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B&W Owners Group "White Paper" Addressing Approach for Establishing Steam Generator Design Tube Loads for OTSGs

In February 2005, the B&WOG and NRC met to discuss the approach being taken by the B&WOG to demonstrate compliance of a LBLOCA scenario with 10 CFR 50.46 and 10 CFR 100. This demonstration was intended to support the present design basis for steam generator tube loads.

The B&WOG committed to prepare a "white paper" describing the basis for the approach being taken to establish the design basis for steam generator tube loads and provide the means by which this basis would be preserved for future operation.

Attached is the white paper for your review. The paper should form the basis for future discussion. We look forward to meeting with the NRC for further discussion on the proposed approach.

Sincerely,

Ronald L. Gardner

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White Paper on the OTSG Design and Licensing Basis

Executive Summary

This document presents the proposed design basis for the B&W plant steam generator tubes, tube-to-tubesheet joints, and tube repair products for the once-through steam generators (OTSG) or replacement OTSGs. The design basis is composed of a combination of both deterministic and best-estimate analyses. The deterministic design basis considers the main steam line break (MSLB), feedwater line break (FWLB), or limiting attached reactor coolant system (RCS) pipe loss-of-coolant accident (LOCA) break size for the qualification of the OTSG tubes, tube-to-tubesheet joints, and tube repair products. The best-estimate design basis includes realistic assessments of the structural and leakage integrity of the OTSG tubes when subjected to the limiting large break LOCA (LBLOCA) in the hot leg U-bend. The assessments include evaluation of the potential number of tubes that might fail under LBLOCA loading and the primary-to-secondary leakage associated with those failures. This approach is consistent with the "working" licensing basis as provided in the historical information regarding the initial OTSG design and testing and evolution of the OTSG licensing basis to where the plants are currently. There is also a commitment to ensure that any future as-found degradation fits within the parameters used to form this combined basis. This basis will be further developed and supported by material included in a revision to Topical Report BAW-2374. Included will be demonstrations that the consequences of the limiting large break LOCA (LBLOCA) in the hot leg U-bend with one or more consequential steam generator tube ruptures (SGTRs) are in compliance with the 10 CFR 50.46 acceptance criteria and also satisfy 10 CFR 100, or 10 CFR 50.67.

White Paper on the OTSG Design and Licensing Basis

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1. Introduction

The purpose of this document is to clearly establish the design basis for the steam generator tubes, tube-to-tubesheet joints, and tube repair products for the once-through steam generators (OTSG) or replacement OTSGs. As defined, the design basis will be made up of a combination of both deterministic and best-estimate analyses. The deterministic design basis considers the main steam line break (MSLB), feedwater line break (FWLB), or limiting attached reactor coolant system (RCS) pipe loss-of-coolant accident (LOCA) break size for the qualification of the OTSG tubes, tube-to-tubesheet joints, and tube repair products. The best-estimate design basis includes realistic assessments of the structural and leakage integrity of the OTSG tubes when subjected to the limiting large break LOCA (LBLOCA) in the hot leg U-bend. The assessments include an evaluation of the potential number of tubes that might fail under LBLOCA loading and the primary-to-secondary leakage associated with those failures. With one or more consequential steam generator tube ruptures (SGTRs) the demonstration of acceptable long-term core cooling and evaluation of the on- and off-site dose rates both have unique considerations that must be addressed. This best estimate evaluation will demonstrate that the consequences of the limiting hot leg LBLOCA will not violate the 10 CFR 50.46, 10 CFR 100, or 10 CFR 50.67 acceptance criteria due to the consequential tube failures.

The combination of deterministic and best-estimate design basis considerations is similar to that used for the current plant licensing basis, as provided by the historical background section. The background section includes discussions on the original OTSG testing and licensing basis and a summary of the evolution of the basis since plant operation began. It also discusses methods for ensuring that the assumptions made in establishing the design basis are preserved for the remaining life of the original OTSGs and replacement steam generators, and provides reasonable assurance for continued safe operations.

2. Historical Background

The OTSGs were designed for use in the B&W-designed pressurized water reactor systems. Three scaled models of the OTSG design were developed to test the following parameters:

1. Heat transfer and capacity tests,
2. Control and dynamic response tests,
3. Structural integrity and vibration tests,
4. Feedwater heating tests,
5. Secondary cleanliness tests,
6. Tube leak tests, and
7. Primary side flow tests.

