NUCLEAR ENERGY INSTITUTE

Alexander Marion
DIRECTOR
ENGINEERING DEPARTMENT
NUCLEAR GENERATION DIVISION

December 21, 2001

Dr. Brian W. Sheron
Associate Director for Project Licensing and Technical Analysis
Office of Nuclear Reactor Regulation
Mail Stop O5-E7
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Generic Implications of the TMI Steam Generator Tube Sever Issue

PROJECT NUMBER: 689

Dear Dr. Sheron:

In your November 26, 2001, letter you requested that NEI organize technical and management meetings to discuss industry actions in response to the TMI tube sever experience. At a November 28th senior management meeting, NEI accepted this responsibility and presented the industry's initial observations on the implications of the issue. This letter describes current industry actions and provides a basis for the acceptability of continuing our in-depth investigation concurrent with continued operation.

The industry has begun the process of determining the activities necessary to evaluate the generic implications of the steam generator tube sever issue at TMI. As we discussed on November 28th, we intend to meet with your staff in late January to present the details of our plan. In the interim, we have completed an evaluation of the significance of the TMI experience. This evaluation has been prepared separately for the "once through" and the "recirculating" steam generator designs. These evaluations show that the TMI tube sever issue is not of immediate concern and that it is acceptable to devote the time necessary to implement the industry's action plan. Copies of the EPRI Steam Generator Management Project and B&W Owners Group letters are enclosed.

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1776 I STREET, NW SUITE 400 WASHINGTON, DC 20006-3708 PHONE 202.739.8080 FAX 202.533.0164 am@nei.org

If you have any questions on this matter please contact Jim Riley (202) 739-8137, jhr@nei.org or me.

Sincerely,

Alex Marion

Alex Marion

JHR/maa
Enclosures

c: Mr. Jack R. Strosnider, Jr, U. S. Nuclear Regulatory Commission
Ms. Louise Lund, U. S. Nuclear Regulatory Commission
Mr. Kenneth Karwoski, U. S. Nuclear Regulatory Commission
Mr. Emmett Murphy, U. S. Nuclear Regulatory Commission

Enclosure 1

EPRI SGMP:

Swollen Tube Effects In Recirculating Steam Generators

Pacific Gas and Electric Company

Lawrence F. Womack
Vice President
Nuclear Services

Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

805.545.4600
Fax: 805.545.4234

December 19, 2001

Mr. Ralph Beedle
Senior Vice President and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street N.W., Suite 400
Washington, DC 20006-3708

Dear Mr. Beedle:

Swollen Tube Effects in Recirculating Steam Generators

This letter forwards a technical justification from the EPRI Steam Generator Management Program concerning the timing of generic actions related to the recent steam generator tube damage event at TMI for non-OTSG steam generators. This was requested by the NRC at an NRC/industry senior management meeting in November. This letter justifies no need for immediate action. We intend to meet with your staff to present a longer term generic approach in January 2002.

Introduction – TMI Event Description & Root Cause

During the 14th refueling outage at the Three Mile Island Unit 1 (TMI) nuclear power plant, four steam generator (SG) tubes were found with wear scars as a result of being abraded by a neighboring plugged tube which had become severed at the bottom of the upper tubesheet during operation. The root cause of the severing of the affected tube was reported to be a sequence of swelling of the plugged tube (which had no throughwall degradation) which led to increased axial stress at the upper tubesheet and tube support plate (TSP) intersections, and which made the tube more susceptible to flow induced vibration (FIV), the combination of which led to a subsequent circumferential fatigue failure, Reference 1. The swelling of the affected tube was concluded to have been caused by the thermal expansion of water which had become trapped within the tube during the lower temperature portion of the cyclic operation of the plant. The subsequent increase in temperature to operating conditions causes swelling if the tube is full or nearly full because the thermal expansion of water is greater than that of the tube material.

The NRC staff requested the support of the Nuclear Energy Institute (NEI) "in organizing technical and management meetings between the nuclear industry, including the

pressurized water reactor (PWR) owner's groups, and the Nuclear Regulatory Commission (NRC). These meetings would serve to facilitate discussions on what actions, if any, the industry plans on taking in response to the TMI-1 experience," Reference 2. In subsequent discussions with the NRC staff, Reference 3, representatives from NEI, EPRI, the utilities and nuclear plant supplier engineering organizations expressed the opinions that:

1) the TMI issue is not an immediate problem for the entire nuclear power industry,
2) plants should continue to operate, and
3) there is sufficient time for the performance of a thorough evaluation of the TMI issue to determine its applicability, if any, to the remaining operating plants.

