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Attn: Mr. Stewart N. Bailey

Subject: Transmittal of Final Report on the Evaluation of PSC 2-98 Relating to Design Basis OTSG Tube Tensile Loads.

Reference:

1. PSC 2-98, "Design Basis OTSG Tube Tensile Loads", filed 205 T4.4, 8/20/98
2. Letter to USNRC, "Report of Preliminary Safety Concern Related to Design Steam Generator Tube Tensile Loads (PSC 2-98), October 19, 1998
3. GPU Nuclear Letter #6710-97-2441, "Leakage Assessment Methodology for TMI-1 Once-Through-Steam-Generator (OTSG) Kinetic Expansion Examination", James W. Langenbach to U.S. Nuclear Regulatory Commission, November 26, 1997.
4. B&W Owners Group Letter #OG-1792, Submittal of Topical Report BAW-2374, 'Justification for Not Including Postulated Breaks in Large Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through-Steam Generators,"', David J. Firth to Document Control Desk, U.S. Nuclear Regulatory Commission, July 7, 2000.

Gentlemen:

The purpose of this report is to advise you of the final results of the evaluation of a potential safety concern (PSC 2-98) documented in Reference 1, and reported in Reference 2. The concern relates to the potential situation where the Once Through Steam Generator (OTSG) tube tensile loads resulting from a Small Break Loss of Coolant Accident (SBLOCA) may be larger than the currently recognized limiting load. The limiting load was originally established as resulting from a Main Steam Line Break (MSLB).

This report documents the results of the evaluation of PSC 2-98, which includes the determination of limiting loads that may result from SBLOCA, MSLB, as well as an examination

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of other events. The operability of the steam generator and impact of loads on steam generator repair products was also assessed in the evaluation of the safety concern.

A. Discovery and Cause

In August of 1998, FTI notified the B&WOG utilities of a concern that the current limiting OTSG tube loads based on the MSLB accident may be exceeded by a different more limiting transient. This transient is the Small Break Loss of Coolant Accident (SBLOCA). This concern resulted in the issuance of Preliminary Safety Concern (PSC 2-98), Reference 1. The PSC was reported to the NRC in Reference 2. Evaluation of the reportability of the concern under the criteria of 10CFR Part 21 required evaluation of the defined event with respect to the structural integrity of the steam generators and its potential effects on the health and safety of the public. SBLOCA thermal hydraulic and structural analyses have been performed under this evaluation. An operability assessment addressing the effects of the SBLOCA event on the structural integrity of the OTSG and on public safety has also been completed.

Additionally, because the SBLOCA had the potential for becoming the more limiting event in both tube load and dose consequence aspects, the analyses for the previous limiting accident, Main Steam Line Break (MSLB), were revisited. Therefore, the Main Steam Line Break (MSLB) thermal-hydraulic and structural analyses have been re-performed and an operability assessment addressing the effects of the MSLB event on the structural integrity of the OTSG and on public safety has also been completed.

At the time in history when the MSLB accident was first established as the limiting design basis event for OTSG tube tensile loads, the LOCA event was considered bounded by the MSLB event based on testing results. Advancements in the state of the art methods for performing complex analyses and the results from later selecting different break locations for evaluation resulted in the issuance of this preliminary safety concern.

B. SBLOCA Evaluation

The LOCA being evaluated for impact on OTSG tube tensile loads under PSC 2-98 is defined specifically as a double ended rupture of the pressurizer surge line. This break size is considered the largest credible break size when not considering a break in the main reactor coolant system piping. Since PSC 2-98 specifically concerns the OTSG tube tensile loads resulting from a SBLOCA in the pressurizer surge line, the results of that evaluation are submitted herein.