The results of this testing performed in the 1960's were reported in BAW-10002 (Reference 2) and BAW-10002 Supplement 1 (Reference 3). The loss of external load, steam pipe failure, loss of station power, and loss of feedwater flow were simulated abnormal transient tests in BAW-10002. The primary side blowdown structural test was performed to evaluate the structural integrity of the SG tubes and the tube-to-tubesheet joints. No failures were revealed during this blowdown testing. Additional results for these tests were provided in Supplement 1 to BAW-10002 and additional information was added from the Oconee 1 field testing program.

In October 1980, Topical Report BAW-10146 (Reference 4) was issued. This Topical Report, BAW-10146, included the analytical and experimental justification for establishing a minimum acceptable steam generator tube wall thickness in accordance with the guidelines in the NRC draft Regulatory Guide 1.121. This report included tube loadings from normal operating and faulted conditions. The faulted conditions considered were a safe shutdown earthquake (SSE) in combination with either a loss of coolant accident (LOCA), a main steam line break (MSLB), or a feedwater line break (FWLB). Three double-ended guillotine LOCAs were considered. They included postulated breaks in the hot leg at the reactor vessel outlet nozzle, the cold leg pump discharge piping, and the cold leg pump suction piping. The double-ended guillotine break in the hot leg at the reactor vessel outlet nozzle was the most limiting at the end of blowdown. Hand calculations were used to estimate the maximum tube-to-shell temperature difference (ΔT) due to the effects of the cold auxiliary feedwater (AFW) injection (also referred to as emergency feedwater (EFW) injection). The MSLB produced the largest accident tube-to-shell ΔT and resulted in a limiting tensile tube load of 3140 lbs. The tube load for the LBLOCA at the reactor vessel outlet nozzle was calculated to be 2600 lbs.

In the early 1980's, the NRC and ACRS concluded that a probabilistic evaluation could be used on the main loop piping to eliminate from consideration certain design basis loads from RCS pipe LBLOCAs. This conclusion was supported by the fact that there were no known mechanisms in the PWR primary piping material for developing a large break without going through an extended period during which the crack would leak copiously. This framework was used to develop BAW-1847 (Reference 5) and BAW-1847, Revision 1 (Reference 6), which was approved by the NRC on December 12, 1985 for use on the B&W-designed plants.

Various reevaluations of the limiting abnormal transients were completed during the late 1980's and early 1990's. These evaluations, which began with the MSLB, LBLOCA, and FWLB from BAW-10146, excluded the LBLOCA loads because it was interpreted that the LBLOCA loads did not need to be considered based on leak-before-break methodology. Therefore, the MSLB accident transient was used to qualify the initial reroll repair methodology and other SG tube repair hardware and processes.

In 1998, a preliminary safety concern (PSC 2-98) was identified. The PSC identified that a break in one of the lines attached to the RC piping could result in a larger tube-to-shell ΔT and possible larger tube loads than those resulting from the MSLB event. To disposition this PSC, analyses of the limiting attached pipe break (the pressurizer surge line) were performed to define the limiting LOCA conditions for use in the tube load evaluations. As an extension of the calculation of new tube loads, plant specific MSLB analyses were also performed.

In March of 2000, a B&W designed plant owner submitted for a technical specification change on a tube reroll repair qualification. After the license change was submitted to the NRC, the Utility recognized that LBLOCA was listed in the plant FSAR as a tube load design basis event but was not explicitly discussed in the tube reroll repair qualification report supporting the license change submittal. Therefore, the utility initiated a condition report. The tube reroll repair qualification had credited LBB as a basis for not considering LBLOCA and therefore did not discuss LBLOCA in the report. The NRC reviewed the material and concluded that LBB could not be used to remove thermal loads associated with LBLOCA from the design basis because the thermal load is not a dynamic load. The NRC approved the reroll submittal, which did not consider LBLOCA, for the B&W-designed plants based on commitments to evaluate the leakage and dose from the as-found flaws on a best-estimate basis using estimated LBLOCA thermal loads until BAW-2374 was submitted, reviewed and approved. The intent of BAW-2374 was to show that all loads associated with the LBLOCA, including those other than dynamic loads, need not be considered in the design basis of the SG tubes and their repair products.

