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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: T. R. QUAY

SUBJECT: STATUS OF DSER OPEN ITEMS AND RAIs RELATED TO LBB

Dear Mr. Quay:

In a letter dated April 11, 1996 from Diane T. Jackson, NRC to Nicholas J. Liparulo, Westinghouse, a summary of open items related to Subsection 3.6.3 of the DSER for which the NRC staff required additional information was provided. This letter is provided in response to that NRC letter and identify the information required for closure of the items. In addition, responses to the NRC requests for additional information 210.202 through 210.212 transmitted in a letter from Kevin Coyne, NRC to Nicholas J. Liparulo, Westinghouse, dated May 5, 1995 are provided. The attachment to this letter provides the information required for closure.

The meeting scheduled with the NRC staff for review of the LBB evaluations is scheduled for July 16 and 17, 1996. Closeout of these items can be discussed at that meeting.

The response to the related DSER open item 10.4.9-2 on the potential for water hammer in the startup feedwater system is also provided.

Revision 7 of the SSAR introduced a change in the approach of the evaluation of pipe break hazards. The high energy pipe break hazard evaluation is performed during Design Certification. This includes the following activities:

1. identify the high energy line break locations and rooms on the nuclear island and in the turbine building adjacent to the main control room.

2. for each room with a high energy break, determine if there are essential systems, structures or components that are needed to mitigate the break.

3. for each room with essential systems, structures or components, identify the regions that are affected by pipe whip and jet impingement associated with the pipe breaks.

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4. for each room, ensure that there is no adverse interaction between the essential items and the whipping pipe or jet. If necessary, identify the locations of pipe whip restraints and jet shields that are needed to protect the essential systems, structures and components.

These activities are available for NRC audit. Westinghouse will be ready to discuss the pipe break hazards evaluation and schedule for NRC review and audit during the meeting with the NRC scheduled for June 25 and 26, 1996 to discuss piping analysis.

If you have any questions, please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachments

cc: D. Jackson, NRC
N. Liparulo, Westinghouse (w/o attachments)

Remaining Open Items in Chapter 3.6.3: Leak-before-break

The following are the responses to the most recent NRC review of status of open items associated with the use of leak-before-break (LBB) in the AP600. The revision of Appendix 3B included in SSAR Revision 7 addresses the majority of these items.

Item No. & NRC Status — Description / Response, W Status

3.6.3.4-1 (608) — LBB Bounding Analysis

Action W

(A) Revise description of methodology and acceptance criteria of the LBB bounding analysis in SSAR Revision 4 to incorporate information provided in the March 15, 1995 handouts and in the July 25-26, 1995 meeting, and to address all LBB evaluation. Especially, how a margin of 2 between the leakage crack size and the critical crack size can be verified and were calculated under what loads respectively are not clearly stated. SSAR Subsection 3.6.3.2 and Appendix 3B should be revised to ensure compliance of LBB methods and acceptance criteria with guidelines stated in NUREG-1061, Volume 3 and SRP 3.6.3.

Response

Sections 3B.3.1.3 and 3B.3.2.3 of SSAR Appendix 3B revision 7, describe the margin on flaw size. Section 3B.3.3 describes combinations for the maximum loads. This item is Closed

(B) The proposed load combination for the LBB evaluation for the Feedwater line is unacceptable. SRP 3.9.3 does not permit load combinations to be developed based on probabilistic arguments. (See evaluation of Open Item 3.6.3.5-5)

Response

Section 3B.3.3 of SSAR Appendix 3B revision 7, shows load combination for the main feedwater line LBB evaluation. Loads due to main feedwater pipe break in the turbine building are included. The Westinghouse proposal to base feedwater line load combinations on probabilistic arguments has been dropped. This item is Closed

(C) As indicated in the July 25-26, 1995 design review meeting, the staff has been unable to verify the leak rate in the Westinghouse LBB evaluation of the 4 inch automatic depressurization system line (See DSER Open Item 3.6.3.6-4). The staff utilized the PICEP computer code in its verification analysis. A further meeting is needed to review the details of Westinghouse LBB bounding analyses to resolve this issue.

Response

Westinghouse does not use the PICEP computer code for the AP600 LBB evaluations. Westinghouse does not use the PICEP code for evaluations of LBB in operating nuclear power plants. The computer code used by Westinghouse to determine leak rate has been successfully benchmarked to leak rate tests. Westinghouse believed that the PICEP code is excessively conservative. Calculations for the AP600 LBB evaluation are available for audit. A meeting, including an audit, is scheduled between Westinghouse and the NRC staff for July 16 and 17, 1996. This item is Closed

Item No. & NRC Status | Description Response, W Status

3.6.3.4-2 (609) Action W

COL applicant to verify LBB bounding analyses (same as COL Action Item 3.6.3.4-1) on materials, as-built analyses, and acceptance parameters.