As part of the evaluation of SBLOCA tube loads, the double ended rupture of the pressurizer surge line for the 177 fuel assembly (177FA) lowered loop plants was analyzed. The inputs and assumptions of the analyses bound the response for all the lowered loop 177FA plants (ANO-1, CR-3, Oconee-1, 2, &3, and TMI-1). For the raised loop plant, DB-1, a double ended rupture of the continuous head vent line (between the steam generator upper plenum and reactor vessel head) and the pressurizer surge line were analyzed.

The purpose of the SBLOCA analyses was to determine conservative primary-to-secondary pressure and tube-to-shell delta T responses for input to OTSG tube load calculations. Sensitivity studies showed that the assumption of minimum Emergency Feedwater (EFW) flow minimizes the cooling of the shell, maximizes the tube-to-shell delta T predictions, and thus maximizes tube loads. The minimum 200 gpm EFW flow has also been approved by the plants and is consistent with the emergency operating procedures during plant LOCA events. The OTSG thermal hydraulic plant transient data (fluid temperatures, film coefficients, metal temperatures, and system pressures) were then used as boundary conditions to calculate the axial tube loads.

The maximum calculated SBLOCA-induced axial tensile load on any tube in a lowered-loop OTSG with up to 25% of the tubes plugged is 2097 lbs. The maximum calculated SBLOCA-induced axial tensile load on any tube in a raised-loop OTSG (Davis-Besse Unit 1) with up to 25% of the tubes plugged is 2276 lbs. for the continuous head vent line break. These loads are higher than the MSLB axial tube load (1402 lbs.) last used as a bounding load for evaluating OTSG in-service test methods, inspection acceptance criteria and applied repair products for all plants except for the Oconee Units, which used 2376 lbs. Therefore, an assessment of the impact of this higher load on the operating plants during the SBLOCA event was performed.

The analyzed SBLOCA accident imposes forces that result in minor deformation of steam generator structural components. The flexure of the tubesheets during the transient, which occurs primarily due to the axial tube loads resulting from differential pressures and temperatures, was also determined. This flexure causes the tubesheet holes in some locations to dilate. The hole dilation has a potential effect on the qualification of repair products, such as plugs and sleeves, which attach to the inside diameter (ID) of the tube or tubesheet hole. The resulting hole dilation was also evaluated.

As stated above, the bounding SBLOCA maximum axial tube load is 2276 lbs. and occurs at the periphery of the tube bundle. The bounding maximum tubesheet hole delta dilation, based on diameter, is 2.39 mils in the radial direction with a coincident 0.45 mils in the tangential direction and occurs near the periphery in the lower tubesheet. (The delta dilation is the difference between the expected inside diameter of a tubesheet bore hole and the expected outside diameter of a tube within that tubesheet bore hole.)

The OTSG components potentially affected by the SBLOCA tube loads were identified and each component was evaluated with respect to the calculated tube loads and tube sheet hole dilations. An operability assessment has evaluated resultant leakage and pullout strengths due to hole dilation. For axial tube loads the critical tube degradation is circumferentially oriented flaws and volumetric flaws such as IGA. Axial flaw degradation is not limited by the tube load resulting from SBLOCA. Tension tests of tubes with flaws of varying circumferential extent showed that the OTSG tubing has adequate strength during the SBLOCA event to prevent tensile failure. IGA degradation has been examined and determined to not pose a risk to tube integrity relative to the SBLOCA axial load. Based on the results of the evaluations, the SBLOCA transient is concluded to have no significant impact on the structural integrity of any of the evaluated components.

Therefore it is concluded that the reanalysis of the SBLOCA transient for plants with OTSG's does not indicate an increased risk to the health and safety of the public.

C. MSLB Evaluation

Plant-specific MSLB thermal hydraulic analyses were also performed by FTI for CR-3, ANO-1, and DB-1. Duke Energy had previously prepared the MSLB thermal-hydraulic analysis for their three Oconee units. TMI-1 performed a conservative reanalysis of the Main Steam Line Break transient in 1997.