The B&W Owners Group prepared and submitted a topical report following a Regulatory Guide 1.174 approach for risk-informed evaluations in July 2000, (BAW-2374 Revision 0, Reference 7) to justify not including LBLOCA in the OTSG licensing basis. Meetings and discussions with the NRC during the review process identified that the topical report could not be accepted as written. The meetings helped identify the information that needed to be added and the restructuring that was needed based on the current interpretation of Regulatory Guide 1.174. Revision 0 of the topical was withdrawn, and it was restructured and additional information added based on NRC review. The material was submitted to the NRC in March 2001 via Revision 1 of BAW-2374 (Reference 1).

Upon receipt of the new material, the NRC reinitiated its review of the topical report. In December 2002, this review identified that demonstration of compliance to 10 CFR 50.46 long-term core cooling and 10 CFR 100 dose criteria was missing. After several phone calls and meetings, the topical report was withdrawn on March 13, 2003 and plans for a replacement topical were developed.

Work has continued since that time with periodic working meetings between the NRC and the B&WOG. Work progressed towards demonstrating compliance with 10 CFR 50.46 and satisfying 10 CFR 100 in a manner consistent with previous NRC review comments. A cladding rupture study was performed and concluded that no clad rupture would occur, which means no gap activity would contribute to dose consequences. In addition, a statistical evaluation was performed which predicted that a very small number of tubes would be expected to sever with rapidly decreasing probability as the number of affected tubes increased.

Submittal of a revised topical report was targeted for December 2005, however; based on the February 2005 meeting with NRC an extension may be needed. The B&WOG method for demonstrating compliance was questioned by the NRC SG mechanical group who wanted to keep the design basis based on a deterministic LBLOCA evaluation with full margins of safety. During the NRC-B&WOG meeting FANP had described that a full deterministic approach may lead to additional in-situ testing and personnel exposure that was not warranted based on the low risk of a LBLOCA. The validity of the in-situ testing was also questioned if the LBLOCA axial loads had to be applied. Uncertainties in the actual applied axial load due to tube deformations and stiffness could inadvertently result in loads several times the intended load being applied to the tube section in question. These potentially large test loads are difficult to monitor and could easily result in a tube failure during the in-situ test. For these reasons, FANP and the B&WOG believe that using the deterministic LBLOCA loads are unrealistic and are not appropriate as a future method for establishing the OTSG design basis.

3. OTSG Licensing Basis

Since the NRC acceptance of the LBB topical report (BAW-1847) in December 1985, the design and qualification of repair hardware and alternative repair criteria for the OTSGs have been based on limiting accident loads other than LBLOCA. For many years the limiting accident was the MSLB event. With the issue of PSC 2-98 in 1998 and subsequent evaluations of breaks in the smaller piping attached (SBLOCA) to the RCS pipes, the limiting accident became more plant specific and can be either MSLB, SBLOCA, or both depending on the criteria being evaluated (leakage vs. axial load). Many of these qualifications, such as the Alternate Repair Criteria (ARC) for tube end cracking, and repair roll methodology have been approved by the NRC for use in the OTSG plants.

With acceptance of the revision of BAW-2374 as it is intended to be submitted as described in this paper, the current design basis that has been used for the SG repair products for the past 20 years will remain valid and any future products would be evaluated using the limiting accident transients from any pipe breaks other than those in the main RCS piping (LBLOCA). These pipe breaks include breaks in the secondary side main steam line and feedwater line and primary attached RCS pipe breaks such as the pressurizer surge line, continuous head vent line, decay heat drop line, and core flood line. A detailed evaluation of the various pipes and locations has been performed to establish which breaks create the largest tube loads. The limiting breaks, (the main steam line break and pressurizer surge line break) are currently being used as the design basis accident loads for the steam generator and its repair products.

The OTSG or replacement OTSG deterministic design basis will be defined as the MSLB, FWLB, or limiting attached RCS pipe LOCA break size. For these design basis accidents, rigorous deterministic analyses will continue to be performed with the full margins of safety to show the structural and leakage integrity are satisfied for tubes, tube-to-tubesheet joints, and tube repair products. Compliance is based upon meeting structural requirements, no tube rupture and the leakage requirements according to applicable sections of the ASME Code, and the new Performance Criteria as included in NEI97-06 and the updated Generic License Change Package.

