Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:093

SEP 2 1 1982

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

Dear Mr. Check:

MEETING SUMMARY FOR MEB/CRBRP SEPTEMBER 8 AND 9, 1982, MEETING

Enclosure 1 of this letter summarizes the resolution of items discussed between the Mechanical Engineering Branch and the CRBRP Project on September 8 and 9, 1982. Enclosure 1 contains a matrix indicating for each item, the resolution and/or additional information or meetings required. Item numbers in this matrix refer to a summary list of open items handed out by the NRC in the September 8 and 9, 1982, meeting (Enclosure 3). For items the Project considered resolved by information supplied at the meeting, Enclosure 2 formally submits the PSAR pages handed out during the meeting. These PSAR pages will also be incorporated into Amendment 71 of the PSAR, scheduled for September 24, 1982. Additional information is being provided as Attachments 1 through 8 of Enclosure 1 or will be supplied at a later date as indicated in the matrix.

To resolve certain more complex issues, the Project proposes three future meetings in addition to those already scheduled. These are meetings to discuss sodium-to-air leak detection during the week of September 27, 1982, inservice inspection during the week of October 4, 1982, and PHTS hot leg piping integrity during the week of October 18, 1982. Specific dates and agendas for these meetings will be discussed with your staff in the near future.

Three additional items discussed at the meeting were in regard to points raised by the ACRS. These items (numbered 72, 73, and 74) will be discussed at the October 25, 1982, meeting.

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Any questions regarding the information provided or future activities can be addressed to Mr. D. Robinson (FTS 626-6098) of the Project Office Oak Ridge staff.

Sincerely,

John R. Longenecker

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

Enclosures

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WORKING MEETING WITH MECHAN .AL ENGINEERING BRANCH

ENCLOSURE 1

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 1 | x | | | | Modified PSAR Table 3.2-2 provided |
| 2 | | | | x (week of 9/27) | Meeting on Na-air Leak Detection |
| 3 | | x (10/18/82) | | | Applicant to provide listing of non-safety related systems and principle codes used |
| 4 | | x (10/18/82) | | | |
| 5 | | | x | | NRC to identify specific drawings of concern |
| 6 | x | | | | Clarified PSAR Table 3.2-5 provided |
| 7 | x | | | | Clarified PSAR Table 3.2-5 provided |
| 8 | x | | | | Modified PSAR Table 3.2-5 provided |
| 9 | x | | | | Modified PSAR Table 3.2-5 provided |
| 10 | | | x | | NRC to confirm subject systems not required |
| 11 | x | | | | Modified PSAR Section 3.6 provided |
| 12 | x | | | | Modified PSAR Section 3.6 provided |
| 13 | x | | | | Modified PSAR Table B-1 provided |
| 14 | x | | | | Modified PSAR pg. B-25 discussed |
| 15 | x | | | | Modified PSAR Table B-1 discussed |
| 16 | | | x | | NRC to review PSAR Appendix H |
| 17 | | x (10/18/82) | | | Identified Refs. from PSAR Appendix A to be provided |
| 18 | | x (10/25/82) | | | Modified PSAR Section 3.7-A to be provided |

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 19 | | x (10/18/82) | | | Modified PSAR Section 3.7 to be provided |
| 20 | | x (10/25/82) | | | Modified PSAR Section 3.7 to be provided |
| 21 | | x (10/25/82) | | | Modified PSAR Section 3.7 to be provided |
| 22 | | x (10/25/82) | | | Modified PSAR Section 5.3 provided; examples to be discussed at Oct. 25 meeting |
| 23 | | x (10/25/82) | | | Modified PSAR Section 3.7 to be provided |
| 24 | x | | | | Modified PSAR Section 3.7.3.12 provided |
| 25 | x | | | | Modified PSAR Section 3.7.3.13 provided |
| 26 | | | x | | NRC to review updated PSAR Section 3.9.1 |
| 27 | | | x | | NRC to review updated PSAR Section 3.9.1 |
| 28 | | | x | | NRC to review updated PSAR Section 3.9.1 |
| 29 | | | x | | NRC to review updated PSAR Section 3.9.1 |
| 30 | x | | | | Modified PSAR pg. 3.9-1hd provided |
| 31 | | | x | | NRC to review Modified PSAR Section 3.9.1 |
| 32 | | x (10/18/82) | | | Modified PSAR Section 3.9.2 to be provided |

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 33 | x | | | | Resolved in Q/R CS 210.14 |
| 34 | | x (10/18/82) | | | Committment to be provided in PSAR Section 3.7 |
| 35 | | | x | | NRC to review PSAR Sections 5.5, 7.5, and 5.3; clarification provided in attachment 1 |
| 36 | x | | | | PSAR Table 4.2-47 discussed |
| 37 | x | | | | PSAR Table 4.2-43a discussed |
| 38 | | x (attachment 2) | | | Clarification provided in attachment 2 |
| 39 | | x (attachment 2) | | | Basis provided in attachment 2 |
| 40 | | x (attachment 2) | | | Guidance provided in attachment 2 |
| 41 | | | x | | NRC to review PSAR Section 4.2 |
| 42 | x | | | | PSAR Section 4.2 discussed |
| 43 | | | x | | NRC to review PSAR Section 4.2.2.1.1.2 |
| 44 | | | x | x (week of 10/4) | NRC to review PSAR Section 3.9.1; meeting on In-Service Inspection |
| 45 | | x (attachment 3) | | | Clarification provided in attachment 3 |
| 46 | x | | | | Modified PSAR Section 4.2.2.4.2 provided |

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 47 | x | | | | Modified PSAR Section 4.2.2.4.2.6 provided |
| 48 | | x (10/01/82) | | | Assurance for testing to be provided |
| 49 | | x (10/18/82) | | | Modified PSAR Table for Section 3.1 to be provided |
| 50 | | x | | | Comparison for "low temperature" components to be provided |
| 51 | | x (10/18/82) | | | Modified PSAR Table for Section 3.1 to be provided |
| 52 | x | | | | Reference to CC 1489 has been deleted |
| 53 | | | x | | NRC to review report ES-LDP-82-009 |
| 54 | | x (10/18/82) | | | Modified PSAR Section 5.1 to be provided |
| 55 | | | | x (week of 10/18) | NRC to write a detailed summary of concerns and a proposed program of confirmation; to be subject of Piping Integrity Meeting |
| 56 | | | | x (week of 10/18) | (Same as Item 55) |
| 57 | | | | x (10/25/82) | Report on application of screening rules included in report PVP-63, "Flow Induced Vibration of Cylindrical Structures;" examination of selected components on 10/25/82 meeting |
| 58 | | x (10/25/82) | x | | Listing of alternative criteria to be provided. NRC review RDT F9-5T and F9-4T |

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 59 | | x (10/25/82) | x | | Listing of areas where inelastic analysis used used will be provided; NRC review RDT F9-5T and F9-4T |
| 60 | | x | | x (10/25/82) | Representative example of elastic follow-up will be provided and discussed at 10/25/82 meeting |
| 61 | | x | x | | Combined with items 58 and 59 |
| 62 | | x (attachment 4) | | | Testing concerning notch effects and other considerations provided in attachment 4 |
| 63 | | | | x (week of 10/18) | Combined with items 55 and 56 |
| 64 | x | | | x (10/25/82) | Applicant committed to discuss impact of new Section III curves at 10/25/82 meeting |
| 65 | | x | x | | Combined with items 58, 59 and 61 |
| 66 | | x | | | Resolved for Reactor Vessel transition joint; plastic strain concentration concerns to be resolved with Item 59 |
| 67 | | x (attachment 5) | | | Project position provided in attachment 5 |
| 68 | | x 10/18/82) | | | Response to item to be provided |
| 69 | | x (10/25/82) | | x (10/25/82) | Project to respond to use of BTP MEB 3-1 instead of Appendix A to O'Leary Letter in 10/25/82 meeting |

| ITEM | RESOLVED Information identified in meeting (See Enclosure 2) | Additional Information to be provided (date) | NRC review required on current docket | Meeting on Details required (date) | Description and Comments |
|---|---|---|---|---|---|
| 70 | | x (10/25/82) | | x (10/25/82) | Modified PSAR Section 3.7-A to be provided; typical specification to be examined in 10/25/82 meeting |
| 71 | | x (attachment 6) | | | Clarification provided in attachment 6 |

Attachment 1

Clarification of Item #35

The CRBRP steam generators have been designed both to minimize the likelihood of leaks and to mitigate the consequences of leaks. Pertinent details of the steam generator design and analysis are given in PSAR Sections 5.5.3.1.5.1 (1) and 5.5.3.11.4.

Those sections have been updated recently in response to formal NRC questions. Other formal question responses were pertinent to this question, but did not change the referenced sections. Since these reponses were submitted as parts of several different transmittals, copies are attached.

Question CS281.8

Provide the design criteria and bases that demonstrate wastage allowance of the CRBR steam generator tubes, caused by sodium-water reaction products, is acceptable. The analysis should include Na water reaction temperature and other major variables in the small water leak situation.

Response

The steam generator tube wastage allowance and provisions to accommodate tube leaks are discussed in the revised PSAR Section 5.5.3.11.4. It should be noted that section 5.5.2.3.4 discusses the function of the tube sheet baffling as wastage baffles. This provides tube protection in the most likely location for leaks.

Question CS 281.9

Describe the sample and instrument readings and the frequency of measurements that will be performed to monitor the feed water purity and need for condensate cleanup system demineralizer resins and filter replacement. State the chemical limits and precaution to be taken to protect steam generator tubes against excessive corrosion and deposition. Also, provide the basis of establishing the chemistry limits.

Response:

PSAR Section 5.5.3.11.4 presents the feedwater and steam drum purity established to protect the steam generator tubes against excessive corrosion and deposition. PSAR Section 5.5.3.11.4 also adds additional information relative to monitoring and controls. The following major factors provided the basis for establishing the chemistry limits:

1. Because of the relatively low evaporator recirculation ratio in CRBRP, it was recognized early in the program that the CRBRP water chemistry limits would need to be similar to those limits which extensive experience in the fossil fired boiler and nuclear steam generator industries with once-through designs had shown to be required. Basically, this requires the use of all volitile treatment (AVT) consisting of a pH adjustment agent (typically amonium hydroxide) and an oxygen scavenging agent (typically hydrazine). The concentration of AVT agents is controlled in the feedwater to minimize corrosion in both the feedwater train and in the evaporator recirculation loop. Therefore, the then existing industry AVT chemistry requirements were established as the basis for CRBRP chemistry control.

2. These chemistry requirements were further refined to address the particular needs of CRBRP relative to materials, i.e., because of the 90/10 copper-nickel condensor, a 0.002 ppm copper concentration was specified. This low limit minimizes the potential for transport of copper to the evaporator tube internal surfaces where it would cause excessive tube corrosion.

3. The low recirculation ratio in the evaporators results in DNB in the 2-1/4 Cr-1 Mo evaporator tubes. This requires close control of the sodium ions to prevent stress corrosion cracking problems and close control of the chloride and sulfate ions to prevent "under deposit" corrosion. For example, the feedwater sodium ion concentration is maintained at 0.001 ppm maximum to achieve 0.006 ppm maximum in the recirculation loop. Similarly, the chloride and sulfate ions are maintained at low values in the feedwater to achieve a 0.015 ppm maximum for both species in the recirculation loop.

In order to meet the evaporator water chemistry requirements described above, requirements for condensate system demineralizer resin regeneration and/or replacement, continuous monitoring/recording and grab sampling of the Condensate Polishing effluent have been established as follows:

| Max. Allowable Impurities | Design Limit Operation Above 5% Power | Monitoring | Grab Sample |
|---|---|---|---|
| Total Suspended Solids | 16 ppb | None | Yes |
| Silica ($SiO_2$) | 5 ppb | Continuous | Yes |
| Iron (Fe) | 5 ppb | None | Yes |
| Copper (Cu) | <1.5 ppb | None | Yes |
| Sodium (Na) | 1 ppb | Continuous | Yes |
| Chloride (Cl) | 2.5 ppb | Continuous | Yes |
| Cation Conductivity at 77$^{o}$F | 0.2 mmho/cm | Continuous | Yes |

Question CS760.36

Concerning the potential sodium/water reaction, the steam generator design considers only a design basis leak consisting of a single tube, double-ended guillotine rupture of a steam tube followed by two additional single double-ended tube guillotine ruptures, spaced at 1.0 second intervals.

a. From the very closely packed CRBR steam generator tube arrangement, with one tube surrounded by six adjacent-tubes, if one steam tube was a double-ended rupture, the six adjacent tubes can be involved. Please discuss this case and include your analysis.

b. In the three tube rupture model, the failures of second and third tube follow at 1.0 second intervals. The effects of this assumption are essentially the same as for a single tube rupture model. Further substantiation as to why adjacent tubes can't rupture at the same time is needed.

c. What is the response to three simultaneous tube ruptures instead of three staggered ruptures?

d. The TRANSWRAP results in the PSAR show the initial pressure pulse fails to burst the rupture discs. The peak pressure in the IHX is 331 psia and the design pressure for the IHX tubes is 325 psig. If more than one tube ruptures at the beginning, can the initial pressure pulse burst the rupture discs? What will be the pressure history in the IHX?

e. The steam generator tube bundle is welded to the tube-sheets. During the $Na/H_2O$ reaction, the tube sheets suffer the highest pressure pulse impact. Due to hockey-stick shape of the tube bundle, the lower tube sheet will be the most affected. If the lower tube sheet fails, can't the water pour into the shell-side and provide further sodium/water reaction effects?