The TMI-1 MSLB analysis resulted in peak axial, tensile loads of 1310 lbs. on tubes in the periphery of the TMI-1 steam generator tube bundles. In addition to tube loads, the TMI-1 calculations included dilations, contact pressures, and leak rates from postulated flaws within the upper tube sheet kinetic expansion tube-to-tubesheet joints. The methodology and results of these MSLB analyses were submitted to the NRC in Reference 3.

The results of the thermal-hydraulic analyses prepared by FTI for the CR-3, ANO-1 and DB-1 plants were used as direct input to the structural analysis of the steam generators. The structural analysis involved the determination of tube loads and tube/tubesheet hole delta dilations for each transient. (TMI's MSLB induced peak axial tube loads and tubesheet hole delta dilations are bounded by those described herein for the other B&W designed plants. Therefore the assessments discussed for the steam generators and repair hardware are also applicable to TMI-1.)

Plant-specific MSLB analyses were performed because of differences in plant equipment that affect the plants' primary and secondary side responses during the event. Equipment differences may include, but are not limited to, feedwater (FW) pumps (flow rates), FW temperatures, valve performance (closure times), and FW isolation control (Feed Only Good Generator Logic, FOGG). In addition, assumptions relative to loss of offsite power (LOOP) can have a significant effect on the responses of the primary and secondary sides. The different responses result in differences in tube-to-shell ΔTs (temperatures) and primary-to-secondary ΔPs (pressures) for each of the plants. The ΔTs and ΔPs affect the tube axial loads and the tubesheet hole dilations.

Like the SBLOCA analyses, the purpose of the MSLB analyses was to determine conservative primary-to-secondary pressure and tube-to-shell delta T responses for input to OTSG tube load calculations. The OTSG thermal-hydraulic time history data (fluid temperatures, film coefficients, metal temperatures, and system pressures) from the thermal-hydraulic analyses were used as direct input to the structural analysis of the OTSG. The structural analysis was performed to calculate the axial tube loads and tubesheet hole dilations.

A summary of the maximum tube loads for the applicable MSLB transients are reported in Table 2-1. The table summarizes the maximum loads for each of the cases evaluated. The cases include a) the affected and unaffected steam generators, b) plugged and unplugged configurations and c) various transient times. The axial tube loads are the values at the periphery of the tube bundle, which are the maximums.

The maximum MSLB tube load for any of the plants analyzed is 3380 lbs. and occurs at the periphery of the bundle for the DB-1 plant-specific event. This load is mostly the result of local cooling caused by the emergency feedwater (EFW) wetting. The maximum load for the un-wetted tubes for DB-1 is less than 900 lbs.

The bounding maximum tubesheet-to-tube Δ dilation, based on diameter, is 2.3 mils in the radial direction accompanied by 0.1 mils in the tangential direction. The maximum dilation occurs at the periphery in the lower tubesheet for the Oconee transient.

The reported dilations represent the dilation between the tube and tubesheet bore and are directly applicable to the original tube-to-tubesheet rolled joint as well as tube re-rolls. They are indirectly applicable to plug and sleeve rolled joints within the tubesheet region of the tube and have no effect on repair products outside the tubesheet region of the tube.

As stated, the dilations reported in the table represent the maximum tubesheet-to-tube Δ dilation during the transient. The Δ dilations vary significantly over the 10 to 15 minutes of analyzed transient time with the maximum occurring over a very short period. As the transient progresses beyond the time of maximum dilation, the dilations will decrease and eventually approach the steady state compressive values.

The initiation of cold EFW into the OTSG can cause significant cooling of the wetted tubes. The cooling of the bundle's periphery tubes can result in tube loads in excess of those for the non-wetted tubes. The effect of EFW in the affected (broken) steam generator is much more severe than that in the unaffected (intact) steam generator. The difference in maximum tube-to-shell ΔT between the wetted and un-wetted tubes in the unaffected OTSG is only a few degrees. The small difference in temperature is considered negligible and, therefore, no separate analysis of the wetted tubes in the unaffected OTSG was performed.

