating to the computer programs used for the static, heat transfer, and dynamic analysis of Seismic Category I structures. Of these, the following computer programs will be used for the analyses of HTS piping within containment:

1. WECAN

2. WESTDYN

3. MARC

4. ELTEMP

5. CHERN

6. ANSYS

7. TFEATS

All the above programs have been described in Appendix A of the PSAR and hence are not repeated in this response. Appendix A also provides information relating to the adequacy of these codes and verifications that have been completed or planned. Where verification studies are in progress, the results will be provided in the FSAR.

TABLE Q110.78-P-1 SUMMARY: ENVIRONMENTAL EFFECTS UPON ANNEALED TYPE 304 AND 316 SST

Stress Intensity Category	Specific Item	Sodium Exposure	Loss of Interstitials Effect	Loss of Interstitials Basis+
Primary Limits	Sr	Degrading	Degrading	Ave
	Sy	-*	Degrading	Ave
	S_u	-	Degrading	Ave
	Creep Eqn	-	DNI	Ave
	Onset Tertiary Creep	-	DNI	Ave
Primary & Secondary (and Buckling)	S^r	Degrading	Degrading	Ave
	S_y	-	Degrading	Ave
	S_u	-	Degrading	Ave
	Creep Eqn	-	DNI	Ave
	Onset Tertiary Creep	-	DNI	Ave
	Creep Hardening	-	DNI	Ave
	Stress-Strain Eqn	-	Modification	Ave
	Cyclic Hardening	-	DNI	Ave
	ϵ_{total}	-	Improvement	Ave
Peak	S_r	Degrading	Degrading	Point***
	S_y	-	Degrading	Point
	S_u	-	Degrading	Point
	Creep Eqn	-	DNI	Point
	Onset Tertiary Creep	-	DNI	Point
	Creep Hardening	-	DNI	Point
	Stress-Strain Eqn	-	Modification	Point
	Cyclic Hardening	-	DNI	Point
	Fatigue Curve	DNI**	DNI	DNI
	Creep-Fatigue Interaction	-	DNI	DNI
	Stress Rupture	-		
	Notch Effect	-	DNI	DNI
	Fatigue Notch Effect	-	DNI	DNI
	Saturation of Hold Time Effects	-	DNI	DNI

*No significant effect expected
**DNI - Design information not included
***Point - Instantaneous value or peak, not average values.

+ Basis - Denotes whether point (C+N) content or average content is used as the basis to establish the effect.

TABLE Q110.78-P-2

BASE TECHNOLOGY PROGRAMS IN SUPPORT OF THE HTS PIPING DESIGN EFFORT

Item	Title (With Objective)	Status
1	Thermal Transient Facility (Analysis and test to verify inelastic predictions of ratchetting of piping - welded pipe and Croloy-to-304/316 SS joints)	Ongoing
2	Transition Weld Development	Ongoing
3	Piping Supports (Establish design of piping supports using load bearing insulation - FFTF type)	Complete
4	Mixing Tee Studies	Complete
	a. Hydraulic Tests of SPTE Mixing Tee Model (Water Tests)	Complete
	b. Mixing Tee Considerations for FFTF (Water and Sodium Tests)	Complete
	c. CRBRP IHTS Mixer Thermal-Hydraulic Model Tests	Complete
5	Fracture Mechanics Studies (To prove leak-before-break assumption)	
	a. Characterize Crack Propagation, Critical Crack Size & Crack Leakage	Complete
	b. Sodium Effects on Fracture Mechanics	Complete
	c. Corrosion Study of Sodium Leaking to Air	Complete
6	Evaluation of Formed & Welded Pipe	Ongoing
7	Multi-Loading-Test Facility (MLTF) (inelastic response of piping, straight pipe, elbow and tee sections)	Ongoing
8	Simplified inelastic design analysis procedure for piping systems, PIRAX-2	Complete

TABLE Q110.78-P-2 (Continued)

BASE TECHNOLOGY PROGRAMS IN SUPPORT OF THE HTS PIPING DESIGN EFFORT

Item	Title (With Objective)	Status
9	High Temperature Time-Dependent Characteristics of Materials in Sodium (Mechanical properties of stainless steels and Inconel 718 in air and sodium)	Ongoing
10	Simplified methods (Program for qualification of analysis models of reduced complexity and dimension)	Complete
11	Piping Restraint Effects	Ongoing

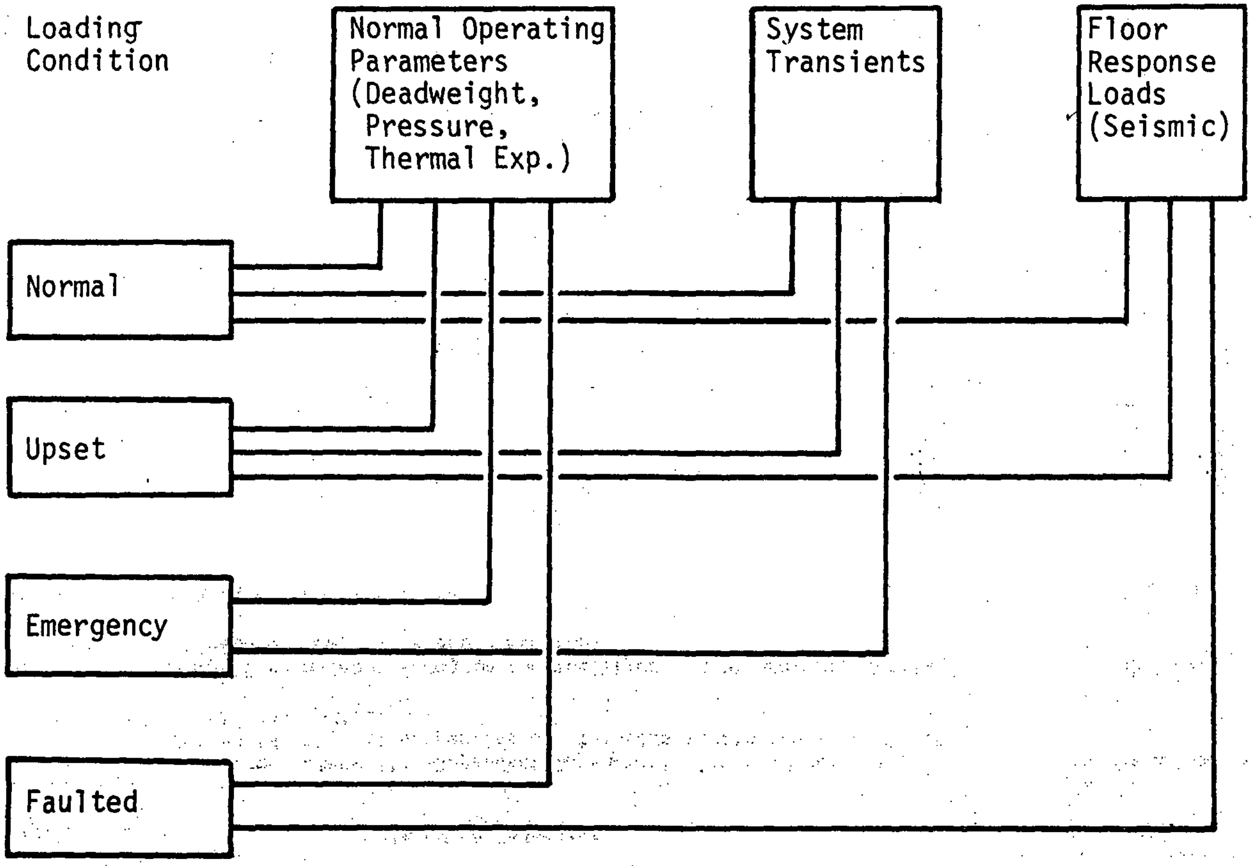

Figure Q110.78-P-1. HTS Piping and Support System Design and Analysis Process

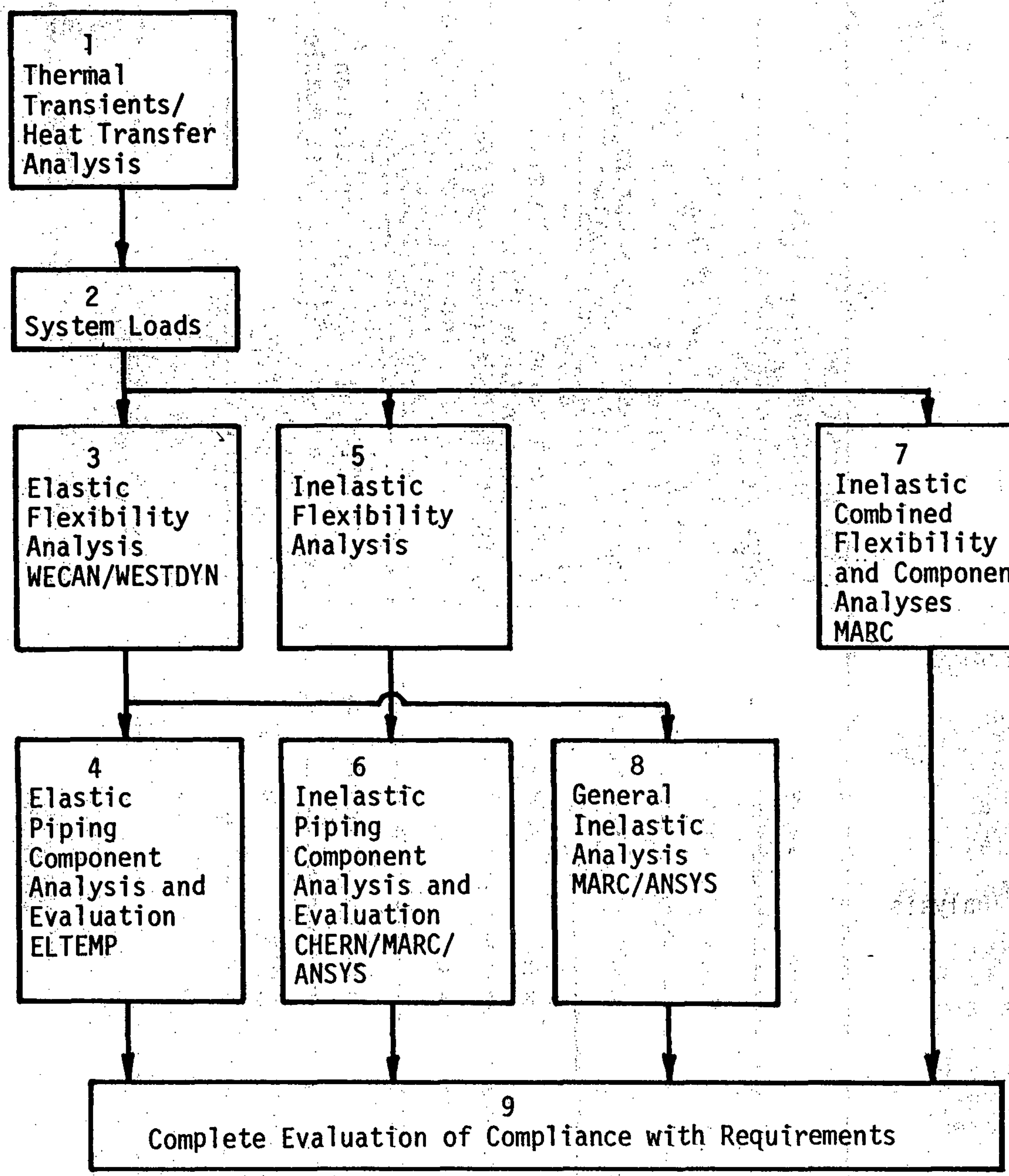

Figure Q110.78-P-2. HTS Piping Analysis Procedures

	PHTS 36" Hot Leg From Downcomer Outlet Elbow To Pump Inlet Nozzle	PHTS 24" Hot Leg From Pump Vessel Outlet Nozzle To IHX Vessel Primary Inlet Nozzle	PHTS 24" Cold Leg From IHX Vessel Primary Outlet Nozzle To Check Valve Inlet Nozzle	PHTS 24" Cold Leg From Check Valve Outlet Nozzle To Downcomer Inlet Elbow	IHTS 24" In-Containment Hot Leg From IHX Vessel Intermediate Outlet Nozzle To Hot Leg Containment Penetration	IHTS 24" In-Containment Cold Leg From Cold Leg Containment Penetration To IHX Vessel Intermediate Inlet Nozzle	PHTS 2" IHX Vent Line Between Pump Nozzle And IHX Nozzle	PHTS 6" Pump Bubbler Line Between Pump Nozzle And Inert Gas System	PHTS 2" Pump Drain Line Between Pump Nozzle And Auxiliary Liquid Metal System
Load Conditions									
A. Design	x	x	x	x	x	x	x	x	x
B. Normal/Upset	x	x	x	x	x	x	x	x	x
C. Faulted	x	x	x	x	x	x	x	x	x
D. Testing	x	x	x	x	x	x	x	x	x
E. Special	x	x	x	x					
Types of Analysis									
A. Thermal	x	x	x	x	x	x	x	x	x
B. Elastic	x	x	x	x	x	x	x	x	x
C. Inelastic									
1. One-D									
2. Two-D					x	x			
3. Three-D		x							
D. Dynamic	x	x	x	x	x	x	x	x	x
E. Buckling		x							
Design Criteria									
A. Sec. III									
1. Low Temp.									
2. High Temp.	x	x	x	x	x	x	x	x	x
B. Other									

Figure Q110.78-P-3 Analysis Checklist for System 51A Piping

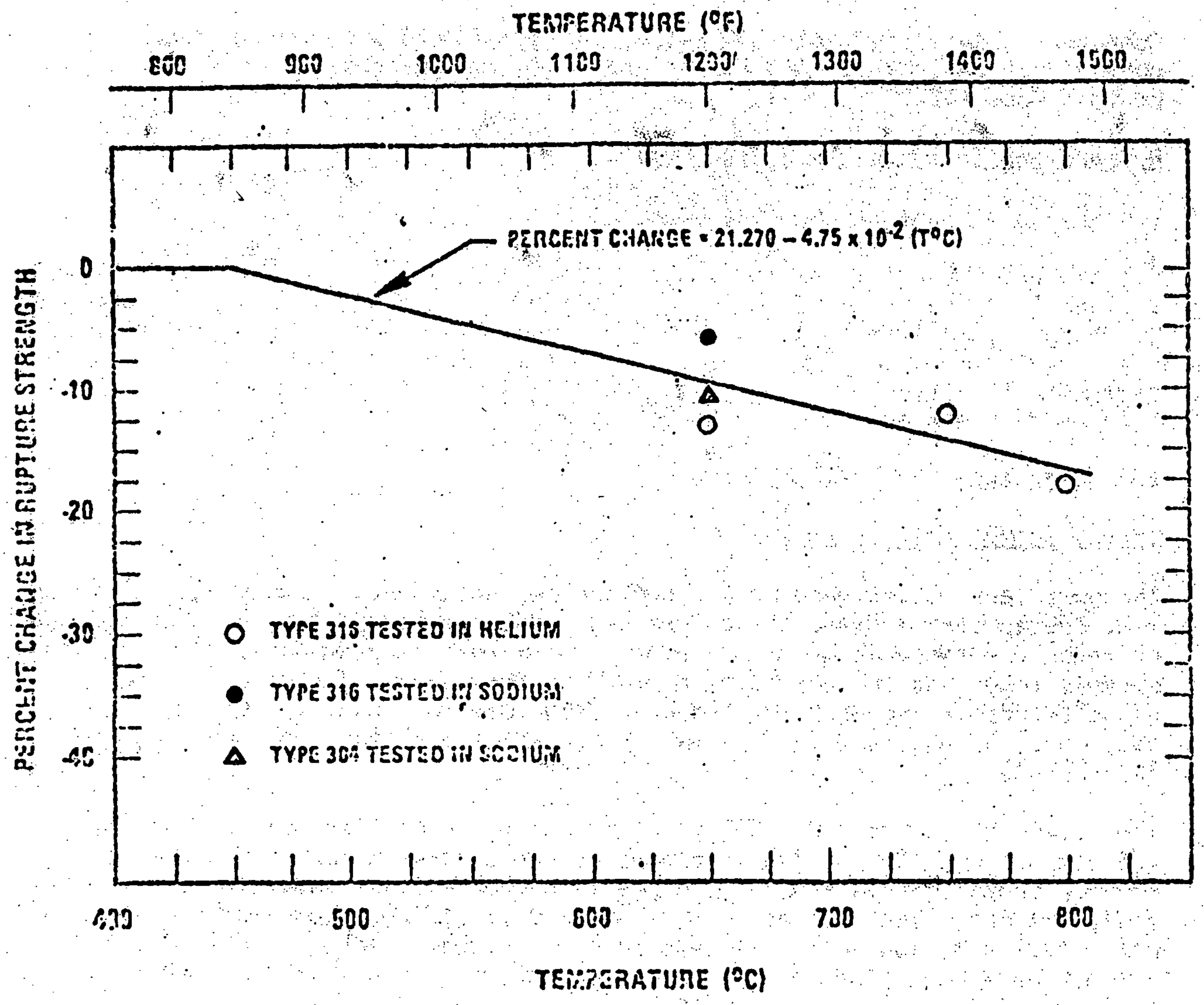

Figure Q110.78-P-4. Decrease in Rupture Strength of Type 304 and 316 SS Due to a Sodium Environment

Response for HTS Pumps:

Primary and Intermediate Sodium Pumps

The information presented below is based on that currently available in the pump structural evaluation plan. In general, these plans provide for the overall philosophy and analytical approach to be followed in the structural analysis and identify the conditions under which certain supplemental analysis may be required.

Components operating above 800^{O}F are considered "elevated temperature components" and have strain limits and creep-fatigue damage limits established by Code Case 1592.

Primary and Intermediate Sodium Pump

1.0 Failure Modes and Loading Conditions

For the purposes of loads and analysis the pump (see Figure Q110.78-PU-1) is divided into four subcomponents: Subcomponent 1 - the pump tank; Subcomponent 2 - upper inner structure including the pressure bulkhead; Subcomponent 3 - the rotating machinery; Subcomponent 4 - the static hydraulics section. The predominant failure modes and associated loading conditions for each subcomponent are addressed in Section 2.0.

2.0 Structural Evaluation Criteria

2.1 Subcomponent 1 - Pump Tank

The pump tank is designed to the ASME Boiler and Pressure Vessel Code Section III, Subsection NB Class 1 and Code Case 1592, where applicable. The cone and cylinder are designed mainly by dynamic stiffness requirements. These include seismic loads and the necessity of keeping the natural frequency of the structure above the operating speed of the impeller to avoid critical resonance during operation. STARDYNE and ANSYS computer codes are used for this analysis.

The lower end of the pump tank will operate above 800^{O}F on the primary pump, whereas all portions of the intermediate sodium pump normally operate below 800^{O}F. Creep effects for the intermediate pump will be shown to be insignificant. The evaluation of that portion of the primary pump above 800^{O}F will utilize methods of finite element coarse model inelastic analysis. These methods have been used on the FFTF intermediate heat exchanger as well as the intermediate heat exchanger of CRBRP.

The pump tank is divided into two groups for analytical purposes; namely, nozzles and the pump tank assembly.

Nozzles

For structural integrity of the pressure boundary, axisymmetric ANSYS models of the various nozzles will be used. Detailed time dependent temperature distributions will be determined with these models. Elastic piping loads will be evaluated using Fourier Series and axisymmetric ANSYS models. Because of the proximity of the discharge and suction nozzles in the pump tank sphere, a coarse three dimensional model will be used to assess their interaction for both mechanical and thermal loads. This work will be used to develop and justify a conservative set of boundary conditions for the axisymmetric analysis. The suction, discharge, and IHX-vent return nozzles and transition ring of the primary pump are creep-fatigue and strain limit critical.

The creep damage problem is magnified due to material decarburization which reduces the time-dependent allowable stress for a given time at 1015°F by 11%. Strain limit problems occur in the weld between the nozzle forgings and the pump tank sphere. These nozzles cannot pass the simplified elastic rules of Code Case 1592-1. Coarse model inelastic analysis using axisymmetric MARC models with axisymmetric loads will be used to demonstrate compliance with the design criteria of Code Case 1592-1. The unit histogram to be used will include six subcycles and will be run for six unit cycles. It is anticipated that the histogram will be composed of two normal cooldown conditions, two U-1a transients, and one U-1b transient on one U-8 and E-5 transient (see PSAR Appendix B for definition of transients).

An additional functional requirement of the discharge nozzles involves the slip fit with the discharge duct. This slip fit must be shown to remain open during the design life and not to enlarge to the point where leakage degrades pump performance to an unacceptable level. The problem will be evaluated using an axisymmetric model of the nozzle.

The standpipe bubbler nozzle of the primary pump is not subject to transients and significant pressure. Simplified methods will be used to demonstrate its structural adequacy. The drain nozzle will be shown acceptable by comparing it to the suction and discharge nozzles. The cover gas vent nozzle is not subject to transients and significant pressure. Simplified elastic methods will be used to demonstrate its structural adequacy.

As noted earlier, the intermediate pump does not involve creep design considerations. Except for U-11a and E-4a*, the transients of the intermediate pump nozzles are very mild. The elastic shakedown limits and fatigue damage will be evaluated using twice the stress range from the E-4a transient. The severity of transients U-11a and E-4a, are such that plastic design evaluation procedures (Paragraph NB 3228 of Section III) must be used to assess distortion and total strain range (for fatigue damage). The MARC models developed to evaluate the primary pump suction and discharge nozzles will be used. The histogram will consist of transient E-4a and will be run for four cycles.

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* E-4a was subsequently deleted as a plant transient, but it has been retained as a pump design requirement since it umbrellas other emergency transients in its severity.

Pump Tank Assembly

The pump tank assembly consists of the remaining parts of the tank (sphere, sealing cone support attachment junction, sphere-to-cone transition piece, cone, cylinder, and mounting flange). These parts can have an effect on the system requirement related to peak-to-peak nozzle vibrations. In addition, deformations of the sphere-to-cone transition piece are directly related to the pump tank impedance requirement.

A three dimensional (180^{o}) thermal and elastic ANSYS finite element model of the tank will be used to assess the three dimensional effects for mechanical and thermal loads on the sphere-to-cone transition piece. The results of this three dimensional analysis will be used to develop and justify axisymmetric models of the transition piece which will be used to evaluate the pressure boundary for safety requirements. This work will involve detailed thermal analysis for two thermal transients and eight mechanical load runs. Also the model will be used to develop the foundation stiffness of the trunnion type supports for the sealing cone assembly.

The axisymmetric analyses and structural evaluation of the sphere-to-cone transition piece and the sealing cone support attachment junction will follow the same general procedure described for the nozzles. The number of transients considered, the unit histogram, and the number of unit cycles run in the coarse model inelastic analyses will be the same. The cone and cylinder sections of the tank are not subject to severe thermal transients. Their design is based upon load controlled considerations (pressure and seismic) and the peak-to-peak nozzle motions. In the sphere the areas which are far from local discontinuities will be shown acceptable using one dimension CHERN inelastic analyses.