Newly added SSAR Section 3.6.4.2, Revision 4 is partly acceptable. Only COL actions regarding reconciliation of as-built piping materials were addressed. Additional COL actions regarding reconciliation of as-built piping design parameters, in addition to verification that as-built stresses are within the limits of bounding analyses, such as pipe diameter, configurations, characteristics of supports, etc. were not included. Thus new Section 3.6.4.2 needs to be revised. DSER COL Action Item 3.6.3.4-1 and DSER Open Item 3.6.3.4-2 remain open and further Westinghouse action is needed.

Response

Section 3.6.4.2 of SSAR revision 7 is revised to include the COL action. This item is Closed. This response also addresses DSER-COL item 3.6.3.4-1 (1883)

3.6.3.5-2 (611) Action W

Class 1 versus. Class 2 differences in analysis, fabrication, and inspection - RAI 252.5

(1) Perform fatigue crack growth analyses for each ASME Code Class 2 system for which LBB is demonstrated.

Response

A fatigue crack growth analysis at the main feedwater nozzle is performed as indicated in Section 3B.2.4 of SSAR Appendix 3B revision 7. This item is Closed

(2) Include the two terminal end welds to the steam generator nozzles for the MS and feedwater lines in the PSI and ISI program.

Response

Inspection criteria are described in Section 3B.5 of SSAR Appendix 3B revision 7. Augmented In-Service Inspection at the main feedwater nozzles connected to the steam generators will be performed as indicated in Section 3B.8 of SSAR Appendix 3B revision 7. This item is Closed

(3) Provide explanation of the significance of differences in analysis, fabrication, and inspection requirements between Class 1 and 2 systems, as requested in Q252.5, should be provided.

Response

SSAR Appendix 3B revision 7, section 3B.4 describes the analysis differences, section 3B.6 describes the fabrication differences and section 3B.5 describes the inspection differences. This item is Closed

Item No. & NRC Status	Description Response, W Status
3.6.3.5-5 (614, 2422 - 2428) Action W	Justification of LBB for MS and feedwater - RAI 252.13 In a letter dated May 2, 1995, the staff requested the following additional information about the feedwater lines:

(A) Discuss the steps taken to ensure that waterhammer is not a concern in feedwater line. During the meeting, W described various design and operating features to address water hammer concerns on the feedwater system. The staff observed that these features would serve to minimize, but not necessarily eliminate, water hammer occurrences in the AP600 feedwater system. In addition, the staff also observed that there was no operating experience for the AP600 feedwater design, and consequently the application of the LBB methodology to the AP600 feedwater system may not be acceptable.

Response

SSAR Appendix 3B revision 7, section 3B.2.3 describes that water hammer is not a concern for the main feedwater line inside the containment. This item is Closed. This response also closes RAI 210.202

(B) Explain why thermal stratification is not a concern and what assurance is there that thermal stratification will not occur in the feedwater line.

Response

SSAR Appendix 3B revision 7, section 3B.2.6 describes thermal stratification of the main feedwater line inside the containment. This item is Closed. This response also closes RAI 210.203

(C) Commit to perform augmented ISI (100% volumetric inspection every inspection interval) at the feedwater nozzles connecting to the steam generator.

Response

SSAR Appendix 3B revision 7, section 3B.8 describes that augmented in-service inspection at the main feedwater nozzles connected to the steam generator will be performed. This item is Closed. This response also closes RAI 210.205

(D) In addition to performing ASME Code Class 1 stress and fatigue evaluation at the nozzle connecting to steam generator, perform a Class 1 equivalent fatigue evaluation for the Class 2 portion of the feedwater line.

Response

SSAR Appendix 3B revision 7, section 3B.2.4 describes that a fatigue evaluation at the main feedwater nozzle equivalent to ASME Class 1 piping is performed. This item is Closed. This response also closes RAI 210.206

Item No. Description
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Status

(E) Discuss any significant differences in the ASME Code Class 1 II and fabrication requirements from the requirements applicable to the Class 2 portion of the feedwater line inside containment that affect LBB assumptions.

Response

SSAR Appendix 3B revision 7, sections 3B.5 and 3B.6 describe the II and fabrication differences. These differences do not affect the LBB analyses assumptions. This item is Closed. This response also closes RAI 210.207

(F) Verify that the dynamic load used for design bounds the effects of feedwater pipe break outside containment (including isolation check valve slamming) and the effects of a postulated water hammer event.