Response

a. Double-Ended Guillotine (DEG)

DEG failure of a steam generator tube is not a credible event. It is rather a convenient and conservative definition on which to base a calculation. Because conditions are not uniform around the initial tube failure, the adjacent 6 tubes will not all be equally affected. Typically, the initial failure will be the consequence of a local effect in the tube wall which results in a directional failure that restricts the reaction zone for potential overheating of adjacent tubes to those tubes that face the initial failure. Statistically, tubes are observed to fail to less than one DEG and to fail asymetrically so that fewer than six adjacent tubes would be subsequently involved.

The Design Basis Leak (DBL) is derived from analysis of bench scale and large leak test data. Bench-scale tests have led to the understanding of how typical small leak progression occurs in the steam generator tube wall. Figure 15.3.3.3-1 in the PSAR illustrates a typical development of a leak within a steam generator tube. These tests have shown; (1) that a small initial leak progresses, resulting in a leak rate of $10^{-2}$ lbm $H_2O$/sec within two hours, and (2) that a leak of $10^{-2}$ lbm $H_2O$/sec magnitude can produce wastage rates of 0.001 to 0.005 inches/second on target material.

Large Leak Test Reg (LLTR) Series II Test A3 was a leak progression test initiated by exposing a pre-drilled 0.0013 $in^2$ hole simulating the self-wastage leak indicated in Step 5 in Figure 15.3.3.3-1. This initiator produced a wastage failure in a tube two rows away after sixty seconds. The failure area was less than 0.017 $in^2$ as compared to the CRBRP SG tube cross-sectional flow area of 0.13 $in^2$. Conservative aspects of this result are: (1) the initiator was aimed and spaced to produce the maximum wastage rate on the target tube*, (2) the sodium was initially static, and (3) the target tube contained initially static water. The leak from the 0.017 $in^2$ failure produced a wastage/overheating failure in the thin-wall (0.025" compared to 0.109" prototypic) injection tube within 25 to 37 seconds. The flalure area in the injection tube was measured post-test as 0.125 $in^2$.

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*The target distance (two rows away) was previously determined by bench scale experiments to yield the maximum wastage rate on the target tube.

Within 18 to 23 seconds after the injection tube failure, three tubes failed due to a combination of wastage/overheating, undercooling, and overpressure. The latter two effects were conservative in that the initially static, subcooled water in the tubes was vaporized and expelled into the water supply system and the pressure in the tubes rose to 2400-2600 PSI prior to failure. These three failures were determined to be 0.1, 0.20 and 0.17 $in^2$.

Japanese large leak tests results have shown that (1) intermediate size leaks produced secondary wastage failures within tens of seconds: failure areas were 0.005 to 0.05 $in^2$, and (2) DEG leaks did not produce secondary failure.

Based upon LLTR and foreign data, a plausible leak progression can be developed for the CRBRP steam generator. Taking the representative leak progression sequence illustrated in Figure 15.3.3.3-1 and assuming (1) a leak magnitude equal to or greater than that indicated in Step 1 of the progression depicted, (2) that this leak does not plug, and (3) that this leak and resultant leaks escape operator action, a plausible sequence is as follows:

1. Within two hours the leak has enlarged as shown in Step 5 of the progression depicted.

2. The enlarged leak produces a wastage failure in another tube after more than one minute. The area of this first secondary failure is 0.005-0.05 $in^2$.

3. The total water injection rate of about one lbm/sec results in burst of the expansion tank rupture disks (150 PSID) within a few minutes. The event is then terminated by isolation and blowdown of the three steam generators in the affected loop.

4. It is conceivable that additional wastage failures could occur during the few minutes in which system pressure is increasing to the rating of the expansion tank disk. Given (1) that a water leak produces a turbulent diffusion flame which is itself situated in a turbulent flow field of high-conductivity, high-heat capacity liquid sodium, and (2) observed wastage failure areas, the size of these potential additional failures would very likely be comparable to the first secondary failure. These potential secondary failures would simply shorten the time to burst of the expansion tank disk. The sequence described above is considered to represent a conservative, plausible leak progression scenario.

   In order to define a clearly conservative DBL (which is not intended to represent either a plausible or mechanistic sequence), it is necessary to include burst of the SWRPRS rupture disks (325 PSID). A rapid Equivalent Double-Ended Guillotine (EDEG) failure serves analytically to burst the SWRPRS disk and also to conservatively bound the failure magnitude. The DBL is defined as follows:

An Equivalent Double-Ended Guillotine (EDEG) failure (0.26 $in^2$) of a steam generator tube followed by two additional EDEG tube failures. The two additional EDEG failures occur as follows:

One additional EDEG failure occurs at one second after the initial EDEG failure.

A second initial EDEG failure occurs at two seconds after the initial EDEG failure.

This sequence of three EDEG failures occurs after an intermediate-size leak (less than a DEG) from a steam generator tube has increased local pressures in the IHTS to the threshold of SWRPRS rupture disk burst. The CRBRP DBL is conservative in both the magnitude of and the timing of secondary failures, compared to the conservative plausible leak progression scenario presented above.

b. A tube failure mechanism already introduced into this discussion is a precursor tube leak, leading to an adjacent tube material wastage/overheating, subsequently leading to pressure rupture of a tube. Figure 760.36-1 shows an array of tubes in cross-section where tube "p" (precursor) is postulated to have an undetected material or manufacturing defect which eventually produces a leak which escapes operator action and causes wastage/overheating on one or more adjacent tubes. The shaded area depicts a potential leak jet, the other surface of which reacts with sodium and thereby develops a high temperature (theoretically as high as 2700$^{\circ}$F, measured as high as 2200$^{\circ}$F in LLTR tests). The source temperature for the overheating is greatest at the reacting interface between the water and the sodium, and less away from the reacting interface.

As the surface of the jet impinges upon the tubes the tube material heats up locally. Fluctuations in the geometry of the jet and the reacting interface during this dynamic event will mitigate the wastage of the adjacent tube but may be insufficient to prevent the metal temperature of an affected tube from rising locally to the point at which the tube wall is too weak to withstand the internal pressure and, therefore, ruptures. Any one of the affected tubes could reach this condition first.

When the pressure rupture occurs, a new, larger water/steam jet is created, with a different profile of tube impingement and localized material wastage/overheating. While the preceding smaller jet and localized material overheating profile may have raised spot temperatures on more than one tube, the pattern of localized overheating is immediately superseded by a new pattern caused by the new, larger water jet. The probability of an additional tube completing its localized wastage/heat-up to a failure temperature before the new overheating profile takes over is considered negligible. (Such an eventuality would be conservatively imposed upon an event which is already extremely unlikely). In any case, two tube failures, both with plausible rupture areas of 20% of an EDEG tube failure, would still be umbrellaed by the one EDEG tube failure recommended for the design basis event definition. For added conservatism, it may be assumed the larger water jet and resulting material overheating pattern, may, like the precursor jet and associated overheating pattern, be sufficiently stable long enough for rupture temperature to be reached on a nearby tube thereby resulting in an additional tube rupture. On the one hand, the larger jet impinges on more tubes than did the precursor jet, thereby increasing the probability of a failure. On the other hand, the much larger jet is more turbulent and diffuse and less likely to permit the reacting surface of the jet to stay on any particular tube area long enough to overheat it to failure. Of more significance than either of these points is recognition that the new, large jet and resulting sodium/water reaction create a rapidly expanding bubble of hydrogen which drives the sodium rapidly away from the tube rupture location. This rapid movement of the sodium interface substantially reduces the potential for a stable reaction zone on the stationary tube surfaces.

c. As discussed previously the CRBRP DBL is clearly conservative in both magnitude and timing of secondary failures. As such, the Project considers it inappropriate to evaluate the simultaneous tube ruptures.

d. Referring to the footnote on Table 5.5-11 of the PSAR, the water injection history input to the TRANSWRAP calculation of the SWR DBL in the evaporator correspond to the following leak sequence:

| Time (Sec) | Event |
|---|---|
| 0.0 - 0.3 | Water flow rate constant at 2.5 lb/sec (this represents the undetected moderate sized leak which has pressurized the system to just below the disk burst pressure - PSAR page 5.5-24b). |
| 0.3 | First Equivalent Double-Ended Gullictine (EDEG) break. |
| 1.3 | Second EDEG. |
| 2.3 | Third EDEG. |

Referring to Figure 5.5-4A and page 5.5-28 of the PSAR, the sharp increase in IHX pressure at 480 milliseconds corresponds to evaporator rupture disk buckling in response to the first EDEG at 300 ms. Predicted peak pressure in the IHX is 331 PSIA as compared to an allowable* range under emergency conditions of 400 to 760 PSIA.

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*Based on ASME Code Case 1331-8 primary membrane stress criteria.

As discussed previously the CRBRP DBL is clearly conservative in both magnitude and timing of secondary failures. As such, the Project considers it inappropriate to evaluate more than one tube rupture at the beginning.

e. The results of the analysis of the $Na/H_2O$ reaction predict that the maximum pressure, 365 PSIA, occurs on the upper tube sheet. The pressure time history at this location is shown in Figure 5.5-4b. The peak pressure at the lower tube sheet during this event is 348 PSIA.

The design pressures on the tubesheets are 325 psig on the sodium side and 1900 to 2400 psig on the water/steam side depending upon the tube sheet location and whether the unit is an evaporator or superheater. Since these $Na/H_2O$ reaction peak pressures would be enveloped by the design pressure differentials across the tubesheet, these loadings can be accomodated with the same degree of structural reliability as normal operation.

Question CS760.102

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

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

Response

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

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

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

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

- Helium leak test.

- Penetrant test requirement limiting defect size to much less than that of the Code.

- Weld geometry requirement limiting concavity, convexity and wall thinning.

- Micro-focus radiographic examination - developed to radiograph tube-to-tubesheet welds with improved resolution.

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

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

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

2. Definition of tight weld geometry acceptance criteria.

3. Post-weld heat treatment thermal stress evaluation.

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

Question CS760.103

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

The prototype for the CRBR hockey-stick steam generator module will be tested in sodium at the Energy Technology Engineering Center in the near future.

Please supply us with a copy of the test request and other pertinent supporting documentation relative to the test article and facility design and types of testing to be conducted.

Response:

The first paragraph of the question appears to be a continuation of question CS760.102, and was answered as such.

The description of the Prototype Steam Generator test is provided in the amended PSAR section 5.5.3.1.5.1 (1).

a. Rupture of steam generator tubing. The design recognizes that the tubing is subject to an internal, high water/steam pressure loading, augmented by steady and cyclical temperature gradients through the tubing wall and by point loadings at support plate contact points.

b. Water/steam leak through tubing or tube-to-tubesheet weld. This failure mode is identified more with material degradation situations than with loading conditions. It can result from water/steam leakage paths caused by weld inclusions or porosity, waterside metal corrosion on cyclic fatigue, sodiumside metal wear at support points, sodiumside metal corrosion by adjacent tube leak.

c. Sodium-to-air boundary rupture. To preclude this eventuality, the design recognizes large sodium/water reaction pressure pulse loadings resulting from rupture of water/steam tubing, low pressure loadings from the sodium system and transient thermal gradients during duty cycle events.

Tests which have been performed are planned to support the steam generator design are:

a. Modular Steam Generator (MSG) Tests, (1972-1974)

Objectives

The objectives of the MSG tests were to confirm both the thermal-hydraulic and structural performance of the hockey stick steam generator design concept. The program tested the MSG unit to it's designed power rating of 32 Mwt maximum and included steady state tests, transient tests and an extended endurance test. The MSG unit had 158 tubes and was operated in a once through mode.