The purpose of this document is to present information supporting that opinion relative to recirculating or U-tube type steam generators (RSGs). The TMI SGs are of a once-through configuration and there are significant differences in design and operating characteristics which favorably influence the structural response behavior of the tubes in RSGs during operation.

Swollen Tube Events in Recirculating Steam Generators

The permanent deformation of SG tubes due to the thermal expansion of entrapped water was apparently first reported to have occurred at the Obrigheim plant in Germany in 1973, Reference 4. In that instance water was trapped in a closed crevice between the tube and tubesheet and resulted in radially inward buckling of the tube. One of the first observations of swelling and bursting of plugged tubes was reported at Ringhals 2 in 1984, Reference 5. A total of six row 1 tubes that had been preventatively plugged in 1979 using Westinghouse explosive plugs were reported to be swollen and one of those was reported to be axially ruptured. A generic evaluation of the phenomenon for all plants with Westinghouse explosive plugs was documented in Reference 6. Two axially ruptured tubes were found in row 1 of SG 4 at Salem 1 in 1991. The tubes had been preventatively plugged with explosive plugs in 1977, and in each case the location of the rupture was at the location of a wear scar from the tube lane blocking device. The wear mechanism was not active because the device had been removed in 1988. A presentation was made to the NRC staff in April of 1991, Reference 7. A letter was prepared for

distribution to all utilities with plants judged to be potentially affected by the phenomenon, Reference 8, containing a technical description summary, an assessment of the safety significance, and a discussion of reportability considerations. A copy of the same technical assessment was transmitted to the NRC staff via Reference 9. The issues associated with these tubes were found to be not safety significant. At that time there were no known occurrences associated with Westinghouse rolled, welded, or mechanical plugs.

In October of 1992, Electricité de France (EdF) reported finding a swollen and axially ruptured plugged tube in row 1 of SG 1 at Dampierre 2, Reference 10. The tube had been preventatively removed from service in 1985 using a Framatome mechanical plug of a design licensed from Westinghouse. The plug had been fabricated from Alloy 600TT material and a remedial modification of the plug had been made at the prior maintenance outage because of the potential for cracking of mechanical plugs. Two axially ruptured plugged tubes were also discovered by EdF at Tricastin 1 in 1993, Reference 11. At Tricastin, one of the tubes was reported as not swollen and the other was reported as swollen. The implication is that one of the tubes had degradation of a depth that it burst before uniform plastic deformation of the tube occurred. In this instance the tubes had been removed from service using welded hollow conical plugs. Throughwall axial cracks were found in the plugs in both instances. It was also noted in Reference 11 that a total of 2,934 mechanical plugs had been inspected, that 77 of those were judged to be not leak tight, that 55 tubes were inspected after removing the plug(s) and that 14 tubes were found to be swollen, but that no tubes were found to be ruptured.

In 1995, an axially ruptured row 1 tube was found in SG 4 at the Haddam Neck Nuclear Power Plant, a.k.a. Connecticut Yankee, Reference 12. The tube had been removed from service using a Westinghouse mechanical plug fabricated from Alloy 600TT material. A remedial modification was made in 1989 by installing a Babcock & Wilcox Company "plug-a-plug" in the mechanical plug because of potential stress corrosion cracking concerns. It is of interest to note that the tube became bent upon bursting and elastically contacted a neighboring tube, leaving a small dent mark on the surface. There was no tube-to-tube contact during the operation of the plant subsequent to the rupture because the adjacent tube did not exhibit any evidence of wear. Finally, an axially ruptured plugged tube was found in SG 4 at the Indian Point 2 plant in 2000. It was also a row 1 tube, but had been plugged in 1973 with a welded plug in each leg.

In summary, tubes that have been found to be swollen and axially ruptured in RSGs have been plugged with explosively welded, rolled, and fusion welded plugs, and in tubes with mechanical plugs, including those which have been subsequently plugged with a plug sealing device. It is not possible to eliminate the plug diode effect by a review of plug types. In the above instances no significant damage to in service tubes has occurred and no circumferential ruptures or cracking or fatigue failures have ever been suspected or observed in RSGs.

Flow Induced Vibration & Fatigue Failure of SG Tubes

The requisite condition for a plugged tube to cause damage to neighboring tubes is that its vibration amplitude must be large enough for contact with the neighboring tubes to occur. This apparently only happens when the plugged tube becomes severed as did the tube in the TMI SG; severed tubes would be expected to be fluidelastically unstable and to exhibit large vibration amplitudes. In order for a tube to sever it must have a combination of reduced damping and environmental conditions conducive to inducing significant amplitude fluidelastic instability. Axial cracking would not be expected from flow induced vibration because the fatigue damage is caused by the alternating bending stress.