The staff reviewed W letter NTD-NRC-95-4564 on the subject of "AP600 Feedwater Line Load Combination." The letter requested that in load combinations for applying LBB to main feedwater lines, effects of dynamic loads due to feedwater line break in the turbine building need not be included, especially the break induced depressurization loads. The staff has the following concerns:

(a) In Enclosure 1 of the letter, W provided a probabilistic approach, which concludes that probability of a pipe break in nonsafety-related main feedwater pipe in the turbine building is negligibly small. Thus the depressurization transient loads induced by such a break is an event of low probability and should be excluded from LBB load combination considerations.
The staff has the following concerns:
a. General Design Criterion 4 requires that structures, systems, and components important to safety shall be designed to accommodate dynamic effects of pipe ruptures. The staff position to postulate pipe rupture is delineated in SRP 3.6.2. The Branch Technical Position MEB 3-1 is a deterministic criteria which governs pipe rupture postulation of high energy lines inside and outside the containment, either designed by the ASME Code or other than ASME Codes. The criteria should also be applicable to the feedwater lines.
b. Although the General Design Criterion 4 permits exclusion of dynamic effects associated with postulate pipe ruptures from design basis when analyses demonstrated that the probability of pipe rupture is extremely low, it is a general requirement without specific acceptance criteria. The staff position on an acceptable approach is as delineated in the NUREG-1061, Volume 3, which indicates that, for justifying such exclusion, a deterministic fracture mechanics evaluation should be performed for demonstrating sufficient margins against pipe failure.
c. Currently, piping design is based on deterministic Code rules with specified loads and load combinations,and the pipe stresses should be deterministically calculated for meeting specific limits under various defined plant operating conditions. No probabilistic approach is allowed in the piping design.

Item No. & NRC Status | Description Response, W Status

(b) W indicated that the feedwater line anchor located at the exterior auxiliary building wall will eliminate transfer of dynamic loads from the feedwater line break in the turbine building. The staff evaluation concludes that the anchor may be effective to prevent transfer of jet thrust loads due to a feedwater line break in the turbine building. However, the portion of feedwater line inside containment will still be affected by the break induced depressurization loads. Based on the above discussion, the staff evaluation concludes that the bases and the probabilistic approach included in the W letter for justifying exclusion of dynamic effects of feedwater line break in the turbine building is unacceptable.

Response

(a)The Westinghouse proposal to base feedwater line load combinations on probabilistic arguments has been dropped.

(b)Section 3B.3.3 of SSAR Appendix 3B revision 7, shows load combination for the main feedwater line LBB evaluation. Loads due to main feedwater pipe break in the turbine building are included.

This item is Closed. This response also closes RAI 210.208

(G) Provide a discussion of the reduced thermal load effects in the feedwater line resulting from rerouting the auxiliary feedwater to a separate nozzle on the steam generator.

Response

SSAR Appendix 3B revision 7, section 3B.2.6 provides a description. This item is Closed. This response also closes RAI 210.209

(H) Discuss how erosion-corrosion effects have been minimized or eliminated in the feedwater line inside containment.

Response

SSAR Appendix 3B revision 7, section 3B.2.1 provides a description. This item is Closed. This response also closes RAI 210.210

(I) Discuss how fatigue effects due to dynamic operational vibration cycles have been minimized in the feedwater line.

Response

SSAR Appendix 3B revision 7, section 3B.2.4(under high-cycle fatigue) indicated that main feedwater pump vibration is isolated from the leak-before-break feedwater line inside containment via the piping and equipment supports. This item is Closed. This response also closes RAI 210.211.