Results

The modular Steam Generator test unit received a total of 9300 hours of sodium exposure with 4000 hours of steaming. Tests were carried out to cover a wide range of operating parameters which enveloped most CRBRP conditions, even though the unit was not operated in the recirculation mode. The test confirmed the design concept, including such design features as the tube/tubesheet welds, use of shroud and inlet thermal liners, and floating spacer plates. The test also confirmed the basic material choice (2 1/4 CR - 2 Mo), the performance capability and analytical prediction techniques. No sodium to water boundary leak indications, cracks or corrosion were discovered during post-test examinations, or subsequent use of the unit for sodium water reaction tests. Overall, the unit performed in a very "well-behaved" manner.

b. Hydraulic Test Model (HTM), (1969-1976)

Objectives

The objective of the HTM test program was to determine the response of sodium side design features to prototypic sodium flows. The test used water as the working fluid in a full scale, but shortened, model of the CRBRP Prototype hockey stick steam generator design.

The HTM unit contained the full internal details of the prototype steam generator including a complete complement of tubes, and the test facility duplicated CRBR inlet and outlet piping configurations.

Results

The tests confirmed the hydraulic acceptability of the prototype steam generator sodium (shell) side. Information on pressure drops, velocities and flow fields was obtained. It also confirmed that the prototype configuration did not have any flow-induced vibration problems. In fact, the test showed that the unit had very minimal tube vibrations. These results confirmed the choices of spacer plate locations, spacer plate flow and tube hold sizing, thermal liner/shroud flow passage size, inlet window sizing, outlet region window and passage sizing and internal clearances.

c. Sodium to Water Boundry Leak Tests (1974 - Present)

Objectives

The objective of these tests was to characterize sodium to water boundary leaks through the use of both small and large scale leak tests. The test program includes investigation of leak initiation, leak propagation tube to tube, leak enlargement within a tube, leak detection both chemical and acoustic, leak mitigation by automatic action, large sodium/water reaction dynamics, large leak damage predictions, and post leak investigations. In addition, the program is intended to support the establishment of adequate design and operational methods to accommodate large sodium-water reactions, and to help develop inservice inspection equipment and techniques.

The test article for these tests is the Modular Steam Generator (MSG) unit discussed in Section 5.5.3.1.5.1.a which was modified to incorporate leak injection tubes.

Results

The results and data obtained from the Large Leak Test Rig (LLTR) Series 1 tests demonstrate that the analytical methods used to predict the pressures and velocities resulting from large SWR.

Events are conservative, and thus confirmed that the design includes loading estimates well in excess of those actually produced. The effects of sodium-water reactions were characterized including leak reaction mechanisms and dynamics, large leak dynamic effects and resulting system pressure pulses, leak propagation mechanisms, and pressure relief system response. Leak detection capabilities were determined test inspection techniques were refined. Inservice inspection equipment was developed which could detect tube degradation from leak growth, leak propagation, tube wear, or tube corrosion. Very good resolution and accuracy of wall thickness and flaw measurements has been demonstrated. For more detail concerning test procedures and results see sections 1.5.1.4.2 and 5.5.3.6.

d. Few Tube Tests (FTT), (1978)

Objectives

The objectives of the Few Tube Test Program were to (1) conduct endurance tests of the Few Tube Test Models (FTTM) to evaluate tube/tube support° wear and the reliability of tube-tubesheet welds under long-term operating conditions, and (2) obtain performance data for operating conditions ranging from natural circulation to full power. Including both steady state and transient operations.

Component Characteristics

The physical configuration of the FFTM's were similar to the reference hockey stick design of the CRBRP steam generator evaporator/superheater configuration. The evaporator employed 7 active tubes and the superheater employed 3 active tubes (4 tubes were plugged during fabrication). The tube-to-tube support interface details were designed to be representative of the CRBRP units. The length and radii of the FTTM tubes were selected to be the same as the CRBRP steam generator shortest row of tubes on the basis that this condition represented the worst combination of tube-to-tube support movement and side forces.

Results

During the tests, the models were exposed to transients of a severity which damaged the internals to a degree that testing had to be discontinued. Post test examinations revealed the following:

1. Thermal performance prior to the transients was as expected but test data was insufficient for thermal analysis studies.

2. It was found that the design of the tube supports and tolerances imposed on the parts did not permit the tube bundle to thermally expand/contract readily and led to binding or jamming of the tubes, resulting in tube buckling and mechanical failure of the shroud support.

3. The Sodium/Water pressure boundary remained intact even under the extreme transient condition and the severe mechanical loads caused by the transients.

The test resulted in a redesign of the plant unit internals, particularly in the geometry and location of the tube supports and prompted a change in the material selection to reduce friction. The test also increased confidence in the tube design and the tube to tubesheet weld design, since both successfully withstood conditions which were considerably more severe than required by plant operating conditions.

e. Department from Nucleate Boiling (DNB) tests, (1975-1976)

Objective

The overall objective of the test program was to verify that excessive damage to the CRBRP evaporator tubes will not be produced by operation with departure from nucleate boiling (DNB) or liquid film dryout in conjunction with maximum specified CRBRP water chemistry conditions.

Test conditions were selected which represented the worst case conditions to which the CRBRP evaporator tubes could be subjected: (a) Maximum ( T) or tube wall temperature oscillations and (b) Maximum sodium hydroxide conditions in the evaporator water.

The test program was comprised of the following phases:

- o Initial thermal conditioning run of 400 hours for establishing proper film condition inside the tube. During this period, thermal/hydraulic test data were taken to establish the endurance test condition.
- o Endurance testing of about 3000 hrs with in-situ nondestructive examinations after about 1000 hrs and after test completion.
- o Post-Test Destructive Examination.

Results

A total of 2820 hours were accumulated at endurance test conditions. In order to achieve these test hours, the test section was exposed to 4181 hours of steaming with CRBRP water chemistry and with thermal/hydraulic parameters varying ± 10% from the nominal.

The principal findings from the destructive post-test examination were:

1) No localized damage or accelerated corrosion attributable to DNB operation was found in the DNB region nor in any other region of the steam tube.

2) The corrosion of the entire tube (including nucleate boiling, DNB, and film boiling) was found to be essentially uniform. The observed maximum loss of wall thickness was found to be 0.40 mils (10.2 pm) for the period of testing which translates to about 0.8 mil/year (20 pm). By conservative extrapolation of the test data, a long-term life in the order of 30 years would be expected for the CRBRP evaporator tubes.

3) Deposition/fouling on the tube wall was minimal and was characterized by nickel alloying of the magnetite scale. Only a small fraction of the corrosion products present in the recirculation water deposited in the steam tube.

Complete test program results are presented in Reference 22.

f. Friction and Wear Tests (1973-1979)

Objectives

As a subtask of the National Friction and Wear Test Program, tests were conducted using Steam Generator tubing and support plates to verify that selected material couples will meet the wear allowance at end-of-life under simulated operating conditions. Tests were used to select the proper material couple to assure tube wear due to differential thermal expansion meets the design limit of .004" and that the wear does not result in galling between the tube and spacer plate.

Several different tests were performed including pin/plate wear samples and simulated tube/space plate geometrics.

Results

Tests performed by two independent organizations produced the following conclusions.

1) The material couple between Inconel 718 and 2 1/4 Cr-1 Mo has the best wear characteristics of the couples tested.

2) When subjected to simulated CRBR conditions, the selected material couple produced wear less than the .004" allowance.

3) For the selected couple, the wear is abrasive and not adhesive.

4) The couple of Inconel 718 and 2 1/4 Cr-1Mo meets the .004" wear allowance with margin for tube sides loads of less than 25 lbs. The couple may be satisfactory for side loads of up to 100 lbs. as indicated by limited test results from one organization.

g. Single-tube Performance, Stability and Interaction Tests (1976-1977)

Objectives

The objectives of these tests were to establish single tube heat transfer correlations, hydraulic stability data and the effect of tube to support interactions on the structural vibration of a tube.

The program utilized single tube tests in sodium and air. The tests in sodium included prototypic temperatures, heat fluxes and flows for a single tube. The tests in air parametrically studied structural vibration of tubes.

Results

The tests provided heat transfer correlations for use in all three heat transfer regimes of an evaporator. These included applicable sodium-side heat transfer information.

Stability results were encouraging but not conclusive. As a result, inlet orifices were added to the evaporator tubes which raise tube differential pressures and thus ensure stability.

The prototype steam generator test program will provide conclusive data which it is anticipated will show that these orifices are not required.

The effects of tube orientation, tube/support misalignment, fluid medium, tube/support-hold clearance, support thickness, exciting force amplitude, and support spacing on the vibrational characteristics and displacement response amplitude of a multi span tube were determined. The test results were compared with the analytical results based on the multi-span beam with "knife-edge" supports. The experimental results showed a small variation of resonant frequencies due to tube orientation, tube/support-hold clearance, support thickness, tube-support spacing and excitation force amplitude. Measured frequencies were close to the calculated natural frequencies. The tube/support-hole clearance was found to be the most sensitive parameter for response amplitude. A significant variation in displacement amplitude was observed for the tube/support hole clearances greater than 20 mils. Short spans placed at both sides of the excitation span reduced the overall displacement amplitude significantly, despite the fact that the lowest resonant frequency is not maximized.

h. Tube to Tubesheet Welds Tests (1976 - 1980)

Objectives

Mechanical testing of tube to tubesheet weld specimens prototypic of CRBRP steam generators was performed to determine the specific effects of microstructure, composition, environment and stress/strain on the failure susceptibility of the 2 1/4 Cr - 1 Mo steel welds. The study investigated the most probable mechanisms likely to cause failure of CRBRP tube to tubesheet welds. Predominantly this work has evaluated "standard" good quality welds, a few lower quality welds have also been tested, since these were considered most likely to have the highest failure susceptibility. The tests included weld and/or HAZ-notched tensile, impact and bend specimens of the tube to tubesheet weld region.

Results

Weld failure susceptibility was not observed during testing at ambient temperature under conditions of high uniform stress and strain, high local stress and strain (all above yield point) and high strain rates. Both microscopic and macroscopic ductility of the weld area was retained under these testing conditions, and specimen rupture occurred only in the base materials. Using the design as employed in the CRBRP steam generator, the following specific results were found with respect to:

1) Caustic stress corrosion (CSC) - Tests at 232°C with 10% and 20% NaOH in pure water indicate that post weld heat treatment (PWHT) at 727°C (1340°F) for twenty (20) minutes or longer, provided a high degree of resistance to CSC of welds.

2) Biaxial stress-rupture - Tests at 510 and 566°C (950 and 1050°F) of tube to tubesheet weld specimens with and without a one hour, 727°C PWHT were performed. Up to 10,600 hours at 566°C and 14 Ksi failures of material with or without PWHT occurred in the annealed tubing in a ductile manner. For durations of greater than 10,600 hours, the failure times were not reduced relative to rupture times for unwelded base material.

3) Four point bend flexural fatigue - Tests at 510 and 566°C of tube to tubesheet weld specimens both with and without PWHT, and with PWHT plus 1,000 hour, 510°C aging, resulted in base metal failure, with secondary HAZ cracking. Tests of welds with induced outside diameter concavity of varying depths showed that the PWHT'd welds crack preferentially in the weld if the concavity is 0.25 mm (0.010 in) deep or greater. Cantilever bend tests (applicable to the upper tubesheet) resulted in tubesheet spigot failure. Weld failures were associated with shorter fatigue lifetimes than for annealed material.

Tests were used to determine the proper post weld heat treatment procedure and weld geometry with respect to concavity and convexity. Procedures have been adopted that limit concavity to .010 inch to preclude failure in fatigue in the base material.

I. Mechanical Properties Tests (1968 - 1981)

Objectives

The overall objective is to verify and supplement ASME Code and RDT standards methods and design information for assuring the structural adequacy of the steam generator.

Results

Properties required to characterize the material in the CRBRP high temperature environment have been obtained. The CRBRP use of 2 1/4 Cr - 1 Mo has involved inelastic analysis and thus a full creep equation has been developed. Recent data of high quality on five heats were used to develop the creep equation. In some time/temperature/stress domains non-classical behaviors were observed and captured.

Fabrication and environmental effects upon the properties of the material have been established. Tests of Post Weld Heat Treated (PWHT'ed) material disclosed that the tensile strength of 2 1/4 Cr - 1 Mo can be reduced by extremely long time post weld heat treatments (40 hours at 1340°F). The loss of carbon (by $N_a$ transport) can reduce the stress-rupture life of 2 1/4 Cr-1Mo.