There are two significant flow field excitation mechanisms that affect the tubes in RSGs, turbulence and fluidelastic. Other mechanisms such as vortex shedding are not significant. In general, turbulence excitation in SGs is not sufficient to result in tube fatigue damage to the extent that circumferential cracking and severing of the tube would be expected to occur unless there is significant circumferential cracking present to begin with.

Large vibration amplitudes occur if the tube interacts with the flow such that a significant amount of energy is absorbed from the flow field, i.e., fluidelastic interaction. Very large vibration amplitudes can occur if the cross-flow velocity is greater than a critical effective value, as determined from the configuration dimensions, the fluid density, and the structural damping. The ratio of the flow velocity to the critical velocity is known as the stability ratio (SR). If the SR is greater than unity, the tube is termed to be fluidelastically unstable. A SR greater than one, however, does not mean that the amplitude of vibration increases without bound. In this regard the phenomenon is not

akin to elastic buckling. Analyses performed for a swollen tube with an axial rupture in typical RSG predicted cross-flow fields demonstrate that even with the low damping associated with locked boundary conditions, which may lead to a SR in excess of unity, the tube would be expected to attain a steady state vibration amplitude compatible with an equilibrium vibration cycle. The equilibrium cycle occurs when the energy extracted by the damping equals the energy absorbed by the tube vibration mode by the fluidelastic interaction. Most of the swollen tubes at TMI, of which the majority were located in the high cross-flow field found on the periphery of the bundle, achieved an equilibrium vibration cycle that did not lead to fatigue failure.

There are two regions in RSGs that would be considered most susceptible to fluidelastic excitation, in the U-bend where the velocities are highest and at the top of the tubesheet where there is significant cross-flow and the density is greatest. In addition, the preheater region on the cold leg side of the steam generator for preheat style SGs would also be considered susceptible because of the cross-flow velocities and high fluid density. The potential for tube severs to occur in these regions is discussed in the following sections.

Top of the Tubesheet Region

Swelling of the tubes results in their becoming locked at TSP locations, resulting in a decrease in the squeeze film damping and attendant increase in susceptibility to flow induced vibration. However, corrosion products that accumulate in tube-to-tube support crevices in SGs with drilled hole TSPs (the early Westinghouse designs) or carbon steel eggcrates (the CE design) lead to the same effect, i.e., reduced damping leading to increased FIV susceptibility (at the same time the natural frequency of the tube is increased which reduces the susceptibility to fluidelastic excitation). In addition, it is likely that tubes in older Westinghouse SGs are locked at the elevation of the first TSPs. It is extremely likely that the locked crevice conditions have been in place for many years of operation without any tubes being fluidelastically unstable. Similarly, several CE SGs operated with tubes locked at the first support for many years. Therefore, locking of tubes at the lower tube supports in RSGs does not result in unacceptable flow induced vibration conditions. This has been confirmed by the number of tubes which have been found to be swollen, and in fact some have been ruptured, but have not developed circumferential fatigue cracks. Hundreds of other tubes have been unplugged and returned to service at several plants with no severed tubes being discovered. The maximum cross-flow velocity coupled with the highest fluid density generally occurs at the top surface of the tubesheet

and along the tube lane, the space between the straight length portions of the tubes with the smallest U-bend radius. There has never been any evidence of flow induced cracking or fatigue of the tubes due to cross-flow at the top of the tubesheet in spite of the locked tube conditions.

Flow induced vibration analyses were performed for Westinghouse Model 51 SG peripheral tubes (the high cross-flow and density combination region) with locked tube-to-TSP boundary conditions and low associated structural damping values. The results of those analyses demonstrated that the tubes would be fluidelastically stable and that fatigue failures of the tubes would not be expected to ensue if the tubes became locked at the first TSP.

Fluidelastic instability analyses were performed for all of the described cases of bulged or swollen tubes in the United States and Sweden. Swollen tubes located anywhere in the bundle of the Ringhals and Connecticut Yankee SGs were calculated to be fluidelastically stable. For the Salem tubes where the ruptures occurred in tubes at the most severe flow location, the steady state stress amplitude associated with the equilibrium cycle was found to be acceptably low. The analysis also demonstrated that the crack associated with the rupture of a tube would not be expected to grow following the rupture. Swollen tubes have been found in row 1 locations without having become severed. In summary, there are no known locations of fluidelastic instability to the extent that fatigue failure would be expected to occur.