Item No. & NRC Status	Description Response, W Status
	(J) Commit to provide instrumentation for monitoring any unanticipated dynamic loads in the feedwater line inside containment. **Response** SSAR Appendix 3B revision 7, section 3B.7 indicated that instrumentation for monitoring unanticipated dynamic loads in the feedwater lines inside containment will be provided in the first plant. This item is Closed. This response also closes RAI 210.212. In OITSD No. 614 on September 15, 1995, W committed to revise the SSAR Appendix 3B to address the above staff concerns to both the MS and feedwater lines. Thus DSER Open Item 3.6.3.5-5 remains open. **Response** The SSAR has been revised as outlined above. This item is Closed
3.6.3.6-1 (615) Action W	Soil conditions for LBB analyses - RAI 210.10 During the meeting on July 27, 1995, W indicated that Appendix 3B will be extensively revised and that the worst case soils condition will be used in the bounding analyses. Thus closure of this open item is pending W action to implement its commitment in the bounding analyses, which will be verified at a future meeting. **Response** LBB bounding analyses have been performed and bounding analysis curves included in the SSAR. The use of bounding analysis curves decouples approval of the LBB approach from the details of the piping analysis. The piping analysis results used to demonstrate the feasibility of the use of LBB in the AP600 is based on the envelope soil conditions. This item is Closed
3.6.3.6-2 (616) Action W	Staff piping design review - RAI 252.11 This item will be evaluated as a part of Open Item 3.6.3.4-1. In Q252.11, the staff requested W to clarify whether the stresses in Tables 3B-3 and 3B-4 of the SSAR used in the LBB evaluation of the reactor coolant loop, were from the analysis of routed or unrouted reactor coolant loop piping. In the December 22, 1992 response to Q252.11, W indicated that the sample analysis for the reactor coolant loop piping was based on routed reactor coolant loop piping supported by primary equipment supports, but interconnected piping (e.g., the pressurizer surge line) was not included in the model. The staff intends to review these stresses in future piping design review meetings. Thus closure of this open item is pending further W action to complete the bounding analyses, which will be reviewed at a future meeting. **Response** Appendix 3B has been revised to use the bounding analysis approach recommended by the NRC review. This question is no longer applicable. This item is Closed

Item No. & NRC Status	Description Response, W Status
3.6.3.6-3 (617) Action W	0.5 gpm versus. 1.0 gpm leakage rate The remaining portion of this issue will be resolved as a part of Open Item 3.6.3.4-1.

(A) Provide a commitment in SSAR that GW-N1-001 will be revised to make it consistent with the SSAR.

(B) W committed to use a margin of 2.0 between the critical crack size and the leakage crack size as specified in the SSAR. According to guidance stated in NUREG-1061, Volume 3, paragraphs 5.2(e) and 5.2(h), the leakage crack size should be calculated using normal loads and the critical crack size should be calculated using normal plus SSE loads. Verify its implementation in the SSAR and in the submitted bounding curves. (See also Open Item 3.6.3.4-1)

Response

SSAR Appendix 3B revision 7, section 3B.3.3 describe the loads to be used for the maximum and normal conditions. Sections 3B.3.1.3 and 3B.3.2.3 describe the leakage flaw size and critical flaw size criteria for the LBB bounding analysis curve generation. Reference to an internal Westinghouse design criteria document is not appropriate for Design Certification. This item is Closed

3.6.3.6-4 (618) Action W	Leakage rate evaluation methodology This issue will be evaluated as a part of Open Item 3.6.3.4-1.

As discussed in the July 1995 meeting, the staff is concerned that the W leakage rate evaluation methodology may not be acceptable for calculating leakage rates in small size piping and especially in the single phase, low temperature flow state. As described in DSER Open Item 3.6.3.4-1, the staff was unable to verify the leakage rate in the W evaluation of the 4 inch automatic depressurization system line.

Response

The computer code use by Westinghouse to determine leak rate has been successfully benchmarked to leak rate tests. The LBB evaluation method and leak rate calculation code has been used in large number of LBB evaluations for piping in licensed nuclear power plants. These evaluations have been approved by the NRC. Note, based on a careful consideration of the effect in a pipe with a leak large enough for detection, there are no cold condition (single phase) 4 inch lines to be evaluated for LBB. This item is Closed

Item No. & NRC Status	Description Response, W Status
3.6.3.6-6 (620)	Waterhammer type loads in LBB analyses (Test results issue)
Action W	W has not yet responded to this open item. Preliminary results from small-break LOCA tests performed at Oregon State University indicate that rapid condensation events have the potential to cause unanticipated dynamic loads to occur in the AP600 reactor coolant system. These water hammer type loads have not been considered in the piping design loads to justify a LBB approach for the AP600 main coolant loop and attached piping. W was requested to address whether these water hammer type loads from condensation events need to be considered in its LBB analyses or, if not, justify why these loads can be excluded. **Response** Rapid condensation events are stipulated to have occurred during the Oregon State University tests in the upper end of the downcomer following a system draindown below the cold leg elevation and during the peak accumulator injection flows. The condensation events resulted in a noise that could have originated as a result of a pressure pulse as the water slug refilled the upper downcomer by pulling water from the upper plenum though the core. No impact or damage from these tests were observed. The pressure pulses from the condensation events are small and occurred primarily in the downcomer region of the reactor vessel, and has no impact on the reactor coolant piping. It is therefore not considered part of the design loading for the piping system.

Revision 7 of SSAR Subsection 3.6.3 also addressed the RAIs associated with LBB. The responses are noted below.

