

January 5, 2005

MEMORANDUM TO: Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Drew Holland, Project Manager, Section 1 */RA/*
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF MEETING HELD ON NOVEMBER 9 AND 10, 2004,
WITH THE BABCOCK AND WILCOX OWNERS GROUP TO DISCUSS
BAW-2374, "RISK-INFORMED ASSESSMENT OF ONCE THROUGH
STEAM GENERATOR TUBE THERMAL LOADS DUE TO BREAKS IN
REACTOR COOLANT SYSTEM UPPER HOT LEG LARGE BORE
PIPING"

The Babcock and Wilcox Owners Group (B&WOG) and NRC met in the Framatome ANP (FANP) offices in Lynchburg, Virginia, on November 9 and 10, 2004, during a closed meeting to discuss the subject topical report. The NRC was represented by W. Lyon and D. Holland. The B&WOG was represented by M. Byram (Entergy Operations, Inc.) and E. Henshaw (Duke Energy). FANP participants included J. Klingenfus, G. Wissinger, S. Levinson, G. Elliott, S. Sloan, D. Costa, J. Begley, and R. Schomaker.

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

The proposed 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 is added to address the long-term cooling criterion in 10 CFR 50.46. Section 3.6 is added to address the dose consequences above and beyond the existing loss-of-coolant accident (LOCA) dose analyses. Section 3.7 will address the water hammer potential resulting from primary fluid leaking into and filling up the secondary side of the once through steam generator (OTSG) and associated piping.

Proposed Table of Contents:

- 1.0 Introduction
- 2.0 Definition of Proposed Change
- 3.0 Engineering Analysis
 - 3.1 Compliance with Current Regulation
 - 3.2 Change is Consistent with Defense-in-Depth
 - 3.3 Change Preserves Sufficient Safety Margins
 - 3.4 Change in CDF and LERF is Small
 - 3.5 Long-Term Cooling
 - 3.5.1 Pressure differential as function of break size
 - 3.5.2 Mechanical loads
 - 3.5.3 Realistic tube flaw distribution
 - 3.5.4 Secondary side isolation
 - 3.5.5 Loss of ECCS inventory
 - 3.5.6 NPSH
 - 3.6 Dose Consequences
 - 3.6.1 Source term
 - 3.6.2 Transport of source term
 - 3.6.3 Dose evaluation
 - 3.7 Secondary Piping Integrity
 - 3.7.1 Steam Line
 - 3.7.2 Feedwater Line

The redevelopment effort for BAW-2374 is currently focused in the following three areas:

1) OTSG 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 OTSG inspection results is predicted to be very low. This information was determined by reviewing the existing information with respect to a condition monitoring and operational assessment type of approach considering:

- Load as a function of OTSG radius
- Past operating experience
- Evaluation of the operating experience for a number of flaws (circumferential and volumetric) in the free span, which would be expected to sever for the large break loss-of-coolant accident (LBLOCA) transient tube load.

2) Clad Rupture Study

The purpose of this task is to determine the number of clad failures expected following a double-ended hot leg break. In a letter to the B&WOG dated May 15, 2003, the NRC accepted 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 preliminary results of this task indicate that no fuel failures will occur.

The current fuel failure assumption for LOCA dose analyses is based upon the footnote in 10 CFR 100.11 which requires the assumption of a maximum hypothetical accident (MHA) source term. However, the proposed approach in this case is to use the results of the clad 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 OTSG tubes can fail and secondary side isolation may be lost due to single failure, but the clad remains intact. This is similar to the approach used for the OTSG tube rupture accident.

3) Monte Carlo Analysis

The purpose of this task is to combine the competing effects of factors influencing the deterministic system analyses used to evaluate tube stress, net positive suction head (NPSH) for the emergency core cooling system (ECCS), 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 OLE_LINK1Ball(r) (Crystal Ball(r)) is added to Microsoft(r) 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, as a spreadsheet calculation, is performed for each case to estimate tube-to-shell temperature and number of OTSG tubes failed. The method was demonstrated for a limited number of inputs using preliminary estimates.

Another Microsoft(r) Excel spreadsheet was developed for extending the RELAP5 transient results to examine the timing of liquid spillover in the steam line following a OTSG 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 number of tubes which may fail to obtain a time at which acceptable NPSH is lost. This spreadsheet is linked into the Crystal Ball(r) spreadsheet.

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

If the number of failed OTSG tubes is very low, consistent with the tube failure results described above, then the use of Crystal Ball(r) 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(r) should continue until the tube failure results are discussed further with the NRC.

The following proposed approaches for developing the 10 CFR 50.46, 10 CFR Part 100, and tube failure arguments for the topical report were outlined:

1) 10 CFR 50.46 Approach

The following options are being considered:

- No best estimate freespan tube failures
- Main steam isolation valve closure before NPSH is lost
- Decay heat removal system 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 (containment sump strainer clogging)

2) 10 CFR Part 100 Approach

- The regulation at 10 CFR 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 tube sheet tube leakage
- Defense in depth argument

3) Tube Failure Argument

- OTSG 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

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 appropriate NRC review groups for the new topical report. 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 in January 2005. Following the meeting, the staff expressed its appreciation to the B&WOG for the presentation. An attendance list is provided in the attachment. No regulatory decisions were made at the meeting. The slides used during the meeting are available in the Agencywide Documents Access and Management System under Accession Numbers ML043210521, ML043210527, and ML043210534.

Project No. 693

Attachment: Meeting Attendees

cc w/att:

Mr. James F. Mallay, Director
Regulatory Affairs
Framatome ANP
Babcock and Wilcox Owners Group
3315 Old Forest Road
Lynchburg, VA 24501

January 5, 2005

- Incremental tube sheet tube leakage
 - Defense in depth argument
- 3) Tube Failure Argument
- OTSG inspection results
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 Babcock and Wilcox Owners Group
 3315 Old Forest Road
 Lynchburg, VA 24501

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NRC-001

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OFFICE	PDIV-1/PM	PDIV-1/LA	SRXB*	SRXB/SC**	PDIV-2/SC
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DATE	12/29/04	12/29/04	1/3/05	1/3/05	1/5/05

MEETING ATTENDEES

MEETING WITH THE BABCOCK AND WILCOX OWNERS GROUP

November 9 and 10, 2004

BABCOCK AND WILCOX OWNERS GROUP

M. Byram (Entergy)
E. Henshaw (Duke)
J. Klingenfus (FANP)
G. Wissinger (FANP)
S. Levinson (FANP)
G. Elliott (FANP)
S. Sloan (FANP)
D. Costa (FANP)
J. Begley (FANP)
R. Schomaker (FANP)

NRC

D. Holland (NRR/DLPM)
W. Lyon (NRR/SRXB)