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DCP/NRC0751
Docket No.: STN-52-003

February 25, 1997

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

TO: T. R. QUAY

SUBJECT: RESPONSES TO OPEN ITEMS ASSOCIATED WITH LEAK-BEFORE-BREAK EVALUATION

Dear Mr. Quay:

In a letter dated February 7, 1997 the NRC provided additional requests for additional information and updates of status for several areas being reviewed by ECGB and EMEB for AP600. The attachment to this letter provides responses to the items associated with the leak-before-break evaluation discussed in subsection 3.6.3 and included in Enclosure 3 of the NRC letter. Several of the items are statused in the NRC letter as resolved and do not require a response. The SSAR changes required as part of the responses will be included in Revision 11 of the SSAR.

Any remaining issues on the AP600 leak-before break evaluations may be discussed at a meeting between Westinghouse and the staff to close out leak-before-break and piping issues planned for mid-March.

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

Brian A. McIntyre, Manager
Advanced Plant Safety and Safety

jml

Attachment

E004 1/1

cc: D. Jackson, NRC w/attachment

1. DSER #3.6.3.4-1 (OITS 608) - LBB Bounding Analysis
Action W/N

- Add a description in SSAR 3B.3.1.3 and 3B.3.2.3 (bounding curve construction procedures), or in 3.6.3.3 (bounding analysis) to explain how bounding curves meet LBB acceptance criteria.

- NRC will audit calculations to ensure that the bounding curves satisfy LBB acceptance criteria.

- Uncertainties in applying LBB to small lines (see NUREG/CR-6443, Section 3.5 on pressure-induced bending effects to leakage flaw size and max stress) needs to be discussed. Westinghouse should perform sensitivity studies.

- Applying the LBB methodology to the FW line is unacceptable. Revisions to the SSAR to delete the main feedwater line from LBB consideration will be tracked under DSER# 3.6.3.5-5 (OITS 614). For further discussion of this issue, see NRC letter dated January 24, 1997.

- Results from the PICEP computer code do not agree with Westinghouse LBB analyses. Need an explanation.

Westinghouse Response

A. As stated in subsection 3B.3, the bounding analysis methodology is consistent with the regulations and guidance on leak-before-break evaluations provided by the NRC.

B. NRC has audited Westinghouse AP600 leak-before-break evaluations twice previously. Westinghouse will provide the bounding analysis curves for an additional review. The meeting to closeout leak-before-break and piping issues is scheduled for mid-March.

C. Westinghouse has previously provided a review that pointed out significant flaws in the assumptions and evaluations contained in NUREG/CR-6443. In the interest of proceeding to a timely decision on final design approval Westinghouse is revising the SSAR to change the minimum pipe size for application of leak-before-break to 6-inch nominal pipe size.

D. See the response to DSER item 3.6.3.5-5.

E. The Westinghouse position on the PICEP code is that it is excessively conservative. This is the position for both AP600 and for leak-before-break evaluations of operating plants. The code used by Westinghouse has been benchmarked to test data and has been accepted for use by the NRC on numerous evaluations of LBB in operating nuclear power plants.

4. DSER# 3.6.3.5-2 (OITS 611) - Class 1 vs. Class 2 differences in analysis, fabrication, and inspection - RAI 252.5
   Action W

Explain why the fatigue crack growth analyses and augmented in-service inspection (ISI), which are performed for the feedwater nozzle connections to steam generator, are not performed at the main steam nozzles. Revision of SSAR Sections 3B.2.4, and 3B.8, Rev 10 may be needed.