For those plants with FOGG-logic (CR-3 and ANO-1) there is little or no injection of EFW into the affected steam generator. Therefore, there is no local cooling of the periphery tubes. For those plants that do not currently have FOGG logic, the injection of EFW into the affected steam generator can result in significant tube cooling and axial tube loads. These local tube loads do not significantly affect tubesheet bow or tubesheet hole dilations. The amount of tube cooling and resulting axial load for the DB-1 transient is amplified by the tripping of the reactor coolant pumps during the transient. The loss of primary flow allows the wetted tubes to cool much more than if the warmer RCS fluid continued to flow through the tubes.

Although the axial load in a plugged tube is typically greater than that of a non-plugged tube for a MSLB transient, the plugged tube load is not used in assessing the structural integrity of the OTSG. Since the tube in question has been plugged, and stabilized where appropriate, failure of the tube will not have a detrimental effect on adjacent tubes specifically, and on the integrity of the OTSG, generally.

There are two categories of items potentially affected by higher OTSG tube loads and/or greater tubesheet hole dilations. These categories are:

- the structural and leakage integrity of tubes, tube-to-tubesheet roll joints, tube-to-tubesheet welds, and
- the structural and leakage integrity of tube repair products.

Based on a detailed review of the OTSG's and their repair hardware, it was concluded that the following items required evaluation or disposition for the effects of the new loads and dilations. The specific items requiring assessment included:

- Undegraded Tubes
- Original Fabrication Tube-to-Tubesheet Rolled Joint
- Original Fabrication Tube-to-Tubesheet Fillet Weld
- OTSG Tube-to-Tubesheet Re-roll Repair (Topical Report)
- 80-inch and 31-inch Rolled Mechanical Sleeves, and Oconee Hydraulic Sleeve
- Tube Plugs (both Mechanical and Welded)
- Tube Stabilizers
- In-service Degraded Tubes
- Tube-to-Tubesheet Repair Welds

An assessment of each of these items has been performed. The assessment evaluated the OTSG for the effects of the newly calculated MSLB tube loads that may be higher than the previous loads used to qualify the item. In addition, the effects of newly calculated tubesheet hole dilations on joint strength and leakage have also been evaluated. The assessment also evaluated the structural strength and leakage integrity of the hardware relative to its impact on health and public safety.

The assessment also evaluated the structural strength and leakage integrity of the hardware relative to its impact on health and public safety.

The OTSG components and related repair hardware potentially affected by the newly calculated MSLB tube loads and dilations have been identified. The OTSG and its repair hardware have been evaluated with respect to the calculated tube loads and tubesheet hole dilations resulting from the plant specific MSLB event. Based on the results of this evaluation, it is concluded that the MSLB transient will have no significant impact on the structural integrity of the steam generator or its repair hardware.

Therefore, it is concluded that the reanalysis of the MSLB transient for the plants with OTSG's does not indicate an increased risk to the health and public safety.

(It is important to note that OTSG repairs and analyses performed by vendors other than FTI are not addressed in this evaluation.)

