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Working Together to Economically Provide Reliable and Safe Electrical Power

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Replacement for BAW-2374, Revision 1, "Evaluation of OTSG Thermal Loads During Hot Leg LOCA"

Ref. 1: BAW-2473, Revision 1, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping."

Ref. 2: Letter, William R. McCollum (Chairman, B&WOG) to Document Control Desk (NRC), "Review of BAW-2374," NRC:03:014, OG-03-1833, March 13, 2003.

The B&WOG and the NRC met in the Framatome ANP offices in Lynchburg, Virginia on November 9 and 10, 2004. The NRC was represented by Warren Lyon and Drew Holland. The B&WOG was represented by Morris Byram (Entergy) and Eric Henshaw (Duke Energy). Framatome ANP participants included John Klingenfus, Gordon Wissinger, Stanley Levinson, Gayle Elliott, Sandra Sloan, Darrell Costa, Jim Begley, and Bob Schomaker. Attachment 1 of this letter summarizes the meeting discussions.

The primary objective of the meeting was to review the proposed approach and content of the replacement topical report for BAW-2374 Revision 1, (Reference 1) and the progress made to date developing the information for this report. A general outline of the approach was previously provided by the B&WOG in a letter to NRC dated March 13, 2003 (Reference 2). The approach was discussed in more detail during a meeting between the B&WOG and the NRC on April 9 and 10, 2004.

At the conclusion of the meeting, several follow-up activities were identified. The NRC and the B&WOG agreed to meet in January 2005 for a pre-submittal discussion involving all of the appropriate NRC reviewers for the new topical report (e.g., reactor systems, steam generator/mechanical, PRA, radiological). The B&WOG agreed to transmit a summary of the proposed approach and the framework for a replacement topical report to the NRC prior to the meeting.

Sincerely,

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Enclosures

cc: D.G. Holland, NRC
Reactor Vessel Working Group
Project 693

Attachment 1

The primary objective of the meeting was to review the proposed approach and content of the replacement topical report for BAW-2374, Revision 1. A summary of the information discussed during the meeting is provided below.

The table of contents for the new topical report is shown below. Sections 1, 2, and 3.1 to 3.4 will be very similar to those of BAW-2374 Revision 1; sections 3.5, 3.6, and 3.7 will be added. Section 3.5 will address the long-term cooling criterion in 50.46. Section 3.6 will address the dose consequences above and beyond the existing LOCA dose analyses. Section 3.7 will address the waterhammer potential resulting from primary fluid leaking into and filling up the secondary side of the steam generator and associated piping.

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The development effort for BAW-2374 is currently focused in three areas:

1) Steam Generator Tube Failure Analysis

The purpose of this task is to define the number of tubes expected to fail for a given tube-to-shell temperature difference at a calculated probability of failure.

The probability of a free span tube failure based upon actual plant data from past steam generator inspection results is very low. Thus, the highest number of tube failures expected is two. This value was determined by reviewing the existing information with respect to a condition monitoring and operational assessment (CMOA) type of approach considering:

- Load as a function of steam generator radius
- Past operating experience

- Evaluation of the operating experience for number of flaws (circumferential and volumetric) in the free span which would be expected to sever for the LBLOCA transient tube load.

This approach and the results must be discussed with the appropriate NRC reviewers (i.e., steam generator/mechanical).

2) Fuel Rod Cladding Rupture Study

The purpose of this task is to determine the number of fuel rod cladding failures expected following a double-ended hot leg break. In a letter to the B&WOG dated May 15, 2003, the NRC accepted the use of a realistic approach to determine the additional dose resulting from the primary-to-secondary tube leakage. (However, the results of this study are not to be used for existing LOCA dose analyses.) The results of this task indicate that no fuel failures will occur.

The footnote in 10CFR100.11 indicates that a maximum hypothetical accident (MHA) source term should be assumed for the LOCA dose analyses. However, the proposed approach in this case is to use the results of the fuel rod cladding rupture study to demonstrate that the source term identified in the footnote is not credible for the event postulated in this analysis. Credible is defined by evaluating the fission product barriers for each type of accident and showing that the dose is calculated based on which barriers remain intact. In this case, one or more steam generator tubes can fail and secondary side isolation can be lost due to single failure, but the fuel rod cladding remains intact. This is similar to the approach used for the Steam Generator Tube Rupture accident.

3) Monte Carlo Analysis

The purpose of this task is to combine the competing effects of the factors influencing the deterministic system analyses used to evaluate tube stress, NPSH, and dose. A Monte Carlo analysis is used because what is limiting for tube integrity is not necessarily limiting for NPSH or dose.

A program called Crystal Ball® is added to Microsoft® Excel to perform the Monte Carlo simulation. Input values are selected based upon distributions of the data for break size and day of the year, and a spreadsheet calculation is performed for each case to estimate tube-to-shell temperature and number of SG tubes failed. The method was demonstrated for a limited number of inputs using preliminary estimates.

Another Microsoft® Excel spreadsheet was developed for the purpose of extending the RELAP5 transient results in order to examine the timing of liquid spillover in the steam line following a steam generator tube failure. This spreadsheet was benchmarked to simplified RELAP5 models. This spreadsheet will be used to define the long-term primary-to-secondary leakage versus the number of tubes which may fail to obtain a time at which acceptable NPSH is lost. This spreadsheet is linked into the Crystal Ball® spreadsheet.

The Monte Carlo approach is still in development, but has already provided useful insights. It will be used to identify to cases to be evaluated in more detail with RELAP5.

If the number of failed steam generator tubes is very low, consistent with the tube failure results described above, then the use of Crystal Ball® may not be necessary. Instead, bounding input could be used in RELAP5 to develop acceptable tube failure probabilities and associated leakage. However, it was decided that input and methodology development for Crystal Ball® should continue until the tube failure results are discussed further with the NRC.

The following proposed approaches for developing the 50.46, Part 100 and tube failure arguments for the topical report were outlined during the meeting:

1) <u>50.46 Approach</u>
The following options are being considered:
   - No best estimate freespan tube failures
   - MSIV closure before NPSH is lost
   - DHR initiation before NPSH is lost

   For the latter two options the following issues would need to be addressed:
   - Probability of tube failures
   - Defense in Depth argument
   - No additional consideration for GSI-191

2) <u>Part 100 Approach</u>
   - Part 100 is met by establishing that the scenario is not credible for MHA source term, based on the clad rupture analysis
   - LBLOCA dose analysis remains bounding
   - Incremental tubesheet tube leakage
   - Defense in Depth argument

3) <u>Tube Failure Argument</u>
   - Steam generator inspection results
   - Define tube load vs. acceptable flaw size
   - Determine the number of tubes to fail
   - Probability of tube failure
   - Total probability of event with tube failure
   - Defense in Depth argument