The analysis and evaluation of the closure and support flange complex will involve an equivalent axisymmetric interaction model. The model will use the gap element capability of ANSYS to assess the changing configuration effects due to surfaces moving in and out of contact. Sliding with friction will be considered. The bolts will be modeled with beams and local flexibility at the nut face and threaded areas will be considered. Non-axisymmetric over-turning loads (seismic and rocking vibration) will be evaluated using the model with Fourier Series. In this instance, the gap element will not be used to simulate the circumferential ring joint. The stiffnesses determined will be available for inclusion in the seismic and pump dynamics models. In addition, the dynamic pressure pulse for the SMBDB and sodium-water reaction will be evaluated.

2.2 Subcomponent 2 - Upper Inner Structure

The upper inner structure will conform to the same code requirements as the pump tank. The design of the upper closure plate and radiation shield is controlled by the design pressure and temperature requirements. Elastic failure is the predominant mode. The thermal shield will have steady state thermal gradients which will be determined by a 2D ANSYS axisymmetric model. The motor stand will be designed by the stiffness requirements of the motor and seismic loads. The principle failure mode will be buckling under SSE seismic load. Parts of the primary pump upper inner structure will operate above 800°F; none of the intermediate pump upper inner structure operates above this temperature.

2.3 Subcomponent 3 - Rotating Machinery

The rotating machinery can be removed and inspected after an emergency or faulted event and repaired before the plant is placed in service again. Therefore, this equipment will be designed and analyzed to the ASME Boiler and Pressure-Vessel Code, Section III, Subsection NB for Class 1 Components and Code Case 1592 where applicable. However, for emergency events when Code Case 1592 is used, the design rules for load controlled stresses (Section 3227) will apply. Strain deformation and fatigue analysis need only be performed up to the emergency event and the limits will apply only to the pumps' ability to operate at pony motor speed after the event. The shaft will be designed by critical frequency requirements, inertial loads, torque and thermal transients. Failure modes will be fatigue, shear failure and creep fatigue in the shaft. The upper journal has a local area which will be analyzed inelastically with a 2D ANSYS axisymmetric model. Loads caused by bearing misalignment will be accounted for. Portions of the rotating machinery in the primary pumps operate above 800°F; all rotating machinery of the intermediate pumps normally operate below 800°F.

2.4 Subcomponent 4 - Static Hydraulics Section

The hydraulic section consists of the lower removable region of the pump inner structure and the mating sealing cone mounting in the pump tank. It will be analyzed to same code rules as Subcomponent 3 (rotating machinery). The principle loads will be thermal transients, hydraulic pressure, containment of a failed impeller, reaction loads against the hydraulic machinery due to deformation of the sphere during the thermal transients and bearing loads due to axisymmetrical heating. Creep and creep-fatigue are the predominant failure modes. Portions of the static hydraulics section in the primary pumps operate above 800°F; all of the static hydraulics section of the intermediate pumps normally operate below 800°F.

For purposes of analyses, the static hydraulics section has been divided into two parts, the sealing cone assembly and the pump case assembly.

Sealing Cone Assembly

In order to satisfy the functional requirements (operability and performance) and to supply structural characterization (stiffnesses) for establishing adequacy with respect to system requirements (peak-to-peak nozzle motions), extensive three dimensional analysis of the assembly's support on the pump tank is required. With respect to the functional requirements, the following gross distortions must be considered:

a) Time dependent (steady state loads and residual stress from plastic action during thermal transient and/or seismic DBE events).

b) Time independent (plastic action during steady state and transient conditions).

c) Time independent (elastic action during steady state and transient conditions).

With respect to system requirements and the pump seismic analysis, the load path and stiffness characteristics between the inner structure and pump tank are needed. The interaction of the sealing cone assembly with its support on the pump tank and the pump case will be established.

A full three dimensional (360°) model of the sealing cone will be used (elastic ANSYS). The gudgeon sleeve and supports will be substructured and included. Gross thermal distributions including circumferential variations due to low flow rate conditions will be determined. The detailed three dimensional analysis will be used to justify two dimensional models which will in turn be used to establish time dependent distortion and stresses.

Two elastic-plastic-creep MARC analyses will be used. The first will be a two dimensional (Rz) analysis to assess the axial distortion of the cone. The second will be a two dimensional (RG) analysis to assess the ovalization of the sealing cone at different elevations. As stated above, the three dimensional analyses will be used to justify the conservative two dimensional models. For the primary pump, a unit histogram of three subcycles will be run for six unit cycles for each of the three models. For the intermediate pump, the potential plastic ratcheting from the severe U-11a transient will be assessed using the same models with plastic action only. An objective of this analysis is to show that the sealing cone/hydraulic assembly radial gap does not increase in a manner which would degrade pump performance with respect to functional requirements.

The structural adequacy (code-type evaluation) will be evaluated for the following:

a) The discharge duct-to-sealing cone junction (modeled as an equivalent axisymmetric problem).

b) The support assembly is basically the same as the discharge nozzle and therefore can be shown adequate by comparison.

c) The cone - the 2D and 3D elastic models will be used for regions far from discontinuities.

Pump Case Assembly

Extensive three dimensional analysis of the pump case assembly is required in order to satisfy the functional requirements (operability and performance) and to supply structural characterization (stiffness) for establishing adequacy with respect to system requirements (peak-to-peak nozzle motions).

With respect to functional requirements, the following gross-type distortions must be considered:

a) Time dependent events (steady state loads and residual stress from plastic action during thermal transients and/or seismic DBE events). Misalignment of bearing housing due to distortion of the volute in the axial direction, and ovalization of the volute in the circumferential direction.

b) Time independent events (plastic action during thermal transients and seismic events).

c) Time independent events (elastic action during steady state and transient conditions).

d) Ovalization of the bearing housing during thermal transients due to the journal being offset from center under low flow conditions.

With respect to system requirements and the pump seismic analysis, the load path and stiffness characteristics between the inner structure and pump tank are needed. The interaction of the pump case (volute, bearing housings, and cylindrical attachment) with the sealing cone assembly and inner structure will be established. The load transfer across the lugs which connect the attachment cylinder and volute casting is important because plastic action would change the as-manufactured alignment of the two housings.

The structural adequacy (code-type analysis and evaluation) will be evaluated for the following items. The general approach for the items to be analyzed is as given for the nozzles:

a) Lower bearing housing lugs and volute-axisymmetric approximation derived from the above three dimensional analysis will be used to justify this approach.

b) Upper lugs, cylinder and volute-axisymmetric approximation will be used in the three dimensional analysis results above.

c) Attachment cylinder, baffle and bolted joint - the axisymmetric approximation will also be used to determine baffle motions for bubbler impedance.

d) Upper bearing housing attachment cylinder, baffle and bolted junction.

e) Lower bearing housing.

MARC inelastic analyses will be performed to evaluate creep damage and the effects of ratcheting strains on bearing operability. This inelastic evaluation will include axisymmetric 2D models of the upper bearing complex and of the upper case housing.

2.5 Seismic Analysis of the Pump

The pumps will be seismically analyzed using both the response spectra and the time history methods. Response spectra solutions will provide upper bound seismic loads for use in general stress analysis. Time history analysis will be performed for evaluation of more critical regions and of interaction effects such as journal/bearing impact during seismic events. Also interactions occur through the pump case and sealing cone assemblies which are critical with respect to functional and system requirements. The interaction is nonlinear due to the gaps which will open and close during the seismic event. The local impacts which result when the gaps close will be considered. The model will be developed using the dynamic options of the ANSYS and STARDYNE computer programs. ANSYS will employ axisymmetric conical shell and continuum elements with non-axisymmetric loads. The STARDYNE model will be a 3 dimensional beam representation. The steps which will be carried out are as follows:

a) Local stiffnesses will be developed at points of internal support. This will be done by means of small static computer models or by hand.

b) The remainder of the linear dynamics model will then be developed. This includes the addition of any fluid masses and external mechanical masses such as the motor and/or piping.

c) The time-history input loadings will be developed on tape from the support foundation time-history acceleration for the DBE and SSE events.

d) The size of the above model will be reduced by substructuring techniques in order to lower solution run times.

e) The STARDYNE model will be run simultaneously in all three directions. The ANSYS model will be run in each of three directions and the results stored on tape. The results will be scanned to determine the maximum response points at various critical points.

f) Internal forces and/or stresses will be derived for use in subsequent stress evaluation.

g) Final results will be developed and tabulated for the entire unit. The results of the analysis along with the details of the model will be summarized in a final report.

2.6 Overall Pump and Foundation System Dynamic Analyses

For the overall system analyses of the pump and the drive motor system, the model will include the coupling effects of the foundation and interconnecting piping. The system is analyzed with the finite element HASTRAN, ANSYS and STARDYNE codes. A detailed model of both the drive motor and pump that includes the foundation elements and main piping spring mass elements will be used.

3.0 Structural Test to Support Analysis

At present it is planned to design by analysis. However, half scale and full scale water tests have been run to determine and adjust the pump performance characteristics. A dynamic analysis of the pump operating in the water test setup was made and the water test results confirmed the analysis in all cases. The pump shaft's mid-span deflection was measured and confirmed the maximum T.I.R. of 0.017" as predicted in the dynamic analyses. The test results indicated hydrostatic bearing lift-off at all operating conditions as predicted by the dynamic analysis and confirmed the hydrostatic bearing load capability of the pump. A prototype pump will be tested in sodium for the upset thermal transients identified in the pump specification, up to the facility capability (temperature increase of 400°F up to 1000°F and decreases of approximately 500°F). Full scale water tests will be run on the plant pumps to determine and adjust their performance characteristics.

4.0 Relevant Programs from Other Facilities

Experience gained from the FFTF sodium pump tests has been applied to the design of the CRBRP pumps where applicable. In particular, bearing clearances in the CRBRP pump are being adjusted to compensate for Type 304 stainless steel shrinkage as observed in the FFTF tests.

5.0 Computer Programs

Computer codes used in the pump dynamics analysis are: ANSYS, HASTRAN and STARDYNE.

The computer codes used in the pump structural analysis are: ANSYS, HAFMAT, LPGEN, MARC, N-1045, N-1050, N-2050, N-2060, PRINCP AND SINDA.

ANSYS, HAFMAT and MARC codes are described in Appendix A of the PSAR. The remaining codes will be added in an upcoming PSAR amendment.

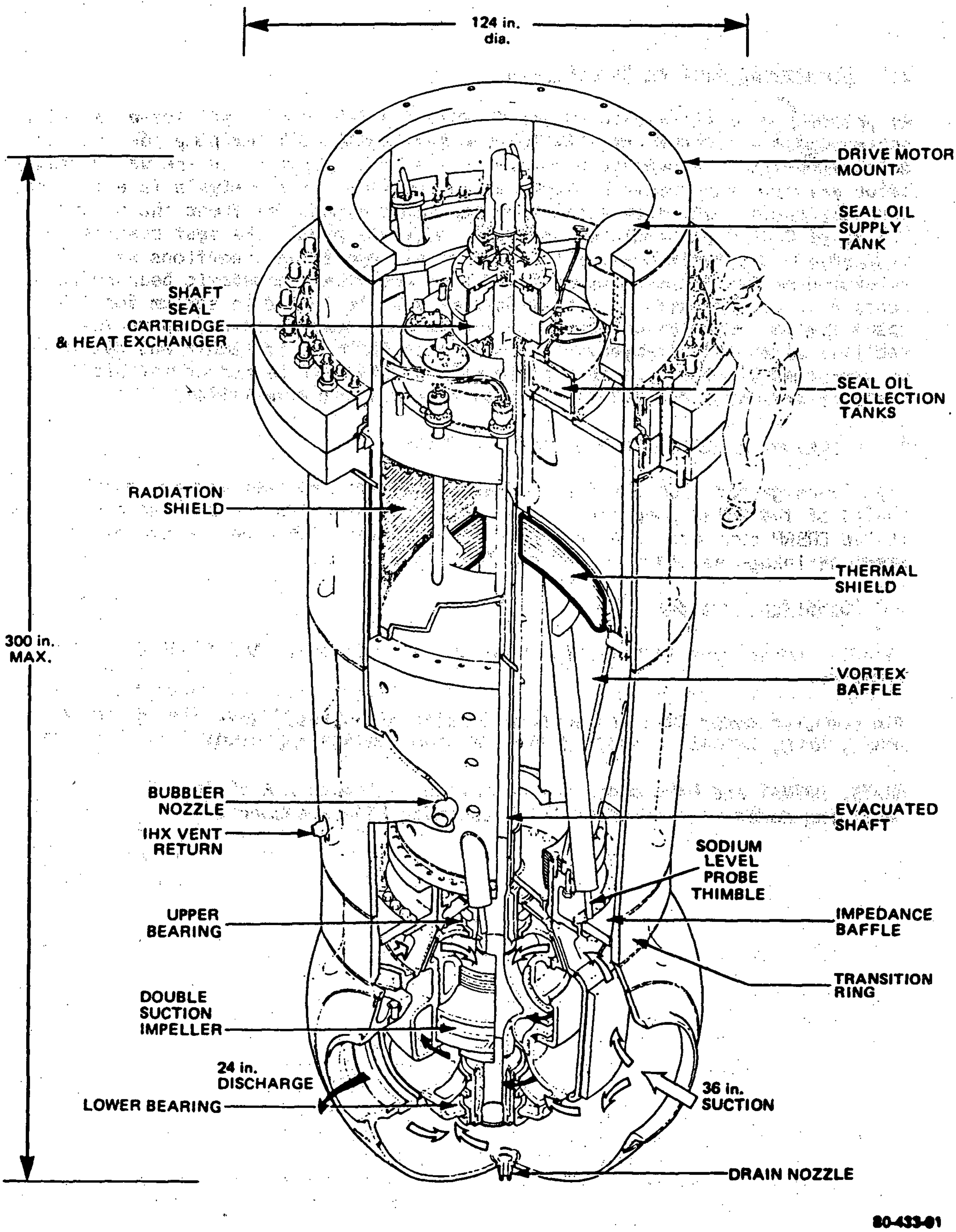

FIGURE Q110.78-PU-1 PRIMARY PUMP ISOMETRIC

Response for IHX:

The methods used in the static and dynamic analysis of the Intermediate Heat Exchanger (IHX) to determine structural and functional integrity are summarized in this response.

The IHX provides the thermal link between the primary and intermediate heat transport piping. The IHX is a straight tube flexible downcomer design using an essentially counterflow arrangement of heated and cooled sodium. Figure 110.78-IHX-1 depicts the salient features of the IHX design. The main support for the IHX is the hanging support cylinder which is fabricated from type 304 stainless steel at the top and type 316 stainless steel at the bottom. It consists of a cylindrical shell that is welded to the IHX shell and tube bundle through a "Z" shape junction forging at the lower edge and has an upper flange which is anchored to the operating floor. The shell is fabricated from type 304 stainless steel in the bottom portions and type 316 stainless steel in the top areas where it is welded directly to the lower edge of the cylindrical hanging support through the "Z" junction. The bottom portion of the shell assembly consists of a lateral support ring with spacer guides to restrain the tube bundle, the lower tubesheet, the hemispherical head, and the primary outlet nozzle. The tube bundle is comprised of two major sub-assemblies: (1) the bundle, consisting of tubesheets, tubes, support plates, tierods and spacers, outer shroud, hemi-head, downcomer, strongback and by-pass seal, and (2) the channel assembly consisting of replaceable bellows, upper head, intermediate outlet nozzle, intermediate vent, inner and outer channel cylinders, upper downcomer pipes and the "Z" forging. The uppermost portion of the channel contains a removable IHX bellows assembly. The bellows permits the differential axial thermal growth between the downcomer and straight tubes and also serves as a portion of the pressure boundary between the primary and intermediate systems. A detailed description of the IHX is included in Chapter 5 of the PSAR.

The IHX was designed and constructed to the Class 1 requirements of the ASME Code, Section III and the supplementary requirements of RDT E15-2NB-T. Design and construction of parts and components for design temperatures exceeding 800°F were in accordance with Code Case 1592 and the supplemental requirements of RDT F9-4T.

LOADINGS CONSIDERED

Two major types of general loading were considered, mechanical and thermal. Mechanical loads consist of internal pressure, nozzle loads (due to piping weight, thermal expansion and seismic effects), dead weight of the component and its content, seismic dynamic loads in the component itself, vibratory dynamic loads from various sources, and rapid high pressure loads as impact dynamic load effects. Thermal loads consist of many thermal transients in the sodium of varying degrees of severity, duration, and temperature change direction, as well as steady state high temperature effects. Some parts of the unit operate at temperature below the creep regime but many parts operate in the creep regime. These loads apply to all pressure boundaries and internal parts.

FAILURE MODES

Analyses were performed on the IHX to reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all the modes of failure listed below:

1. Ductile rupture from short-term loadings

2. Creep-rupture from long-term loadings

3. Creep-fatigue failure

4. Gross distortion due to incremental collapse and ratchetting

5. Loss of function due to excessive deformation

6. Buckling due to short-term loadings

7. Creep buckling due to long-term loadings

Critical failure modes for specific areas of the IHX are identified in the discussion of analysis methods.

ANALYSIS METHODS

The following paragraphs provide a brief summary of the considerations involved in the identification of highly loaded areas and the analysis methods applied to show compliance with applicable criteria.

Stress Analysis

The intermediate channel, upper tubesheets, lower portion of the hanger, lower tubesheet complex and the upper portion of the primary shell were included in a single finite element thermal and stress model. The thermal transient analysis was conducted on two separate sub-models composed of the lower tubesheet complex and the intermediate channel complex. The stress analysis was conducted on the combined model through the use of substructuring methods. The stress analysis methods used for the different areas of the IHX are described below:

Lower Tubesheet Complex: For this region, the governing failure mode is fatigue. Creep strain and stress rupture damage are of minor concern as this area operates above 800°F for less than 100 hours during its 30 year design life. While the rules of RDT F9-4T, which permit exemption from satisfying strain limits, can be used, the added restriction of the elastic shakedown limits of ASME Code, Section III presented difficulties for a few of the upset and emergency transients where the $3S_m$ limits were exceeded. In such cases, inelastic analysis based on the approach given in Paragraph NB-3228 of Section III was performed to show that the amount of accumulated strain was within acceptable bounds. The remaining transient cycles were treated using elastic rules.

Intermediate Channel Complex: This area is composed of the intermediate channel, upper tubesheet, lower portion of the hanger and upper portion of the primary shell. Creep-fatigue was the major failure mode with strain accumulation only a secondary concern. The creep damage was evaluated with material properties being modified to account for decarburization which reduces the stress rupture strength of the material. Gross deformations due to creep were not of particular concern as the normal operating temperature is 975°F. A creep buckling analysis of the inner cylinder was performed to confirm that it had adequate thickness.

Several lumped thermal transients were used for creep-fatigue evaluations, including worst up, worst down, moderate up, and moderate down. The moderate thermal transients were used to envelope the large-number-of-occurrence transients, while the most severe transients were used to envelope all transients which are worse than moderate transients. This general transient lumping procedure and usage was employed for transient analysis in the other parts of the IHX also. The total number of lumped transients used is the same as the total number of actual transients.

In addition to elastic analysis, extensive simplified inelastic analysis was employed on the upper tubesheet and outer "Z" and inner "Wye" junctions of the intermediate channel complex to optimize the design prior to performing a detailed inelastic analysis to confirm the design. Detailed inelastic analysis was performed using two dimensional axisymmetric models of these three areas. This general technique of proceeding from an elastic to a simplified inelastic to a detailed inelastic analysis was employed in other areas where necessary.

The detailed inelastic analysis of the upper tubesheet involves the evaluation of several cycles composed of worst up and down thermal shocks and moderate up and down thermal shocks. An equivalent set of material properties was developed and both plastic and creep correlations were made for the solid region used in the model in place of the actual perforated region. The model used contains the tubesheet and the attached cylinders.

For the inelastic analysis of the outer "Z" junction, many cycles composed of severe and moderate up-down thermal shocks were evaluated. Also, several cycles composed of severe and moderate up-down thermal shocks were used in the inelastic analysis of the inner "Wye" junction.

Hanger: The lower portion of the hanger, which operates above 800°F during normal operation, is considered an integral attachment to the ASME Code Class 1 IHX pressure boundary and hence subject to the same design rules as the IHX. The upper portion of the hanger including the anchor bolts was designed to the rules of Subsection NF of the ASME Code, Section III. A jurisdictional boundary between the two areas was established, based upon the temperature level. The major consideration involved the adequacy to withstand primary stresses, the important contributors to these stresses being seismic loading and deadweight. The affect of the axial temperature gradient from the IHX shell to the support flange was taken into account.

Primary Shell: The primary shell is divided into three regions. The upper region involves the primary shell forging and is considered with the intermediate channel complex. The middle region involves the high temperature primary inlet area. Here the critical failure mode was creep-fatigue. The creep damage was evaluated with material properties being modified to account for decarburization. Strain limits were not a problem as large thermal discontinuity stresses were not present. This region was evaluated with simplified inelastic analysis using an infinitely long thick walled cylinder model. The lower region consists of the seal ring, lateral support ring, head and primary outlet nozzle. Here, the critical failure mode was fatigue and the evaluation was handled in the same manner as the lower tubesheet complex.

Additional considerations in the analysis of the primary shell included worst-case weld configurations and their effect on creep-fatigue evaluations. This involved considering the increase in secondary stresses due to maximum mismatch between sections that were joined. In addition, the peak stresses due to the local discontinuity in a weld were considered. Another consideration was dry and wet heat up and cooldown of the primary shell in the area of the primary closure. These cycles control the design of the primary closure seal, and hence reasonable heating rates were established and the effect of local loss of heaters was considered. Also, evaluation of the non-axisymmetric temperature distributions due to maldistribution of flow, such as at the primary vent elevation, was done.

Nozzles: The primary inlet nozzle, the intermediate outlet nozzle, the vent nozzles and the hand hole nozzle including its cap were evaluated in detail. Three vent nozzles were evaluated. The primary shell vent has sodium flowing through it during operation while the other vent nozzles do not. The critical failure modes for the primary inlet, intermediate outlet and hand hole nozzles were creep-fatigue and strain accumulation. The creep damage was evaluated with material properties being modified to account for decarburization. Several lumped thermal transients were analyzed using axisymmetric model approximations. The nozzle loads were evaluated using axisymmetric approximations as subjected to non-axisymmetric loads.

In addition to elastic analysis, simplified inelastic analysis was used to optimize the nozzle configuration so that a minimum of detailed inelastic analysis was required to confirm the design. The detailed inelastic analsis involved the evaluation of several lumped transients composed of severe and moderate up-down thermal shocks.

Internals: The support plates, shroud, tie rods, nuts, spacers, primary by-pass seal complex, primary inlet plenum baffle, top baffle plate and strong-back are included in this category. The mandatory rules of Code Case 1592 were applied even though these components are not part of the pressure boundary. These rules cover the limits on primary stress. Dynamic loadings due to fluid-borne pressure transients and flow-induced vibrations were considered.

Tube and Tube-to-Tubesheet Weld: The critical failure mode was creep-fatigue for both the tube and the tube-to-tubesheet weld. In addition, all loadings on the tube were considered to evaluate the potential for column buckling. As in the case of the intermediate channel complex and the lower tubesheet complex, several thermal transients were evaluated. The tube was modeled using an infinitely long cylinder and simplified inelastic methods were then applied. An elastic analysis of the tube-to-tubesheet junction was performed using an axisymmetric model. As both the tube and tube-to-tubesheet junction are subject to decarburization, creep damage was evaluated with modified material properties. However, excessive strain accumulation was not of concern.