Extensive further testing of 2 1/4 Cr - 1 Mo has accompanied the CRBR steam generator program. Since ASME Code Case N-47 does not contain fatigue data for 2 1/4 Cr - 1 Mo there have been a number of further fatigue and creep-fatigue tests. Data from in-sodium tests at the Argonne National Laboratory reveal that the lack of air (and thus lack of severe exfoliation) removes the compressive hold time damage. These data are now being evaluated in order to provide an appropriate creep-fatigue limit for 2 1/4 Cr - 1 Mo in each environment of the CRBR Steam Generator.

The design limits of the CRBR Steam Generator were modified to reflect the effects of the service specific environment as outlined below:

1) The design limits were reduced to account for the potential effect of PWHT on mechanical properties.

2) In those areas where significant carbon loss could occur the stress-rupture based limits were adjusted accordingly.

3) A design fatigue curve is in use that reflects these data. In most cases fatigue itself is not a concern; the stress-rupture damage dominates.

j. Scale Hydraulic Model Feature Tests (1980 - Present)

Objectives

Tests of 1/6 scale models of various regions of the steam generator will be carried out to support the final design effort. These tests were designed to confirm and refine the sodium-side internals structures design. Features and phenomena modeled are evaporator sodium outlet thermal striping, superheater sodium inlet thermal striping, sodium inlet region flow distribution, sodium side mixing in elbow region, elbow region thermal striping and sodium inlet nozzle inlet liner seal effectiveness.

Results

Thermal striping has been determined to be acceptable at the outlet and is being investigated for the inlet and elbow regions. Flow distributions in the inlet plenum have been obtained. Elbow region flows have been characterized and the elbow shroud design requirements will be determined. The sodium inlet nozzle seal behavior has been confirmed.

Boundary conditions have been established for thermal/hydraulic analysis models.

Design refinements will be tested as necessary.

Thermal/hydraulic analysis boundary conditions and analytical models will be correlated to test data. A disk type sodium inlet nozzle seal has been included in the design to restrict flows and transients on the shell at the inlet nozzle. Detailed modifications will be made to tube spacer flow hole patterns to improve inlet/elbow flow mixing. Thermal striping data to be used in structural analysis will be correlated to a test data base. Performance of current design features in outlet, inlet and elbow regions will be verified.

k. Flow Induced Vibration (FIV) Tests (To begin in 1983)

Objectives

The objective of the test program being planned is to confirm the absense of any degrading flow induced vibration effects in the CRBRP plant unit design. The test program consists of three (3) phases.

Phase I uses low temperature water ($\leq$180$^{o}$F) as the test fluid in a 0.42 scale model with fluid on the shell side only. Test conditions will conservatively envelope all critical velocities and forces expected in the plant with sufficient over testing of simulated flow conditions (approximately 25%) to show that no nearby thresholds exist where problems may develop.

The tests and assessment of data are scheduled to be available prior to the start of significant plant unit fabrication. Although this is a confirmation test, it's schedule will allow results to be incorporated into the plant unit design if required.

Phase II uses low temperature water as the test fluid in an externally instrumented spare plant unit steam generator with fluid on the shell side only. These tests will also run to 125% of simulated full power flow and will conservatively envelope plant unit operating conditions. Phase II is scheduled for 1985.

Phase III uses instrumentation installed on one of the CRBRP Superheaters during startup and pre-operational testing to provide final confirmation of the absense of any degrading flow induced vibration effects in the CRBR steam generators under actual operating conditions.

I. Prototype Steam Generator Tests (To begin in 1982)

Objectives

The objective of the prototype steam generator test is to verify certain performance characteristics. By testing the full-scale SG at high temperatures under steady-state conditions, the effects of numerous design and operating parameters can be determined. This test program supports the overall verification of the CRBRP steam generator units for plant use. The test program includes operation of a single plant prototype unit under steady-state conditions over a range of power from 1 to 70 $MW_t$ to obtain the data necessary for determining steam generator performance characteristics. Operating experience will be obtained to provide input for establishing plant start up and operating procedures.

Steady-state thermal hydraulic performance data will be obtained at power levels representative of steam generator operation under part power and plant emergency decay heat removal conditions. Data will be obtained to evaluate steam side two phase flow stability and sodium side temperature stratification characteristics. Steady-state thermal-hydraulic performance data will be obtained over a range of recirculation ratios. Tests will also be performed to establish $Na/H_2O$ chemical leak detection system response, to measure steam generator acoustic characteristics and demonstrate acoustic leak detection system performance, to measure fouling, to measure tube vibration and to measure system natural circulation capability. Fouling tests will be performed in parallel with the performance tests.

The Prototype test data will also be used to confirm the applicability and accuracy of the analytical models used to prodict the units performance. Because of the nearly identical heat transfer designs of the Prototype test module and the ten plant units, the test data and verified analytical models can be directly applied to predicting plant unit thermal-hydraulic performance over the entire CRBRP operating range.

m. In-Situ Evaporator Performance Tests

Objectives

These tests are intended to provide a final check of the steam generator performance through data acquired from instruments built into one of the steam generator modules installed as an evaporator in CRBR. The required performance data will be obtained during plant pre-operationnal and start-up testing.

5.5.3.1.5.2 Valves

The steam generator system control valves shall be designed to the alternative rules defined in ND 3512 of the ASME Code, Section III. In addition, thermal transient stress analysis, transient pressure analyses, and seismic response analyses were performed for appropriate valve components and, as applicable, for the valve operators. The analyses, which demonstrated that the valve assembly will function as designed and in accordance with the criteria specified in the ASME Code and the valve equipment specification, were provided by the valve manufacturer, after review and approval of their analytical methods.

the intermediate pump discharge and at the intermediate IHX outlet. This measurement, when calculated piping pressure losses between the pressure taps are deleted, will be evaluated against vendor estimates of the intermediate side pressure drop.

The IHX leak tightness and isolation of the primary system from the intermediate system will be demonstrated during the evacuation prior to initial sodium fill and in the sodium inventory observations during Phase 2 testing.

### 14.1.4.6 STEAM GENERATOR MODULE

#### A. FEATURE TESTING

The design of the Steam Generator will be supported by several test programs designed to verify assumptions and provide quantitative data to confirm the adequacy of design analyses.

These tests include (1) the Hydraulic Test Model (HTM), (2) Large Leak Tests (LLT), (3) Few Tube Tests (FTT), (4) DNB tests (departure from nucleate boiling), (5) tube support wear tests, (6) material mechanical properties tests. (7) Modular Steam Generator Tests, (8) Single Tube Performance, Stability and Interaction Tests, (9) Tube to Tubesheet Weld Tests, (10) Scale Hydraulic Model Feature Tests, (11) Prototype Steam Generator Tests, and (12) Flow Induced Vibration Tests. See PSAR Section 5.5.3.1.5 for a description of these tests.

#### B. PREOPERATIONAL AND STARTUP TESTING

A series of tests will be performed on the steam generator modules after they are installed at the site. These tests will be designed to show that the steam generators are properly installed, that they meet all the requirements for safe operation, and that they meet the expected performance requirements.

1. Pre-Operational Tests

   The position and alignment of each module will be checked after it is installed. The module will be checked for leak tightness on both the tube side and the shell side before the sodium and water systems are filled. The water side will be filled first and pressure tested in conjunction with the entire loop (the shell side of the steam generator module will be pressure tested prior to installation). System tests of the water side will provide data on pressure loss vs. flow rate through the module at temperatures up to 400°F. Operability of the module isolation valves and water dump and blowdown subsystem will be tested before the sodium side is filled.

   After all of the IHTS and SGS components are heated to 400°F, the sodium side of the steam generator modules will be filled. System testing of the IHTS will provide data on pressure loss vs. flow rate through the shell side of the steam generator modules.

2. Startup Tests

With the reactor operating, heat transfer and hydraulic performance data will be obtained at several power levels from zero power to 100% of rated power. These data will be used to verify the heat transfer capability and pressure loss calculations. System stability under transient conditions will be used to verify the heat transfer capability and pressure loss calculations. System stability under transient conditions will be demonstrated by changing power levels at the maximum planned rate.

Data will be acquired through Flow Induced Vibration (FIV) Instrumentation externally mounted on a superheater and thermal performance in instrumentation built into an evaporator (see section 5.5.3.1.5.1 (K) and (M)).

The objectives of these tests are to:

a) Demonstrate steam generator performance

b) Determine the overall heat transfer coefficient and module pressure losses at rated power and operating conditions.

c) Demonstrate stable operation at low power levels.

d) Demonstrate stable operation at the maximum planned rate of change in power level.

e) Demonstrate the absence of damaging flow induced vibrations.

A. Steam Generator Module

There are substantial materials properties and weld development programs which support the development of a reliable heat transfer surface for the steam generator module. For descriptions of the test programs see section 5.5.3.1.5.1.

## Clarification of Items 38, 39, and 40

Item 38. Stator checks refers to electrical resistance checks of the stator windings and insulation. Acceptance criteria will be developed for all shutdown tests.

Item 39. The 30 year design life duty cycle for the PCRDM required 732 scrams and 17,000 feet of travel. In the PCRS tests, four prototype mechanisms experienced the following history:

| Mechanism | 001 | 002 | 003 | 004 |
|---|---|---|---|---|
| Scrams | 1868 | 1561 | 1234 | 859 |
| Feet of Travel | 35451 | 20192 | 11816 | 18978 |

At the completion of this test program, each leadscrew was examined for chips and wear in the latch area and along the entire stroke length of the leadscrew. In all cases, some wear was noted in the latch area. However, this wear was never sufficient to degrade the leadscrew to the point where the latch or travel functions were affected. Throughout the entire test program, no mechanism ever failed to latch or scram upon command.

Item 40. During the fabricator acceptance tests, each PCRDM was tested to show that it could provide the necessary torque to produce a 1000 lb. insertion force to free a stuck rod. If this force does not free the stuck rod, the driveline will be separated from the Primary Control Assembly at the breakaway joint (see PSAR Figure 4.2-104) and the entire assembly will be replaced. The breakaway joint has been tested and found to fall within the design load requirement.

Attachment 3

Clarification of Item 45

There is no RDT F9-4 nor F9-5; any reference in the PSAR to these standards refers to RDT F9-4T and F9-5T respectively.

Attachment 4

Comments on Item 62

No specific assessment of typical LMFBR design cases have been made using the simplified elastic analysis method developed by Konish. However, some general cases have been evaluated and reported as noted below:

(1) H.J. Konish, "Inelastic Analysis and Creep Damage Evaluation of a Thin Plate Tensile Specimen Containing a Central Circular Hole," WARD-HT-3045-36, January 1979.

(2) H.J. Konish, "Inelastic Analysis and Creep Damage Evaluation of a Circumferentially Notched Circular Bar Tensile Specimen," WARD-HT-3045-37, February 1979.

(3) H.J. Konish, Simplified Evaluations of Creep Damage in Notched Tensile Specimens," WARD-SD-94000-3, January 1980.

These documents are available under UC-79h distribution.

Attachment 5

Clarification of Item 67

Further analysis is planned to determine the capability of the critical IHTS transition joints to meet ASME Code criteria for thirty years' service. Appropriate actions will be identified in the FSAR if adequate lifetime cannot be demonstrated.

Attachment 6

Clarification of Item 71

A clarification was requested regarding the application of Code Subsection NG to fabrication as well as to the design and analysis of reactor intervals. Code requirements are generally applied throughout the total construction process, i.e., design, analysis, fabrication, etc. However, the specific subsections or modifications thereof may vary among the various reactor internal components. Details will be presented at the October 25 meeting.

