Meeting Purpose: Assess progress toward resolution of the BAW-2374 issue regarding hot leg breaks leading to steam generator tube failure in B&W-designed nuclear power plants

Meeting Date and Location: April 27 - April 29, 2004 at Areva Facilities in Lynchburg, Va.

Meeting Attendees, All Meetings including Exit Meeting:
Eric Henshaw, Head, B&W Owners Group Thermal Analysis Committee
John Klingenfus, Areva Technical Lead on BAW-2374 issue
Gordon Wissinger, Areva (reports to John)
Others for short times

Meeting Attendees, Exit Meeting:
Robert Schomaker, Project Manager for B&W Owners Group Thermal Analysis Committee
Sandra Slone, Areva, Licensing

Discussion

Resolution of this issue has been determined to be more complex than originally envisioned. Each B&W plant design is unique and there are about 50 to 100 variables that potentially influence the plant state. Some variables are dependent; others are independent. Variable selections that are conservative with respect to parts of the transient are sometimes non-conservative with respect to other parts. Further, variable dependencies and predicted behavior lead to combinations where it is not possible to select a conservative approach because of combinations of conservative and non-conservative aspects during the transient.

The approved Appendix K models that potentially apply to this issue are the existing short-term LOCA model, containment modeling, and the long-term boric acid analysis model. Approved models do not exist for much of this issue. However, there are difficulties regarding application of even the approved models. For example, a low heat generation rate is conservative with respect to tube challenge, but non-conservative with respect to cladding challenge and perhaps containment pressure. (Higher containment pressure forces more water out the steam lines if tubes break, but increases pressure at the LPSI pump suction during recirculation operation.) Further, Appendix K requires 1.2 times the 1971 ANS decay heat standard, a non-conservative requirement with respect to tube integrity. However, there are precedents for a lower decay heat generation rate during long-term cooling and for changing input to approved LOCA evaluation models when the previously approved inputs were found to be inappropriate. Therefore, there are suitable precedents for development of an acceptable Appendix K evaluation model to address the BAW-2374 issue.

The planned approach to close the BAW-2374 issue is to separate the plant response analyses from the assessment of the impact of tube degradation on the potential for breaking tubes. The traditional approach to 10 CFR 50.46 Appendix K modeling of selecting break sizes and locations in a detailed thermal-hydraulic representation is impractical because of the large number of variables that must be considered. The plant response analyses therefore are to be developed as follows:

1. Develop a spread-sheet representation of plant behavior as a function of appropriate input variables. (Some of this exists with a cross-check against RELAP 5 model predictions.)

2. Compile the input variable values and appropriate distribution functions (including dependencies where applicable). In many cases, it will be necessary to assume the function (typically a uniform or a normal distribution).

3. Conduct Monte Carlo investigations using the Item 2 inputs into the Item 1 representation. One investigation is anticipated for Oconee and an attempt will be made to represent the other plants with a second, bounding investigation. The output is anticipated to be a distribution function of success as a function of the number of tubes that are assumed broken and as a function of whether or not the worst single failure is assumed.

4. Compare a selected case that is considered to be of significant challenge with a modified RELAP 5 evaluation model and other calculations as necessary to assess adequacy of the spread-sheet calculation.

5. Develop tube response by predicting the number of tubes that break due to a degraded tube condition from consideration of experience, justification of tube wall flaws, steam generation inspections, and assessment of RCS conditions predicted via Item 3. The uncertainty of this prediction is expected to be high.

The final step is to compare Item 3 and Item 5 results. The anticipation is that sufficient conservatism will be established to reasonably ensure that (a) the single failure assumption does not contribute any benefit and (b) the anticipated number of tubes that may break does not significantly challenge core cooling and protection of public health and safety.

<u>Schedule</u>

This issue was recognized in 1999 and remains unresolved. Although the safety significance is small, it is a compliance issue that must be addressed. Recent progress has been slow and I stated that a continuation of this progress rate will not be acceptable.

The other participants in the meeting agreed to provide a schedule for completion of addressing this issue within a reasonable time. I anticipate this schedule, that should be considered to be a commitment, will be sufficiently short that the staff does not have to initiate further action.