Additional considerations include the buckling of the straight tubes due to radial variation in bulk tube temperature during dry heatup and cooldown of the tube bundle. This consideration includes the stress-strain characteristics of the tubes due to loss of carbon and nitrogen and contributes to fatigue damage in the tubes and imposes deflection requirements on the expansion joint. Also, flow maldistribution is of importance in the area of the tube and lower tubesheet to downcomer junction. A nonaxisymmetric bulk temperature distribution in the tubes caused significant fatigue damage in the junction between the downcomer and tubesheet. In addition, compressive axial thrusts on the tubes were accounted for in the tube buckling potential assessment.

Expansion Joint Complex: This area is composed of the expansion joint (bellows), attachments, intermediate inlet nozzle, hand hole and outer cylinder. The critical failure mode is fatigue for all of these components. The intermediate inlet nozzle, hand hole and outer cylinder operate below 800°F for a considerable fraction of the design life. Thus, the rules of RDT F9-4T providing an exemption from the strain limits were used. Several lumped thermal transients were evaluated. For the expansion joint and attachment region, various loading conditions for detailed inelastic analysis were developed. Even though the expansion joint operates below the creep range for most of its design life, the inability to demonstrate elastic shakedown necessitated the use of inelastic analysis. Many cycles of various axial deflections were analyzed. The plastic strain ranges obtained from the inelastic analysis were used to establish the fatigue life of the expansion joint.

SEISMIC ANALYSIS

Seismic analysis of the IHX was performed by the response spectrum method, using the ANSYS dynamic-seismic capability. The hanging support flange was modeled using continuum type finite elements. This accounted for some additional flexibility of this region. Both SSE and OBE vertical and horizontal cases were analyzed. Non-linear effects were factored into the analysis where applicable. The results of the seismic analysis provided input to the structural evaluations described in the previous section of this response.

VIBRATION AND DYNAMIC ANALYSIS

Other dynamic and vibration considerations which were evaluated involve sodium-water reaction, check valve slam, fluid and structural borne vibrations and flow induced vibrations.

The expansion joint was analyzed under the action of the pressure transient entering the intermediate inlet nozzle due to a sodium water reaction. The effect of this pressure transient was also considered for the tubes, as they have low flexibility. The approach was to use a static analysis with an amplification factor, which was determined by comparing the natural frequencies of the bellows and tubes in the appropriate modes to the time rate of change of the pressure.

The transient pressure resulting from check valve slam was analyzed for its effect on the shroud, tubes and expansion joint using an analysis procedure similar to that for the sodium-water reaction.

Calculations for fluid and structural borne vibrations were made as well as assessments for flow induced vibrations. It was found that fluid borne and structural borne vibrations (e.g. from the pump through the cross over piping to the IHX primary inlet nozzle) were insignificant. The effect of flow induced vibration was verified by tests as discussed later.

MATERIAL PROPERTIES

The elastic material properties used in the structural evaluation of the IHX are specified in the ASME Code documents. The Nuclear Systems Materials (NSM) Handbook (TID-26666) was used as the authoritative source for material properties not specified in the applicable Code Documents. All material properties used in the design and analyses of the IHX are specified in the Code Documents or the NSM Handbook.

Thermal and mechanical properties were considered in the selection of materials for use in the IHX. Further, consideration was given to material properties in connection with fabrication procedures as noted below in the section on thermal aging effects. A thickness allowance was provided, in the manner described in the ASME Code, Section III, Subsection NB-3120, to account for the effects of corrosion and erosion.

Where material properties are significantly uncertain, minimum, average, or maximum properties were used as appropriate to obtain a reasonably conservative result. For example, in certain critical situations the evaluations of deformation limits were based upon minimum stress-strain curves. The selection of appropriate properties was guided by RDT F9-5T.

Material Degradation

Most of the data used to define the allowable design stresses in the ASME Code were obtained from tests conducted in air. No attempt is made in the Code to account for the effects of other service environment. The LMFBR development program has focused attention on the mechanical behavior of reactor materials when exposed to high-temperature liquid sodium, in addition to fast neutron irradiation and to long time aging at elevated temperatures. The effects of the service environment upon the response and failure characteristics of the structural materials are summarized in the following paragraphs.

Thermal Aging Effects on Mechanical Properties

Types 304 and 316 stainless steels are non-age hardenable alloys. Thus, no significant changes in strength or hardness of annealed material accrue from long term aging at temperatures up to 1200°F, unlike the precipitation-hardened stainless steels. Some slight increases in strength and decreases in ductility may occur due to carbide formation, together with a reduction in the room temperature impact strength. Of more significance is the fact these alloys will sensitize during long term service in the temperature range from 800° to 1500°F. In this phenomenon, carbide precipitation occurs at the grain boundaries, the adjacent matrix becomes depleted in chromium and the grain boundary regions become susceptible to attack by corrosive media. Such attack is not likely to occur in sodium, which, if pure, is a relatively inert environment. However, cracking may initiate during fabrication and the other pre-operation periods when the component is not exposed to sodium, due to the environmental conditions (presence of water and halides). Because of this, precautions must be taken during such periods to ensure that contact between sensitized material and potentially corrosive media is minimized, if not entirely avoided. Hence no allowances have been made for the effects of thermal aging on the properties of types 304 and 316 stainless steels used in the IHX. This did, however, demand that control be specified and exercised during the fabrication process to prevent stress corrosion and intergranular attack.

Neutron Irradiation Effects on Mechanical Properties

Neutron shielding is provided between the reactor cavity and the HTS cells containing the IHX. Neutron fluences in the vicinity of the IHX are therefore negligible and no fluence effects on mechanical properties of IHX materials are expected.

Effects of (Nitrogen + 2% Oxygen) Atmosphere on Mechanical Properties

The selection of nitrogen gas as the atmosphere for the reactor cavity and HTS cells was based on the desire to prevent chemical reactions should molten sodium leak into the cavity and HTS cell from any source. However, the exposure of austenitic stainless steel to pure nitrogen for extended periods of time at elevated temperatures may lead to the formation of a thin nitrided layer. This is considered undesirable because of the brittleness of such layers. To minimize the formation of such a layer, a small percentage of oxygen (<2%) will be introduced into the nitrogen.

Effects of (Argon-Plus-Sodium-Vapor) Atmosphere on Mechanical Properties

Very limited information is available on the effects of exposure to an argon-plus-sodium-vapor atmosphere on the mechanical properties of a material. It is possible that, if the sodium vapor is continually condensing on the material surface and rejoining the main reactor coolant, there could be some interstitial transfer. However, because of the scarcity of data, it is not possible to provide quantitative assessments of such effects at this time. Practically, the potential for significant mass transfer via condensation is insignificant. It was judged that exposure to the cover gas should be considered the same as exposure to liquid sodium without loss of interstitials. Loss of interstitials due to liquid sodium exposure in other circumstances is discussed below.

Surface Effects of Liquid Sodium on Mechanical Properties

Compared with air testing, liquid sodium may cause certain metallic elements to be transferred from the hotter to the cooler regions of LMFBR systems. In addition, surface oxidation in liquid sodium is greatly reduced when compared to air testing. It is believed that these surface effects are insignificant in their influence on short-term tensile properties.

For time-dependent deformation, such as stress-rupture and fatigue, the effects of a liquid sodium environment are complex and need to be considered in detail. In the case of stress-rupture, it has been shown that for a given temperature and stress, rupture times in air are longer than those in liquid sodium. A sodium-environment correction factor was applied to the rupture strength data specified in ASME Code Case 1592 for type 304 and 316 austenitic stainless steel. This effect was used in all evaluations where stress rupture was involved.

Fatique properties of materials can be greatly affected by the environment in which the properties are measured. The avoidance of excessive surface oxidation by testing in sodium (or inert gas) instead of in air increases the cycles-to-failure for a given strain range. No increase in the design fatique limits due to exclusion of oxygen effects was employed in the analyses as a conservative approach.

Interstitial Transfer Effects on Material Properties

In the Heat Transport System, interstitial carbon and nitrogen are tranferred from the hotter to the cooler regions. This leads to weakening in the decarburized and denitrided regions and to strengthening in the carburized and nitrided areas. In the case of fatigue behavior, however, the effects of interstitial absorption at the surface are complicated because of two concurrent mechanisms. On the one hand carburization can lead to enhanced crack nucleation at carbide particles and, on the other, surface strengthening during strain-controlled fatigue will increase the proportion of elastic straining which is less damaging than plastic deformation. In general, the austenitic materials will be carburized and the ferritic materials will lose interstitials. However, the crossover from carburization to decarburization is system dependent and it is likely that in certain systems at least some of the austenitic material will be decarburized. Procedures have been established for the CRBRP by which the extent of interstitial transfer for types 304 and 316 stainless steel can be determined and from this the effect on mechanical behavior was calculated. The procedures include calculations of surface and average interstitial concentrations and interstitial gradients under decarburizing and denitriding conditions. Because of the shortage of data on nitrogen diffusion, the rates of nitrogen transfer were estimated from available carbon transfer data. Thus, the effects of interstitial transfer on the mechanical behavior of structural materials used were taken into account in the analysis or shown to be insignificant in effect at the region involved mainly due to thickness considerations. For example, the effects of decarburization of the thin walled tubes was considered because of the thin section of metal involved.

STRUCTURAL VERIFICATION TESTS

IHX Expansion Bellows Development Program

The straight tube design of the IHX required that a flexible joint be provided in the intermediate inlet region to accommodate differential thermal expansion. The bellows is thermally isolated from the primary sodium by virtue of its location and because the stagnant primary sodium on its exterior is cooled by the intermediate sodium in the downcomer. This keeps the operating temperature below the creep range, at about 635°F.

A development test program was conducted to verify the structural calculations, design parameters and fatigue life of the IHX expansion bellows. The testing consisted of three parts: a squirm test, a strain gage test and a fatique test. The squirm test was performed in accordance with NC-3649.4 of the ASME Code, Section III and the fatigue test was performed in accordance with Appendix II of Section III. The fatigue test consisted of cycling the bellows through a prototypical plant histogram in a 100 PSI nitrogen atmosphere at 635°F. The tests confirmed the adequacy of the bellows for the IHX service requirements.

IHX Model Flow Induced Vibration Test Program

A model flow test of the IHX tube bundle was conducted to determine the tube vibration characteristics. The objectives were:

1. To determine the amplitude and frequency of flow induced tube vibrations at various elevations of the IHX tubes.

2. To ascertain that the maximum amplitude of tube vibration does not exceed 25% of the nominal distance between the outer surfaces of adjacent tubes.

3. To ascertain that peak tube deflection stress levels do not exceed the allowable tube material endurance limits.

4. To ascertain that unsupported tube span natural frequencies are at least 50% higher than the calculated vortex shedding frequencies.

A full scale replica of a 30° sector of the tube bundle was used to establish geometric similarity. The test results verified analytical predictions and confirmed that the tube bundle would not experience flow induced vibration problems in service.

COMPUTER PROGRAMS

Responses to NRC Questions 110.27 and 110.58 provided information relating to the computer programs used for the static and dynamic analyses of seismic Category I structures. Of those, the following computer programs were used for the analyses of the IHX:

1. ANSYS - For thermal, stress and seismic analyses in all areas of the IHX.

2. CHERN - For simplified inelastic analysis to optimize design of those areas where elastic analysis was not adequate and detailed inelastic analysis was eventually necessary, and for use in those areas where the program was applicable and adequate by itself (mainly tubular configurations).

3. MARC - For detailed inelastic analysis of those areas where required (primary inlet nozzle, Wye junction, "Z" junction, upper tubesheet, etc.).

A description of each of these programs is included Appendix A of the PSAR and hence is not repeated in this response. Appendix A also provides information relating to the adequacy of these codes and verifications that have been completed. Both ANSYS and MARC are extensively used throughout the nuclear industry. The CHERN program has been verified for use on the FFTF-IHX with a high level of confidence.

INFORMATION USED FROM OTHER PROGRAMS

The design of the IHX for CRBRP used information developed in the design of the IHX for the FFTF. Relative to structural assessments, the stress analysis of the FFTF IHX components required the development of analytical techniques and computer programs (e.g., the CHERN program for simplified inelastic analysis was developed for use in the FFTF IHX structural analysis). Analytical techniques for evaluating the effects of thermal transients by lumping and for appropriate application of simplified and detailed inelastic analysis were forthcoming. The use of complex thermal and structural finite element models to represent the physical situations in complicated geometries was also developed. The information and expertise gained from the FFTF IHX design and analysis was used and expanded for application to the CRBRP IHX design and analysis. Since the type of service required for both of these IHX components was very similar, the carry-over and use of techniques established for FFTF was a natural consequence for CRBRP.

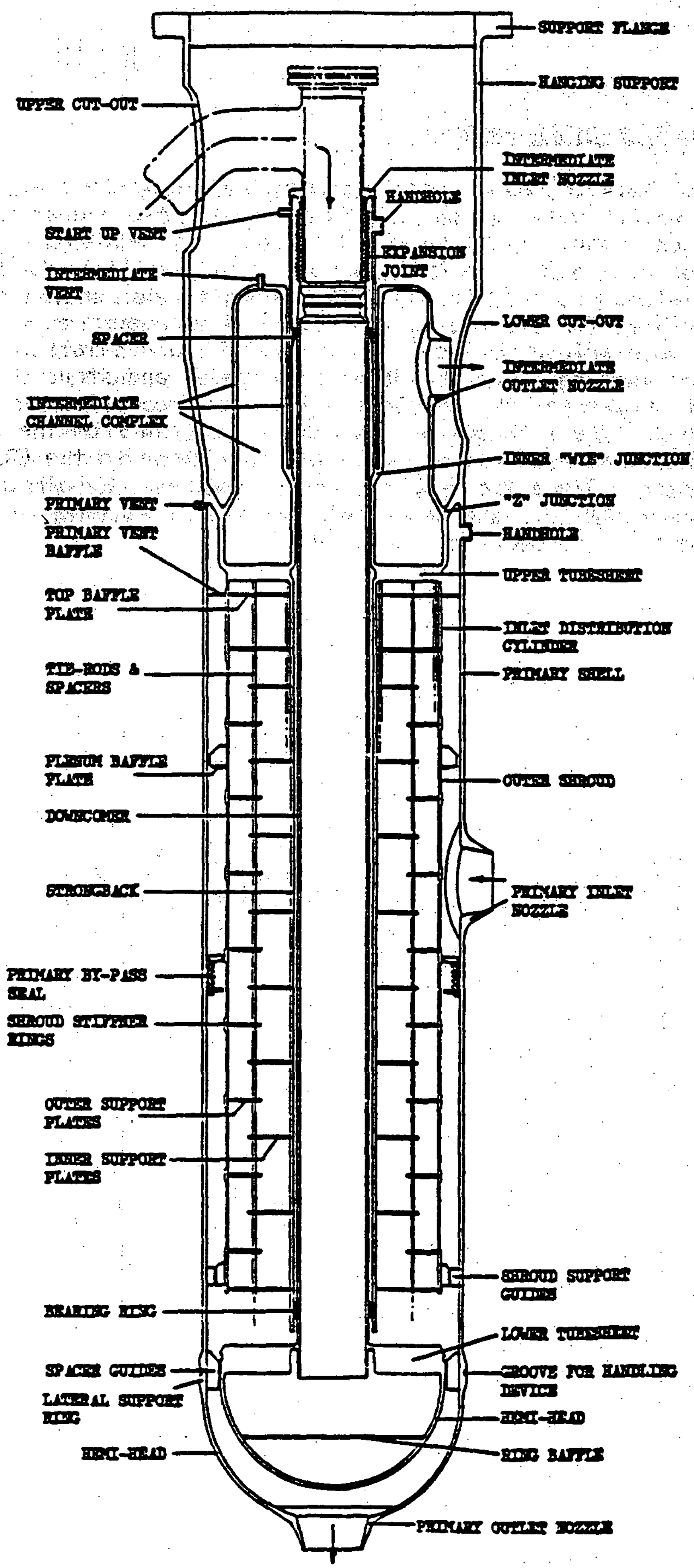

Figure Q110.78-IHX-1 General Arrangement of the IHX

Response:

The methods used in the static and dynamic analyses of the Steam Generator (SG) Modules to determine structural and functional integrity are summarized in this response.

The SG system in each of the three IHTS loops includes two evaporator modules (EV) connected in parallel and one superheater module (SH). The SG modules transfer the reactor generated heat from the intermediate sodium to the water/steam. The main components of a SG module (the EV and SH are identical in almost all respects) are shown in Figure Q110.78-SG-1. The SG module comprises a tube bundle assembly and water/steam heads. The tube bundle assembly consists of the main shell assembly, tube-tubesheet assembly, nozzles, SG internals, and SG externals. The main shell assembly provides containment for the sodium, support for the SG internals and mounting points for the tubesheets, nozzles, and SG externals. The components support used for mounting the SG module is an integral part of the main shell. The tube-tubesheet assembly provides the primary sodium-water barrier and consists of the steam tubes and tubesheets. One sodium inlet and two sodium outlet nozzles are welded to the main shell assembly. A vent nozzle is provided in the upper stagnant region for venting the module during initial sodium filling and for detecting any tube leak. One sodium drain nozzle is attached to the lower part of the shell. The SG internals consist mainly of the shroud, thermal liner, tube-spacer plates, vibration damper and tube suport bars. The SG external attachments are used as lifting devices and insulation supports. A lower restraint fixed to the SG building at the lower tubesheet level is used to provide lateral stability for the SG module. The steam/water heads provide transition betweeen the SG tubes and the steam/water lines of the SG System. A bolted manway in each head provides access for internal inspection of tubes and tube plugging. More detailed description of the SG modules is included in Section 5.5 of the PSAR.

The SG modules are designed and analyzed in accordance with the rules of Class 1 Nuclear Components in ASME Section III (1974 Edition with Addenda through Winter 1974) as supplemented by RDT Standard E15-2NB-T (1974 Edition with Amendments 1 and 2) and in ASME Code Case 1592-4 (with changes in Supplements 8 and 9) as supplemented by RDT Standard F9-4T (1976 Edition). The SG analysis also conforms to the guidelines and procedures in RDT Standard F9-5T(1981 Edition). The aforementioned codes and standards are directly applicable to the SG pressure boundaries. The SG internals are not under the jurisdiction of the ASME Code, they are however analyzed using the guidelines of ASME Section III and Code Case 1592. The SG component support and lower restraint are analyzed as Class 1 component supports in accordance with Subsection NF. The parts of the components supports that experience elevated temperature conform to the design criteria of Code Case 1592.

FAILURE MODES

Analyses are performed on the SG modules to reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all the modes of failure listed below:

1) Ductile rupture from short-term loadings,

2) Creep-rupture from long-term loadings,

3) Creep-fatigue failure,

4) Gross distortion due to incremental collapse and ratcheting,

5) Loss of function due to excessive deformation,

6) Buckling due to short-term loadings,

7) Creep buckling due to long-term loadings, and

8) Non-ductile failure.

Critical failure modes for specific areas of the SG modules are identified in the discussion of analyses. The nonductile failure mode is of particular importance in the SG analysis because the SG pressure boundaries and most of its other components are made of 2 1/4 Cr-1Mo low-alloy ferritic steel. The capability of these ferritic components to withstand nonductile failure is demonstrated by applying the linear elastic fracture mechanics procedures in Appendix G of ASME Section III, taking into account the stresses due to applied loads as well as any residual stresses.

LOADINGS

The principal loadings that are considered in the analysis of the SG components are: a) steam/water and sodium pressures, b) deadweight and seismic loads (OBE and SSE) of the SG module and including the fluid contents and thermal insulation, c) deadweight, thermal expansion and external mechanical loads at the nozzles, d) fluid thrust loads, e) fluid induced vibration loads and other dynamic loads, f) loads generated within the SG due to temperature difference between various parts, g) loads generated by thermal transients, h) sodium-water reaction loads (DBL and DBR) in the event of steam/water tube rupture, i) loads resulting from rupture of steam/water or sodium piping (PB), j) thermal fatigue due to mixing of interfacing steams of fluids at different temperatures (striping), and k) thermal fatigue in the evaporator tubes due to departure from nucleate boiling (DNB).

Since the EV and SH are required to be interchangeable, the SG modules are analyzed so that they can withstand the loadings of both the EV and SH.

ANALYSES

To facilitate the structural analysis, the SG module is divided into design zones in accordance with Paragraph 3112(d) of Code Case 1592. For each zone, the boundaries are defined and the pressures, temperatures, and mechanical and thermal loads are specified. Table Q110.78-SG-1 is a checklist of the analyses performed for the SG various components or design zones. The table shows for each zone the type and method of analysis, loading categories, fabrication and environmental effects, and design rules and limits applied. In the analyses of these zones, the loads are generally drawn from four main sources; a) results of structural assembly analyses, b) results of thermal-hydraulic assembly analyses, c) interface loads with other CRBRP systems, and d) structural tests.

Structural Assembly Analyses

The purpose of these analyses is to simulate the interactions between the components and to provide the loads required for subsequent analyses of the individual components. The main structural assembly analyses performed are the seismic and misalignment analyses. The SG is classified as Seismic Category I component that must withstand five Operating Basis Earthquakes (OBE) and one Safe Shutdown Earthquake (SSE) during the lifetime of the plant. Detailed finite element response spectrum and time history analyses are performed using ANSYS finite element code. The purpose of the response spectrum analysis is to provide conservative seismic loads for all the SG components. The time history analysis is to provide more precisely the seismic loads for the steam tubes. Analyses are performed for OBE and SSE loadings with 2 percent and 3 percent of the critical damping, respectively.

The steam tube bundle could be subjected to large internal mechanical loads if it is misaligned with the numerous supports along its length. A finite element analysis is performed using the WECAN Computer Code to determine these internal loads with the worst possible misalignment permitted by the tolerances. Spring-gap elements are used to represent the interaction between the tube-spacer plate, spacer plate-shroud and shroud-shell. The same finite element model is used to calculate the internal loads resulting from the deadweight and differential thermal expansion between the tube bundle and the main shell.

Thermal-Hydraulic Assembly Analyses

The fluid conditions in the SG modules during steady state and thermal transients must be known to conduct stress evaluations for the SG components. The water/steam and sodium pressures are part of the mechanical loads and the fluid flow rate and temperature determine the boundary conditions for the thermal analyses of the individual components.

Fluid conditions in the EV and SH are calculated by using the SETS, TRUMP and TRANSQ Computer Codes. Flow distributions within the modules are determined by using TEMPEST Computer Code or scale model testing. The postulated event of an equivalent double-ended guillotine break (EDEG) of a single tube followed by two additional sequential EDEG tube failures is a Design Basis Leak (DBL) faulted condition in the affected module and is a Design Basis Reaction (DBR) emergency condition in the other two modules of the same IHT System. These conditions are analyzed with the TRANSWRAP II Computer Code. The analytical model for this Code includes detailed representation of the two evaporators, super-heater and other components of the IHT System, such as Intermediate Heat Exchanger, Sodium pump, and interconnecting piping. The major input to TRANSWRAP II Code is the transient water/stream flow rate from the failed tubes. The program output provides the pressure and velocity of the sodium throughout the faulted and emergency modules. (See PSAR Section 5.5.3.6.2 for details on TRANSWRAP)

<u>Design Zone Analysis</u>

A brief summary of the major analyses follows.