Preheater Region

The Westinghouse preheater or counter-flow design SGs were the subject of flow induced vibration degradation in the early 1980's. One of the remedial actions performed for these SGs was to expand the tubes at multiple TSP locations in regions where the highest cross-flow velocity was present. The tubes were analyzed prior to this action being taken to verify that they would be fluidelastically stable following the expansion process. The key point is that the expanded tube has a very small gap relative to the TSP, resulting in almost non-existent squeeze film damping, but is still fluidelastically stable. The same characteristic would be expected for a tube expanded or swollen as the result of a plug behaving as a flow diode, hence the tube would be expected to be fluidelastically stable and no fatigue failure would be expected to occur.

The SGs of the CE System 80 employ a preheater design, the tubes in which have experienced vibration induced wear. Studies performed and submitted to the NRC via Reference 13 reported on the evaluation of a flow condition at the intersection of the recirculating water entrance window and the tube lane where high radial velocities could exist. An extensive fluid-hydraulics analysis indicated that the SR for tubes in that region could be greater than unity. The same analysis also showed that the high flow velocities drop off rapidly away from the affected region, demonstrating that the region was small and well defined both radially and vertically. The behavior of the flow and tubes in that region has been well understood since the 1988 evaluation. Due to the aggressive nature of this phenomenon, tubes removed from service in this region include the installation of a cable stabilizer regardless of the mechanism for which plugging was required. Therefore, appropriate protection against tube severance is provided.

U-Bend Region

Damage resulting from flow induced vibration of tubes in the U-bend region was the subject of a significant engineering investigation in 1987 as the result of a tube rupture at the North Anna plant. In that case the tube had become locked at the top TSP, which resulted in a significant reduction in vibration damping, and a local region of high velocity flow had been created by non-uniform anti-vibration bar (AVB) penetrations, NRC Bulletin 88-02, Reference 14. The combination of reduced damping, which decreased the critical velocity, and increased local flow led to a SR in excess of unity. Without the locally high flow velocity, the fatigue failure would not have occurred. Following the North Anna event, inspection campaigns were mounted in potentially affected plants to ascertain the as-built positions of AVBs in SGs where the U-bend tubes were considered susceptible to the fluidelastic excitation phenomenon. Most of the susceptible tubes had vibration dampeners installed to eliminate their susceptibility and were removed from service. The others were plugged with sentinel plugs and the surrounding tubes were also plugged to isolate any potential damage. Thus, while tube swelling could occur that would lead to locking at the top TSP, the tube would not be susceptible to high amplitude flow induced vibration.

A reduction in structural damping and a locally high flow velocity is believed to have been associated with the fatigue cracking of a large row U-bend tube, that did not sever, at Indian Point Unit 3. Analyses performed by Westinghouse following removal of the tube from the SG considered a range of support conditions, including the effects of the

observed cracked weld, concluded that the probable cause of the fatigue crack was fluidelastic vibration of the tube due to denting of the tube (reduced damping) and potentially inadequate tube support resulting from the broken AVB/retainer ring weld, Reference 15. In the total operating history of the Westinghouse square pitch steam generators with 0.875" diameter tubing with two sets of AVBs, the Indian Point 3 tube cracking was unique. No other similar events have been observed in tubes with similar tube bend radius. Thus, the operating database for comparison is many thousands of tubes both dented and not dented at the top TSP in similar flow fields. Therefore, it was and is concluded that the specific conditions applicable to the Indian Point 3 cracked tube, that is, denting, stress corrosion cracking, and a broken AVB structure weld, do not represent a generic issue for other operating SGs.

The design of the upper bundle supports in CE SGs was analyzed for the North Anna type U-bend failure event following the issuance of Reference 14. It was been determined that the design differences provide greater in-plane and out-of-plane tube support and that the SR of the tubes is less than unity for this condition regardless of the boundary condition at the top tube support. In summary, no failures were predicted and none have occurred in the operation of the CE design RSG.

Conclusions

The combination of operating experience of US recirculating SGs along with the analysis of swollen tube conditions provides convincing evidence that the issues arising from the TMI swollen tube event do not pose an immediate concern regarding the continued operation of those SGs in US plants. To reiterate the industry opinions expressed at the meetings with the NRC staff,

1) The TMI issue is not of immediate concern to operating power plants with recirculating SGs.