Table 2-1

MSLB Maximum Tube Loads and Diametrical Tube-to-Tubesheet Δ Dilations

Plant	Time (hr),	SG Plugging status, 0 or 25%	Axia l Tube Load (lbs)	Diametrical Tube-to-Tubesheet Δ Dilations (mils)			
				Upper Tubesheet Primary Face		Lower Tubesheet Primary Face	
				Δ radial	Δ circ	Δ radial	Δ circ
AFFECTED STEAM GENERATOR (WITH BROKEN STEAM LINE)							
CR	0.0233	Unplugged	**870**	1.3	-0.1	0.9	-0.3
	0.1967	Unplugged	860	0.8	-0.5	0.7	-0.6
	0.0233	Plugged	780	1.4	-0.1	1.0	-0.3
	0.1967	Plugged	660	1.0	-0.6	0.9	-0.7
ANO	0.0117	Unplugged	560	1.0	-0.3	0.8	-0.3
	0.2018	Unplugged	**670**	0.6	-0.6	0.6	-0.7
	0.0117	Plugged	450	1.1	-0.3	0.9	-0.3
	0.2018	Plugged	640	0.7	-0.6	0.6	-0.7
DB	0.1667	Unplugged	**3380**	0.9	-0.5	1.5	-0.2
	0.3088	Unplugged	350	0.4	-0.6	0.2	-0.7
	0.1667	Plugged	3300	1.0	-0.5	1.6	-0.3
	0.3088	Plugged	310	0.4	-0.6	0.2	-0.7
ONS	0.1681	Unplugged	2720	2.1	-0.3	2.3	0.0
	0.1681	Plugged	**2870**	2.0	-0.3	2.3	0.1
UNAFFECTED STEAM GENERATOR (INTACT STEAM GENERATOR)							
CR	0.0342	Unplugged	970	1.2	0.1	1.0	-0.2
	0.1967	Unplugged	1050	0.8	-0.5	0.8	-0.6
	0.0342	Plugged	990	1.2	0.1	1.0	-0.2
	0.1967	Plugged	**1080**	0.8	-0.5	0.7	-0.6
ANO	0.1194	Unplugged	**960**	0.8	-0.3	0.9	-0.3
	0.2018	Unplugged	740	0.7	-0.5	0.7	-0.5
	0.1194	Plugged	940	0.8	-0.3	0.9	-0.3
	0.2018	Plugged	720	0.7	-0.5	0.7	-0.5
DB-1	0.2731	Unplugged	660	0.4	-0.6	0.8	-0.4
	0.3088	Unplugged	610	0.4	-0.6	0.7	-0.5
	0.2731	Plugged	650	0.4	-0.6	0.9	-0.5
	0.3088	Plugged	590	0.4	-0.6	0.7	-0.5
ONS	Duke provided data for the broken steam generator only since this generator experiences a significantly higher tube to shell temperature difference.						

D. Evaluation of Other Events

The evaluation of PSC 2-98 has analytically demonstrated that the pressurizer surge line break is a severe transient with respect to OTSG tube overcooling. For those plants that have FOGG logic, the pressurizer surge line break predicts more limiting tube-to-shell temperature differences that those predicted for the main steam line break (MSLB) accident. This determination necessitated a reevaluation of other plausible plant transients. Although it is believed that the limiting events for thermal loading (SBLOCA and MSLB) have been determined and analyzed as part of the evaluation of PSC 2-98, an examination of other potential events has been performed and the disposition has been documented.

Overcooling transients and faulted conditions identified for evaluation have been shown to be bounded by either the MSLB, the pressurizer surge line break or continuous upper head vent line break (DB-1 only) events. Several additional events, including a break in the decay heat drop-line, were also considered and shown to be less limiting. The Large Break LOCA events evaluated were limited to breaks in the attached RCS piping and did not include consideration of breaks in the main RCS piping. As discussed with the NRC staff the basis for not including the main RCS piping was addressed in a separate B&WOG document, Reference 4.

For the purposes of the previously defined scope of this Preliminary Safety Concern (PSC 2-98), it is therefore concluded that the limiting overcooling events are the MSLB, pressurizer surge line break, and continuous upper head vent line break (DB-1 only) with respect to maximum tube-to-shell temperature and primary to secondary pressure differences.

E. Conclusions

As a result of the evaluations, the concerns of PSC 2-98 do not constitute a defect or substantial safety hazard as defined in 10CFR Part 21. An adequate margin of safety is maintained. This concern is not reportable under Part 21.

If there are any question on the evaluation results please contact Robert Schomaker at 804-832-2917 or the undersigned at 804-832-2964.

Sincerely,

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