<u>Steam Tubes</u>: The tubes are subjected to severe and numerous types of loading and at the same time are exposed to a detrimental service environment. Common pressure vessel methods are used to evaluate primary stresses due to pressure. Primary stress evaluation also includes seismic loads, deadweight, and flow induced vibration loads with the emphasis on the critical areas at the tube-sheet welds and the spacer plates. Secondary stress evaluation considers the misalignment loads, temperature difference between tubes and shell, and temperature gradient through the tube wall. The DNB fatigue evaluation is based on testing to determine the extent of temperature fluctuations and on combined thermal-stress analyses to assess the largest thermal stress fluctuation. Only elastic stress analyses are required for these loadings except for the temperature gradient through the tube wall during severe thermal transients where simplified inelastic analysis is used.

<u>Main Shell</u>: The primary stresses in the main shell are evaluated with the shell subjected to the run and nozzle loads. These loadings include sodium pressure for steady state and thermal transient events, seismic loads for OBE and SSE and the loads generated during DBL, DBR and PB. The nozzle loadings include the thermal expansion loads. Conservative loading combinations are adopted to bound all loading categories. The stresses due to the run loads are determined by using common pressure vessel methods for the cylindrical portions and by using the stress indices method given in NB-3685 for the elbow portion. The stresses due to to the nozzle loads are calculated by using Bijlaard method. The structural evaluation also addresses the shell buckling and potential thermal striping in the elbow and lower portion of the shell.

Tubesheet and Steamhead Assembly: This design zone includes the tubesheet, steamhead, manway hole and part of the main shell in the vicinity of the tubesheet. The relatively high steam temperature at the SH upper tubesheet and the high steam/water pressure, combined with severe sodium transients at the EV and SH lower tubesheets, indicate potential creep-fatigue damage that requires both upper and lower assemblies to be carefully analyzed. The stress evaluation is conducted through a number of finite element analyses in which WECAN Computer Code is used. These analyses include asisymmetric thermal, elastic, and inelastic analyses and three-dimensional 180 degree elastic analysis. The three-dimensional analysis is used to determine accurately the local stresses arising from the interactions between the tubesheet, steamhead, manway hole, and water/steam nozzle. The perforated region of the tubesheet in these analyses is represented by an equivalent homogeneous material. A variable film coefficient is introduced in the radial direction of the equivalent material to represent the heat convection by the water/steam in the tubesheet holes. The elastic properties of the equivalent material are determined in terms of the ligament efficiency in accordance with the procedures of A-8000 in ASME Section III. The inelastic properties of the equivalent material are generated by a series of inelastic analyses for the smallest repetitive part of the perforation. The thermal and elastic analyses are performed for the steady state and selected umbrella transients. Based on the results of these analyses a leading histogram for a subsequent inelastic analysis is conservatively constructed.

Nozzles: The SG model has eight different nozzles; sodium inlet, sodium outlet, steam inlet, steam outlet, steamhead manway, inspection ports, sodium vent, and sodium dump. For each nozzle the load controlled stresses are first calculated using common pressure vessel and simple beam formulas. The loadings consist of fluid pressure and interface mechanical and thermal loads. The reinforcement of the openings are also checked based on the procedures in Subsection NB and Code Case 1592. Because of the high sodium temperature during full power operating state, in addition to the severe upset and emergency thermal transients at the SH nozzles, the creep-fatigue failure is a plausible failure mode. Inelastic analyses are performed for these nozzles to demonstrate that the creep-fatigue damage is suitably limited by meeting the criteria in Appendix T of Code Case 1592. The temperature distributions in the nozzle wall and surrounding shell are first determined by using the WECAN Computer Code. An axisymmetric model is constructed with the nozzle attached to a sphere that has a radius two times that of the cylindrical shell. For the case of the sodium inlet nozzle, the thermal liner and the sodium flow through the gap are also simulated in the thermal analysis. The thermal and subsequent elastic stress analyses are performed for a number of selected transients that show large and fast sodium temperature change. Based on the results of these analyses, the critical zones are identified and inelastic analyses are performed for each critical zone separately. The loading histogram for these inelastic analyses is conservatively chosen based on the results of the full nozzle elastic analysis and the results of a one dimensional inelastic analysis for the most critical cross section.

MATERIAL PROPERTIES

The mechanical material properties used in the structural evaluation of SG modules are specified in the ASME Code Documents. The Nuclear System Materials Handbook (NSMH), Reference Q110.78-SG-1. is used as the authoritative source for material properties not specified in the applicable Code Documents. Where material properties are required but are not available in the ASME Code or NSMH, the properties will be specified and the sources and procedures for obtaining them explained.

Material Degradation

The fabrication and environmental effects on the properties of the 2 1/4 Cr-1Mo Steel and Alloy 718 used in the SG design are addressed in the response to NRC questions CS250.4 and CS250.9, Reference Q110.78-SG-2. Only the properties of 2 1/4 Cr-1Mo Steel are found to be influenced. The SG service environment affects the 2 1/4 Cr-1Mo Steel in two ways: physically removing the material and degrading the material strength. The removing of the material is of particular importance in the thin steam tubes where allowances are specified to compensate for corrosion, cleaning and wear between the tubes and spacer plates. The material strength degradation by post weld heat treatment (PWHT), thermal aging, and decarburization is accounted for in the design evaluation.

Creep-Fatigue Damage Evaluation

The creep and fatigue damage of the SG components in the elevated temperature regime is suitably limited by meeting the criteria in T-1400 of Code Case 1592-4. The structural integrity of the components made of 2 1/4 Cr-1Mo Steel and subjected to severe thermal loading will generally be demonstrated by simplified and detailed inelastic analyses. The design fatigue curves and creep-fatigue envelope, however, are not provided in the Code Case for 2 1/4 Cr-1Mo Steel. Extensive development programs have been carried out to provide these data. Based on the results of these programs, a conservative procedure for the creep-fatigue evaluation has been developed. The procedure first reduces the experimentally determined fatigue curves in the NSMH to design curves by applying the common two-and-twenty factors. The design curves are then reduced further by applying a factor that depends on the service environment and loading conditions. The stress to rupture values provided in the Code Case are reduced to account for PWHT, thermal aging and decarburization in addition to applying the Safety Factor K' = 0.9 to the applied stresses. The creep damage plus fatigue damage is then limited to the value of 1.

STRUCTURAL VERIFICATION TESTS

The SG design is based on a very extensive testing program that encompassed the material mechanical properties tests and structural and thermal-hydraulic performance tests. A comprehensive description and evaluation of this testing program is presented in Section 5.5.3.1.5.1 of the PSAR.

A list of the tests (completed, in-progress, and planned) is given in the following:

- Modular Steam Generator (MSG) Tests, (1972-1974)
- Hydraulic Test Model (HTM), (1969-1976)
- Sodium to Water Boundary Leak Tests, (1974-present)
- Few Tube Tests (FTT), (1978)
- Departure from Nucleate Boiling (DNB) Tests, (1975-1976)
- Friction and Wear Tests, (1973-1979)
- Single-Tube Performance, Stability and Interaction Tests, (1976-1977)
- Tube to Tubesheet Welds Tests, (1976-1980)
- Mechanical Properties Tests, (1968-1981)
- Scale Hydraulic Model Feature Tests, (1980-present)
- Flow Induced Vibration (FIV) Tests, (to begin in 1983)
- Prototype Steam Generator Tests, (to begin in 1982)
- In-Situ Evaporator Performance Tests

COMPUTER PROGRAMS

The computer programs (ANSYS, SETS, TRANSQ*, TRANSWRAP II, TRUMP, TEMPEST and WECAN) used in thermal-hydraulic and structural analyses of the SG modules are described in Appendix A of the PSAR and hence are not reported in this response. Appendix A also provides information relating to the adequacy of these codes and verifications that have been completed or planned.

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*An amendment to PSAR Appendix A addressing validation of the "TRANSQ" Computer Code will be provided by November 30, 1982.

REFERENCES

Q110.78-SG-1. TID-26666, Nuclear Systems Materials Handbook.

Q110.78-SG-2. CRBRP Material Data Base - CRBRP Engineering Study Report, ES-LPD-82-008, Westinghouse Advanced Reactors Division.

Response for Line Valves

The only large-diameter Seismic Class 1 valves in the PHTS and IHTS are the cold leg check valves in the PHTS. The design limits and rules for these valves are given in PSAR paragraph 5.3.2.3.3 and the analytical methods are given in PSAR paragraph 3.9.1.6.

Question 120.1 (4.2.2)

A list of the materials used for the reactor vessel internals should be provided.

Response:

A list of the materials used for the reactor vessel internals has been provided in revised Section 4.2.2.2.1.

Question 120.2 (4.2.2)

The compatibility of the materials with the coolant should be stated. Also, the applicant should provide assurance that the modules in the core supports and the fuel element aids will not self weld in the low-oxygen sodium.

Response:

The response to this question has been incorporated in revised Section 4.2.2.3.3.2.1 and 4.2.2.3.3.3 and additional Figures 4.2-48A and 4.2-48B.

Question 120.3 (4.2.2)

The welding and seizing of rotating or moving parts of the reactor internals should also be discussed.

Response:

The design considerations for welding and seizing of rotating or moving parts for reactor internals are presented in new Section 4.2.2.5 and new Table 4.2-64. 25

Question 120.4 (4.2.2)

Environmental Effects - Provide references and data on the effects of sodium on mechanical properties of the materials, including the roles of carbon and nitrogen.

Response:

Refer to revised Section 4.2.2.3.3.2, "Environmental Effects on Material Properties". References for this material are provided in Reference Section 4.2. 25

Question 120.5 (4.2.3)

Provide information on the self-welding phenomena associated with couplings and the safety consequences of failure to decouple.

Response:

Primary Control Rod System

The response to this part of the question is contained in revised Section 4.2.

Secondary Control Rod System

The response to this part of the question is contained in revised pages 4.2-168, 204, and 204a. 25

Question 120.6 (4.2.3)

The PSAR indicates that the primary control assembly uses B_4C. Provide assurance that the confirmatory irradiation testing described in the PSAR will actually be performed.

Response:

A discussion of committed B_4C irradiation tests is provided in revised PSAR Section 4.2.3.3.1.5. 15

Question 120.7 (4.2.3)

The PSAR indicates that 17-4 PH materials are used in highly stressed areas such as segmented arms, roller nuts, anti-ejection pawls and leadscrews because of its high strength. Provide the aging temperature that will be used for 17-4 PH material and provide assurance that the material will not embrittle from long term reaging phenomenon.

Response:

The requested information is provided in revised Section 4.2.3.1.7. 25

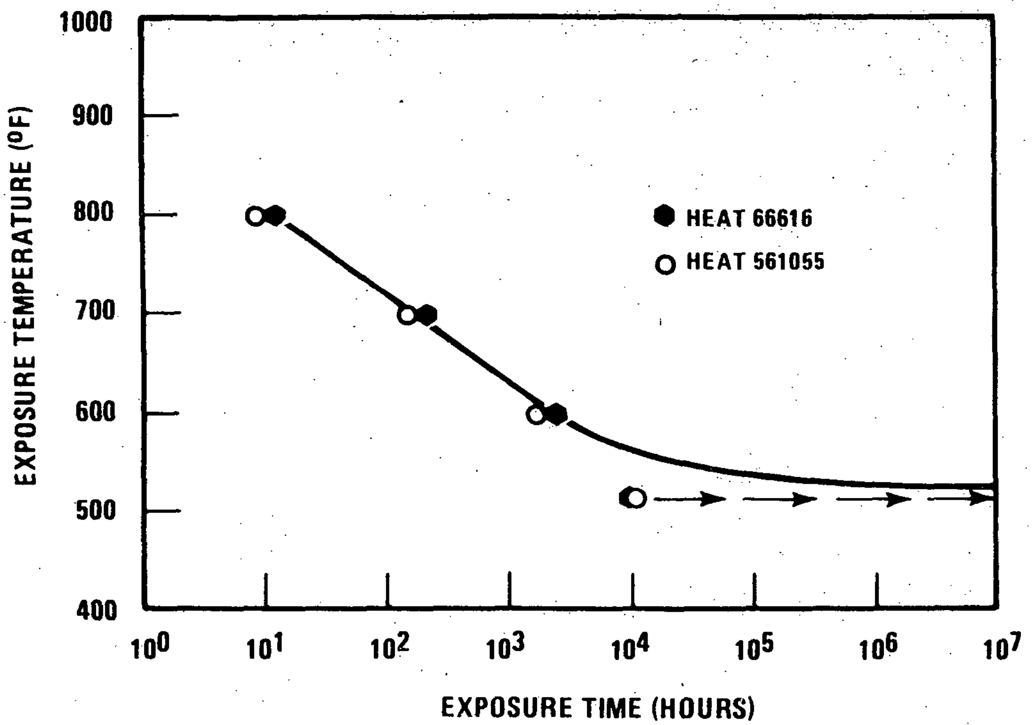

Figure Q 120.7-2 Exposure Time Required to Deplete Room Temperature Toughness by One Half

7683-2

REFERENCES:

Q120.7-1. W.C. Clarke, Jr., "A Study of Embrittlement of a Precipitation Hardening Stainless Steel and Some Related Materials", Trans. AIME, 245, 1969, p. 2135.

Q120.7-2. M.H. Jones, et. al, "Effects of Variation on Normalizing and Tempering Procedure on Stress Rupture Strength, Creep Embrittlement and Notch Sensitivity for a CR - Mo - V and a 17 CR - 4 Ni - Cu Steel", Trans ASM, 47, 1955, p. 926.

Q120.7-3. Armco 17-4 PH Precipitation-Hardening Stainless Steel Sheets and Strip, Product Data Bulletin S-9A, Armco Steel Corporation, Middletown, Ohio, 1966.

Q120.7-4. P.W. Johnson, Jr., "Stainless, Bar and Wire, 17-4 PH-Effect of Exposure at 500/800°F on Mechanical Properties", Armco Steel Co. Report No. 1, June 11, 1969.

Q120.7-5. P.W. Johnson, Jr., "Stainless, Bar and Wire, 17-4 PH-Effect on Expansion at 500/800°F for Mechanical Properties", Armco Steel Co. Report No. 2, June 19, 1970.

Question 120.8 (4.2.3)

In the control assembly analysis, reference is made to the peak absorber temperature of 2870°F for the B_4C hot spot. Provide information on the acceptability of this temperature.

Response:

The response to this question is contained in revised pages 4.2-200 and 200a.

Question 120.9 (4.2.3)

Information is needed to provide assurance that condensation of sodium vapor in the reactor head area will not adversely affect ease of movement or operation of the reactivity control system.

Response:

The requested information is in revised Section 4.2.3.1.6. Additional information has been incorporated in Section 4.2.3.2.1.2. 25

Question 120.10 (5.1.1)

The section on the primary heat transport system describes a thermal liner for by-pass coolant flow to keep the vessel below 900°F. Provide information on the thermal stress analysis of this liner and how thermal fatigue will be avoided where the by-pass and reactor coolant streams remix.

Response:

The information requested is contained in revised Section 5.2.1. 25

Question 120.11 (5.2)

Special welding techniques and processes are specified for welding the reactor vessel and the core structures. There is no reference cited except that they were developed for FFTF. Details of these procedures and processes should be provided.

Response:

Section 5.2.3.2 has been revised to discuss the details of this welding technique. 17

Question 120.12 (5.2)

The reactor vessel cover seal is an Omega seal and is the main seal between the reactor vessel and its support system. The details provided are inadequate for evaluation.

Response:

The Omega seal feature has been deleted from the reactor vessel design since preparation of the PSAR. The related portion of PSAR, Section 5.2 has been modified to reflect this change.

Question 120.13 (Table 5.2-3)

The material used for the guard vessel should be clearly indicated in Table 5.2-3.

Response:

The material to be used is SA 240, Type 304 and is indicated in revised Table 5.2-3 and Section 5.2.2.3.

Question 120.14 (Table 5.3-6)

Table 5.3-6 lists primary system check valve materials. Provide information on satisfactory service experience in sodium for Inconel 718, Stellite-6B, Alloy B8M, and Stellite-6 (C-1).

Response:

Refer to the footnotes on Table 5.3-6 for the supplemental information. Numerous references to FFTF testing added.

Question 120.15 (5.2)

The methods to be used to control delta ferrite need clarification, especially regarding production testing of welds for delta ferrite content.

Response:

Clarification is provided in revised Section 5.2.6. 25

Question 120.16 (5.0)

The accessibility for in-service inspection of piping components inside guard pipes should be discussed.

Response:

A discussion of in-service inspection of piping within guard pipes is provided in Revised Section 5.2.4.5. 25

Question 120.17 (5.3.2.1.3)

The consequences of failure of a weld is considered a more valid reason for determining the extent of inservice inspection than the reasons indicated in the PSAR, e.g., plant outage duration, and availability of people. Provide the basis for listing these items in the PSAR as priorities for weld inspection.

Response:

The inservice inspection will be planned and conducted as appropriate, according to the requirements of ASME Code Section XI, Division 3 (under preparation) as shown in PSAR Section 5.3.2.1.3.

Question 120.18 (5.5)

There does not appear to be sufficient access to the steam generator tubes for inservice inspection. Inservice inspection should include steam generator tubes. This should be discussed in more detail.

Response:

The requested information is provided at the end of revised Section 5.5.2.3.4 of the PSAR.

Question 120.19 (5.4)

Provide additional information on how the inservice inspection program for the intermediate heat transfer system will be implemented and conducted in accordance with the intent of the ASMC Code, Section XI. Explain what exceptions to Section XI Code will be taken as a result of using liquid metal coolant for the intermediate heat transport system.

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC staff in a meeting on September 8, 1976 and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 33

Question 120.20 (5.2.4.6)

Provide details of the contemplated surveillance program.

Response:

Details of the surveillance program are provided in response to
19 Question 120.30.

Question 120.21 (5.3.2)

Under description of design in 5.3.2, reference is made to protection against accelerated corrosion and material degradation. In connection with this effort provide a statement describing what materials are discussed. Also furnish more details or commitments to future detailed discussion of avoidance of potentially corrosive environments.

Response:

The information requested is discussed in revised Section 5.3.2.1.4 of the PSAR.

Question 120.22 (5.1)

Statements are needed on the need for rapid cooling rates to avoid sensitization. Indicate which components cannot be cooled at a rapid rate.

Response:

At this point it is not possible to specify detailed heat treatment schedules. Each component will be evaluated independently to determine whether heat treatment is necessary. If it is required, the heat treatment will be optimized to give the maximum degree of stress relief together with the minimum degree of sensitization. The information requested is further discussed in revised Section 5.3.2.1.4 of the PSAR.

Question 120.23 (5.1)

Provide assurance that field compounded thermal insulation will maintain low leachable chlorides.

Response:

Design features and procedures to assure compatibility of the piping (and components) and external insulation are discussed in revised Section 5.3.3.10.4 of the PSAR.

Question 120.24 (5.5)

The steam generator design description does not have enough supporting reference material for evaluation. Details on the Atomics International Modular Steam Generator (AI-MSG) should be provided.

Response:

Further description of the AI unit is provided in revised Section 5.5.2.3.4 and a reference is provided for further information.

Question 120.25 (5.3.2.1.3, 5.4.2.1.3, 5.5.2.1.3)

The entire subject of inservice inspection of primary heat transport systems (PHTS), intermediate heat transport systems (IHTS) and steam generator system (SGS) must be presented and should include, as a minimum, specific locations to be inspected, method of inspection, and frequency of inspection. Access should be provided by the design to permit performing inspections.

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC Staff in a meeting on September 6, 1976, and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 34

Question 120.26 (5.1.2)

The response to Question 001.58 is not satisfactory. Since CRBRP is a demonstration plant, the integrity of the reactor vessel and nozzles should be monitored and demonstrated. It is the staff's opinion that this can best be accomplished by periodic volumetric examinations. Describe the critical areas in the reactor vessel on the basis of stress analysis, and provide access for future volumetric inservice inspection.

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC Staff in a meeting on September 8, 1976, and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 33

Question 120.27 (5.3.2.1.3)

Provide the following additional information discussed in 5.3.2.1.3 Surveillance and Inservice Inspection.

(1) Describe the metallurgical inspections that will be conducted.

(2) Inspections should be performed on the basis of consequences of failure rather than on the convenient access provisions. Describe specific areas to be inspected.

(3) What are "pump bearings of IHX tube to tube support plate interface?"

Response:

The NRC position with respect to preservice and inservice inspection for CRBRP has been provided in Round 2 Question (RSP) 120.66. The Project discussed its plans with the NRC Staff in a meeting on September 8, 1976, and is subsequently revising its Inservice and Inspection Plan to more fully accommodate the NRC position. The plan and any necessary supporting discussion have been provided in response to Question 120.66. 33

Question 120.28 (5.2.3.4)

Describe methods that will be used to verify the integrity of the core support structure of the vessel during service life..

Response:

The integrity of the core support structure during service life will be verified by material surveillance.

The CRBRP material surveillance program provides for core support structure material specimens and will include tensile specimens. These specimens will be placed in surveillance locations having a higher flux than in the region of the component whose material properties are being verified. Verification of the expected material behavior in conjunction with analysis is the best known means of verifying the integrity of the core support structure. The material behavior, as determined from the surveillance specimens, will lead the actual component neutron exposure.

The core support structure tensile specimens are selected on the basis of a minimum total residual elongation of ten percent. The first measurable loss of ductility in austenitic stainless steel occurs at 1×10^{21} n/cm^2 >0.1 MeV with increasing loss of ductility at higher fluence levels. Thus, tensile specimens are selected for all regions of the core support structure where the end of life fluence is equal to or greater than 1×10^{21} n/cm^2. Notch ductility degradation, as well as strength property changes, are also progressive with increasing fluence above 1×10^{21} n/cm^2.

Question 120.29 (5.2.4.5)

Provide assurance that the mobility of the transporter is adequate to permit observation of all welds in the vessel. Provide greater detail on the sensitivity of a TV camera (i.e., size of fissure).

Response:

Revised Section 5.2.4.5 provides more detailed information regarding transporter mobility and TV camera sensitivity. 25

Question 120.30

Provide in detail, the philosophy of the overall surveillance program of FFTF which, as indicated in the response, will be followed by the CRBRP. In particular, describe the coupon surveillance program, especially with regard to the materials, and testing conditions under which these data will be generated, and how these data will be used to monitor or evaluate the mechanical properties of the materials in question.

Response:

PSAR Section 5.2.4.5 has been expanded to discuss the coupon surveillance program in more detail.

Question 120.31 (5.1.7.e.2)

Provide supporting evidence that a sodium leak in an austenitic stainless steel welded system can only originate as a small or weeping type leak, and that it will propagate very slowly.

Response:

This question is answered in detail in the "CRBRP Primary Pipe Integrity Report," which was submitted to NRC on December 19, 1975. Section 4 in general and Sections 4.1, 4.5 and 4.6 specifically cover this question.

