Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:109

OCT 2 0 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:

ADDITIONAL INFORMATION RESULTING FROM THE SEPTEMBER 8-9, 1982, MECHANICAL ENGINEERING BRANCH (MEB)/CLINCH RIVER BREEDER REACTOR PLANT (CRBRP) MEETING

Reference: Letter, HQ:S:82:093, J. R. Longenecker to P. S. Check, "Meeting Summary for MEB/CRBRP September 8 and 9 Meeting," dated September 4, 1982

Enclosed is additional information for several items identified in the referenced letter from the September 8-9, 1982, meeting between the Project and the MEB.

Enclosed are responses to items 17, 19, 32, and 68. Responses to items 3, 4, 34, 49, 51, and 54 from the referenced letter will be provided by October 25, 1982.

Sincerely,

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

Enclosures

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17. NRC QUESTION:

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 verifiction cannot be fully assessed. (Item 5 pg. 3.9.1-9)

RESPONSE:

The Project received and reviewed on September 1 the list of computer programs identified by the NRC (EG&G) as not having readily available verification documents. Below, each of the listed programs is identified by the reference number in PSAR Appendix A. The disposition for each request is shown:

A.5 APSA	Provided 10/13/82 to EG&G Representative
A.23 DRIPS	Provided 10/13/82 to EG&G Representative
A.35 FESAP	Publicly available in the referenced ASME Publication
A.61 MRI/STARDYNE	Sent to EG&G representative on 10/20/82
A.71 PLAP	Provided 10/13/82 to EG&G Representative
A.81 SNAP	Provided 10/13/82 to EG&G Representative
A.86 SUPERPIPE	This and NUPIPE, which supercede it, provided 10/13/82 to EG&G Representative
A.100 WECAP	Previously provided to the NRC for LWR code verification
A.101 WESDYN	Previously provided to the NRC for LWR code verification

The owner of the copyrighted report for STARDYNE has agreed to permit a limited number of copies to be provided to the NRC. These will be available to NRC by October 19.

19. NRC QUESTION:

Are the hydrodynamic loads associated with partially filled tanks (sodium and water) considered in the CRBR design?

(Item 2 pg. 3.9.2-25)

RESPONSE:

The load analyses of CRBRP tanks will include any significant hydrodynamic effects from the contents. A modified PSAR page 3.7-8 is attached, stating this conclusion.

with mathematical representation of the system or components. A sufficient number of masses with their appropriate degrees of freedom are used in the model to adequately describe the behavior of the structural system, and to insure an accurate determination of the dynamic response. Significant non-linearities, such as gaps or clearances between RCRS components, are included in the mathematical model. In this case, a nonlinear time history analysis is performed, which considers the impact forces generated at the gap locations. Non-symmetrical features of geometry, mass, and stiffness, are modeled to include their torsional effects in the analysis. Hydrodynamic effects of partially filled tanks will be evaluated wherever they are significant in magnitude. Descriptions of a preliminary reactor system linear model and a preliminary RCRS non-linear model are given in Section 3.7.3.15.

The methods of response spectra analysis and time history analysis are described in a number of publications. A description of these analyses techniques is provided in Appendix 3.7-A.

The system or component is analyzed with the seismic input (floor response spectra or time histories) derived at the particular points of support on the structure. All significant modes of the mathematical model are included in the analysis. The significant, dynamic response modes are those predominant modes which contribute to the total, combines modal response of the system. Other modes, whose inclusion in the square root of the sum of the squares modal summation have anegligible effect on the total response would not necessarily be used. With this procedure the number of modes included will be such that inclusion of additional modes will not result in more than a 10% increase in responses. Where the response spectrum method is used, the individual modal responses are combined by the square root of the sum of the squares, except for closely spaced modes (frequencies less than about 10% apart) where the modal responses are combined by the absolute sum. The analysis is performed independently in each of the two horizontal directions, and the vertical direction. Similar effects obtained for each of the three directions are combined by the square root of the sum of the squares. This is consistent with Regulatory Guide 1.92.

A simplified analysis based on a single mass model or an equivalent static load method may be used when it can be demonstrated that the simplified analysis provides adequate conservatism. For the simplified analysis, the equivalent static force, F_s, is distributed proportional to the mass of the component, and is calculated by the following equation:

$$F_s = 1.5\ W\ A_s$$

where W is the total weight of the component, and A_s is the maximum peak acceleration of the response spectra, which apply at the points of support of the component. Components whose fundamental frequencies are greater than 33 Hz in any direction, are assumed to be rigid in that direction and may be designed for at least the maximum acceleration at their supports.