2) Operating power plants do not have to cease operation in order to address potential concerns associated with severing of plugged tubes and consequential damage to neighboring active tubes.

3) There is sufficient time for the performance of a deliberate evaluation, or investigation, of the TMI issue to determine its applicability, if any, to the remaining operating plants.

The bases for these conclusions is that remedial programs that followed issuance of NRC Bulletin 88-02 on rapidly propagating cracks in U-bends were sufficient to address concerns for locking of plugged tubes at the top TSPs, among all of the observed plugged diode effects only swelling and axial failures have occurred, hundreds of plugged tubes have been returned to service with no known instances of tubes being severed, and thousands of tubes locked by denting have been operating in the highest cross-flow velocity flow fields in RSGs without experiencing one fatigue failure. Thus, it is also considered to be likely that the outcome of a detailed evaluation will conclude that there is an acceptably low risk of consequential damage to neighboring active tubes and that remedial actions to address potentially ruptured plugged tubes are not necessary to the continued safe operation of the plants.

Sincerely,

Chairman, Steam Generator Management Program

LFW:cll

xc Jim Riley, NEI
Alex Marion, NEI
David Steininger, EPRI
Jeff Ewin, INPO
Gary Fader, INPO

References

1. Presentation, "TMI Once Through Steam Generator (OTSG) Inspection/Repair," Exelon to NRC staff (November 9, 2001).
2. NRC Staff Letter, "Potential for Plugged Steam Generator Tubes to Impact the Integrity of Active Tubes," B. Sheron to A. Marion (NEI), (November 25, 2001).
3. NRC and Nuclear Industry Senior Management Meeting, White Flint One Building, Rockville, MD (November 28, 2001).
4. Schenk, von H., "Erfahrungen mit den Dampferzeugern im Kernkraftwerk Obrigheim nach uber 100000 stunden Betrieb und deren Austausch als unzerlegte Einheiten," Der Maschinenschaden, 57, Heft 4, pp. 111-119 (1984).
5. SG-84-07-002, Rev. 1, " Ringhals 2 Steam Generator Plugged Tube Evaluation Report," Westinghouse Electric Corporation, Pittsburgh, PA (July 1984).
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9. NS-NRC-91-3602, "Rupture of Plugged Steam Generator Tubes," S. R. Tritch to J. E. Richardson (USNRC), Westinghouse Electric Corporation, Pittsburgh, PA (June 18, 1991).
10. Telephone conversation, "EdF Fishmouth Tube," J. Wootten, et al. (Westinghouse) with P. Cornet (Electricité de France) (October 30, 1992).
11. TR-105371, "EdF Maintenance Strategy for the Alloy 600 SG Tube Plugs," Proceedings: Steam Generator Strategic Management Workshop, EPRI, Palo Alto, CA (July 1995).
12. NTD-NRC-95-4415, "Engineering Evaluation of Burst/Plugged Tube(s) at Connecticut Yankee," Westinghouse Electric Corporation, Pittsburgh, PA (March 13, 1995).
13. LD-88-049, "Steam Generator Tube Vibration Issue Closeout on the CESSAR-F Docket," A. Sherer (CE) to G. Vissing (USNRC), Combustion Engineering, Windsor, CN (July 1, 1988).
14. NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," United States Nuclear Regulatory Commission, Washington, DC (February 5, 1988).

15. SG-91-05-055, "Fatigue Fracture of Steam Generator Tube R45-C50 from Indian Point Unit 3," Westinghouse Electric Company, Pittsburgh, PA (April 1991). Introduction – TMI Event Description & Root Cause.

Enclosure 2

B&W Owners Group:

Implications of TMI-1 Plugged Tube Sever For Other B&W OG Plants

December 20, 2001
OG-01-1814

Mr. Ralph Beedle
Senior Vice President and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street N.W., Suite 400
Washington, DC 20006-3708

References:
1. B.W Sheron (NRC) letter to A. Marion (NEI) dated Nov. 26, 2001, subject: "Potential For Plugged Steam Generator Tubes To Impact The Integrity of Active Tubes"
2. J.H. Riley (NEI) e-mail dated Dec. 3, 2001, subject: "NRC Meetings On SG Issues – November 27th & 28th, 2001"

Dear Mr. Beedle:

In response to the requests in the correspondence referenced above, please find attached the B&W Owners Group summary of the generic implications of the TMI Once Through Steam Generator (OTSG) plugged tube sever event. This information is being submitted by Framatome ANP on behalf of the B&WOG. The NRC has requested an industry response for this matter by December 21, 2001. It is our understanding that NEI will forward this information to the NRC by that date, along with a similar Westinghouse summary response for the Recirculating Steam Generators (RSGs).