Question 120.32 (5.5.1.1 & 7.5.5.3)

In the second "b" of 5.5.1.1, the design basis leak for a steam generator module is addressed, but small leaks are not. However, in 7.5.5.3 the subject of small leaks is addressed and it is stated that the operator will take corrective action to prevent the leak rate from increasing. Describe this corrective action in sufficient detail to explain how it functions to control the leak rate.

Response:

Small leaks are not addressed in the second "b" of 5.5.1.1 since the paragraph pertains to the SWRPRS design, which is based on the steam generator module design basis leak. The paragraph titled Leak Detection Subsystem in section 5.5.1.1 refers to section 7.5.5.3 for subsystem details. The Operation Requirements paragraph of section 7.5.5.3 indicates the operator action based on detected leak size. The leak growth in the affected module is minimized by depressurization of the water/steam side. In addition, depressurization limits the quantity of water/steam available at the leak site.

Question 120.33 (5.5.3.7)

Provide engineering verification for the statement "The existence of unidentified leaks are not anticipated . . ." Provide leakage history for other liquid metal reactor systems that substantiate this statement, or provide other justification for this statement.

Response:

Section 5.5.3.7 has been revised in response to this question. 25

Question 120.34 (E.3.2)

Since the sensitivity of a leakage detector depends strongly on the location of the detector, cell volume, cell environment (atmospheric composition and temperature), sodium temperature and flaw shape, describe what criteria for the above variables will be used to ensure that a small leak in the PHTS or IHTS system can be successfully detected.

Response:

The detection of small sodium leaks in the PHTS and/or IHTS can be successfully accomplished by the Sodium Leak Detection System since its design criteria are based on cell environment, sodium temperature and leak location. The proposed Leak Detection System for CRBRP consists of aerosol monitors, cable detectors, contact detectors, particulate radiation monitors (for ^{24}Na), smoke detectors and pressure and temperature sensors. Emphasis is placed in the PHTS where protection is needed to avoid a large leak from the piping; however, in the IHTS the emphasis is reduced, since the vaults are being designed to accommodate a double ended pipe rupture, and there are no safety implications, just economic considerations.

Aerosol monitors are used to monitor the annular space between PHTS and IHTS piping and its insulation. In this configuration, cell size has no effect on the detectors' sensitivity. In addition, guard vessels and major components are provided with aerosol type detectors and cable detectors. The vault atmosphere of both Primary and Intermediate systems are monitored by pressure and temperature sensors; Primary System vaults are also monitored by particulate radiation monitors, and Intermediate Vaults are monitored by smoke detectors. Small cells in both Primary and Intermediate Systems are monitored by aerosol detectors.

Test data to date indicate that the effect of atmosphere composition or leak detection sensitivity is small when compared to the effect of sodium temperature. At the higher temperatures, $\geq 600^{\circ}F$, aerosol formation increases substantially so the use of aerosol detectors is planned. At the lower temperatures, aerosol formation is reduced, so other methods of leak detection are utilized such as particulate radiation monitors (concentrations of 10^{-15} to 10^{-16} gm/cc of sodium can be detected by these instruments).

Flaw shape is a variable that affects the amount of sodium leakage; however this is an uncontrollable variable. Detection criteria have been established for detection of small leaks in a specified period of time and will be provided in response to Q222.75.

Question 120.35 (7.5.5.3.1)

The thin walled nickel membrane used in the hydrogen detectors to pass hydrogen will be subject to Na corrosion, given as 3.9×10^{-3}/in/yr per Table 9.2.1.2-1 of Vol 2 of BNWL 1901. Justify the thickness of the Ni membrane, and/or plans and schedules for replacement.

Response:

The 3.9×10^{-3} in/yr nickel corrosion rate indicated in BNWL 1901 is based on exposure to 650°C (1200°F) liquid sodium flowing at 30-40 ft/sec. The conditions are not comparable to the hydrogen detector operating conditions of 950°F liquid sodium flowing at less than 1 ft/sec. Figure 9.2.1.2-1 of BNWL 1901 shows the effect of temperature on the corrosion rate of nickel in sodium and Reference Q120.35-1 includes data indicating the effects of sodium velocity on corrosion rate.

Based on the lower liquid sodium temperature and velocity, the corrosion rate is predicted to be 8×10^{-6} in/yr. Therefore, corrosion of the nickel membrane is not expected to be a problem based on an initial membrane thickness of 1×10^{-2} in.

References

Q120.35-1 Summary Report, Mass Transfer Program, GEAP-10394, August, 1971.

Question 120.36 (5.3.3.6.1.1) (5.4.3.6.1.1)

Identify the critical loop and locations within the primary and secondary system pipings based on stress analysis. Describe what measure(s) will be taken to ensure the integrity of those piping systems during the service lifetime of the plant.

Response:

Revised Sections 5.3.3.6.1.1 and 5.4.3.6.1.1 identify critical loop and locations within the primary and secondary system piping based on stress analysis. Revised Section 5.4.3.6.1.1 describes the measures taken to insure piping integrity. 25

Question 120.37 (5.3.3.6.2)

Provide justification for using the upper-bound curves presented in Figures 26-28 to estimate the fatigue crack growth of 304 stainless steel considering possible synergistic effects of temperature, frequency, stress ratio, sodium environment, etc., on crack growth at service conditions.

Response:

The "CRBRP Primary Pipe Integrity Status Report" submitted to NRC on December 19, 1975, gives a detailed illustration on the influence of environment and temperature, stress ratio, thermal aging, cold work, crack orientation, heat-to-heat variation, grain size, irradiation, biaxial stress state, loading waveform, welds and static loadings on fatigue-crack growth behavior for 304 stainless steel (Sections 4.1.1 thru 4.1.12).

In the PSAR, the upper-bound curves in Figures 5.3-26 thru 28 are used to obtain the most conservative estimate of crack growth.

Question 120.38 (E3.1.4)

Since it is stated that a double-ended hot-leg rupture not only will cause the pressure and temperature transients on the PHTS cells or reactor cavity but might also have the possibility of introducing gas bubbles into the core, describe the service experience and what measure(s) will be taken to ensure the structural integrity of these portions of piping to preclude a double-ended hot-leg rupture.

Response:

To insure integrity of the PHTS hot leg piping, the design criteria to be employed are the Class 1 requirements of the ASME B&PV Code, Section III, RDT Standard E15-2NBT and Seismic Category I requirements. In addition, the high temperature design criteria given in Code Case 1592 and RDT Standard F9-4T will be applied and the envirnoment surrounding the pipe will be inerted. (See PSAR Section 5.3.)

A program to verify the integrity of ASME Class 1 Piping has been undertaken. The Primary Pipe Integrity Status Report (Reference 2 to PSAR Section 1.6) issued in December 1975, describes results with analysis on the cold leg piping. A revision to this report which will include the analysis of the 24" diameter hot leg piping will be available in Jan. 1977. This report will demonstrate the adequacy of the structural integrity of these portions of the HTS piping to preclude a double-ended piping rupture. 33

Question 120.39 (4.2, 4.3, 5.3)

It is necessary to assure operability of reactivity control systems and to provide for disassembly after operation of certain reactor internals. Literature references at the conclusion of Section 4.2 are included to provide information on the antigalling characteristics and irradiation stability of mating materials in the reactor internals. The references include data obtained at lower than the LMFBR operating temperature. Provide test data to ensure operability of mechanisms under all reactor operating conditions.

Response:

The information requested is provided in revised PSAR Sections 4.2.2.3.3.3 and 4.2.3.1.3.

Question 120.40

The ductility of metallic materials is reduced by neutron exposure and thermal aging. Provide experimental data to ensure that the ductility of the material used for the reactor internals will be sufficient to maintain the integrity of the core through a thermal transient at the design end of life of the reactor.

Response:

The experimental data is provided in revised Section 4.2.2.2.1 of the PSAR.

Question 120.41 (4.2, 4.3, 5.2, 5.3)

Types 304 and 316 stainless steels are used extensively for fabricating the reactor vessel, major system components, and piping. The susceptibility to hot cracking of welds in these materials is decreased by adjusting the composition of the welds to contain delta ferrite in conformance to the requirements of Section III, ASME Boiler and Pressure Vessel Code, 1974 Edition. When exposed for extended times at elevated temperatures, delta ferrite may convert to the Sigma phase. Provide experimental verification to show that the fatigue and creep-fatigue properties of welded austenitic stainless steel components are not reduced below safe values at EOL, by the presence of Sigma phase in the microstructure.

Response:

There are no data that would directly relate the effect of sigma phase on either fatigue or creep-fatigue properties of Type 308 or 16/8/2 stainless steel weldments at the EOL of the CRBRP. There are some data which are presented below that indicate little effect of short-term aging on the fatigue properties of austenitic stainless steel weldments. Experimental verification to show the effect of sigma phase on the fatigue or creep-fatigue properties as requested would require a test program to generate such data.

The amount of sigma phase which could be formed as a result of transformation from delta ferrite will be low because the amount of delta ferrite is restricted in CRBRP. The restrictions on delta ferrite are imposed through invoking ASME Code Section III, Code Case 1592 and RDT standard requirements on CRBR welds. Delta ferrite will be held to 5 to 9 percent in filler metals for use with Type 304 material and to less than 5 percent in filler metals used with Type 316 material (provided 16-8-2 filler metal is used).

It is expected that some of the delta ferrite in the 304 and 316 weldments may convert to sigma when exposed to high temperatures over the operating life of the CRBRP. Ratz (Reference Q120.41-1) observed a small amount of sigma (approximately 2 percent) in a weldment made in Type 304 pipe. The weldment had been exposed as part of a sodium loop to temperatures between 1200° and 1470°F for 20,000 hours. Baker and Soldan (Reference Q120.41-2) have observed approximately 1.5 percent sigma in 16-8-2 weld metal after 60,000 hours of laboratory exposure at 1150°F. 10,000 hour, 900°F aging of 308 CRE (controlled residual element) weld metal containing 5 to 7 percent delta ferrite did not result in sigma formation (Reference Q120.41-3). Similarly, aging of Type 308 weld metal for 1000 hours at 1100°F did not result in sigma formation (Reference Q120.41-4).

Limited data obtained after short term aging show no detrimental effect on weldment fatigue properties. Brinkman and Korth (Reference Q120.41-4) observed little effect of 1000 hour, 1100°F aging on the fatigue life of Type 304 weldments made with Type 308 filler metal. Similarly, 5923 hour, 900°F aging of Type 308 weld metal had no significant effect on fatigue properties (Reference Q120.41-6). James (Reference Q120.41-7) has observed slightly decreased fatigue crack propagation rates (relative to 304 base metal) in Type 308 weld metal after 3000 hour aging at 1000°F while Type 304 weldments aging at 1100°F for 800 hours observed to have a beneficial effect on fatigue crack propagation rates.

In summary, the available observations, based on relatively short-time aging coupled with the time temperature transformation data of Cole, et.al. (Reference Q120.41-8) leads to the conclusion that sigma effects on fatigue and creep-fatigue effects over the operating life of CRBRP are not detrimental. However, it must be recognized that no long term aging data are available to confirm this conclusion.

References

Q120.41-1. A. Ratz, Effect of Long-Time Exposure to Liquid Sodium on a Welded AISI Type 304 Stainless Steel Pipe, Nuclear Technology, Vol. 17, February 1973, p. 153-159.

Q120.41-2. R.A. Baker and H.M. Soldan, "Service Experiences at 1050°F and 1100°F of Piping of Austenitic Steels, Proc. Inst. Mech. Engrs., 178, Pt. 3A, 1963-1964, p. 4-85.

Q120.41-3. Fuels and Materials Development Program Quarterly Progress Report for Period Ending September 30, 1973, ORNL-TM-4405, p. 155-161.

Q120.41-4. C.R. Brinkman and G.E. Korth, Heat-To-Heat Variations in the Fatigue and Creep-Failure Behavior of AISI Type 304 Stainless Steel, and the Fatigue Behavior of Type 308 Stainless Steel Weld Material, ANCR-1097, May 1973, p. 45.

Q120.41-5. Mechanical Properties Test Data for Structural Materials Quarterly Progress Report for Period Ending October 31, 1974, ORNL-5103, p. 5.

Q120.41-6. Mechanical Properties Test Data for Structural Materials Quarterly Progress Report for Period Ending July 31, 1975, ORNL-5106, p. 2.

Q120.41-7. L.A. James, Effect of Thermal Aging Upon the Fatigue-Crack Propagation of Austenitic Stainless Steels, Metallurgical Transactions, Vol. 5, April 1974, p. 831-838.

Q120.41-8. N.C. Cole, G.M. Goodwin and G.M. Slaughter, "Effects of Heat Treatments on the Microstructure of Stainless Steel Weld Metal", Microstructural Science, Vol. 3, 1975, p. 789 to 822.

Question 120.42 (4.2, 5.2)

The transport of a portion of the radioactive structural materials from the core to colder areas in the reactor systems is anticipated. Estimate the maximum radioactivity with operating time in particular areas of the primary heat transport system, and show that this radioactivity will not interfere with the in-service inspection procedures. Provide sufficient evidence that deposits of mass transported material will not reduce the heat transfer capability of the IHX to an unacceptable degree, including the results of the study cited (but not referenced) on page 5.3-70.

Response:

The CRBRP in-service inspection program is discussed in PSAR Section 5.3.2.1.3. The principal emphasis in the program is placed on visual condition inspection. Capabilities are being developed to permit remote viewing of the primary coolant boundary in the PHTS cells and pipeways. This will include the ability to view the annuli between an IHX or pump, and its respective guard vessel. This type of inspection will result in minimal radiation exposures. Further discussion of in-service inspection radiation exposures can be found in revised Section 12.1.5.

The study cited on page 5.3-70 has been identified in revised Section 5.3.3.10.3.1. Discussions of IHX performance including allowances for sodium corrosion product deposition are provided in Section 5.3.3.5 "Intermediate Heat Exchanger Characteristics".

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Question 120.43 (4.2.3.1.7):

Describe the test program(s) performed on the Inconel 718 CRDM bellows to ensure that adequate margins of strength will be maintained in the high temperature sodium and irradiation environment.

Response:

The information requested is found in amended PSAR Section 4.2.3.1.7.

Question 120.44 (5.2.2.1)

Describe the effect of internal heating of the boron carbide neutron shield ring surrounding the vessel near the flange on the flange temperature.

Response:

The information requested is provided in revised Section 5.2.2.1.

Question 120.45 (5.2)

The reply to Question 120.11 referred to a "block welding technique". In using this technique, discuss how the sensitization of welds is prevented. Discuss how weld defects are prevented at the weld "block" overlaps or start-stops.

Response:

The term "block welding" used in the response to Question 120.11 is a misnomer. Section 5.2.3.2 has been revised to clarify this.

Question 120.46 (5.3)

Published data (NRL Memorandum Report 2752) suggests that notch ductility of Type 316 stainless steel submerged arc weldments may be quite low (56 ft-lb upper shelf value) and that it may be significantly further degraded by neutron irradiation. Provide data to verify that fabrication processes used in the manufacture of the CRBR pressure vessel and other components will produce sufficient notch ductility to enable these structures to withstand thermal transients during the plant design life.

Response:

A discussion of the components' capability to withstand thermal transients is provided in new PSAR Section 5.3.3.10.1.5.

Question 120.47 (5.3)

The present programs for tests to verify high temperature design criteria are described in Tables 5.3-19 through 5.3-22. In the descriptions of the samples to be tested, none are indicated as containing welds or weld metal. Describe which samples in the presently funded program will test welded specimens with delta ferrite to verify the adequacy of welds made with weld rod purchased to the RDT M1-1T and M1-2T Standards.

Response:

The delta ferrite content of stainless steel welds will be controlled by strict compliance with ASME Code Section III Division 1. Subsection NB for Class 1 Components (Section NB2433) and Code Case 1592 (Section 2433). In complying with the code, tests of production filler metal or welded metal are required as stipulated therein and delta ferrite limits are specified. More information is provided in revised Section 5.3.3.10.2.3. 25

Question 120.48 (5.3)

Information presented in the PSAR indicates that the fatigue life of austenitic stainless steel is reduced by the presence of a carbide phase in the microstructure. Provide experimental data and discussion to verify that long time exposure at high temperature will not reduce the fatigue life (due to the formation of chromium carbide) below the design requirements.

Response:

Section 5.3.2.2.3 has been expanded to provide the information requested. 25

Question 120.49 (5.3, 5.5)

The reply to 120.15 states that there will be no testing for delta ferrite in production welds. The amount of delta ferrite in the weld will be controlled by comparing the composition of the filler welding rod to the Shaeffler diagram. It is the staff's position that the amount of delta ferrite in the production welds must either be determined by actual measurement or an acceptable program of testing preproduction welds be established to demonstrate uniform and predictable weld ferrite content under production conditions. Describe your method of conformance with the above position.

Response:

The information on delta ferrite content is provided in revised Section 5.3.3.10.2.3. 25

Question 120.50

The primary pump shaft is supported at the impeller end by a sodium bearing. Provide data to show that during start-up and shut-down, self welding of bearing and journal surfaces will not occur. Describe the mock-up tests conducted to show the conditions under which self welding, galling or hard-surface deterioration do not occur.

Response

Self-welding is a phenomenon which depends upon the simultaneous interaction of a number of parameters such as:

- Surface Cleanliness
- Metal Diffusion Rate
- Surface Temperature
- Contact Pressure
- Contact Time
- Material Couple
- Surface Finish
- Sodium Purity

In general, self-welding requires that the above parameters have the following characteristics:

1) Surfaces be atomically clean.

2) High metal diffusion rate. Metal temperature above $800^{\circ}F$. The higher the temperature, the higher the metal diffusion rate.

3) Contact pressure between the interfacing surfaces approaching the yield strength of the mateiral.

4) Long contact time (static contact).

5) Material couples of similar material which are relatively soft, of low yield strength, and prone to galling.

6) A rough surface which provides many points of very high contact stress (small true area of contact). The effect of surface finish is not well understood.

7) Sodium of very high purity, 1 ppm of O_2 or less.

In case of the CRBRP pump at start-up or during coastdown, some of the above parameters are in the direction of no self-welding as follows: 33

1) At $600^{o}F$ (more likely $400^{o}F$) temperature is too low for metal diffusion.

2) Contacting pressure (hertzian stress) is very low because bearing contact loads during start-up and coastdown are very low (stress well below the yield strength of the hardfacing material).

3) Static contact time during periods of pump shut-down could be long, but contact stress is very low (less than 1000 psi).

4) Hardfaced surfaces (Stellite, Colmonoy) in general are non-galling, high strength surfaces.

5) Surface finish of 16 microinches AA provides a uniform smooth surface and a relatively large area of true contact (low stress).

In addition, there is the following test experience which provides confidence that self-welding of the pump bearing will not occur:

1) The continuing Sodium Technology Program on Friction Wear and Self-Welding at Westinghouse ARD reports that a review of the self-welding tests conducted on various material couples, including hard-facing materials for bearings, showed that for all materials tested no self-welding occurs at temperatures of 850°F or lower for static contact times up to 6 months. At temperatures of 600°F or less, expert opinion is that self-welding probably does not occur under these contact conditions regardless of contact duration.

2) Friction and wear tests at LMEC with pin and disk machines operating in sodium at representative contact stresses (500-800 psi) and temperatures from 400 to $1200^{o}F$ showed no evidence of self-welding of various hardfacing materials tested. Coefficients of sliding friction averaged in the range from 0.3 to 0.6 with relatively low wear.

3) Westinghouse ARD FFTF pump bearing mock-up tests in sodium to investigate the effect of 100 start and coastdown cycles of the FFTF pump, represented a total of 10,000 revolutions of rubbing under typical load. Bearing rotation with contact was at a temperature of $400^{o}F$ (typical coastdown temperature from pony motor speed). Inspection of the bearing surfaces (Stellite 6B vs Stellite 6B) indicated only slight abrasion in a narrow band and no evidence of any self-welding whatsoever. The degree of surface abrasion observed (very light galling) does not deteriorate the performance of the bearings since the bearing is very tolerant of considerable surface damage. 33

4) Forthcoming FFTF pump tests in sodium in the SPTF at LMEC will provide additional valuable information on the performance of a typical bearing hardfacing material couple, Stellite 6B vs. Stellite 6B, under actual operating conditions.

The significant amount of experimental and actual operating evidence cited above leads to the firm conclusion that the CRBRP pump hydrostatic bearing hardfacing material will not self-weld under the conditions of pump start-up or coastdown after a shutdown. Additional confirming evidence will be obtained from the FFTF pump tests in sodium at LMEC in SPTF, including effects of surface deterioration or galling, if any, on bearing performance.

No additional mock-up tests are contemplated at this time for the CRBRP pump bearing pending review of the FFTF pump SPTF sodium test data relating to bearing surface conditions and performance. 33

Question 120.51 (5.2.1)

The closure head has an average temperature of 400°F under both normal operation and refueling conditions. It is stated that because the closure head is at 400°F the elastomer seals can be kept at 150°F for adequate life. Provide information to justify the use of 150°F as the operating temperature at the region of the elastomer seals at the top of the risers.

Response:

Revised Sections 5.2.2.2 and 5.2.4.5 provides the information requested. 25

Question 120.52 (5.3.1, 5.4, 9.3)

Strip heaters are used to heat sections of the pipe in the primary, intermediate heat transport system, and auxiliary liquid metal systems. Electric shorting may result in pipe damage. Describe the precautions that are taken to ensure against this occurrence and/or the inspection to be performed following such an occurrence. Additionally, discuss the effect of thermal stresses resulting from the failure of one or more of the strip heaters.

Response:

Section 9.4, Piping and Equipment Electrical Heating discusses the design of the heaters to prevent damage to piping or components due to electric shorting. A discussion of the effects of failure of one or more trace heaters is provided in revised PSAR Section 9.4.3, "Safety Evaluation".

Question 120.53

Welds joining ferritic and austenitic alloys may be subject to accelerated degradation when exposed to long-term, high temperature sodium environment and thermal cycling. Describe the methods and processes used in making bimetallic joints, indicate the location of such joints, and provide evidence that the design basis mecnanical properties of these weldments will be retained throughout their anticipated service life.

Response:

A discussion of the conservative approach being taken with respect to ferritic-austenitic welds is provided in revised Section 5.5.3.11.2.

Question 120.54 (5.3.3.6)

The following statement is made in Section 5.3.3.6: "The potential for piping degradation due to caustic corrosion caused by a postulated leak has been investigated experimentally. Experiments have been performed to investigate sodium leakage from cracks in test pipes." Provide a more detailed description of the experiments in support of the foregoing statement and the results of these experiments.

Response:

This question is answered in detail in the "CRBRP Primary Pipe Integrity Status Report". The corrosion aspects of the tests are covered in Section 5.5, and the leakage characteristics in Section 5.4.

Question 120.55 (5.3.2.2.1.5)

Long-term exposure of heat transport system structural materials in sodium will result in a loss of interstitial elements from hotter sections of the system, and thus lead to a loss in strength. Provide the experimental basis for the derivation of the interstitial loss equations and demonstrate the validity of the extrapolation to end of life.

Response:

Section 5.3.2.2.5 has been retitled "Mass Transfer Properties" and the last paragraph deleted. New Section 5.3.2.2.6, "Interstitial Transfer Properties," has been added in response to the question.