Item 1

ENCLOSURE 2

TABLE 3.2-2

PRELIMINARY LIST OF SEISMIC CATEGORY I MECHANICAL SYSTEM

COMPONENTS AND ASSIGNED SAFETY CLASSES (3)

| Components | Safety Class (1) | Quality Group (11) | Location (2) |
|---|---|---|---|
| Reactor Vessel & Primary Heat Transport System | | | |
| Reactor Vessel & Closure Head | 1 | A | RCB |
| Primary Sodium Pump | 1 | A | RCB |
| Intermediate Heat Exchanger (IHX) | 1 | A | RCB |
| Piping | 1 | A | RCB |
| Reactor Guard Vessel | 2 | B | RCB |
| Pump and IHX Guard Vessels | 2 | B | RCB |
| Upper Reactor Vessel Internals | 1 | A | RCB |
| Lower Reactor Vessel Internals | 1 | A | RCB |
| Fuel, Blanket and Control Subassembly Structures | 1 | A | RCB |
| Primary Control Rod Drive Mechanisms Structures | 1 | A | RCB |
| Secondary Control Rod Drive Mechanism Structures | 1 | A | RCB |
| Auxiliary Liquid Metal System | | | |
| Primary Sodium Overflow Tank | 1 | A | RCB |
| Primary Sodium Makeup Pumps | 1 | A | RCB |
| Overflow and Primary Sodium Makeup Piping and Valves (6) | 1 | A | RCB |
| Overflow Heat Exchanger | 1 | A | RCB |
| Airblast Heat Exchangers | 2 | B | RSB |
| EVST Sodium and NaK Forced Convection Loop Components, Piping and Valves | 2 | B | RSB |
| EVST Natural Convection Sodium Loop Components and Piping | 2 | B | RSB |
| EVST Natural Convection NaK Loop Components, Valve, and Piping | 3 | C | RSB |
| Natural Draft Heat Exchanger | 3 | C | RSB |
| Primary Loop Drain Line (6) | 1 | A | RCB |
| Primary Cold Traps (7) | 3 | C | RCB |
| In-Containment Pri Na Storage Vessel | 3 | C | RCB |
| Ex-Cont. Pri Na Storage Vessel | 3 | C | SGB |
| EVST Na & NaK Drain Piping (8) | 3 | C | RSB |
| PHTS Drain Lines (9) | 3 | C | RCB |
| IHTS Na Processing System | 3 | C | SGB |
| EVST Cold Trap | 3 | C | RSB |
| Intermediate Heat Transport System | | | |
| IHTS Piping Extending from IHX | 2 | B | RCB, IB, SGB |

TABLE 3.2-2 (Continued)

PRELIMINARY LIST OF SEISMIC CATEGORY I MECHANICAL SYSTEM

COMPONENTS AND ASSIGNED SAFETY CLASSES(3)

| Components | Safety Class(1) | Quality Group(11) | Location(2) |
|---|---|---|---|
| Intermediate Sodium Pumps | 2 | B | SGB |
| Dump Valves | 2 | B | SGB |
| Expansion Tanks | 2 | B | SGB |
| IHTS Drain Lines (6) | 2 | B | SGB |
| IHTS Drain Lines (9) | 3 | C | SGB |
| Impurity Monitoring and Analysis System | | | |
| Primary Plugging Temperature Indication Package | 3 | C | RCB |
| Primary Sodium Sampling Package | 3 | C | RCB |
| Ex-Vessel Plugging Temperature Indication Package | 3 | C | RSB |
| Ex-Vessel Sodium Sampling Package | 3 | C | RSB |
| IHTS Sodium Characterization Package(3) | 3 | C | SGB |
| Fuel Failure Monitoring System | | | |
| Cover Gas Monitoring Subsystem | 3 | C | RSB |
| Failed Fuel Location Subsystem Continuing Reactor Cover Gas | 3 | C | RSB |

TABLE 3.2-5

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

| Component | Code/Code Class (1) | Location (2) |
|---|---|---|
| Reactor Vessel & Primary Heat Transport System | | |
| Reactor Vessel & Closure Head | ASME-III/1 | RCB |
| Primary Sodium Pump Casing | ASME-III/1 | RCB |
| Intermediate Heat Exchangers, IHX (Tubes and Shell) | ASME-III/1 | RCB |
| Primary Piping | ASME-III/1 | RCB |
| Reactor Guard Vessel | ASME-III/2* | RCB |
| Pump and IHX Guard Vessels | ASME-III/2 | RCB |
| Upper Reactor Vessel Internals | ASME-III/1 | RCB |
| Lower Reactor Vessel Internals | ASME-III/1 | RCB |
| Fuel, Blanket and Control Subassembly Structures | ** | RCB |
| Primary Control Rod Drive Mechanisms Structures | ASME-III/1 | RCB |
| Secondary Control Rod Drive Mechanism Structures | ASME-III/1 | RCB |
| Auxiliary Liquid Metal System | | |
| Primary Sodium Overflow Tank | ASME-III/1 | RCB |
| Primary Sodium Makeup Pumps | ASME-III/1 | RCB |
| Primary Sodium Overflow Piping | ASME-III/1 | RCB |
| Primary Sodium Makeup Piping and Valves | ASME-III/1 | RCB |
| Overflow Heat Exchanger | ASME-III/1 | RCB |
| Airblast Heat Exchanger | ASME-III/2 | RSB |
| EVST Sodium and NaK Forced Convection Loop Components, Piping, and Valves | ASME-III/2 | RSB |
| EVST Natural Convection Sodium Loop Components and Piping | ASME-III/2 | RSB |
| EVST Natural Convection NaK Loop Components, Valve, and Piping | ASME-III/3 | RSB |
| Natural Draft Heat Exchanger | ASME-III/3 | RSB |
| Primary Loop Drain Line (5) | ASME-III/1 | RCB |
| Primary Cold Traps (6) | ASME-III/3 | RCB |
| In-Cont. Pri Na Storage Vessel | ASME-III/3 | RCB |
| Ex-Cont. Pri Na Storage Vessel | ASME-III/3 | SGB |
| EVST Na & NaK Drain Piping (7) | ASME-III/3 | RSB |

*Classified 2, constructed to Class 1 Requirements ("constructed" used as in Subsection NCA1110, Section III of the ASME Code).

**Designed to special criteria. See Section 4.2.1.1.2.2.

TABLE 3.2-5 (Continued)

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

| Component | Code/Code Class (1) | Location (2) |
|---|---|---|
| PHTS Drain Lines (8) | ASME-III/3 | RCB |
| IHTS Na Processing System | ASME-III/3 | SGB |
| EVST Cold Trap | ASME-III/3 | RSB |
| Intermediate Heat Transport System | | |
| IHTS Piping Extending from IHX | ASME-III/2* | RCB,IB,SGB |
| Intermediate Sodium Pump Casings | ASME-III/2* | SGB |
| IHTS Expansion Tank | ASME-III/2 | SGB |
| IHTS Drain Lines (5) | ASME-III/2 | SGB |
| IHTS Drain Lines (8) | ASME-III/3 | SGB |
| Steam Generator System | | |
| Evaporators | ASME-III/2** | SGB |
| Superheaters | ASME-III/2** | SGB |
| Steam Drums | ASME-III/3 | SGB |
| Sodium-Water Reaction Pressure Relief Systems (internal to steam gen. bldg.) | ASME-III/3 | SGB |
| SWRPRS Rupture Disc Assemblies | ASME-III/2** | SGB |
| S.G. Water and Steam Components, Piping and Valves | ASME-III/3 | SGB |
| IHTS Na Dump Tank | ASME-III/3 | SGB |
| Steam Generator Auxiliary Heat Removal System | | |
| Air Cooled Condensers | ASME-III/3 | SGB |
| Auxiliary Feedwater Pumps | ASME-III/3 | SGB |
| Protected Water Storage Tank (PWST) | ASME-III/2 | SGB |
| Connecting Piping and Valves (Extending from PWST to and including the First Valve) | ASME-III/2 | SGB |
| (Remaining Portions) | ASME-III/3 | SGB |
| Containment Isolation Valves (Within their associated fluid systems) | ASME-III/2 | RCB,IB,RSB |
| Containment Cleanup System | ASME-III/3 (See Note 3,10) | RSB |
| Containment Annulus Air Cooling System | ASME-III/3 (See Note 1,10) | |

*Classified 2, constructed to Class 1 Requirements ("constructed" used as in Subsection NCA1110, Section III of the ASME Code).

**Classified 2, constructed to Class 1 Requirements.

TABLE 3.2-5 (Continued)

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

| Component | Code/Code Class (1) | Location (2) |
|---|---|---|
| Containment Annulus Filtration System | ASME-III/3<br>See Note 4 | RSB |
| Refueling System | | |
| Ex-Vessel Storage Tank (EVST) | ASME-III/2 | RSB |
| EVST Guard Vessel | ASME-III/3 | RSB |
| EVTM Containment Pressure Boundary | ASME-III/3 | RSB |
| Spent Fuel Transfer Station | ASME-III/3 | RSB |
| Inert Gas Receiving and Processing System | | |
| Primary Cover Gas Lines (Recycle Argon) | ASME-III/2 | RCB |
| Equalization Line Between Reactor Vessel, Primary Pumps, and Overflow Vessel | ASME-III/2 | RCB |
| RAPS (Outside Containment) | ASME-III/3 | RSB |
| RAPS (Inside Containment) (7) | ASME-III/3 | RCB |
| CAPS (Outside Containment) | ASME-III/3 | RSB |
| Emergency Plant Service Water System | ASME-III/3 | SGB,DGB |
| Emergency Chilled Water System | ASME-III/3 | SGB,CB,DGB,<br>RSB,RCB |
| Normal Chilled Water System | ASME-III/3 | RCB |
| Auxiliary Mechanical Systems for Diesel Generators | ASME-III/3 | DGB |
| Fuel Oil Storage and Transfer System Including: | | |
| Diesel Fuel Oil Storage Tanks | ASME-III/3 | YARD |
| Fuel Oil Transfer Pumps | ASME-III/3 | DGB |
| Fuel Oil Day Tanks | ASME-III/3 | DGB |
| Cooling Water System Including: | | |
| Water Expansion Tank | ASME-III/3 | DGB |
| Jacket Cooling Heat Exchanger | ASME-III/3 | DGB |
| Water Temperature Regulating Valve | ASME-III/3 | DGB |
| Starting Air System Including: | | |
| Air Storage Tanks | ASME-III/3 | DGB |

TABLE 3.2-5 (Continued)

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

| Component | Code/Code Class(1) | Location(2) |
|---|---|---|
| Lubrication System Including: | | |
| Lubricating Oil Heat Exchanger | ASME-III/3 | DGB |
| Lube Oil Filters and Strainers | ASME-III/3 | DGB |
| Control Room Heating, Ventilating, and Air Condition System Isolation Valves | ASME-III/3 | CB |
| Non-Sodium Fire Protection System | | SGB,CB,DGB |
| Seismically Qualified Water Supply Piping, Valves, and Valves I&C | ASME-III/3 | DGB |
| RCB Penetration, Valves, and Valves I&C | ASME-III/2 | SGB,RCB |
| Standpipe System (Nuclear Island) Piping and Valves | Note (9) | RSB,RCB |
| Standpipe System Seismic Category I Pumps | Note (9) | DGB |
| Fuel Failure Monitoring System | | |
| Cover Gas Monitoring Subsystem | ASME-III/3 | RSB |
| Failed Fuel Location Subsystem Containing Reactor Cover Gas | ASME-III/3 | RSB |

Notes:

(1) Including applicable code cases.
(2) RCB - Reactor Containment Building
IB - Intermediate Bay of the SGB
SGB - Steam Generator Building
RSB - Reactor Service Area of the RSB
CB - Control Building
DGB - Diesel Generator Building
(3) Piping from containment isolation valves to the filter intake; filters and discharge ductwork per Reg. Guide 1.52.
(4) System will meet the requirements of Reg. Guide 1.52
(5) Out to First Isolation Valve
(6) Within Dual Isolation Valves
(7) Downstream of Isolation Valve
(8) Downstream of First Isolation Valve
(9) Non-Safety Related, Seismic Category I
(10) ASME-III/3, but not Safety Class 3 as explained in Table 3.2-2

3.6.4.4.1 PDA - Pipe Dynamic Analysis Computer Model

The pipe break is analyzed using the Pipe Dynamic Analysis (PDA) computer program Reference 3. Pipe movements are described in one plane, i.e. two dimensions will fully describe the pipe section modeled as a single beam. Figure 3.6-3 shows a typical pipe configuration that may be analyzed. Depending on the pipe characteristics, the pipe may be represented by either a built-in (Figure 3.6-4) or a fixed, simple support-pinned-end (Figure 3.6-5) at B. When the thrust force (F(t)) is acting on the end of the pipe, angular acceleration will occur about point B (Figures 3.6-4 and 3.6-5). As the pipe moves, a resisting bending moment will reduce the net angular acceleration. A restraining device at C will help reduce the angular acceleration. As the resisting moment of the pipe about B increases to the point where it exceeds the applied thrust, angular deceleration occurs. Kinetic energy is absorbed by restraint deflection and by bending of the pipe. The forcing function (F(t)) may be described as one of the three models shown in Figure 3.6-6. All values of F(t) are calculated using Moody's method of calculating blowdown forces in Reference 2.