Westinghouse Response

The augmented inspection of the weld at the feedwater nozzle was included to provide the NRC staff with additional confidence that the feedwater line would not be subject to a sudden failure. This location was selected based on the experience of feedwater line cracking in early generation operating nuclear power plants. The cracking in the feedwater weld was attributed to conditions including the design of the feedwater ring and stratification in the feedwater line that are not applicable to the AP600. Since Westinghouse is withdrawing the request to include the feedwater line in the evaluations of LBB at this time, (see the response to DSER Item 3.6.3.5-5) this augmented inspection is also being deleted. The weld of the steam line to the steam generator steam nozzle has not had a history of cracking in operating plants and is not subject to the conditions that lead to cracking in feedwater lines. An augmented inspection of that weld is not warranted

7. DSER# 3.6.3.5-5 (OITS 614) - Justification of LBB for MS and FW - RAI 252.13 (OITS 2422 to 2428)
   Action W

The Westinghouse proposal to apply LBB methodology to the FW line is unacceptable. Westinghouse needs to revise the SSAR to delete the main feedwater line from LBB consideration. For further discussion of this issue, see NRC letter dated January 24, 1997.

Westinghouse Response

In the interest of proceeding to a timely decision on final design approval Westinghouse is withdrawing the request to include the feedwater line in the evaluations of LBB in design certification of the AP600. Westinghouse remains convinced that the design of the steam generator and feedwater line preclude significant waterhammer and application of LBB to the feedwater line is appropriate. However, since the staff indicated in the January 24 letter that it will no longer evaluate the merits of this issue, Westinghouse is withdrawing the feedwater line from the list of LBB lines at this time. Westinghouse has provide the staff with a markup of the changes in the SSAR that will implement this change. SSAR Revision 11 will reflect this change.

9. DSER# 3.6.3.6-2 (OITS 616) - Staff piping design review - RAI 252.11
   Action W

This item will be evaluated as a part of DSER Open Item 3.6.3.4-1.

Westinghouse Response

This item is about NRC staff review of the reactor coolant loop stress results and was previously closed. The results of the reactor coolant loop analysis and the associated bounding analysis curves will be available for NRC staff review during the closeout meeting on LBB issues planned for mid-March.

11. DSER# 3.6.3.6-4 (OITS 618) - Leakage rate evaluation methodology
Action W

This issue will be evaluated as a part of DSER Open Item 3.6.3.4-1.

Westinghouse Response

As indicated in the response to Item 3.6.3.4-1, the PICEP code is excessively conservative. Westinghouse uses a code that has been benchmarked to test data and has been accepted for use by the NRC in numerous evaluations of LBB in operating nuclear power plants.

13. DSER# 3.6.3.6-6 (OITS 620) - Water hammer-type loads in LBB analyses (Test results issue)
Action W

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 RCS. 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. Westinghouse 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 and incorporate relevant discussions in the SSAR.

Westinghouse Response

Results from the Oregon State University (OSU) tests indicate the potential for small loads due to steam condensation events. The evaluation of loads has determined that they are of no consequence in the evaluation of leak-before-break. A report on these events has been prepared and provided to the NRC. The report and subsequent evaluations performed by Westinghouse conclude that the events in the OSU tests were caused by rapid steam condensation in the upper downcomer region of the reactor internals resulting in a water level rise. This rise in the water level resulted in an impact on the upper core support plate. The resulting loads have been evaluated to be minimal for the AP600 reactor coolant loop piping and are well within the design limits for the reactor vessel internals.

A meeting between Westinghouse and the NRC Research Branch is planned for March 1997 to discuss the report.

15. Meeting Open Items (OITS 2423 - 2429)
Action N

The Westinghouse proposal to apply LBB methodology to the FW line is unacceptable. Westinghouse needs to revise the SSAR to delete the FW line from LBB consideration. Revisions to the SSAR to delete the FW line from LBB consideration will be tracked under DSER# 3.6.3.5-5 (OITS 614). The staff will complete the review of concerns, other than the FW line, in these open items during the closure of DSER# 3.6.3.4-1.

Westinghouse Response

These meeting open items are all related to information and commitments provided to give the NRC staff with additional confidence that evaluation of the feedwater line for leak-before-break is acceptable based on the criteria in General Design Criteria 4. This information and commitments demonstrated that the feedwater line would not be subject to a sudden failure or unanticipated failure mechanisms. Since Westinghouse is withdrawing the request to include the feedwater line in the evaluations of LBB for design certification, (see the response to DSER Item 3.6.3.5-5), these open items are now immaterial and should be closed out.