Extrapolation of the equations, particularly to low temperatures, contains some uncertainty. However, experimental tests have verified the data for periods up to 2000 hr. within the 850-1360°F temperature range. The early estimates of the carbon potential C_s, to be anticipated in the reactor, now appear to be overly conservative, and a value of C_s of not less than 50, is anticipated rather than $C_s = 30$. This will effectively raise the carburization/decarburization crossover temperatures for T 304 SS and T 316 SS to about 975 and 900°F respectively. Additional 10,000 hr. sample exposures are currently being initiated; when the data are available in 1977, extrapolations of the basic equations will be further validated.

Question 120.56 (5.4.2.1)

The PSAR references ASME Code, including Code Case 1592 and RDT standards in discussing materials for IHTS pressure-containing components. A statement is made that "the use of additional or alternative material properties shall require the approval of the purchaser." Provide additional discussion regarding the intent of this statement and define the term "alternative material properties" in relation to the ASME code design basis.

Response:

This statement is clarified in revised PSAR Section 5.4.1.2.

Question 120.57 (5.5.3.11):

The statement in Section 5.5.3.11 that the "compatibility of austenitic stainless steel with external insulation is assured as set forth in 5.3.3.10.4" is made. The referenced section does not provide documentation. Provide experimental data to justify the above statement.

Provide limits for "excessive moisture in insulation materials," and describe details of the "quality controlled installation" used to prevent "excessive moisture."

Response:

As shown in Table 5.5-3, most components in the Steam Generator System (SGS) are constructed of materials other than austenitic stainless steels. Compatibility of austenitic stainless steels with external insulation shall be assured by requiring the thermal insulation materials to be tested and analyzed in accordance with RDT Standard M12-1T, October, 1972. The standard gives requirements for conducting corrosion tests and chemical analysis using samples of insulation material selected from production lots and for certification of results of chemical analyses and corrosion tests.

Most thermal insulation materials do not in themselves cause stress corrosion cracking of austenitic stainless steels. However, the presence of leachable chlorides and moisture can cause the chloride ion concentration at a stress point sufficient to catalyze crack propagation. As explained above, the leachable chlorides will be controlled by application of RDT Standard M12-1T for all insulation to be installed on austenitic stainless steel and 2-¼ Cr-1Mo components and pipes.

A selection program will demonstrate that the insulation selected meets the required criteria. Specifically the tests will determine the following for selected candidate insulations.

a. moisture content
b. compressive strength
c. thermal conductivity
d. sodium compatibility
e. compatibility with materials of construction
f. leachable chlorides
g. chemical off-gases during heat-up

The above tests will be performed on 3" x 3" x 3" test specimens. The following insulation types have been selected for initial evaluation:

a. Babcock and Wilcox - Kaowool (Alumini-silica)
b. Owens Corning - Kaylo - 10 (Calcium silicate)
c. Pittsburgh Corning Foam Glass (Steam Generator Application)

Question 120.58:

The material presently specified for the Steam Generator Auxiliary Heat Removal System (SGAHRS) is carbon steel. Describe the procedures to be taken to ensure against caustic gouging, stress corrosion cracking, pitting corrosion, incompatibility with insulation and methanation.

Response:

Specific procedures for maintaining and monitoring SGAHRS water chemistry are yet to be developed, however, the system has been designed to allow recirculation mixing (if required) to assure that representative samples are taken during sampling. The minimum water purity level specified is as follows:

Cation conductivity (at 70°F) <10 micro Mho/cm
pH of 9.5 to 10.0 by ammonia addition
hydrazine (catalyzed) 150 ± 25 ppm
Suspended solids <5 ppm

This water chemistry is a common wet lay up chemistry used in commercial plants which is designed specifically to prevent pitting corrosion. The extremely low corrosion rates associated with this chemistry along with the low operating temperatures preclude methanation. Caustic gouging is not a problem since the SGAHRS contains no caustic. SGAHRS water will contain no sodium hydroxide,only ammonium hydroxide which will not cause stress corrosion cracking in carbon steel.

Since the usual effect of increased velocity is an increase in corrosion rate, flow velocities have been limited to the approximate ranges specified below:

Type of Service	Maximum Velocity - fps
Pump Suction	10
Pump Recirculation	70
Steam Drum Feed	20
Saturated Steam-PACC	22
Saturated Steam-Turbine Supply	200
Turbine Exhaust	500
Superheater and Steam Drum Vent	500

The minimum corrosion allowance, including cleaning, in terms of additional thickness of material, shall be 0.10 inches. This shall be deducted from the available structural material before strength calculations are performed.

Question 120.59 (4.2, 5.2 & 5.5.3.11)

Thermodynamic stability of structural materials used in the primary, intermediate, and secondary heat transport systems is an important consideration. Show that the steam generator tubing will not be degraded by the interstitial mass transfer. Provide background data on the decarburization of 2-1/4 Cr-1 Mo steel at 965°F in steam generator sodium to justify the statement in Section 5.5.3.11.4 that the carbon level will not drop below 0.03% during 30 year design life of the tubes.

Response:

Section 5.5.3.11.4 has been revised in response to this question.

Question 120.60 (5.3.2.2)

The statement is made in 5.3.2.2 that, "limits will be placed on the carbon level, ranging from 0.04% to 0.055% to ensure that the steel does not fall below the ASME Code requirement of 0.04% for high temperature service at end of life." Provide the experimental data to justify these limits, and to verify that the strength requirements of the Code will be met throughout the material cross-section at the end of the design life.

Response:

Clarification of the ASME Code requirement regarding minimum carbon content for Types 304 and 316 stainless steels to be used in Code Case 1592 applications, is warranted. Table I-14.1 of Code Case 1592 lists permissible material specifications for structures, other than bolting, and Note 1 of this table specifies a minimum carbon content of 0.04 percent for Types 304 and 316 stainless steel. This is interpreted to be a "start-of-life" requirement, since the Code does not provide specific guidance on deterioration of materials in service. Paragraph NA-1130 of Section III, subsection NA states:

"(a) The rules of this Section provide requirements for new construction, and include consideration of mechanical and thermal stresses due to cyclic operation. They do not cover deterioration which may occur in service as a result of radiation effects, corrosion, erosion, or instability of the material. These effects shall be taken into account with a view of realizing the design or the specified life of the components".

Paragraph NB-2160 of Section III, subsection NB for class 1 components states further:

"Consideration of deterioration of materials caused by service, is generally outside the scope of this Section. It is the responsibility of the Owner to select materials suitable for the conditions stated in the Design Specifications (NA-3250), with specific attention being given to the effects of service conditions upon the properties of the materials".

"Any special requirements shall be specified in the Design Specification (NA-352 and NB-3124). When so specified, the check analysis shall be made in accordance with the base metal specifications..."

Section 5.3.2.2 of the PSAR was thus misleading in referring to an "ASME Code requirement of 0.04% for high temperature service at end of life due to interstitial loss in sodium", and the Section has been revised. As noted above, the ASME Code specifies a minimum carbon content of 0.04% to ensure the material properties given in the Code Case 1592.

The Code requirement for a "start-of-life" minimum carbon content of 0.04% for Types 304 and 316 stainless steels in applications governed by Code Case 1592, is being met by specifying a carbon content of 0.04% - 0.08% in the Equipment Design Specifications.

As noted in the excerpts quoted above, the Code requires the designer to evaluate the effects of the service environment on material properties. Guidance for performing this evaluation is provided in an Appendix to the Equipment Specifications.

Briefly, for interstitial transfer effects in sodium, the analysis consists of the following:

1. Determine the carbon + nitrogen (C+N) concentration for the component material in question at the end of life. Equations are provided for determining the actual interstitial gradient through the material cross-section as well as the average interstitial content.

2. Based on results of Step (1) above, revised material properties (tensile stress-rupture) are calculated and corrections are made to the Code allowables where necessary.

For a more detailed treatment of the effect of sodium on mechanical properties of materials, refer to Section 4.2.2.3.3.2.1 of the PSAR.

The experimental basis for Step (1) above, is the subject of the response to PSAR Question 120.55. The equations for determining the effect of (C+N) on tensile properties of 304 and 316 stainless steel have been derived in Reference Q120.60-1. The basis for calculating stress-rupture properties can be found in Reference Q120.60-2.

In the majority of cases, cross-sectional thicknesses are large, and operating temperatures are such that interstitial losses are small, thus resulting in a negligible impact on design. However, in other cases (e.g. primary piping, IHX tubing, etc.), the material cross-sections are relatively thin and interstitial losses could be sufficiently high to cause reductions in properties which must be considered in design (Reference Q120.60-3). To minimize the impact of these reductions on design, higher "start-of-life" minimum carbon contents (e.g., 0.055% for the primary piping) are or will be specified in the Piping Design Specification for the primary piping components.

Question 120.61 (5.5.3.11.3)

Describe the rust protecting methods that will be used for 2-1/4 CR-1 Mo and carbon steel. Provide evidence that the material(s) used for rust protection and solvents required for their subsequent removal during or after erection will not produce deleterious effects.

Response:

Section 5.5.3.11.3 has been revised to discuss the method of rust prevention.

References:

Q120.60-1 P. Soo and W.H. Horton, "The Effect of Carbon and Nitrogen on the Short-term Tensile Behavior of Solution-Treated Types 304 and 316 Stainless Steels", WARD-NA-3045-2, July 1973.

Q120.60-2 S.A. Shiels, S.L. Schrock, and L.L. France, "Interstitial Transfer Program Impact Assessment Report - Part II", WARD-NA-3045-3, Sept. 1973, pp. 76-82.

Q120.60-3 P. Soo, "Selection of Coolant-Boundary Materials for the Clinch River Breeder Reactor Plant", WARD-D-0010, August 1974, Table 14, pp. 58-59.

Question 120.62 (120.15, 120.41, 120.47, 120.49)

Responses to previous questions are not sufficient to conclude that the mechanical properties of welded austenitic stainless steel will not degrade during the life of the plant. The information submitted does not address adequately the long term thermal aging effects. The specific weld filler rod and welding procedures to be used in CRBRP will affect the weld ferrite content, and, after thermal aging, the sigma phase morphology. Sigma phase can degrade the weld joint mechanical properties.

It is our position that tests should be initiated prior to plant construction to evaluate the long term thermal aging effects upon the mechanical properties, toughness and crack propagation of welds using materials and procedures specified for CRBRP.

Response: (Interim)

Qualitative assessment of the limited available data regarding microstructural and property changes of austenitic stainless steel weldments as related to service temperature and exposure time indicate that the materials utilized in CRBRP have a high likelihood of acceptable mechanical performance. However, the need is recognized to expand the data base, particularly the data on weldments thermally aged for suitable times at temperature and/or proven methods of extrapolating available short term data.

An experimental program to examine the effects of long term thermal aging on welded austenitic stainless steel is included within the base technology program. This program would utilize the specific base metals, weld filler metals, and weld processes to be utilized in CRBRP fabrication. Specimens will be evaluated in the condition prototypic of start of plant life and after various thermal aging times. The duration of thermal exposure would be terminated on a case-by-case basis. Material combinations selected for investigation would include prototypic primary hot and cold leg piping welds as a minimum. The properties to be evaluated would be selected to provide insight on property degradation as related to likely failure modes of the component involved and may include any or all of the following:

- microstructural evaluation
- tensile properties
- creep properties

- crack propagation
- toughness as determined by the J-integral method

The schedule for the experimental program will be dependent upon the fabrication schedules of a number of related components, for some of which vendors are yet to be selected (such as the PHTS piping). A detailed schedule for this experimental program therefore cannot be defined at this time. However, NRC will be informed when such a schedule is available.

Question 120.63 (3.2, 5.3.2.1.3)

ASME Code Case 1594 is applicable for the examination of elevated temperature Section III, Class 1 components only. Provide a listing and technical basis for preservice nondestructive examination requirements that you are specifying for ASME Section III, Class 2 and Class 3 components which have not been upgraded to Class 1, and where metal temperature exceeds those for which allowable stress values are given in Section III.

Response:

The Auxiliary Liquid Metal System and the Impurity Monitoring and Analysis System have components designated as ASME Section III, Classes 2 and 3, in which metal temperatures exceed those for which allowable stress values are given in Section III. The following components by system are Section III, Class 2 or 3, and must be designed for elevated temperature service in accordance with an applicable Code Case.

EVST NaK air blast heat exchangers

NaK piping from overflow heat exchangers to the EVST ABHX including valves

Primary Na cold trap economizers

Piping between primary Na cold traps and the first isolation valve

Primary Plugging temperature indicator (PTI) and associated piping & valves

Primary sodium sampling package (SSP) and associated piping and valves

Intermediate sodium cold trap pumps

Intermediate sodium cold trap economizers

Intermediate sodium cold trap piping and valves (applies only to normally flowing circuit-not applicable to drain lines, transfer piping, or piping between cold trap economizer and crystalizer)

Intermediate sodium characterization package and associated piping and valves

EVS Multipurpose Sampler

ASME Section III, Class 2 or 3 components of the Steam Generator System (SGS) and Steam Generator Auxiliary Heat Removal System (SGAHRS) which will see elevated temperature service are as follows:

- Superheater Outlet Steam Piping
- Superheater Outlet Isolation Valve
- Superheater Outlet Check Valve
- Superheater Relief Valve Inlet/Outlet Piping
- Superheater Relief Valve
- Reaction Products Separator Tank
- Sodium Rupture Discs to Reaction Products Separator Tank Piping
- Reaction Products Separator Tanks Equalizer Piping
- Superheater Steam Vent Inlet/Outlet Piping
- Superheater Steam Vent Valve
- Superheater Steam Vent Isolation Valve

Preservice inspection for all ASME coded plant components is addressed in response to NRC Question 120.66.

Question 120.64 (RSP)(5.6)

Your response to Question 120.18 is not complete. A general description and technical basis for your program for periodic inservice inspection of steam generator tubing was not addressed.

It is our position that periodic volumetric inservice examination of a representative sample of the steam generator tubes is required. This requirement is intended to assure the continuing structural integrity of the sodium-water boundary, to mitigate the consequences of the unlikely event of a significant sodium-water reaction and to identify potential long term degradation mechanisms that may result from plant operating conditions. The initial sampling program and examination frequency should be based on results from your steam generator development program subject to modification depending on results from periodic volumetric inservice inspection experience.

Response:

Access for future inservice inspection of Steam Generator tubing is provided through the use of removable steam heads that allow access to the entire tubesheet at both ends of each steam generator module. The extent and interval of inservice inspection of steam generator tubes is dependent on results of ongoing steam generator development programs and on the sensitivity and accuracy of existing and developing ISI methods for CRBRP steam generator tubing. Actual inspection intervals will be selected after the steam generator development programs are complete.

Question 120.65 (5.5, 15.3)

In your response to Question 120.18 you state in Section 5.5.2.3.4 that the inner diameter of the steam generator heat transfer tube is readily available for inspection by ultrasonics, eddy current, and/or other suitable means which will be determined at the conclusion of a development program now in progress.

Identify any inaccessible regions in the bend radius of the U-tube to internal probe inspections by eddy current or ultrasonic methods due to probe interference or potential loss of coupling.

Conventional inside diameter eddy current probe techniques are not normally used to examine ferromagnetic tubing, such as 2¼ CR-1 Mo material, with wall thicknesses as great as 0.109-in. to detect tube wastage or through wall penetration. Discuss the effectiveness of the eddy current method considering the mechanism for the development of a large leak from a small steam leak by defect enlargement on the sodium side as shown in PSAR Figure 15.3.3.3-1. Provide the technical basis for your conclusions.

Discuss potential limitations to the ultrasonic method for the detection of discrete radial flaws (pin-holes), circumferentially oriented cracks in the butt weld region and tube deformation in support areas. Define your developmental program objective for minimum detectable flaw size and rejection (tube plugging) criteria in terms of percentage of tube wall wastage and minimum crack size. Provide the technical basis for your conclusions.

Discuss the scope of your development program for steam generator tube leakage location using acoustic emission techniques.

Response:

The CRBRP Steam Generator modules are in a hockey stick configuration with a "worst case" head radius of 24 inches. Eddy current and ultrasonic probes should both be able to negotiate this bend while making measurements, although the CRBRP Project expects ultrasonic inspection techniques to be adequate without reliance on eddy-current techniques.

The ultrasonic method using commercially available probes has demonstrated the ability to detect pits a few mils deep, through wall flaws 0.010 inches in diameter, and notches 0.003 inches deep or less, in prototypical tubing. The goal for minimum detectable notch of five percent of the tube wall and the accuracy of wall thickness measurement is ± 0.005 inches. The measurement of tube deformation greater than 0.005 inches has been demonstrated using ultrasonic methods. The tube-to-tubesheet butt welds are examined by a separate ultrasonic technique developed at ORNL. The sensitivity to circumferential cracks is strongly dependent on details of weld crown geometry on the inner surface of the tubing. This notch sensitivity is thought to be about ten percent of the tube wall based on prototypical welds determined to date.

Acoustic emission leak location has been demonstrated in static steam generator modules, but is not currently practical in operating steam generators or in full scale steam generators. Development at the National Laboratories as well as commercial developments are being closely followed. A decision to use this approach for leak location is not warranted at this time. If development is successful, this method can be applied to the steam generator modules.

Question 120.66 (RSP) (5.0, 5.1.2, 5.3.2.1.3, 5.4, 5.4.2.1.3, 5.5.2.1.3)

Questions 120.19, 120.25, 120.26 and 120.27 related to the inservice inspection program have not been answered.

It is our position that, as a minimum, a preservice nondestructive examination and inservice inspection (ISI) program including the examination categories, inspection methods, and governing documents or acceptable alternative methods is required. These methods are summarized below:

Preservice Nondestructive Examination

Examination Categories	Inspection Method	Reference Documents
Primary heat transport system including reactor and closure head, connected piping	Volumetric examination of 100% of the welds and adjacent base metal with methods capable of characterizing the throughwall dimension of indications.	ASME Code Case 1594
System, primary pump tank, check valve and intermediate heat exchanger shell. Also reactor vessel core internals.	To assure the initial structural integrity of the primary heat transport system and reactor vessel core internals, we will require the following information about all flaws over 5% in throughwall dimension retained after the preservice examination: 1. Location of acceptable flaw. 2. Definition of the throughwall dimension as a percentage of wall thickness. 3. Characterization of the flaw in terms of nature, enclosed volume, and actual orientation. 4. Complete documentation suitable for fracture mechanics evaluation.	
Balance of Class 1 components.	Volumetric examination of 100% of welds and adjacent base metal	ASME Code Case 1594

Examination Categories	Inspection Method	Reference Documents
Intermediate heat exchanger tubing.	Eddy current method for preservice baseline.	To be established by CRBR development program
Guard vessels	Volumetric examination of 100% of the welds and adjacent base metal.	ASME Code Case 1594
Intermediate heat transport system including intermediate heat exchanger connected piping system, pump, expansion tank, dump valves and flowmeter	Volumetric examination of 100% of the welds and adjacent base metal.	ASME Code Case 1594
Balance of Class 2 components	Volumetric examination of welds and adjacent base metal.	ASME Code Section III. We will require the identification of governing codes in cases where metal temperatures exceed those for which allowable stress values are given in Section III
Steam generator tubing	Volumetric method for preservice baseline.	To be established by CRBR development program

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Examination Categories	Inspection Method	Reference Documents
Class 3 components	Examination of welds and adjacent base metal.	ASME Code Section III. We will require the identification of governing codes in cases where metal temperatures exceed those for which allowable stress values are given in Section III or when Section III is not applicable

Inservice Examination Program

We will require that an inservice inspection program be established and submitted for review. This program should include periodic volumetric examination of a significantly representative sample of vessel and piping welds in both the primary heat transport (PHT) system and the intermediate heat transport (IHT) system. The periodic volumetric examination is intended to provide assurance of the continuing structural integrity of the ASME Class 1 and Class 2 components and to reliably identify potential long term degradation mechanisms that may result from plant operation.

Provide a detailed discussion of your volumetric examination program for both the PHT system and the IHT system including the technical justification for the selection of specific welds. Pertinent subjects that should be addressed in your response are the safety significance of the weld, potential degradation processes in the system, and the areas of highest operating stress levels, and highest temperature gradients. Regions of structural discontinuity and terminal ends of the piping systems, such as vessel nozzle to pipe welds, should be included in your inservice inspection program. Your discussion should include the identification of the location, materials, and accessibility of selected welds in each loop of both the PHT system and the IHT system.

We will require that for those welds selected for the representative volumetric sampling program, the design and arrangement of the system components include allowance for adequate clearances to conduct 100% volumetric examination of the selected welds and adjacent base metal. Since potential degradation mechanisms and unique operating conditions are still to be firmly defined during development programs, examination accessibility should be an important consideration in all component designs. We require that sufficient accessibility be provided for visual inservice examination of the reactor vessel nozzle to pipe weld for each inlet downcomer.

After the preservice examination is completed, acceptable retained flaws with the greatest percentage throughwall dimension should be added to the volumetric inspection program. We will require that the methods, equipment and techniques used during the periodic volumetric inservice examination program be equivalent to those used to establish the preservice baseline.

Response:

NRC question 120.66 requested that information be submitted related to inservice inspection of CRBRP including details of the preservice inspection program. We had previously described our inservice inspection program in the document, "CRBRP Plan for Inservice Inspection - August 1976", which was discussed in the joint NRC - CRBRP Project meeting on September 8, 1976. We have subsequently revised our inservice inspection program, and a summary of the revised program is described below. The CRBRP Inservice Inspection Plan has been revised and is provided in Appendix G to the PSAR. The plan includes details of the inservice inspection described below.

General Approach

The general approach of inservice inspection of CRBRP is based on ensuring, through continuous monitoring and periodic inspections, the integrity of systems and components whose failure could adversely affect core reactivity control or core cooling, or could result in an unacceptable release of radioactivity to the environment. Continuous monitoring will be provided by a diverse, redundant leak detection system capable of detecting very small amounts of sodium leakage. Periodic visual inspections will be performed at specified intervals to examine components for signs of degradation. Visual inspection also will be performed to locate and evaluate leaks detected by the leak detection system. These leak detection and visual inspection programs are considered to provide the level of monitoring necessary for the safety of the plant to detect any significant breach in the structural integrity of the pressure boundary. However, periodic volumetric examinations will be performed on intermediate system dissimilar metal welds and adjoining base metal to determine if (1) flaws are propagating or (2) other degradation is occurring such as to impair the structural integrity of the system.

The design and arrangement of piping and components will be such as to allow access for the specified inspections.

We consider that the requirements specified herein provide the requisite degree of safety for assuring the structural integrity of CRBRP systems and components. This inservice inspection program is to detect degradation processes and the onset of failure mechanisms well in advance of the time that a serious breach of the coolant boundary could occur.*

Steam/water components of ASME Class 2 or 3 will be inspected in accordance with the requirements of ASME Section XI, Division 1.