Three different types of time dependent loads may be applied, only one of which can be used for any given analysis. The first type is depicted in Figure 3.6-6 (a) and is a three-step function. Type two is shown on Figure 3.6-6 (b) and is a constant force to time $t_1$ and then can be any function described by $A = B(x)^n$ between $t_1$ and $t_2$. At $t_2$ the force becomes constant. The third type is a step function shown in Figure 3.6-6 (c).

The forcing function (F(t)) as derived in a generalized form in Section 3.6.4.1 represents only the steady state portion of the pipe blowdown force. Until steady state is reached (see Reference 2), the forcing function on the pipe during this transient period is the sum of the initial blowdown force and the initial wave force. The instant the pipe ruptures a depressurization wave travels at sonic speed toward the reservoir. The wave force applies only to the portion of the pipe it is traveling through. After reaching the reservoir the wave may reflect as a re-pressurization wave. When this re-pressurization wave enters a pipe segment where pressure is reduced, a pressure differential exists across the pipe segment which results in net forces being applied. The initial blowdown force applies only to the portion of pipe where the fluid discharge occurs. Depending on the fluid characteristics, many wave transmissions and associated fluid acceleration may occur as represented in Figure 3.6-6 (c) or approximated by 3.6-6 (b). Figure 3.6-6 (a) represents an example when the depressurization wave travels to the reservoir, the fluid flashes to steam in the line, sonic velocity decreases to about 100 ft/sec after flashing, and the wave does not return to the break segment until steady state is reached.

When the pipe break is postulated to occur, the loop conditions are assumed to be those associated with plant stretch conditions. The pipe is assumed to be positioned in the pipe whip restraint so that the pipe velocity after the break is that calculated with the forcing function and the maximum clearance (between the pipe and the restraint) which could exist during various operating conditions.

It is assumed that the pressure, enthalpy and volume of the fluid in the reservoir (the steam drum) remain unchanged. It is also assumed that the reservoir and pipe break are connected by an ideal nozzle through which the flow is isentropic. Friction is not a factor in the calculation of the transient wave forces, but friction is a factor in the calculation of the blowdown and steady state forces (Reference 2).

TABLE B-1 (Continued)

| Event | | Frequency |
|---|---|---|
| 4. Faulted Events | | |
| F-1 | Deleted | |
| F-2 | DHRS Activation Without SGS Cooldown | |
| F-3 | Feedwater Line Ruptures | |
| F-3a | Feedwater Line Rupture Between Steam Drum and Inlet Isolation Valve | |
| F-3b | Feedwater Line Rupture In Main Incoming Header | |
| F-4 | Steam Line Ruptures | |
| F-4a | Saturated Steam Line Rupture | |
| F-4b | Main Steam Line Rupture | |
| F-4c | Rupture Between Superheater Module Outlet and Superheater Outlet Isolation Valve | |
| F-4d | Rupture Between Superheater Outlet Isolation Valve and Main Steam Line | |
| F-5 | Recirculation Line Breaks | |
| F-5a | Recirculation Line Break Between Drum and Recirculation Pump Inlet | |
| F-5b | Recirculation Line Break Between Evaporator Outlet and Drum Inlet | |
| F-6 | Intermediate Loop Sodium-Air Leak | |

## TABLE B-1

## PRELIMINARY DESIGN DUTY CYCLE EVENT FREQUENCIES

| Event | | Frequency |
|---|---|---|
| 1. Normal Events | | |
| N-1 | Dry system heatup and cooldown, sodium fill and drain | 5 total system + 8 per loop + 17 additional for entire intermediate loop exclusive of IHX |
| N-2a | Startup from refueling | 140 |
| N-2b | Startup from hot standby | 700 |
| N-3a | Shutdown to refueling | 60 |
| N-3b | Shutdown to hot standby | 210 |
| N-4a | Loading and unloading | 9300 (each) |
| N-4b | Load fluctuations | 46500 (each, up and down) |
| N-5 | Step load changes of ± 10% of full load | 750 (each) |
| N-6 | Steady state temperature fluctuations | $30 \times 10^6$ |
| N-7 | Steady state flow induced vibrations | $10^{10}$ (sodium) |
| 2. Upset Events | | |
| U-1a | Reactor trip from full power with normal decay heat | 180(1) |
| U-1b | Reactor trip from full power with minimum decay heat | 0(1) |
| U-1c | Reactor trip from partial power with minimum decay heat | 0(1) |
| U-2a | Uncontrolled rod insertion | 10 |
| U-2b | Uncontrolled rod withdrawal from 100% power | 10 |

(1) - The total frequency for U-1 is associated with normal decay heat so as to balance the trips associated with partial decay heat for events U-2 through U-23.

performance test was run using water as the pumped fluid. This test provided information on the pump NPSH and internal leakage flows. Inelastic analysis of the upper journal impeller weld region of the rotating assembly using ANSYS, was required to show adequate ratchetting strain margins for various upset events.

Subcomponent 4 consists of the lower removable region of the pump inner structure. It was analyzed to the same Code rules as Subcomponent 3. The principle loads are 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 during asymmetrical heating. Principle failure modes associated are elastic failure, creep and creep fatigue. The hydraulic casting has been analyzed by a 3D global model using NASTRAN. The bearings are fed directly from the pump discharge so they are exposed to thermal transients. They have been analyzed with a 2D axisymmetric analysis to develop loads and stresses. An axisymmetric 2D model was used to calculate the stresses in the static shroud around the impeller. Inelastic analysis was required in the bearing support region using MARC and ANSYS.

<u>Piping</u>

The incontainment sodium piping shall be designed and analyzed to the Class 1 requirements of the ASME Code, Section III and Code Case 1592. The piping will be designed to assure that piping stresses, strains and deformations are within the applicable Code criteria and system functional limits. The analyses to satisfy these limits shall reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all of the relevant 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

To perform the structural evaluation of the primary piping, the loadings on the piping loop that result from the usual load effects including internal pressure, deadweight, support movements, thermal expansion, seismic, and thermal temperature gradients must be obtained at particular locations in the piping system (usually at piping components such as elbows, tees, reducers, girth welds, etc.).

Formulae given in the ASME Code, Section III are used to determine stresses throughout the piping resulting from internal and external pressure.

General purpose finite element codes are used to perform piping system flexibility analyses which determine the forces and moments acting on the piping system due to various loading conditions. Even though there are no specific guidelines for modeling runs of pipe using pipe or beam elements, most codes check the assembled model for disparities in the assembled stiffness matrix such as large stiffness differences between elements, small stiffnesses or lack of symmetry. Typically such codes can handle a large range of stiffness values. One commonly used code checks the ratio of the maximum to minimum stiffnesses and prints a warning message if this ratio exceeds $1 \times 10^8$.

Stresses that result from these loads will be considered in evaluating the failure modes of the piping and piping material.

The types of analysis required to verify the design of the piping will include elastic, simplified inelastic and detailed inelastic. 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.

response spectra; and (3) the system is of a regular nature without large discontinuancies and not of a highly irregular configuration. This simplified analysis considers the fundamental frequency of the subsystem to be within the range of the predominant frequency of the supporting system. However, this simplified analysis would be usually performed for relatively rigid subsystems or for subsystems for which it has been demonstrated that this type of analysis provides adequate conservatism. The applicable method of analysis is discussed in Sections 3.7.2.1.2 and 3.7.3.5.

3.7.3.10 Modal Period Variation

The response spectra to be used in the mathematical models for Seismic Category I components are modified spectra which take into consideration variations that may affect where peaks occur. As described in Section 3.7.2.1.1 seismic analyses will be performed using the upper and lower bound of the soil (rock) properties. The spectra produced will be widened by ±10% by frequency to account for uncertainties in the structural model and input. Design spectra will be constructed by enveloping the corresponding spectra for the two analyses.

For all equipment, the maximum acceleration is obtained from the spectrum response curves developed at the applicable elevation.

3.7.3.11 Torsional Effects of Eccentric Masses

The seismic mathematical model for any piping system will include consideration of the effects of torsion when applicable. Nonsymmetrical features of geometry mass, and stiffness will be modeled to include their effects in the analysis.

In general, the torsional effects of eccentric masses will be modeled in the piping mathematical model as cantilevered from the pipe flow axis by weightless rods of infinite stiffness.

3.7.3.12 Piping Outside Containment Structure

The seismic analysis of Category I piping buried or otherwise located outside of the containment structure will consider the condition of the foundation at the plant site. See also Sections 2.5.4.5.3, 2.5.4.10 and 2.5.4.13.4. The appropriate displacements obtained from the soil-structure dynamic model will be used in the seismic analysis of the piping system. Two effects will be considered in the seismic analysis of Category I buried pipes and conduits: "free-field" behavior, and relative displacement of pipe ends due to building motions. The two effects occur simultaneously, however, to facilitate the analysis simplifying assumptions will be made to separate the effects in a conservative manner. The "free-field" stresses are critical for long straight portions of buried pipes. The effects due to relative displacement of pipe ends due to building motions are critical at the ends and at bends of the line. For the "free-field" behavior, the maximum axial stresses will be assumed to be due to a wave traveling in the ground along the longitudinal axis of the pipe and producing a ground motion also in that direction.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

For Category I piping having non-Category I piping systems connected, the analysis of the Category I piping will include, as a minimum, the section of the piping system to the first effective seismic restraint or anchor point beyond the classification boundary.

In any given fluid system, a valve will serve as the seismic Category I and non-Category I boundary. The valve capability to maintain a pressure boundary in the event of a seismic event is to be assured by designing piping on the non-Category I side through the first seismic restraint or anchor beyond the valve for that same seismic event.

For the seismic restraints, the piping system analysis includes the structure or building interaction by considering the appropriate stiffness values in the analytical models. The structure/building mass is usually not considered since its dynamic response is negligible. For the anchors, the piping system is modeled to the anchor with the appropriate stiffness values considered. The resultant anchor loads are summed to form the design loads for the anchor.

3.7.3.14 Field Location of Supports and Restraints

For the analysis of multiple supported subsystems, the effects of relative displacements between piping and support points at different elevations on the supporting system are considered as discussed in Section 3.7.2.7. The response spectra for the different elevations were superimposed to yield an envelope response spectrum to be used in the response spectrum analysis of multiple supported subsystems.

3.7.3.15 Seismic Analyses for Fuel Elements, Control Rod Assemblies and Control Rod Drives

The seismic analyses that will be used to establish the seismic design adequacy of the reactor internals, assemblies, control rod drives, etc., is discussed in Section 3.7.2.1.2. For components such as the assemblies and control rod drives where clearances exist between adjacent members, a non-linear time history analysis has been performed, see Section 4.2.3.3.1.4. The mathematical model consists of the whole reactor system. Preliminary models for linear analysis are discussed below.

Density Ratios

The value of density ratios, $(\rho s/\rho)_m(\rho s/\rho)_p$, from Table 3.9-7 is 0.851. The hydraulic modeling in IRFM is designed to reproduce the flow fields and thus the fluid forcing functions which will occur in the actual reactor. The difference in density ratio between model and prototype results in model structures having a lower natural frequency than comparable prototype structures when the effects of the virtual mass of the fluid in the model are considered. By applying equivalent fluid forcing functions on the model and prototype, the onset of unstable vibrations, if present, will occur in the model before the prototype. The increased density of the test fluid also provides a slightly higher driving energy in the model as opposed to the prototype. Both effects are small but make the model testing conservative.

Vibration Displacements

With respect to the modeling based on the requirements that the ratio of model-to-prototype Strauhal number be unity, the previously cited regimes of DU will establish the scalability of model results. In those regimes where the model is conservative, the effect of density ratio damping should be further conservatism. Then the ratio $(y/D)_m = (y/D)_p$ is considered conservative. The model results obtained in the non-conservative regime and the regime wherein $S_m/S_p \neq 1$ are not directly scalable to the prototype, and the results must be further analyzed based upon the test circumstances to establish applicability.

Model to Prototype Scaling Ratios

Based upon the values of Table 3.9-7 and the geometric scaling ratio of 0.248, the following are model-to-prototype ratios of measured parameters:

$f_m/f_p$ = 4.432 (frequency)

$\Delta_m/\Delta_p$ = 0.248 (displacement)

$F_m/F_p$ = 0.076 (force)

$x_m/x_p$ = 4.871 (acceleration)

The deformation controlled stresses and strains were determined by a 2-D axisymmetric model of a cross section of the ring. The model used for both the thermal and thermal stress analyses of the ring is shown in Figure 4.2-77.