Further, it is the B&WOG intent to meet with NEI and EPRI/SGMP on January 10, 2002 and on or about January 30th with NEI plus January 31st with the NRC to address the industry long term action plans for this matter.

If you have any questions or require additional information, please contact me at (434) 832-2613 or Mr. Jeff Brown at (434) 832-3925.

Sincerely,

G. K. Wandling
Project Manager
B&W Owners Group Services

Framatome ANP B&W Owners Group
3315 Old Forest Road
Lynchburg, VA 24501
Phone: 804-832-3635 Fax: 804-832-4121

c: SG Committee
Steering Committee
DJ Firth
JH Riley – NEI

Framatome ANP B&W Owners Group
3315 Old Forest Road
Lynchburg, VA 24501
Phone: 804-832-3635 Fax: 804-832-4121

1.0 INTRODUCTION

During the October 2001 outage at Three Mile Island Unit 1, a previously plugged steam generator tube was found to have severed in the top span and impacted four adjacent in-service tubes, causing wear scars on the tubes. As a result of this discovery, TMI-1 conducted an extensive evaluation to determine the root cause of the severance. The plant also performed a significant amount of inspection and repair work to eliminate any detrimental effects of similar failures on in-service tubes during future operation of the plant.

As the investigation into the TMI tube sever progressed, information was regularly shared with the NRC and other utilities having B&W Once-Through Steam Generators (OTSGs). As a result of this investigation it became apparent that plugged tube locations in other OTSGs could potentially be subject to similar failures. As a result, Oconee-3, which had a regularly scheduled outage in November 2001, inspected a number of previously plugged locations that were postulated to be susceptible to the same failure scenario. No similar occurrences were found.

Additional investigative work is being planned to fully define the susceptible populations and appropriate corrective actions to be implemented at the remaining plants during the next scheduled outage and beyond. In the short term, a qualitative assessment of the likelihood of similar failures at other plants has been performed. The purpose of this letter is to provide the results of that assessment.

2.0 SUMMARY OF TMI INSPECTION RESULTS

During the bobbin coil ECT inspection of the TMI-1 B-OTSG in October 2001, wear indications were detected on four of the six tubes surrounding previously plugged tube B66-130. The wear indications were located in the uppermost span just below the secondary face of the upper tubesheet (UTS). The plug installed in the UTS of B66-130 was removed so that the tube could be visually inspected, and it was confirmed to be severed at the secondary face of the UTS. Portions of this tube and two of the surrounding tubes were eventually removed from the OTSG and examined in the laboratory. Based on observations of the fracture face, the failure mode was concluded to be high cycle fatigue. B66-130 was also determined to have swelled to 0.699" OD, compared to a typical OTSG tube outside diameter of 0.627". The diameter of the swelled tube is larger than the bore diameter in both the drilled and broached tube support plate (TSP) holes. It was later concluded that this swelling resulted from internal overpressurization due to trapped water in the tube.

Tube B66-130 was plugged in 1986 (TMI-1 5R refueling outage) due to eddy current indications in the 6th tube span (above the 5th TSP). The tube was plugged using a B&W alloy 600 ribbed plug in the lower tubesheet (LTS) and a B&W alloy 600 rolled plug in the UTS. Based on the location of the indications, tube

B66-130 did not require stabilization when removed from service, thus no stabilizer was installed. The alloy 600 UTS plug was replaced with a FRA-ANP alloy 690 rolled plug during the 12R outage (1997). This replacement was part of an ongoing TMI-1 program to replace UTS (hot leg) alloy 600 plugs with alloy 690 plugs due to the susceptibility of alloy 600 plug material to PWSCC. No tube inspection of B66-130 was required or performed during 12R when the plug was replaced.

As part of the effort to define the extent of condition at TMI-1, a total of 870 plugs were removed from the upper head in both OTSGs to allow inspection of the plugged tubes. This inspection resulted in the discovery of one additional severed tube (A2-24), which also was reported to be swollen based on ECT profilometry. This tube had been plugged in 1983 with Westinghouse Alloy 600 rolled plugs in both the UTS and LTS. An additional 27 tubes were also found to be swollen, but not severed.

All of the swollen tubes, including the two severed tubes, were contained in two hardware categories of plugged tubes, as indicated below:

1. Tubes with Alloy 600 rolled plugs manufactured by Westinghouse, which had been installed in 1982-1983. Several of these plugs were observed to have ejected after their initial pressurization cycle, and the remaining plugs were subsequently re-rolled in place to improve the integrity of the rolled joint. This specific plug design is unique to TMI-1 among OTSGs.