Requirements

Requirements for examinations to be performed during the construction and erection phase are included in Table QT20.66-1. Principal features of the CRBRP program are as follows:

1. Construction Phase - Examinations and Tests

   Welds and adjoining base material in the coolant boundary of components of the primary and intermediate heat transport systems, including the reactor vessel, the reactor vessel closure head, piping, pump tanks, IHX exterior, expansion tanks, and valves, will be radiographically examined in two directions (one normal to the surface of the weld and the second taken along the line of fusion between one side of the weld and base material) and also will be inspected by the liquid penetrant or magnetic particle method. Inspection of the reactor internals will also include radiographic examinations as well as liquid penetrant examinations. In addition, all welds will be ultrasonically examined to the extent practical by use of techniques that yields results similar to those that would be expected should such examinations be required after the plant is placed in service.** It is planned to perform these ultrasonic examinations either in the fabrication shop where the component is made or at the construction site shortly after erection. The purpose of these ultrasonic examinations is to

*For complete discussion of degradation processes, stress levels, and likelihood of postulated failure, see Reference 2 of PSAR Section 1.6.

**The ultrasonic inspection method used will reflect the latest current practical techniques. However, since ultrasonic inspection of austenitic stainless steel welds is highly developmental, particularly at high temperature, the techniques used at a later date are likely to be significantly different. A meaningful comparison between the initial and later inspections will require an extrapolation factor to account for sonic attenuation differences in the metal if the temperature of the later examination differs from the baseline examinations.

obtain records for comparison should future inservice examinations (over and above those examinations described in Sections 2 and 3 below) be necessary. The results of radiographic examination will be the principal basis of acceptance of the welds; indications found that are classed as unacceptable by the governing code will be repaired.

Other flaws that are detected and determined to be nongeometric will be characterized to the extent determinable by the reference technique, sufficient to enable appropriate evaluations by use of fracture mechanics methods. Welds in components of other systems will be examined in accordance with requirements of the governing (ASME) construction code.

2. Preservice Examination Program

All coolant boundary welds will be examined visually at room temperature. The coolant boundary welds will also be inspected at 400°F with techniques that are the same as those to be used in the inservice inspection program. In addition, dissimilar metal welds in the intermediate system will be examined volumetrically at room temperature and at 400°F by use of ultrasonic techniques (or other proven volumetric examination techniques presently being developed for this program) that are the same as those to be used in the inservice inspection program. The dissimilar metal welds in the IHTS piping are also among the highest stressed in that system. The extent of examination will include the circumferential welds (heat affected zone) plus the adjoining one-foot section of longitudinal pipe welds.

System leak tests will also be performed prior to system fill, in accordance with ASME Section XI, Division 3, article IMA-5210. The pressurizing medium will include helium as a constituent.

3. Inservice Examination Phase

The primary heat transport system and the portion of the intermediate heat transport system that is inside the containment building will be monitored continuously for leakage of sodium. Diverse redundant leak detection capability will be provided by aerosol-type leak detectors, radiation monitors, smoke detectors, continuity-type detectors, and level sensors. The detection capability of the leak detectors is described in Section 7.5.5 of the PSAR.

Leak detection devices will also be provided for all ASME Code Class 2 and 3 liquid metal systems. Periodic inspections include visual examinations of welds throughout the primary heat transport system, the intermediate heat transport system, and other ASME Code Class 2 and 3 systems.

A volumetric examination will be performed on the dissimilar metal welds in the intermediate heat transport system. Other welds will be visually inspected in accordance with the ASME Code Section XI, Division 3.
The frequency of examination of these welds will be in accordance with ASME Code Section XI, Division 3.

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TABLE Q120.66-1

CRBRP SUMMARY OF NONDESTRUCTIVE EXAMINATIONS OF WELDS
CONSTRUCTION, PRESERVICE, AND INSERVICE PROGRAMS

Examination Categories	Construction*	Preservice	Inservice
	(All Welds)	(All Coolant Boundary Welds)	(Essentially All Coolant Boundary Welds)
Reactor Vessel and Nozzle Welds (Similar Metal Welds)	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT (2) and LT	Remote Visual, Room Temperature and also @ 400°F	Remote Visual @ 400°F; Continuous Monitoring
Reactor Vessel (Dissimilar Metal Weld)	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT (2) and LT	Remote Visual, Room Temperature and also @ 400°F	Remote (4) Visual @ 400°F; Continuous Monitoring
Reactor Internals	VT+2Directional RT+ PT; Also UT (2)	CRBRP is in the course of evaluating the ASME Section XI	Division 3 requirements and will develop CRBRP requirements based upon the results of this evaluation.
Closure Head	VT+RT+PT/MT per ASME-III Class 1; Also UT	Continuous Monitoring	Continuous Monitoring
Reactor Coolant Piping Outside of Guard Vessels	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual Room Temperature and also @ 400°F; and LT	Direct or Remote Visual 400°F; Continuous Monitoring
Reactor Coolant Piping Inside Guard Vessels	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring

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TABLE Q120.66-1 (Cont'd.)

CRBRP SUMMARY OF NONDESTRUCTIVE EXAMINATIONS OF WELDS
CONSTRUCTION, PRESERVICE, AND INSERVICE PROGRAMS

Examination Categories	Construction*	Preservice	Inservice
	(All Welds)	(All Coolant Boundary Welds)	(Essentially All Coolant Boundary Welds)
Primary Pump Tank	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT (2)	Remote or Direct Visual Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F, + Continuous Monitoring
Check Valve	VT+2Directional RT+ PT Per ASME-III Class 1: Also UT (2)	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F, + Continuous Monitoring
Balance of Class 1 Components	VT+2Directional RT+ PT Per ASME-III Class 1: Also UT	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual + Continuous Monitoring
IHX Shell	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT (2)	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F, + Continuous Monitoring
IHX Tubing & Tube-to Tubesheet Welds	UT+ECT Per ASME-III Class 1	LT per IMC-2100 (3)	Continuous Monitoring (3) per IMB-5300
Class 2 Guard Vessels	VT+RT Per ASME-III Class 2	Remote or Direct Visual, Room Temperature and PHTS @ 400°F	Remote or Direct Visual PHTS @ 400°F

TABLE Q120.66-1 (Cont'd.)

CRBRP SUMMARY OF NONDESTRUCTIVE EXAMINATIONS OF WELDS
CONSTRUCTION, PRESERVICE, AND INSERVICE PROGRAMS

Examination Categories	Construction*	Preservice	Inservice
	(All Welds)	(All Coolant Boundary Welds)	(Essentially All Coolant Boundary Welds)
Intermediate Heat Transport System Piping, Except Dissimilar Metal Welds	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring
Intermediate Heat Transport System Piping Dissimilar Metal Welds	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Direct Visual and UT @ Room Temperature; Direct Visual @ 400°F; and LT	Direct Visual and Continuous Monitoring @ 400°Ft; UT at Room Temperature
IHTS Pump Tank	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring
IHTS Expansion Tank	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring
IHTS Valves, Flow-meter, Etc.	VT+2Directional RT+ PT Per ASME-III Class 1 + ASME Code Case 1594; Also UT	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring

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TABLE Q120.66-1 (Cont'd.)

CRBRP SUMMARY OF NONDESTRUCTIVE EXAMINATIONS OF WELDS
CONSTRUCTION, PRESERVICE, AND INSERVICE PROGRAMS

Examination Categories	Construction*	Preservice	Inservice
	(All Welds)	(All Coolant Boundary Welds)	(Essentially All Coolant Boundary Welds)
Balance of Class 2 Liquid Metal Components	VT+RT Per ASME-III Class 2	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT	Remote or Direct Visual @ 400°F + Continuous Monitoring
Steam Generator-Evaporator and Superheaters	VT+2Directional RT+ PT Per ASME-III Class 1 + Code Case 1594; Also UT or ECT tubing	Remote or Direct Visual, Room Temperature and also @ 400°F; and LT and Volumetric per IMC-2100	Waterside of Tubing & Tube-Tube Sheet Welds-UT or ECT; Sodium (Shellwelds) Visual + Continuous Monitoring
Class 3 Liquid Metal Components	VT+RT, MT, or PT per ASME-III Class 3	Visual @ 400°F; and LT	Visual @ 400°F; + Continuous Monitoring

*VT = Direct Visual Examination
RT = Radiographic Examination
PT = Penetrant Examination
ECT = Eddy Current Examination
UT = Ultrasonic Examination
MT = Magnetic Particle Examination
LT = Leak Test (Per IMA-5000 ASME Section XI, Division 3)

(1) Circumferential weld joints including the adjoining one foot sections of longitudinal welds.
(2) Base material prior to construction.
(3) Reference Response to NRC Question 120.67
(4) Exception to ASME Code - See Appendix G
(5) Nondestructive examinations during construction will be done at room temperature.

Question 120.67 (5.3.2.3.2)

Question 120.19, associated with the in-service inspection program for the intermediate heat transfer ststem, was not answered.

It is our position that sufficient design accessibility should be provided in at least one intermediate heat exchanger (IHX) to conduct periodic in-service examination of the IHX tubing. The intermediate heat exchanger is an essentially prototypical design that will be subjected to unique plant-operating conditions and transients. In=service examination of a representative sample of the intermediate heat exchanger tubing is necessary to assure the continuing structural integrity of the radioactive sodium-intermediate sodium boundary to preclude the possibility of gradual tube degradation proceeding undetected to an unacceptable stage and to reliably define potential long-term degradation mechanisms that may result from plant operating conditions. The initial sampling program and examination frequency should be based on results from your IHX development program subject to modification depending on results from accumulated in-service examination experience.

Response:

I. Background

The CRBRP - IHX design has benefitted from both the successes experienced and also the failures in sodium-to-sodium heat exchange operation. A case of a failed IHX tube in the Hallam plant was due to a poor design and a lack of vibration testing. The tube failure was due to flow-induced vibration at a point where the tube passed through a baffle plate and was subjected to a locally high cross-flow of four times nominal cross-flow. This failure was rather rapid and probably would not have been avoided by a periodic inspection program. The failure may have been avoided if either vibration analysis or vibration testing of the unit had been performed. Even in this instance, the event which revealed the design deficiency was local and not massive.

The CRBRP plant operation is designed so that the pressure on the intermediate side of the unit exceeds the pressure on the primary side during all steady-state conditions anticipated. Tube leaks, if any, would result in an increase in primary sodium inventory and a decrease in intermediate sodium inventory. Radioactive sodium escape is, thereby, precluded.

II. CRBRP - IHX

Building upon industry experience in designing sodium-to-sodium IHXs, the CRBRP unit and specifically the tubing is not expected to suffer from material degradation to unacceptable levels by reason of the following design considerations:

A. Vibration Testing

A full-scale model of a 30° sector of the IHX unit has been tested from zero to 120% of full flow in water at elevated temperature. The model includes all distribution devices, tie rods, baffle plates, etc. Results of this test demonstrate that the vibration of IHX tubing will not be a mechanism for failure or gradual degradation through fretting, impact, etc.

B. Vibration Analysis

A vibration analysis was applied to the IHX bundle including tubing. The results of this analysis concur with and support the finding of the vibration test in that tube vibration will not be a problem in the CRBRP - IHX.

C. Flow Design

Flow distribution devices have been designed and scale models tested to ensure an even flow distribution throughout the tube bundle. Although these devices have been incorporated into the IHX design to minimize stresses due to uneven fluid flow and heat transfer, they also assure that there will be no local areas of high cross flow which caused a tube failure in the Hallam IHX.

D. Stress Analysis

The IHX tubing has been analyzed in accordance with the following criteria:

1. ASME Code Section III
2. Code Cases 1592, 1593, 1594, 1595 & 1596
3. RDT Standard E15-2NBT
4. RDT Standard F9-4T

Stresses due to all plant conditions, transients, etc., are below the limits set by the above criteria. Potential buckling problems have been eliminated by employing a floating lower tubesheet and expansion bellows to allow differential thermal expansion between the shell downcomer, etc., and the tubes. The stress analysis of the tubes has shown no mechanism for gradual degradation of the tubes to an unacceptable level.

E. Corrosion

Curves defining corrosion rates in mils/year/PPM O_2 versus temperature for 304 & 316 stainless steels have been supplied to the IHX designer. These rates are for sodium flowing at velocities encompassing CRBRP flow rates. The corrosion allowance was included in sizing the IHX tubing to ensure acceptable wall thickness at end of life.

F. Radiation

Material degradation due to radiation has been investigated for the IHX. The total fluence expected over the plant life at the IHX is below the level which would affect the material. Therefore, no degradation of the IHX tubing is expected due to radiation.

G. Material

The material used for the IHX tubes is 304 stainless steel. This material has been used in several other sodium-to-sodium heat exchangers and has been found acceptable. The tubing used in the CRBRP - IHX conforms to requirements of the following criteria:

1. ASME Code Section III
2. ASME Code Section II, SA213
3. RDT E15-2NBT
4. RDT M3-2T

In addition to these material requirements, supplemental requirements have been included in the material order.
These include additional chemical content limitations and cleanliness requirements. Due to long industry experience and the quality of the material purchased, it is not expected that degradation of the tubing will occur.

III. Possible Tube Surveillance

Access to the tubesheets and inside of the tubes will be provided for IHX tube leak location and plugging. Tools will be designed which will enable instruments to be inserted into an individual tube during IHX maintenance in the event a leak occurs. Access to the O.D. of the tubes is clearly impossible since the bundle is surrounded by the shell and shroud and since the tube spacing denies access to an interior tube.

If the IHX must be opened for tube plugging, great care must be taken to avoid contaminating the unit due to inleakage of air, water vapor or other deleterious matter. The IHX will be drained, but all surfaces will remain sodium wetted. Contact of a contaminant with sodium may lead to caustics or corrosion products which could be detrimental to the IHX.

The possibility of this contamination occuring will be kept to an absolute minimum by employing inert gas purging, temporary seals, cleaning, strict maintenance procedures, etc. Nevertheless, it is felt that the opening of the IHX unit, for any reason, should be kept to an absolute minimum in order to avoid possible contamination. This risk is acceptable only to a leaking or defective unit. Exposing a sound unit to this risk to search for unknown mechanisms is unacceptable to the project as detrimental to maximizing overall safety.

IV. Conclusion

It is strongly asserted that the IHX tubing has been procured, designed, analyzed, and tested to such a degree that we feel certain that there will be no material degradation of the tubing to unacceptable levels over the life of the CRBRP Plant. It is also believed that unnecessary opening of the IHX for tube inspection may introduce the harmful contaminants into the IHX unit. This risk of degrading the IHX is judged to far outweigh the benefits of inservice tube inspection of an operating nonleaking unit.

Question 120.68

The description of your materials surveillance program is not adequate to conclude that all potential degradation processes which may occur inservice will be monitored. We require justification for the use of materials in the lower and upper internal structure receiving irradiation fluence from 10^{17} to 10^{21} n/cm^2. In addition to your proposal of only sub-size tensile specimens withdrawn at one-quarter plant life intervals (plus one contingency capsule) from areas receiving a fluence > 1 x 10^{21} n/cm^2 we require additional specimens withdrawn at the same interval to monitor other degradation processes that may occur in the primary heat transport loop. The properties to be monitored include strength, toughness mass transport related changes and phase transformations of both base and weld metals from components and structures located in the lower and upper core internal regions, reactor vessel and primary heat transport piping. Specimens from bimetallic welds should also be included in your proposed surveillance program. Provide the description of a revised materials surveillance program which reflects the requirements of this position.

Response

A comprehensive material surveillance program for the CRBRP Reactor and Primary Heat Transport Piping materials is being developed and detailed documentation on this program will be provided to NRC as it is developed. The program will include in-reactor surveillance of material coupons to study neutron irradiation effects and examination of components to determine mass transport effects as well as laboratory test programs on base metal-weld metal combinations to monitor thermal aging effects at prototypic plant temperature and steady state stress conditions. These and other specific concerns raised in the above question are addressed in the following paragraphs:

Neutron Irradiation Effects: End-of-life residual ductility concepts are used in designing the CRBRP Reactor and Primary Heat Transport System components. Available fast neutron irradiation data provided in References Q120.68-1, -2 and -3 show that no significant loss in tensile ductility in austenitic stainless steels and nickel base Alloy 718 occurs until total neutron fluences exceed 10^{21} n/cm^2 (E > 0.0 MeV). This is illustrated in the attached Figures Q120.68-1, -2, -3, and -4. Hence, the use of in-reactor surveillance specimens for components in the upper and lower reactor internals receiving neutron fluences less than 10^{21} n/cm^2 are not considered necessary. Similarly, in-reactor surveillance of the Primary Heat Transport piping and components is not necessary as the neutron fluence corresponding to plant lifetime is significantly lower than 1 x 10^{21} n/cm^2. However, surveillance of the reactor vessel midband base metal and weldments will be done even though the fluence is less than 10^{21} n/cm.^{2}

In-reactor surveillance of CRBRP ferritic steels is not considered to be necessary since these steels are located in regions of the plant where the total fluence is less than 1 x 10^{17} n/cm^2 which is the threshold limit given in 10CFR50, Appendix H for ferritic steels in the reactor belt-line region.

Fracture Toughness: Fracture toughness surveillance of the ferritic steels in the CRBRP heat transport system is not included in the surveillance program because of the low fluence levels. The fluence levels at both the head and the vessel transition section weld regions are below 10^{17} n/cm^2 (E_{max}>1 MeV), and there are no ferritic materials in the reactor vessel below the transition section. Therefore, no fracture toughness surveillance is required by 10CFR50, Appendix H. 33

Fracture toughness surveillance specimens of the austenitic stainless steels and nickel base alloys have not been included in the program. A test program is being developed with the goal of demonstrating acceptable fracture toughness behavior of irradiated austenitic stainless steel base metals, weldments and nickel base structural materials. Depending on the results obtained from this program, CRBRP surveillance of fracture toughness will be done for those components which are exposed to fluence levels causing significant degradation.

Thermal Aging Effects: Assessment of the available data regarding microstructure and property changes as related to service temperature, time and stress indicates that the CRBRP primary coolant boundary materials have a high likelihood of acceptable performance and thus in plant surveillance is not considered necessary. This position is based on the data from on-going base technology programs concerned with providing a firm basis for appraising possible degradation that could occur to LMFBR materials. To augment and broaden the base of the existing technology efforts, typical materials and weldments utilized in the CRBRP primary heat transport system will be provided for inclusion in this on-going effort.

Included in the program will be hot leg and cold leg piping welds together with their respective base metals and the reactor vessel bi-metallic weld (SA-508 Class 2 to SB 168, Inconel 600). Mechanical properties and microstructural examinations will be performed on both base metals and weld metals after long-term exposure to temperature and steady-state stresses which simulate operating conditions. Test specimens will be fabricated by vendors providing CRBRP component hardware. Thermal aging will be performed in air at prototypic plant temperature and steady-state stress for pre-selected periods of time. Mechanical property evaluations will be performed at the operating temperature in the as-fabricated condition and after each period of thermal aging, i.e., after 1, 2, 3, 6,...etc., years. The properties to be evaluated will be selected after a comprehensive review of available data on material degradation mechanisms and could include microstructure, tensile properties, creep properties, fatigue properties, crack propagation and notch toughness. A limited evaluation will also be performed at the plant refueling temperature of 400^{o}F.

Mass Transport Related Effects: Allowances for mass transport related changes to the reactor and Primary Heat Transport System piping and components have been provided for by imposing appropriate limits on raw material composition, e.g. carbon content and in the design of components. These allowances are based on experimental data obtained from sodium test loops and are considered representative of mass transfer phenomena anticipated in the CRBRP. In addition, data from on-going test programs at ARD and elsewhere to examine the facts of mass transport on mechanical properties of LMFBR materials will be utilized to assess potential impact on CRBRP plant performance.

In-plant surveillance of mass transport related changes will be performed on selected components, with the IVTM port plug being the prime component. The IVTM port plug is fabricated from 316SS and is exposed to flowing sodium and temperatures representative of the reactor vessel outlet. Changes to this component due to mass transport can be considered to be representative of those in most of the reactor vessel and Primary Heat Transport System components.

The IVTM port plug is available for examination when it is removed during each refueling operation. Mass-transfer evaluations will be performed after exposure times which will cause measurable changes in surface and bulk chemistry and microstructure or, as a minimum, at times corresponding to the withdrawal schedule identified in the surveillance program. The results from these evaluations will be compared with the data used to establish the design allowances and with the data from the ongoing programs to assess any degradation in properties.

Mass transport effects on Alloy 718 will be obtained from the control rod drivelines which are scheduled for removal after 10 years. Examinations of these components will be included in the material surveillance program.

Microstructure: The surveillance program will include microstructural examinations to determine phase transformations. These examinations will be made for the aging and mass transport programs noted above on unstressed regions, such as the grip regions, of the irradiation surveillance specimens.

References:

Q120.68-1 T. T. Claudson, Semi-Annual Progress Report-Irradiation Effects on Reactor Structural Materials-March 1975 to July 1975, HEDL-TME 75-95, December 1975, pp HEDL 102-HEDL 111.

Q120.68-2 J. M. Steichen and A. L. Ward, Effect of Strain Rate on the Tensile Properties of Irradiated Inconel 718, HEDL-SA-1059, January 1976.

Q120.68-3 A. L. Ward, Austenitic Stainless Steel Weld Materials-A Data Compilation and Review, HEDL-TME 74-25, May 1974, pp. 22-36.

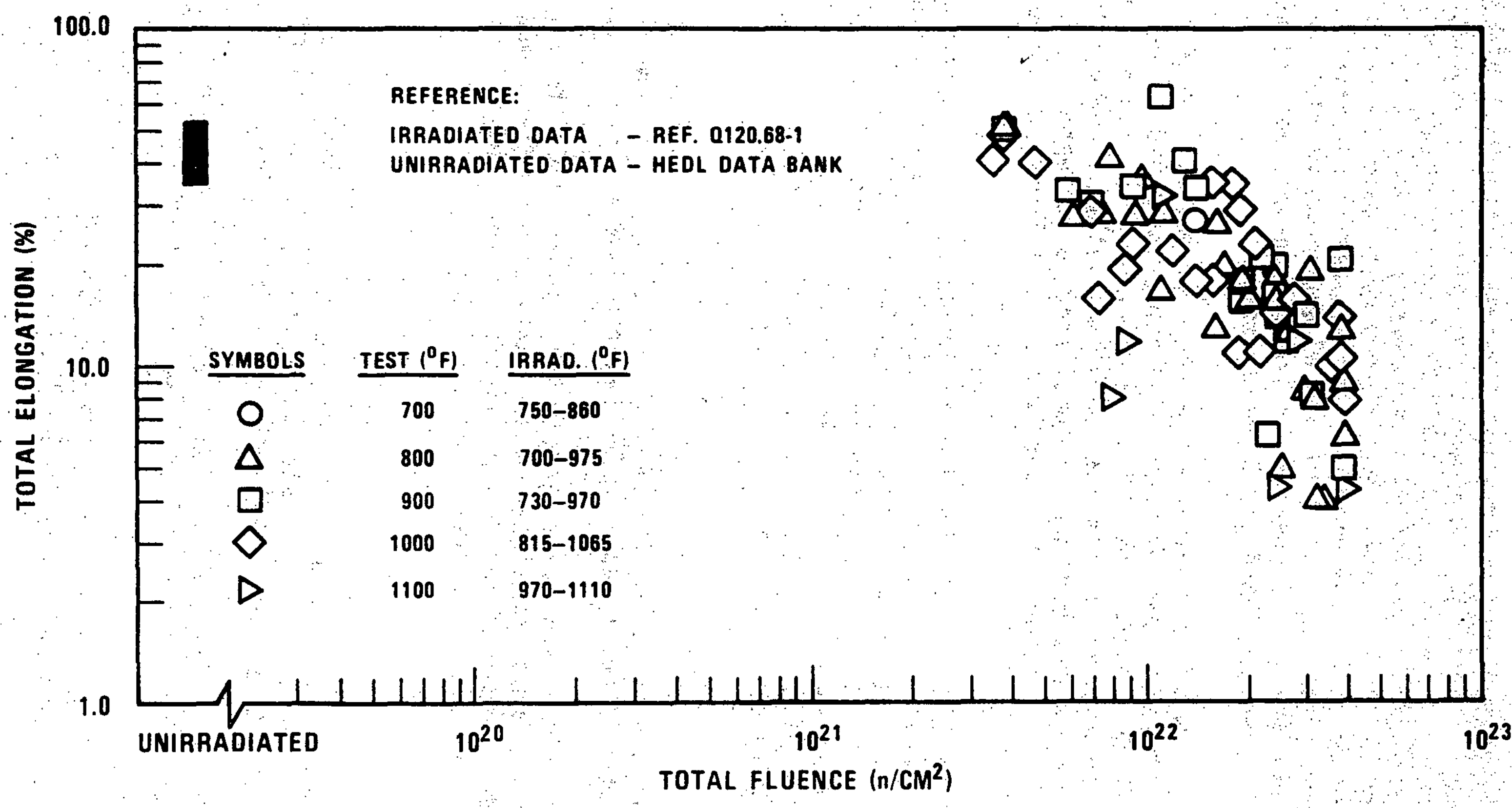

FIGURE Q120.68-1 RESIDUAL DUCTILITY OF ANNEALED TYPE 316 STAINLESS STEEL AS A FUNCTION OF TOTAL FLUENCE.