Thermal boundary conditions applied to the upper core former model were fluid temperatures applied through a convection coefficient in several different regions of the model as shown in the figure. The structure was analyzed elastically for the U-1b, U-2b, U-18 and E-16 thermal events which may be conservatively used to umbrella all other loadings.

All regions of the CFS were shown to be adequate using elastic analysis methods except the top surface of the upper ring. This area was shown to be adequate by simplified inelastic methods. The fatigue damage at this location was .414 with a creep damage of .239. This combination of damages falls within the creep fatigue interaction envelope of Code Case 1592.

4.2.2.4.2 Upper Internals Structure

This section presents the analysis performed in support of the final design of the Upper Internals Structure (UIS) and used to demonstrate the adequacy of this component for the expected service conditions and environment. The adequacy of the design is based primarily upon meeting the criteria of Section III of the ASME Boiler and Pressure Vessel Code, including Code Case N-47, and supplemented by RDT Standards F9-4T and F9-5T and special project structural design rules presented in Section 4.2.2.3.2 and 4.2.2.3.3. These special project structural design rules have been developed based on material properties testing. A summary of the components analyzed, material properties, structural design criteria, mechanical loads, thermal environment, methods of analysis and structural analysis is presented herein.

4.2.2.4.2.1 Components Analyzed

The major components of the UIS are identified in Figure 4.2-45. A brief outline of the functions of the UIS is given in Section 4.2.2.2.1.7. A list of the components of the UIS analyzed to demonstrate structural adequacy of the design are:

- o Lower Plate and Ligament
- o Upper Plate
- o Support Columns
- o Shear Webs
- o Core Barrel Key

For most areas of the UIS the most severe thermal transient among the Upset (U) and Emergency (E) Duty Cycle events is an uncontrolled rod withdrawal from full power. For the lower shroud tube, the E-16 emergency transient, three loop natural circulation, is also severe. All other UIS transients are grouped with respect to severity under these transients. The fluid temperature changes are less severe farther from the fuel exit as a result of mixing with control assembly flow and blanket assembly flow. These other assemblies also have less severe changes occurring at their exits. The heat transfer analyses of different areas of the UIS account for all these differences.

Faulted Loads

Two faulted events are identified in the UIS duty cycle. Only one occurrence of either of these events is considered. Faulted events are not considered in cumulative damage calculations.

4.2.2.4.2.6 Methods of Analysis

Elastic analysis, simplified inelastic and rigorous inelastic analysis have been used to develop the detail design which meets all its structural requirements. The simplified inelastic analysis used for the UIS are 1) Neubers method, this method is presented in Code Case 1592 (N-47) in Section T-1430, and 2) Simplified inelastic analysis of plates and cylinders under thermal transient loadings. This technique is used in the program HOTDAMG described in Appendix A. This method utilizes a strain correction factor which is a function of the elastically calculated stress and the yield stress to account for plasticity.

The rigorous inelastic analysis for the UIS was performed using finite element analysis methods. ANSYS and WECAN (both described in Appendix A) have been utilized for this type of analysis. Verification problems have shown that both programs are adequate for detailed inelastic analysis.

Computer Codes

The following computer codes are utilized in the heat transfer and structural analysis of the upper internals structure:

ANSYS
HOTDAMG
WECAN
TAP-A
TRUMP
VARR-II
TEMPEST

Descriptions of these computer codes are given in Appendix A.

4.2.2.4.2.7 Structural Analysis

The detail rigorous analysis can be divided between overall analysis and detail part analysis. The seismic analysis, duty cycle evaluation, and overall thermal stress analysis are overall analyses. Other items discussed are detail part analyses.

TABLE 5.5-2

MANDATORY CODE CASES FOR SGS AS APPLICABLE

| Code Case | |
|---|---|
| 1473-1 | Short Time High Temperature Service for Section VIII, Division 2<br><br>-Modifications to Section VIII, Division 2, are provided for vessels which are to operate during part of their service life (less than 2500 Hrs.) at temperatures above those now provided for in Section VIII, Division 2. |
| 1481 | Elevated Temperature Design of Class 2 and 3 Nuclear Components<br><br>-Modifications to Section III are provided for Class 2 and 3 components with normal operating temperatures above those provided for Section III. |
| 1592 | Components in Elevated Temperature Service Section III, Class 1. |
| 1593 | Fabrication and Installation of Elevated Temperature Components, Section III, Class 1. |
| 1594 | Examination of Elevated Temperature Nuclear Components, Section III, Class 1. |
| 1595 | Testing of Elevated Temperature Nuclear Components, Section III, Class 1. |
| 1596 | Protection Against Overpressure of Elevated Temperature Components Section III, Class 1. |
| 1606 | Stress Criteria Section III Classes 2 and 3, Piping Subject to Upset, Emergency, and Faulted Operating Conditions.<br><br>-Design criteria are provided for Class 2 and 3 piping subject to upset, emergency, and faulted conditions. |
| 1607 | Stress Criteria Section III, Class 2 and 3, Vessels Subject to Upset, Emergency, and Faulted Operating Conditions.<br><br>-Design criteria are provided for Class 2 and 3 vessels subject to upset, emergency, and faulted conditions. |

ENCLOSURE 3

CRBR MECHANICAL DESIGN REVIEW OF THE PSAR

SUMMARY LIST OF OPEN ITEMS

Reference: C. Kido, et al., "Clinch River Breeder Reactor Project, Mechanical Design Review of Chapters 3, 4, and 5 of the Preliminary Safety Analysis Report," EGG-EA-5881, EG&G Idaho, Inc., Idaho Falls, Idaho, July 2, 1982.

The following open items are from the low temperature portion of the review.

1. Two items have been omitted from the list of Seismic Category I Mechanical System Components (Table 3.2-2):

    a. Reactor Core and Internals

    b. Reactor Shutdown Systems. (Item 1 pg. 3.2.1-4)

2. Justification should be provided for not classifying the Liquid Metal/Gas Leak Detection System as a Seismic Category I system. (Item 2 pg. 3.2.1-4)

3. In PSAR Section 3.2.2, the non-safety related components and piping are not clearly identified, nor are the corresponding industry standards for design, construction, and operation clearly presented. (Item 1 pg. 3.2.2-3)

4. Do any mechanical systems and components correspond to Quality Group D requirements as contained in Regulatory Guide 1.26? (Item 2 pg. 3.2.2-3)

5. In general, the fluid system boundaries are not clearly indicated on the piping and instrument drawings. (Item 3 pg. 3.2.2-4)

6. On Table 3.2-5, the Applicant presents the selected ASME Code classifications for the principal system conponents of Seismic Category I. The Applicant should more completely explain the footnote "Classified 2, Designed and Constructed to Class 1 requirements." (Item 4 pg. 3.2.2-4)

7. In PSAR Section 3.2.2.2, the Applicant lists examples of Safety Class 2 fluid system components, which includes the Intermediate Heat Transport System (IHTS) piping extending from the Intermediate Heat Exchanger (IHX). However, this section of piping has been footnoted in Table 3.2-5 as being designed and constructed to Class 1 requirements. The Applicant should clarify the discrepancy. (Item 5 pg. 3.2.2-4)

8. Table 3.2-2 notes that the containment annulus cooling system

and cleanup system shall meet the safety Class 3 requirements, but are not classified as safety Class 3. Table 3.2-5 does not list the containment annulus cooling system but does note that portions of the cleanup system shall meet ASME Class 3 and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered--Safety--Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled in Nuclear Power Plants." The Applicant should clarify this apparent discrepancy. (Item 6 pg. 3.2.2-5)

9. Similarly, the containment annulus filtration system is listed as Class 3 in Table 3.2-2 and as meeting the requirements of Regulatory Guide 1.52 per Table 3.2-5. The Applicant should clarify the safety classification of this system. (Item 7 pg. 3.2.2-5)

10. It is not clear why the Applicant's definition of safety classifications presented in PSAR Section 3.2.2 does not include requirements of postaccident containment heat removal and containment atmosphere cleanup systems.
(Item 8 pg. 3.2.2-5)

11. The pipe whip analysis assumes that the pipe break occurs with the pipe centered in the restraint. This results in an average initial clearance between pipe and restraint. The maximum possible clearance is required.
(Item 1 pg. 3.6.2-5)

12. Detail the relationship between the time variation of the jet thrust forcing function and pressure, enthalpy and volume of fluid in the reservoir driving the jet. This is required by 3.6.2.III.2.C (3) of the Standard Review Pl. .
(Item 2 pg. 3.6.2-5)

13. Table B-1 of Appendix B lists faulted event F-1, whereas in Section B.1.4.1 that transient has been apparently deleted. The Applicant should correct Table B-1 to be consistent with the duty cycle description. (Item 1 pg. 3.9.1-9)

14. Table B-2 indicates zero (0) frequency for upset events U-1b and U-1c. The Applicant should correct this apparent omission. (Item 2 pg. 3.9.1-9)

15. The Applicant should clarify footnote 1 of Table B-1, by specifying which events "balance the trips associated with partial decay heat". Whatis the meaning of the use of "each" associated with events N-2, N-3, and N-5?
(Item 3 pg. 3.9.1-9)

16. NUREG-0718 (Revision 2), January 1982, states that consideration of anticipated transient without scram (ATWS) conditions shall be included in the Applicant's test program to qualify reactor coolant system relief and safety valves.

In Appendix B of the PSAR the Applicant has not included the ATWS test conditions. (Item 4 pg. 3.9.1-9)

17. One-third of the computer program verification documents reviewed in the PSAR made reference to documents not readily available. A list of the missing documents was sent to the CRBR Project Office in April 1982. Until those documents have been received and reviewed, the adequacy of computer program verification cannot be fully assessed. (Item 5 pg. 3.9.1-9)

18. The definition of adequate modal content is poorly stated in 6.2 of Appendix 3.7-A, p. 3.7-A.8 of the PSAR. It should be rewritten to correspond to that in 3.7.2.2.1, p. 3.7-8 of the PSAR. (Item 1 pg. 3.9.2-34)

19. Are the hydrodyamic loads associated with partially filled tasks (sodium and water) considered in the CRBR design? (Item 2 pg. 3.9.2-25)

20. A more detailed description of the criteria which justify the equivalent static load method of analysis is required. This affects 3.7.2.1.2, and 6.1 of Appendix 3.7-A of the PSAR. (Item 3 pg. 3.9.2-25)

21. The description of simplified analyses should state the floor spectra are valid only for support points that are either explicity included in the structural analysis or rigidly attached to such a point. This affects the same areas of the PSAR as item 20 above. (Item 4 pg. 3.9.2-25)

22. Is there a maximum permissible length ratio for adjacent elements on a straight run of pipe? The piping models depicted in Figures 4.1-5 and 4.1-7 of the PSAR appear to have adjacent elements with large length ratios. (Item 5 pg. 3.9.2-25)

23. Calculation of dispacements for support points not included in the structural models is not discussed. What are the procedures for this calculation? This question concerns 3.7.2.7 of the PSAR. (Item 6 pg. 3.9.2-25)

24. Are the effects due to local soil settlements, soil archings, etc., considered in the analysis of Category I buried piping systems? This question concerns 3.7.3.12 of the PSAR. (Item 7 pg. 3.9.2-25)

25. Shouldn't the analysis of Category I piping systems be extended beyond the seismic restraints or anchors at boundaries a sufficient distance to insure accurate support load calculations for the seismic restraints or anchors? This question concerns 3.7.3.13 of the PSAR. (Item 8 pg. 3.9.2-25)

26. What is the acceptance criteria for FIV tests in PSAR Section 3.9.1? Will there be numerical limits on allowable deformation and/or vibration? (Item 9 pg. 3.9.2-25)

27. Part of the test plan involves installing accelerometers of the CRBR during pre-operational testing. What is the justification that the instrumentation is sufficient and adequate to correlate these test results with the analysis Models, and FFTF results? What are the acceptance criteria to ensure similarity of results? These questions concern Section 3.9.1 of the PSAR. (Item 10 pg. 3.9.2-26)

28. What is the justification that the parameter ratios between the model and the CRBR are adequate to ensure proper modeling (PSAR Section 3.9.1)? What are the acceptance limits for these ratios? (Item 11 pg. 3.9.2-26)

29. On page 3.9-1h (Amend. 30) Table 1 is referenced under <u>Vibration Displacements.</u> Where is this table? (Item 12 pg. 3.9.2.26)