2. Tubes with UTS Alloy 600 rolled plugs manufactured by FRA-ANP originally installed between 1985 and 1988, which had been removed and replaced with FRA-ANP Alloy 690 rolled plugs.

In addition to inspecting for signs of swelling, the presence of water in the tubes was recorded. The following table summarizes those results for the various populations of plugged tubes.

Table 1. TMI Plugged Tubes with Water

Plug Category	Total Number Examined for Water Content	Total with >50% Water	% with >50% Water
1. Re-rolled Westinghouse Alloy 600 Roll plugs	455	237	52%
2. FRA-ANP UTS Alloy 600 Roll Plugs Replaced with Alloy 690 Roll Plugs	249	24	10%
3. Originally installed, un-repaired Alloy 600 roll plugs on both ends	67	2	3%
4. Originally installed, un-repaired Alloy 690 roll plugs on both ends	87	0	0
Total:	**858**	**263**	**31%**

Therefore the two categories of plugs that account for all the swollen tubes also account for all but 2 of the tubes containing over 50% water.

3.0 SUMMARY OF OCONEE-3 INSPECTION RESULTS

During the November 2001 outage at Oconee-3, 108 tubes were deplugged and inspected. This population included all un-stabilized tubes from category 2 above (37 total), plus tubes plugged with original Alloy 600 plugs in the upper and lower tubesheet. A total of 8 tubes were found to contain water greater than half the volume of the tube. No severed or swollen tubes were found at Oconee-3.

4.0 CONCLUSIONS FROM THE TMI INVESTIGATION

As indicated earlier, an investigation was performed to determine the root cause of the circumferential failure of B66-130. The investigation included analysis of tube pull samples of severed tube B66-130 and two of its adjacent tubes, extensive eddy current inspection of deplugged tubes, flow-induced vibration (FIV) and low cycle fatigue analyses, thermal hydraulic analysis, and evaluation of the causes and effects of internal pressurization of plugged tubes.

The investigation concluded that the circumferential failure of tube B66-130 was a combined result of the following factors:

- A significant volume of primary water entered the tube during operation due to mechanical plug leak-by.
- The tube had sufficient structural integrity to withstand an internal pressure (in excess of 7000 psi) that could swell the tube enough to contact the TSP.

- The tube pressurized during plant heatup due to the water being trapped in the tube, causing the tube to swell and become restrained at both the top TSP and the UTS secondary face.
- The upper span region of tube B66-130 is located in a region of high steam flow velocity at the bundle periphery. When the tube became restrained at its upper span, the top span became isolated from the rest of the tube, thus decreasing the tube's damping and increasing the amplitude of vibration due to the steam cross flow. This resulted in increased stresses due to FIV, which led to failure of the tube by fatigue.
- Shallow tube degradation, OD IGA, near the UTS secondary face became an initiating location for the FIV-induced fatigue failure. The presence of this degradation in tube B66-130 is believed to have provided a small area of stress concentration, which to some extent reduced the fatigue life for this particular tube. Presence of a flaw is not believed to be a necessary pre-cursor for failure of other swollen, peripheral tubes.

As previously indicated, two populations of plugged tubes were observed to be susceptible to swelling at TMI-1. The commonality between the two populations is the execution of a repair that improved the integrity of one or both plug joints. In the case of the Westinghouse population (Category 1), the repair consisted of re-rolling the plugs in place. In the case of the FRA-ANP population (Category 2), the repair consisted of removing the UTS Alloy 600 rolled plug and replacing it with an Alloy 690 rolled plug, which is known to have better leakage performance. In both cases, the repair potentially trapped existing water inside the tube at cold conditions that could not leak out through the tighter plug joints as the plant subsequently heated up. This sequence of events is concluded to have set up the conditions necessary for tube overpressurization that led to the swelling of the 29 tubes observed at TMI-1.

Both severed tubes (B66-130 and A2-24) are located near the periphery of the tube bundle. This is significant for two reasons. First, the cross flow velocity of the secondary side steam is highest at the periphery, which leads to lower margins to flow instability and reduced fatigue life. Second, the outer 2 to 4 rows of tubes pass through drilled holes in the uppermost TSP. Tests have been performed to investigate the effects of swelling on the damping of the tubes in both drilled and broached TSP holes. These tests clearly show that the reduction in damping due to the swelling of the tube is more significant for a drilled hole than a broached hole.