7683-223

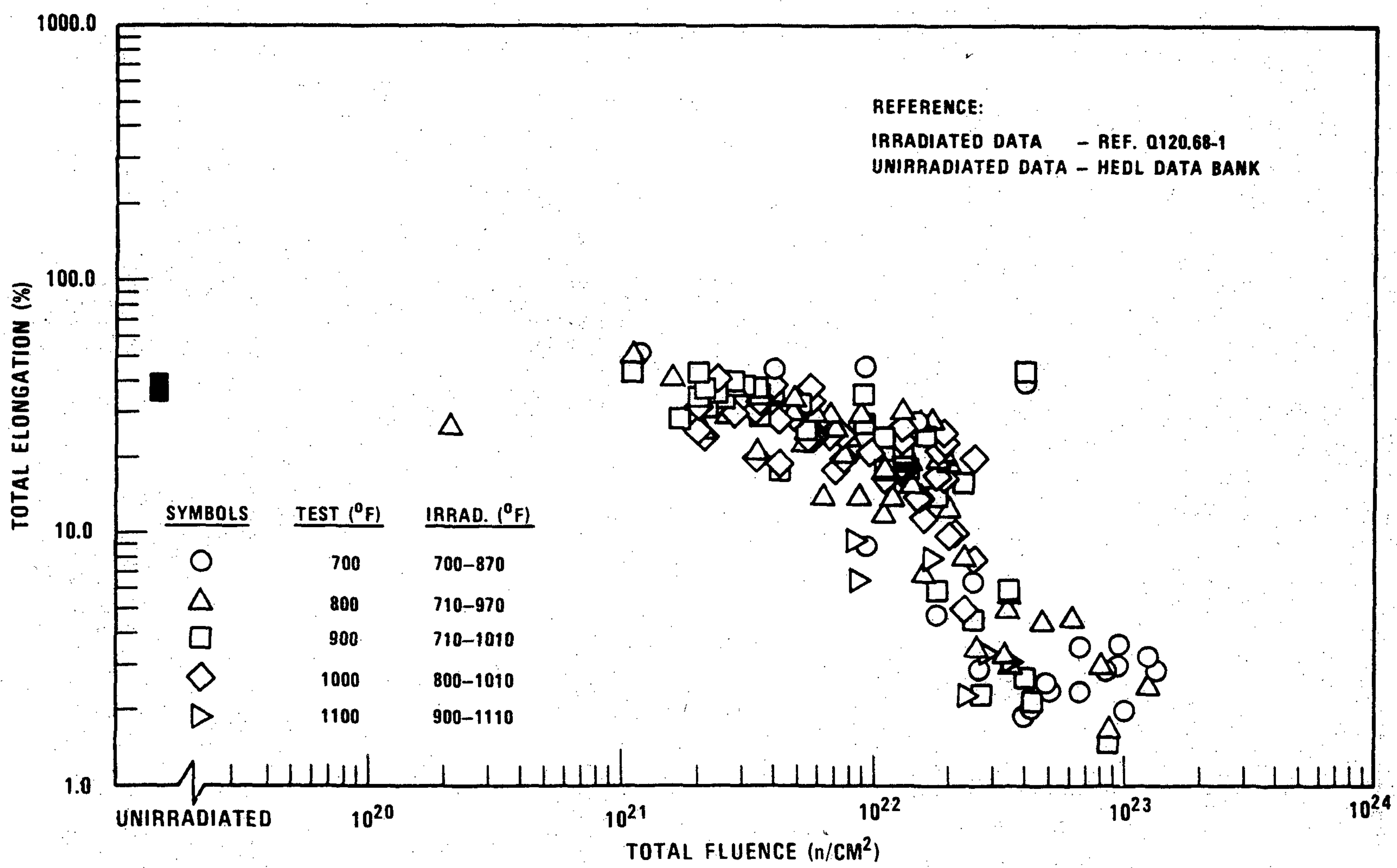

FIGURE Q120.68-2 RESIDUAL DUCTILITY OF ANNEALED TYPE 304 STAINLESS STEEL AS A FUNCTION OF TOTAL FLUENCE.

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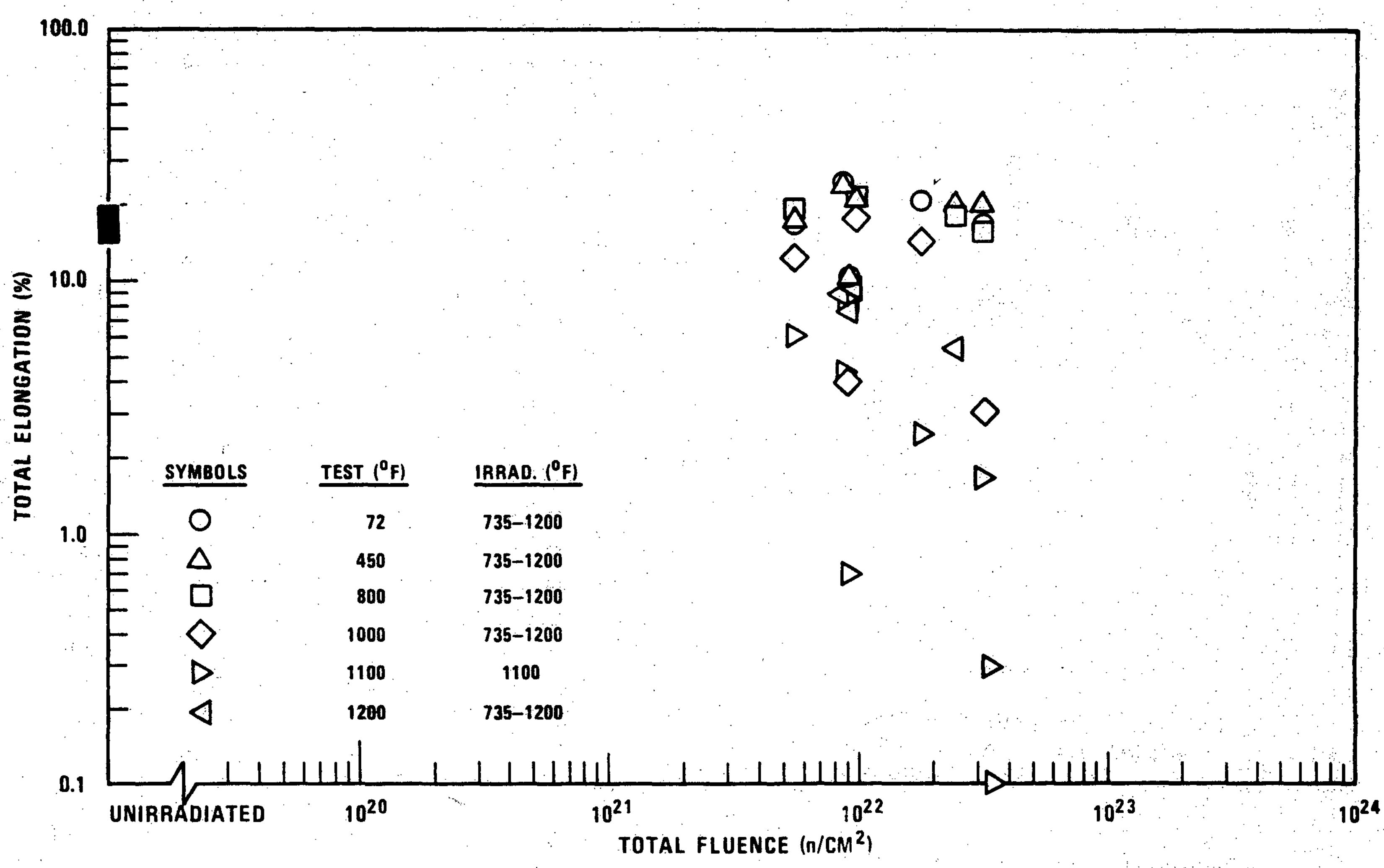

FIGURE Q120.68-3 RESIDUAL DUCTILITY OF ALLOY 718 AS A FUNCTION OF TOTAL FLUENCE (REF. Q120.68-2)

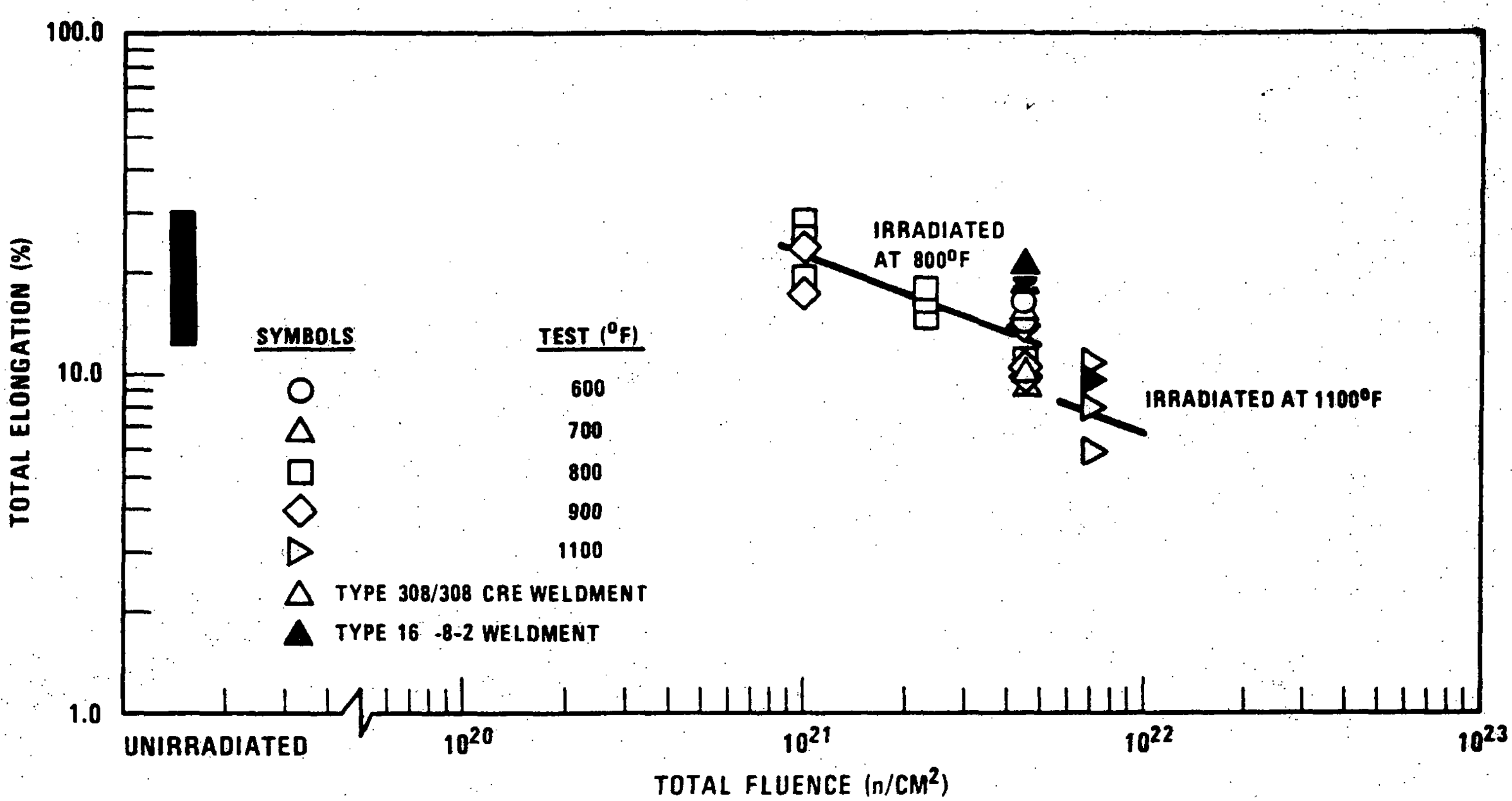

FIGURE Q120.68-4 RESIDUAL DUCTILITY OF STAINLESS STEEL WELDMENTS AS A FUNCTION OF FLUENCE (REF. Q120.68-3)

Question 120.69 (5.5)

The PSAR description of the steam generator is not complete. We require a detailed description for evaluation. The description should include: design requirements; material specification; description of welding and heat treating procedures; methods used to ensure preservice material integrity; mechanical properties of basic materials; mechanical properties of welds, including bi-metallic welds; mechanical properties of anticipated back-up materials, mechanical properties anticipated after projected inservice degradation, such as mass transfer of interstitial elements from the nonstabilized 2-1/4 Cr-1 Mo steel to minimum levels.

Response:

Section 5.5.2.3.4 has been modified in response to this question.

Question 120.70 (5.2.4.5)

In response to question 120.29 you described the resolution capability of the TV camera and monitor inspection system. We require that design accessibility and transporter(s) mobility be provided such that the in-service inspection program requirements for detailed visual examination of Class 1 and Class 2 welds can be performed at sensitivities sufficient to resolve weld fissure(s) and/or to verify the crack dimension associated with 100 gm/hour sodium leak. The definition of the visual examination method should not be interpreted as a general visual survey.

Response:

The design accessibility and transporter(s) mobility has been described in the previous response to question 120.29.

The remote visual inspection system as proposed is the current state-of-the-art for closed circuit television (CCTV). The sensitivity of the CCTV is specified in PSAR Section 5.2.4.5. As noted in that Section, the resolution is better than that required by the ASME Code, Section XI, paragraph IWA-2210(b). There are presently no requirements of the ASME code or the CRBRP project which require the resolution of a weld anomaly associated with 100 gm/hour sodium leak. It is expected that the proposed systems resolution is sufficient to detect sodium leakage of 100 gm/hour. However, it is expected that the capability to resolve base or weld metal anomalies associated with a 100 gm/hour sodium leak is beyond the present state-of-the-art.

Question 120.71

The responses to Questions 120.31, and 120.34 are not sufficient to conclude that an adequate and reliable sodium to gas leakage detection system will be incorporated in the CRBRP. Further test work will be required to verify that the leakage detection thresholds and signal response times of these systems are reproducible and predictable. We required design verification tests conducted in a representative mock-up that simulates portions of the primary and intermediate heat transport systems. Mock-up design verification tests of leakage detection capability in guard vessels will also be required. The simulation should include sniffing tubes for aerosol detectors that have the same length, complexity and thermal gradients of those proposed for CRBRP.

The staff requires that the following areas be addressed in the development program:

(1) Adequacy of coverage, redundancy, diversity and range of the sensor units.

(2) In-situ calibration and operability tests. Develop a sensor qualification program.

(3) "Plate-out" of aerosol in long sniffer tubes with thermal gradients.

(4) Diversity, redundancy, and sensitivity of sensors over the temperature range of 400^{o}F to 1100^{o}F and in inerted and/or ambient atmospheres.

(5) Sensitivity, diversity, and redundancy of sensors monitoring the IHTS within the RCB.

We require that the leak detection system installed in CRBRP be capable of in-place operational verification and calibration. We also require that leak detection system diversity and redundancy be maintained at all plant conditions. Multiple sensors at the end of one sniffer tube are not considered diverse and redundant.

Response:

The Project and the NRC staff attended a meeting, June 18, 1976, to discuss the summary of design and development status of the Liquid Metal to Gas Leak Detection System for CRBRP. At that meeting, the Project provided NRC a complete discussion of the Liquid Metal to Gas Leak Detection System and its supporting development program. The NRC staff concerns were individually addressed by the Project, and supporting documentation was sent to the staff for their evaluation. The pertinent information has been incorporated in PSAR Sections 1.5 and 7.5. A brief summary is provided below.

An adequate and reliable Sodium to Gas Leak Detection System will be provided for the CRBRP. The system utilizes very sensitive leak detectors located at strategic locations in the plant and these are backed up by other types of detectors which are also very sensitive to liquid metal leaks.

Further test work is planned in the form of Verification Tests (at LMEC) and Long-Term Environmental Performance Test (at EBR-II). These tests will verify that leak detectors performance and response times are reproducible, predictable, and within sensitivity requirements. The tests will be conducted on realistic representative mock-ups of both primary and intermediate piping.

Mock-up of guard vessels is not necessary since tests have been conducted by monitoring a cell which has a volume equivalent to that of guard vessels. These tests have shown that aerosol diffusion/convection takes place throughout the cell with no preferential path, enabling detection by aerosol detectors located at ceilings or floor of cell at about the same time.

Specific NRC concerns are addressed below using the same (1) through (5) numerical identification as used in the question. (For more detail, see revised PSAR Sections 1.5 and 7.5.5.)

(1) Adequacy of coverage, redundancy, diversity and range of sensors are addressed in the test program to be conducted at LMEC and EBRII during FY 1977.

(2) In-situ calibration and operability tests will be developed to meet the intent of applicable sections of Regulatory Guide 1.45.

(3) The "plate-out" of aerosols in long sniffing tubes, having typical LMFBR Thermal gradients, has been negligible in past tests. This will be confirmed during the verification test series.

(4) Past and proposed test programs have addressed the sensitivity of redundant and diverse sensors operating in both air and inerted environments over the temperature range of 400°F-1000°F. Revised PSAR Section 7.5.5 gives details of the diversity provided in the reactor cavity area which was identified by NRC staff as an area of special concern.

(5) The sensors monitoring the IHTS within the RCB are similar with respect to sensitivity, diversity and redundancy to that provided for the Reactor Vessel and PHTS.

The system for CRBRP will be designed to provide redundancy/diversity for all required plant conditions. It will also be designed to allow for in-place operational verification and calibration.

Multiple sensors at the end of one sniffer tube are neither considered diverse nor redundant, and the leak detection system does not plan to take credit for diversity/redundancy should this approach be used. At this time, the Project has no plans to put multiple sensors at the end of one sniffer tube, each sensor will have its own sniffer tube.

Question 120.72

The PSAR discusses the detection of gross leakage through the intermediate heat exchanger to the primary sodium system. However, undetected leakage may occur, and under certain conditions, a higher cover pressure on the primary than on the intermediate coolant may exist. We require a leakage detection system to identify and quantify leakage from the primary to the intermediate coolant system.

Response:

Section 7.5.5.2 has been expanded to address the concerns expressed with the potential for a reverse pressure differential and the need for a primary to intermediate leak detection method.

Question 120.73 (7.5-30, 15.3.3.3.1)

Page 7.5-30 indicates that a water to sodium leak of 10^{-4} lb/sec in the steam generator will be detected and identified by a hydrogen rate of rise detector in the IHTS. Figure 15.3.3.3-1 postulates the development of a steam leak with size increase from $4x\ 10^{-5}$ to 3×10^{-2} in 30 seconds. Figure 7.5-5 presents curves for hydrogen concentration versus time for various water leak sizes and indicates a relatively slow hydrogen rise for a steam leak of 10^{-4} lb/sec during the first 2000 seconds. The above data is inconsistent for conclusive leak detection evaluation.

Provide an analysis to show that a 10^{-4} lb/sec steam to sodium leak will be detected and distinguished from background level in the IHTS. In your analysis consider the normal background level plus an increased level due to a potential undetected leakage from the primary system. Consider in your analysis variations in cold trap operations on background level and the variation of background level with changes in reactor power level and core life.

Response:

As indicated on page 7.5-30, the range of leak sizes detectable by the rate of rise method is from 10-5 to 10^{-4} lb/sec. Figure 7.5-5 indicates that a 10^{-5} lb/sec leak will be detected in approximately 1/2 hour. Figure 15.3.3.3-1 addresses the development of a steam leak with size increase from 4×10^{-5} to 3×10^{-2} lb/sec in approximately 2 hours as indicated in steps (4) and (5) of the figure. Therefore, the leak would be detected with sufficient time remaining to allow for operator corrective actions.

As shown in Figure 7.5-4 leak rates on the order of 10^{-4} lb/sec will be detected on the first pass rather than the rate of rise method thereby allowing ini-initiation of rapid loop shutdown. If leaks should enlarge suddenly, and the leak detection system cannot detect the leak in time to implement corrective action, the SWPRS rupture discs provide the required protection.

The sensitivity established for the in-sodium hydrogen detector, 3 ppb in a background of 100 ppb, corresponds to a leak rate of 10^{-4} lb/sec as indicated in Section 7.5.5.3.2. During normal steady state operation the cold trap will be operated at an appropriate flow rate and temperature to maintain the IHTS hydrogen background concentration below 100 ppb. After establishing the background concentration for a given operating mode, only minor variations in cold trap operation are anticipated which will still allow a resolution of 3 ppb change in the hydrogen background concentration during steady state operation (Section 7.5.5.3.2.)

Primary to intermediate sodium leakage will not occur during normal operation since the IHTS pressure is required to be a minimum of 10 psi above the PHTS pressure in the IHX (Section 7.5.5.2.1.). Only minor variations in hydrogen background level with changes in reactor power level and core life are expected. These changes would not appreciably affect the capability of the leak detectors to resolve a small leak signal.

Amend. 29
Oct. 1976

Question 120.74

Report WARD-D-0127 (Primary Pipe Integrity Status Report-December 1975) discusses lower bound critical crack lengths determined in model elbow tests and states that additional tests are planned to investigate the potential rupture area opening sizes at the plant operating temperature. We require that information and analysis developed from these tests be submitted for our review. Your analysis should include and justify maximum anticipated rupture areas (leak sizes) which can be developed in the primary and intermediate heat transport systems considering the system environment and design conditions.

Response:

The development test program discussed in Subsection 4.6.5 of the referenced report is presently in the planning stage.

Pertinent information and analysis developed from these tests will be submitted to NRC when available.