30. On page 3.9-1h (Amend. 30) under <u>Density Ratios</u>, what is the basis for conservations? (Item 13 pg. 3.9.2-26)

31. The Applicant should specifically note differences between the testing requirements of Regulatory Guide 1.20 and CRBR testing. The effects of high temperatures on instrumentation should be included. (Item 14 pg. 3.9.2-26)

32. The description of the piping startup test program found in Chapter 14 of the PSAR is inadequate. See Subsection V.1 above for a list of the elements which should be included in an adequate description. (Item 15 pg. 3.9.2-26)

33. This open item has been resolved by the answer to question CS 210.14. (Item 1 pg. 3.9.3-7)

34. In Sections 5.3.2.3.4 through 5.3.3.1.2 of the PSAR, the Applicant has not committed to develop and utilize a snubber operability assurance program as required by Section II.3-b of SRP Section 3.9.3. (Item 2 pg. 3.9.3-7)

35. The sodium/water heat exchangers are of a unique configuration designed to minimize the probability of tube leakage. The Applicant should provide a detailed discussion of tube leakage, and features included to deal with this potential problem area. (Item 3 pg. 3.9.3-7)

36. In Table 4.2-47, what is the criterion for the allowable loads on bearings? What is the basis for contact (Hertz) stress between the balls and races? What is the margin for the thrust bearing? (Item 1 pg. 3.9.4-5)

37. Where is Table 4.2-43a referenced, and what is its meaning?

(Item 2 pg. 3.9.4-5)

38. On page 4.2-307 what is meant by "stator checks"? Will there be acceptance criteria for shutdown tests?
(Item 3 pg. 3.9.4-5)

39. What is the basis for determining that the CRDS mechanism latching will not ship or otherwise degrade the lead screw so that continued operating will be impared?
(Item 4 pg. 3.9.4-5)

40. No tests to determine CRDS capabilities to overcome a stuck rod have been included? (Item 5 pg. 3.9.4-5)

41. The removable radial shielding (RRS) is in a preliminary phase of design. Stress analysis, taking into account the effects of environmental conditions, has not been completed. The Applicant has not provided sufficient information for the staff to complete its evaluation of the RRS component.
(Item 1 pg. 3.9.5-5)

42. The Applicant should define the "mechanical discrimination features" which are designed into the lower internals components to ensure proper support and alignment and to accommodate thermal expansion. (Item 2 pg. 3.9.5-5)

43. The Applicant should specify the criteria for change out of nonpermanent reactor internal components, such as the lower inlet modules (LIM). Present information is insufficient to conclude that structural interference will not occur during LIM withdrawal. (Item 3 pg. 3.9.5-5)

44. On Table 5-1 of Appendix G of the PSAR the Applicant has not provided a program of testing and inspection of the reactor internals structures. (Item 4 pg. 3.9.5-6)

45. There is an apparent inconsistency in specifying the use of RDT F9-4 and F9-5 versus RDT F9-4T and F9-5T. The Applicant should clarify the discrepancy. (Item 5 pg. 3.9.5-6)

46. PSAR Section 4.2.2.4.2 states that special project structural design rules were used to determine adequacy of the upper internals structure. The Applicant should provide a description of and basis for the use of these rules.
(Item 6 pg. 3.9.5-6)

47. The Applicant should specify the methods of simplified and rigorous inelastic analysis mentioned in PSAR Section 4.2.2.4.2.6. (Item 7 pg. 3.9.5-6)

48. The Applicant has not provided sufficient details of the inservice testing program for pumps and valves to allow the staff to complete its evaluation at this time.

a. In those instances where the CRBR inspection and testing requirements are different from the ASME Code Section XI, the Applicant should identify those differences and provide justification for the variance.

b. In those instances where requirements have been specified which are not in the ASME Code Section XI, those requirements should be clearly identified. (Item 1 pg. 3.9.6-7)

49. Provide an amended version of Table 3.1-1, "Components which Comprise the Reactor Coolant Boundary", which includes the following for each item in the current table:

a. ASME Class

b. ASME Edition

c. ASME Addenda. (Item 1 pg. 5.2.1-8)

50. Does the reactor coolant boundary design, which was made to Code Editions and Code Cases at least five years old, provide a comparable level of safety to a similar design made to current Code Editions and Code Cases. (Item 2 pg. 5.2.1-8)

51. A table identifying all ASME and ANSI Code Cases applied to Section III, Division 1 and 2 components should be included in the PSAR. (Item 3 pg. 5.2.1-8)

52. Code Cases 1473-1, 1481, 1489, 1521, 1606, and 1607 should be reviewed by the NRC to determine acceptability for use in the CRBRP design. Such reviews should include consideration of the unique features of a sodium design. (Item 4 pg. 5.2.1-8)

53. Does the current design of the elevanted temperature portion of the core support structure to Code Case 1592-7 (as supplemented by RDT standards) achieve a comparable level of safety to a design done to the current Code Case N-201, "Class CS components in Elevated Temperature Service, Section III, Division 1"? (Item 5 pg. 5.2.1-8)

54. Section 5.1.2 of the PSAR states that part or all of the Auxiliary Liquid Metal System and the Cover Gas System are included in the reactor coolant boundary, yet components of neither system are mentioned in Table 3.1-2, "Components Which Comprise the Reactor Coolant Boundary". Clarify this discrepancy. If components of these systems are not to be added, justify this action. (Item 6 pg. 5.2.1-9)

The following open items are from the high temperature portion of the review.

55. Weldment Safety evaluation must consider early crack initiations, metallurgical notch effects, the reduction of material ductility and facture toughness, the strength of the heat affected zone, residual stresses and the cooldown between welding and annealing. The Applicant should commit to a series of tests to answer these questions prior to plant startup. (Finding 1 pg. 1 of Attachment 1)

56. Leak-before-break concept in the hot leg (over 800$^{o}$F) piping requires additional justification by the Applicant. Circumferential cracking has been observed in the heat affected zones of weldments in non-nuclear and test hardware when subjected to CRBR temperature and loads. Acceptable criteria for each justification is required prior to issuance of a CP if the Applicant intends to rely upon the leak-before-break safety concept for the hot leg piping. (Finding 2 pg. 3 of Attachment 1)

57. Seismic events impose high short-term primary stresses on a structure. The seismic loads affect the inelastic strain accumulation and may also produce plastic strain accumulation. Consequently, the sequence of loading becomes important in the creep regime. The Applicant should commit to use acceptable methods and criteria to account for creep enhancement of strain accumulation and creep rupture damage due to potential realistic sequences of seismic events prior to issuance of a CP. (Findings 3 pg. 5 of Attachment 1)

58. In elevated temperature structural analysis, a complex interface between engineering design and materials behavior must be considered. The Applicant needs to additionally consider: (i) that the materials in critical structures have the minimum creep properties, (ii) the effects of thermal aging accompanied by service-induced strain, (iii) the effects of carbon surface redeposits introducting a brittle surface condition (iv) the use of austentic materials with grain boundary porosity which pass all acceptance specifications but have very low creep fatique strength and ductility, (v) the justification for use of the methods of RDT Standard F9-5T, and (vi) the justification for the use of any alternate criteria that may have been used. (Findings 4 pg. 7 of Attachment 1)

59. The simplified code tests for high temperature piping are not sufficient the ensure that the creep effects are insignificant. At locations where the Applicant is performing detailed inelastic analysis, the location should be described, the logic and sequence of events should be described, extrapolation to end of life should be described, and the effects fo strain hardening, cyclic hardening, thermal annealing, creep hardening and thermal cycling should be described. (Findings 5 pg. 9 of Attachment 1)

60. The Applicant should provide the method and critria used to

account for creep redistribution of stress (elastic follow-up) in the piping system and the criteria for including the redistribution loads in combination with the elastically calculated loads in sizing the supports for NRC review and concurrence.
(Findings 6 pg. 11 of Attachment 1)

61. Creep rupture damages at stress raisers was evaluated by the ratios of the time at stress to the minimum time to rupture at that stress. Since creep rupture damage is such a highly nonlinear function of stress, the damage occurring after cycle hardening can be orders of magnitude higher. These effects should be included in the creep rupture damage evaluation. (Findings 7 pg. 12 of Attachment 1)

62. Stress raisers introduce a site where creep rupture damage could cause early crack initiation and more rapid crack propagation (a notch weakening effect). An evaluation is needed to determine what geometric, loading and material parameters could cause significant notch weakening. Loading conditions such as transverse shear have contributed to weldment cracking at structural discontinuities. The Applicant should commit to an acceptable program to quantify the extend and seriousness of the problem prior to issuance of CP. (Finding 8 pg. 13 of Attachment 1)

63. The flaw sensitivty of the hardware should be evaluated considering the reduced fracture toughness and ductility due to temper embrittling. Carbon surface deposits, prior creep rupture damage and irradiation effects where relevant. The Applicant should commitment to an acceptable program to evaluate flaw sensitivtiy prior to an issuance of a CP.
(Finding 9 pg. 14 of Attachment 1)

64. The Applicant used modified creep-fatique damage rules for non-Code stamped austentic stainless steel components. The modified rules assumed that in compressive hold, the creep damage is only 20% as damaging as that caused by the same sustained stress in tension. Other studies indicate that this may not be conservative and so the Applicant should justify the 20% factor.

Stainless steel materials in some components may be subjected to high cycle fatique beyond the Code Case 1592 curve limit of $10^6$ cycles. In 1982 ASME adopted new curves extending to $10^{11}$ cycles, for material temperature below 800°F. Similar curves are also available for temperatures above 800°F. The Applicant should confirm that his criterion is in agreement with the new data.

For 2-1/4 CR-1Mo new fatique design curves which account for environmental effects have recently been approved by Code Committees. The Applicant should ensure that his criteria are consistent with the new curves.

(Finding 10 pg. 15 of Attachment 1)

65. The Applicant should determine what areas of elevated temperature systems and components do not meet current N-47 Code Case requirements, identify these areas, and provide justification that these areas satisfy the general safety margins of the ASME Code. RDT Standard F9-4T and F9-5T have not had the benefit of independent review as national consensus standards nor review by NRC. Therefore they could not be treated as validated acceptance criteria for the conduct of this review. NRC is conducting detailed review of these Standards prior to issuance of the CP and will inform the Applicant of any revisions or further technical justification which may be required in the design analysis methods, constitutive relations or design acceptance criteria therein. (Finding 11 pg. 16 of Attachment 1)

66. The lower reactor vessel transition weld was analyzed to the NB 3228.3 of the Code. However, the Code includes no plastic strain concentration effects and is not conservative under this stress state. The Applicant should commit to the use of acceptable criteria prior to issuance of a CP.
(Finding 12 pg. 18 of Attachment 1)

67. Stress analysis of the Intermediate Heat Transport System (IHTS) transition joints showed that the hot joint (936$^{o}$F) could meet the ASME Code criteria for only a fifteen-year life. Also, the Applicant's conclusion is based on an anticipated minimum carbon content which does not fall below 0.05%. The Applicant must provide assurance that all of the carbon cannot be depleted from the worst cross-section, or the structural integrity with zero carbon should be examined. Also, the Applicant should provide an acceptable plan regarding justification of the joints for thirty years' service or for replacement after fifteen years prior to issuance of a CP. (Finding 13 pg. 19 of Attachment 1)

68. Large thermal stresses arise in the outer region of the perforated area of the steam generator tubesheet to the rim. Creep rupture damage combined with fatique due to relaxation of high residual stresses limits life of the component. The ASME Code does not provide acceptance criteria for the design of the perforated plates in elevated temperature service.

NRC is planning to provide a position on the acceptance criteria for a perforated tubesheet operating in elevated temperature service. (Finding 14 pg. 21 of Attachment 1)

69. Applicant should review the current version of the MEB 3-1 (Revision 1) to assure that other documents used for specifying pipe break locations provide an equivalent level of conservatism (p. 3.6.2-5).

70. The information on load combinacions and emergency limits in

PSAR Sections 3.9.2/3.7A is incomplete. Coverage equivalent to that in current SARs (e.g., the Byron PSAR) should be provided.

71. The applicant states that the design and analysis of the reactor internals will be in accordance with Subsection NG of the ASME Code, Section III. The applicant should clarify whether or not this subsection is applied to construction (p. 3.9.5-4).

72. Applicant should provide additional justification on the integrity of the Core Support Structure - Support Cone Weld.

73. The Applicant should address an optimization of the number of snubber, such that any snubbers failure effects will be minimized.

74. Provide a discussion for the selection of the plant duty cycles.