32. NRC QUESTION:

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)

RESPONSE:

PSAR Section 3.9 discusses the startup functional testing which will include vibration, thermal expansion, and dynamic effects (operational transients) on specified high and moderate energy piping, and the associated supports and restraints. The test program, to be supplied by the FSAR, will include:

(1) A list of systems to be monitored.

(2) A listing of flow modes and transients.

(3) A list of selected locations in the piping systems where visual inspections and measurements (as needed) will be performed.

(4) A list of snubbers on systems that will be measured for snubber travel from cold to hot positions.

(5) A description of the thermal motion monitoring program.

(6) A description of actions necessary to correct vibration that is beyond the acceptance levels.

PSAR Section 3.9 will be modified to reflect the above in Amendment 73 scheduled for November.

68. NRC QUESTION:

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).

RESPONSE:

A. The steam generator design has been changed from a bolted streamhead to an integral (welded) steamhead for both the superheater and evaporator. The PSAR is currently being revised to include the present welded steamhead and an update will be supplied in Amendment 73 scheduled in November.

B. The NRC question is generally true with regard to the type of loading and the associated failure mode at the outer rim of the perforated regions of the Steam Generator tubesheets. By eliminating the tubesheet stud holes in the present (welded) steamhead design, the areas of the largest stress concentration are removed and the level of the peak stress in the entire tubesheet is reduced. In a comparative study between the bolted and welded designs, with the same radial temperature distribution, the elastically calculated peak stress is reduced from about 74 ksi at the stud-hole circumference of the bolted design to 43 ksi at the outermost tube-hole circumference of the welded design.

The question, however, is incorrect with regard to the acceptance criteria for perforated plates in elevated temperature service. The acceptance criteria for ASME Class 1 components provided in Code Case 1592-4, as supplemented by RDT Standard FS-4T, are the acceptance criteria for all structural members including perforated regions of tubesheets. Obviously, for such regions, creep and fatigue damage evaluation will require more intricate types of stress analyses to determine the largest possible creep-fatique damage that could result around the circumferences of the outermost holes. The acceptance will remain the same as described in Appendix T of the Code Case. The same rationale has been adopted by the ASME Code in setting the acceptance criteria for the design of perforated plates at service temperature below the creep regime. Article A-8000 in Section III provides analysis methods that greatly facilitate the analysis of perforated plates, but in the same time the article adheres to the same acceptance criteria of Class 1 components given in

Subsection NB.

C. The tube-to-tubesheet joint is a critical location in the steam generator. However, the design of this joirt conforms to the same acceptance criteria used for the primary coolant pressure boundary (ASME Class 1, Code Case 1592-4, as supplemented by RDT Standard F9-4T).

Bosses are machined on the face of the tubesheet to permit full penetration butt welds with the tubes. The welds are located at 1-3/4" from the tubesheet surface, i.e. more than $8\sqrt{Rt}$, and thus are beyond the range of any local stresses that arise due to the interaction between the tube and tubesheet. Therefore, the joint stresses do not occur at the same location as the weld.

The analysis method for the tube-to-tubesheet joints are more complicated than that for common pressure vessels. The analysis have to account for the interactions between the tube and tubesheet and the loadings on both parts should be simultaneouly applied. But the acceptance criteria (for creep-fatique damage and inelastic strain accumulation) will remain the same as provided in Appendix T of the Code Case.

D. Tests were planned as part of the structural evaluation of a previous bolted steamhead design which used Alloy 718 studs to attach the steamhead to the main tubesheet. Accelerated testing of the Alloy 718 studs and the female 2 1/4 CR-1 Mo steel was planned. In the present steam generator design, however, the welded steamhead concept is adopted and the studs are no longer used. Also, the present structural evaluation of the steam generator does not include any other proof testing where creep could affect the outcome of the testing.

E. The acceptance criteria presently available in the ASME Code for Class 1 components are sufficient to demonstrate the structural integrity of the perforated tubesheet and the tube-to-tubesheet joints operating in elevated temperatue service. Alternative or additional criteria in this regard do not seem necessary.

Since the structural evaluation of the steam generator tubesheets does not presently rely on any accelerated scale model testing in which thermal creep could be a factor, there is no need for acceptance criteria in this area.