The effect of tube swelling on the FIV response of the tube was evaluated by computing the fluid-elastic stability margin (FSM) for the tube in the swollen condition. The FSM is defined as the ratio of the critical cross flow velocity divided by the actual velocity. The critical velocity is defined as the point where the tube becomes unstable, which causes very large amplitudes of vibration and imminent failure by fatigue. If the FSM is greater than one, the actual velocity is less than the critical velocity, and the tube is stable.

The FSM for a swollen outermost periphery tube was estimated by analysis to be 1.1. This value was calculated using lower bound damping values appropriate for a tube that is swollen to the extent that it is restrained in the drilled hole of the 15th TSP. An FSM of 1.1 indicates a very small margin to instability. Therefore tube B66-130 was concluded to have failed because it was near instability (indicated by an FSM <1.0), which caused amplitudes of vibration in excess of those that would be predicted if the tube were completely stable. By comparison, the FSM for the worst located swollen tube with a broached opening in the top TSP is estimated to be >1.6, using lower bound damping. This 45% increase in stability margin is significant because it provides confidence that tubes passing though broached holes at the uppermost TSP, even if swollen, are significantly less susceptible to severance due to FIV fatigue.

5.0 EVALUATION OF RISK TO OTHER PLANTS

Based on the information in the previous section, the population of plugged tubes most susceptible to severs at the remaining B&W plants is limited to tubes with the following characteristics:

1. The tube has a plug that has been repaired in such a way that the joint integrity has been improved without removing existing water from the tube, AND

2. The tube passes through a drilled hole in the uppermost tube support plate.

Based on the above criteria, a query of the plugged tube history for all the B&W plants was performed. The data show that prior to the repairs conducted at TMI-1 and Oconee-3 in the fall of 2001, a total of 546 plugged tube locations fell into this category. Of this number, 492 (90% of the total) were at TMI-1. Another 11 tubes meeting these criteria were deplugged and inspected at Oconee-3 as part of their investigation in November 2001. Therefore, after the repairs made at TMI-1 and Oconee-3, 43 potentially susceptible tubes remain in the other plants.

Because of the condition monitoring results related to the tubes with wear scars, TMI-1 performed an evaluation to assess the risk implications of the tube sever and resulting damage to in-service tubes. The first step in this evaluation was to estimate the rate of wear on the in-service tubes due to the impact of the severed tube. The wear rate was then used to estimate the length of operating time a tube could be in service in a condition such that it would be predicted to fail under MSLB conditions. The changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) due to this condition were calculated and determined to be acceptable. Using the TMI results as input, the resulting change in CDF and LERF for the remaining plants was estimated. This estimation included an assumed probability of tube sever equal to 2/492 = 0.004 per susceptible tube, based on the TMI-1 results (2 confirmed severs in 492 susceptible locations). The result showed that the estimated changes in CDF and

LERF for the remaining plants would be in Region III as defined in Figures 3 and 4 of NRC Regulatory Guide 1.174, which is classified as a very small increase.

The two categories of plugged tubes evaluated for this initial assessment of risk were chosen because they have been identified as those most susceptible to overpressurization in OTSGs. The commonality between these two populations is the potential for trapping water inside the tube during repair to one or both plugs. It is noted that all categories of plugged tubes are potentially susceptible to overpressurization based on industry experience, since the effects of overpressurization have been observed in a small number of RSG plugged tubes. However, evidence of overpressurization (e.g., plug collapse or wear on adjacent tubes) in other populations was not identified at TMI-1, nor have they been identified at other B&W plants. The additional risk of consequential tube damage from these other populations of plugged tubes is considered to be insignificant when compared to risk associated with the two populations of repaired plugs discussed above, and therefore is considered to be inconsequential to the short term operation of the plants. The B&WOG will consider all plugged tube populations when the final determination of long term corrective actions is made to address this issue.

6.0 CONCLUSION

The preceeding sections have summarized the pertinent observations from TMI-1 and Oconee-3, the failure scenario attributed to the severed tubes at TMI-1, and the populations of plugged tubes at the remaining plants that are potentially susceptible to similar failures. It has been shown that the susceptible population at the other plants is significantly less than that at TMI-1. Based on this information, the B&WOG has performed a preliminary estimate of the risk associated with current operation and has concluded that the increase in CDF and LERF are acceptably small per Reg. Guide 1.174. The B&WOG therefore concludes that continued operation of the plants is acceptable while inspections continue and a long term strategy is developed for inspection and repair to mitigate similar occurrences in the future.