To assess the potential impact on steam generator structural and leakage integrity in the unlikely event of a LBLOCA, a detailed best estimate operability assessment of the SGs has been performed. Although prepared to assist the utilities in performing their own plant specific assessments, the document provides a comprehensive assessment of the SGs and their repair products. The document includes:

- definition of LBLOCA event and estimation of limiting tube-to-shell ΔT and primary-to-secondary ΔP,
- estimation of limiting loads and dilations, and
- assessment of structural and leakage integrity.

The operability assessment provides information that shows by best estimate practices that the steam generator structural and leakage integrity would be maintained in the event of a LBLOCA.

The only thing that could potentially affect this conclusion is a potential increase in the number and size of circumferential flaws in the OTSGs. As discussed in the following sections, this potential scenario is addressed in BAW-2374 and by commitments by the utilities or license conditions to perform a defined assessment of outage inspection results for LBLOCA loadings.

4. Supplemental Licensing Basis for LBLOCA

To address the safety significance and consequences of the low probability break in the hot leg U-bend (or LBLOCA break above surge line attachment to hot leg), detailed assessments for long-term core cooling and dose consideration as defined in 10 CFR 50.46 and 10 CFR 100 and 50.67 (Alternate Source Criteria) will be performed. The results of the evaluations will be provided in BAW-2374. BAW-2374 will be used to justify and define realistic analyses methods or evaluations for the LBLOCA at the top of the hot leg U-bend piping to demonstrate compliance to affected criteria (10 CFR 50.46, 100, 50.67 etc.). The evaluations for compliance will be based on best estimate projections. SGTR and leakage beyond the normal design basis allowable limits will be allowed with a LBLOCA in the RCS piping. Calculations will be prepared to show an acceptable number of tube severs that can be tolerated while maintaining assurance that the limits of 10 CFR 50.46 and 10 CFR 100 (and 50.67) are maintained. The leakage calculated from this analysis will become the basis for future operability evaluations.

A LBLOCA leakage assessment will be performed prior to commencing power operation from each refueling outage using plant specific inspection data to show that the as-found degradation in the SG tubes would not result in failure (sever) of more tubes than determined to be acceptable in the compliance evaluations. A brief summary of the best-estimate operability assessment performed for the SG hardware subjected to a LBLOCA will be provided.

The typical current limit of 1.0 gpm will continue to be the allowable accident leakage from all other events besides the LBLOCA event. Acceptable long-term core cooling based on the as-found flaws and best-estimate leakage considerations will be used as the alternate acceptance criteria for LBLOCA.

5. Future Commitments

To assure that potential increases in number and size of circumferential indications in the OTSG tubing would remain bounded by the conclusions of BAW-2374, the utilities will be required to provide plant specific assessments of the circumferential degradation for limiting LBLOCA loadings each outage. The structural integrity assessment will be performed in conjunction with the current condition monitoring and operability assessments (CMOA) similar to the way the "for information" LBLOCA leakage assessments are currently performed. The evaluation will look at outage inspection techniques, inspection results, and growth rates to determine the probability of single and multiple tube failures. The results will be compared to those used in BAW-2374 for determination of acceptability.

The best approach for incorporating the LBLOCA assessment into the licensing basis is to use a Technical Specification change. Time permitting, the LBLOCA assessment requirement may be tied with the Generic License Change Package (GLCP) submittal.

6. Conclusions

Reasonable assurance is provided by maintaining the design basis as the worst combination of MSLB, attached pipe LOCA or FWLB for the steam generator tubes, tube-to-tubesheet joints, and tube repair products. The licensing basis will be expanded to include a best-estimate analyses of the most limiting hot leg U-bend LOCA to demonstrate compliance with all regulations. This combination of both deterministic and realistic analyses will be continued in future evaluations to demonstrate that the number and nature of as-found flaws can remain acceptable.

7. References

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2. FANP Proprietary Topical Report BAW-10002, "Once-Through Steam Generator Research and Development Report," August 1969.
3. FANP Proprietary Topical Report BAW-10002, Supplement 1, "Once-Through Steam Generator Research and Development Report," June 1970.
4. FANP Proprietary Topical Report BAW-10146, "Determination of Minimum Required Tube Wall Thickness for 177-FA Once-Through Steam Generators," October 1980.
5. FANP Proprietary Topical Report BAW-1847, "The B&W Owners Group Leak-Before Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W-Designed NSS," September, 1984.
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7. FANP Proprietary Topical Report BAW-2374, "Justification for Not Including Postulated Breaks in Large Bore RCS Piping in the Licensing Basis for Existing and Replacement OTSGs," July, 2000.