RAI 210.202 Westinghouse should discuss in the SSAR the steps taken to ensure that waterhammer is not a concern in the feedwater lines.

See the response to DSER open item 3.6.3.5-5 Part (A)

RAI 210.203 Westinghouse should explain in the SSAR why thermal stratification is not a concern and what assurance that thermal stratification will not occur in the feedwater lines.

See the response to DSER open item 3.6.3.5-5 Part (B)

RAI 210.204 Westinghouse should demonstrate in the SSAR that the feedwater nozzles at the steam generators is the controlling location for stress and fatigue effects for the feedwater line inside containment.

A review of the geometry and lay out of the feedwater line demonstrates by inspection that the connection to the feedwater nozzle is the controlling location in the feedwater line for stress and fatigue effects. The geometry and lay out may be reviewed during the piping stress analysis audit. This information is not appropriate for SSAR. The use of a separate startup feedwater line and nozzle in the steam generator significantly reduces the transient introducing cold water through the feedwater line.

RAI 210.205 Westinghouse should commit in the SSAR to perform augmented II (100 percent volumetric inspection every inspection interval) at the feedwater nozzles connected to the steam generators.

See the response to DSER open item 3.6.3.5-5 Part (C)

RAI 210.206 In addition to performing ASME Code Class 1 stress and fatigue evaluations at the nozzle connecting to steam generator, Westinghouse should perform a Class 1 equivalent fatigue evaluation for the Class 2 portion of the feedwater line. The results of the evaluation should be included in the SSAR.

See the response to DSER open item 3.6.3.5-5 Part (D)

RAIs 210.207 Westinghouse should discuss in the SSAR, any significant differences in the ASME Code Class 1 II and fabrication requirements from the requirements applicable to the class portion of the feedwater line inside containment that affect LBB assumptions.

See the response to DSER open item 3.6.3.5-5 Part (E)

RAI 210.208 Westinghouse should verify that the dynamic load used for the design bounds the effects of feedwater pipe break outside containment (including isolation check valve slamming) and the effects of a postulated water hammer event. The supporting information should be included in the SSAR.

See the response to DSER open item 3.6.3.5-5 Part (F)

RAI 210.209 Westinghouse should provide in the SSAR a discussion of the reduced thermal load effects in the feedwater line resulting from rerouting the auxiliary feedwater to a separate nozzle on the steam generator.

See the response to DSER open item 3.6.3.5-5 Part (G)

RAI 210.210 Westinghouse should discuss in the SSAR how erosion-corrosion effects have been minimized or eliminated in the feedwater line inside containment.

See the response to DSER open item 3.6.3.5-5 Part (H)

RAI 210.211 Westinghouse should discuss in the SSAR how fatigue effects due to dynamic operational vibration cycles have been minimized in the feedwater lines.

See the response to DSER open item 3.6.3.5-5 Part (I)

RAI 210.212 Westinghouse should commit in the SSAR to provide instrumentation for monitoring any unanticipated dynamic loads in the feedwater lines inside containment.

See the response to DSER open item 3.6.3.5-5 Part (J)

Question: DSER-OI 10.4.9-2

STARTUP FEEDWATER SYSTEM, FWS, WATER HAMMER) Westinghouse should address the issue of plant damage due to water hammer during startup.

Response: DSER-OI 10.4.9-2

Information contained in Revision 6 of SSAR Section 10.4, describes a startup feedwater system different from that evaluated in the DSER. Revision 7 of SSAR Appendix 3B, addresses the potential for water hammer in feedwater lines.

The potential for water hammer during startup is minimized by the layout and configuration of the startup feedwater system. During startup feedwater operation, the maximum temperature within the startup system is limited by the deaerator temperature to approximately 250^{0} F. The elevated location and temperature of the deaerator at the upstream end and the steam generator (SG) at the downstream end of the piping will keep the line subcooled. The startup feedwater connection to the steam generator does not enter the feedring and thus feedring voiding is not a water hammer initiating concern. Additionally, the nozzle location is above the normal SG water level, with the piping turning downward close to the SG and without piping raisers within containment or auxiliary building to limit the potential for void formation and collection. Isolation of potential high temperature leakage from either the SG or main feed system is accomplished via redundant isolation valves during normal power operation. Finally, the startup feedwater control valve is specified to limit control instabilities and opens in a controlled manner so that it does not induce water hammer. The "Main Feedwater Line" portion of SSAR subsection 3B.2.3 addresses a number of design features included in the main and startup feedwater system, piping, components and control that minimize the potential for water hammer.

This item is closed.