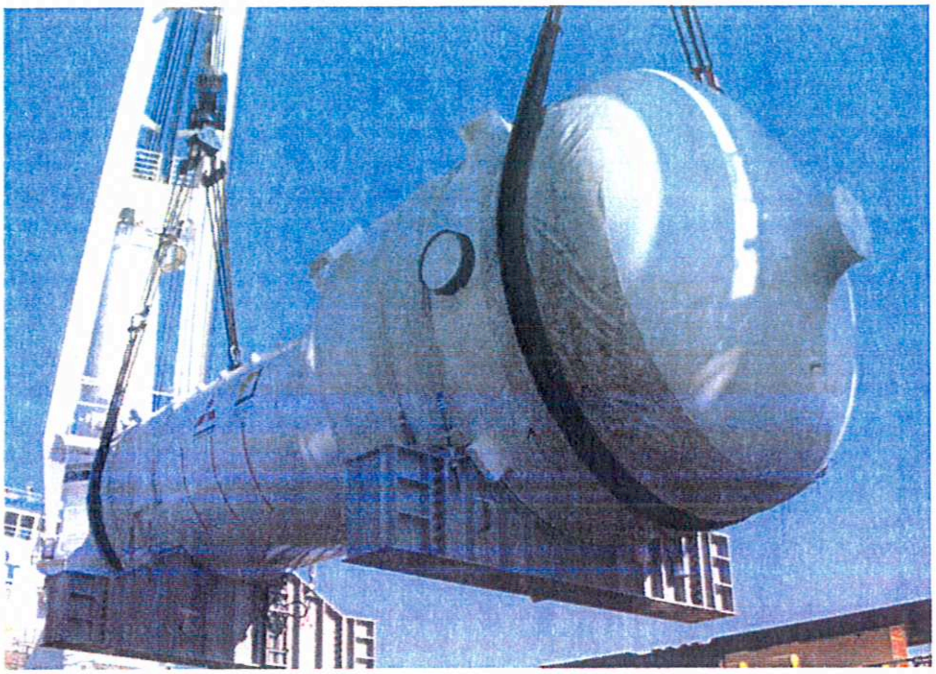

Management Sponsor:

Gary Kline, Engineering & Technical Services

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Approvals:

RCE Team Leader	5-7-2012 Date
Management Sponsor	5/7/12 Date
Corrective Action Program Coordinator (CAPCO)	5/7/12 Date
Corrective Action Review Board Chair	5.14.12 Date

DISCLOSURE STATEMENT

Consistent with the San Onofre Nuclear Generating Station's corrective action program, this cause evaluation evaluates, through the use of an after-the-fact hindsight-based analysis, conditions adverse to quality and the causes of those conditions. The information identified in this cause evaluation was discovered and analyzed using all information and results available at the time it was written. These results and much of the information considered in this evaluation were not available to the organizations, management, or individual personnel during the time frame in which relevant actions were taken and decisions were made. Consistent with the requirements of 10 C.F.R. Part 50, Appendix B, Section XVI, Edison's cause evaluations have been established as a means to document and "assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected," and, as necessary, to ensure that actions are taken to prevent recurrence.

This cause evaluation does not attempt to make a determination as to whether any of the actions or decisions taken by management, vendors, internal organizations, or individual personnel at the time of the event were reasonable or prudent based on the information that was known or available at the time they took such actions or made such decisions. Any individual statements or conclusions included in the evaluation as to whether errors may have been made or improvements are warranted are based upon all of the information considered, including information and results learned after-the-fact, evaluated in hindsight after the results of actions or decisions are known, and do not reflect any conclusion or determination as to the prudence or reasonableness of actions or decisions at the time they were made.

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EXECUTIVE SUMMARY

SONGS Unit 3 started commercial operation in April 1984 with new Combustion Engineering (CE) steam generators. The Unit was safely operated at 100% nominal power for 15 refueling cycles. In October 2010, Unit 3 was taken offline for the Cycle 16 Refueling and Steam Generator (SG) Replacement Outage. The CE SGs were replaced with Mitsubishi Heavy Industry (MHI) SGs, and the Unit was returned-to-service at 100% nominal power in February 2011.

On January 31, 2012, Unit 3 was operating at nominal 100% full power and approximately half way through the first operating cycle after SG Replacements. SONGS Operators identified a primary to secondary system leak in Unit 3 SG E088. The tube leak was approximately 75 gallons per day and rising. Operators promptly shut down the Unit before Technical Specification leakage limits were exceeded, with no public health or safety consequences. SONGS maintained compliance with shut down requirements. There was no practical opportunity to identify the tube degradation prior to the tube leak. (Component designators 2E088, 2E089, 3E088, 3E089 apply to Unit 2 A and B SGs and Unit 3 A and B SGs respectively. These designators are used throughout the report.)

With secondary side pressurized, visual inspection on the primary side identified the particular SG E088 tube with the leak, and subsequent Eddy Current Testing (ECT) identified unexpected tube-to-tube wear. 100% ECT examination of tubes in both Unit 3 SG E088 and E089 (19,454 tubes combined) identified more locations of unexpected tube-to-tube wear. In-situ pressure testing of tubes with higher levels of tube-to-tube wear resulted in a total of eight tubes in Unit 3 SG E088 that did not meet leakage and/or structural performance criteria as required by Technical Specifications. ECT testing also identified tube wear at Anti-Vibration Bar (AVB), Tube Support Plate (TSP) and Retainer Bar locations. The magnitude of wear indications at some TSP locations exceeded the industry-standard-acceptance criteria included in the SG Program. The primary focus of this cause evaluation is the unexpected tube-to-tube wear. The report also addresses AVB, Retainer Bar, and TSP tube wear for completeness.

SONGS established comprehensive SG Recovery and Root Cause Evaluation (RCE) Teams including the services of industry experts in the fields of SG design, manufacturing, operation, and repair to ensure a complete understanding and verification of the condition, extent, cause, and corrective actions. The RCE Team used a systematic approach to identify the mechanistic cause of the tube-to-tube wear including Failure Modes Analysis (Kepner Tregoe) and Barrier Analysis. The team identified a list of possible causes and then using supporting and refuting facts identified areas warranting more detailed analysis, including primary flow induced vibration, divider plate weld failure, fluid elastic instability (FEI), fabrication/manufacture, shipping, and thermal-hydraulic (T/H) conditions.

MHI is a 10 CFR 50 Appendix B and ASME qualified supplier and as such will be performing an independent root cause analysis. The MHI analysis will address design, organizational, programmatic, and technical aspects that contributed to the observed tube wear. Upon completion of the further causal analysis by MHI, SONGS will update this RCE to include a discussion of MHI's findings and further analysis of Edison's oversight of MHI's design and manufacturing process. Because MHI will be determining the organizational/programmatic causes, this RCE determined the mechanistic cause.

The mechanistic cause of the tube-to-tube wear was identified as FEI, involving the combination of localized high steam/water velocity (tube vibration excitation forces), high steam void fraction (loss of ability to dampen vibration), and insufficient tube to AVB contact forces to overcome the excitation forces. The corrective actions to prevent recurrence of FEI include lowering power operations to reduce tube excitation forces and improve the ability to dampen vibration. This action includes establishing T/H Models and Flow Induced Vibration (FIV) Models capable of predicting SG velocities and void fractions within tolerances to determine operational limits to avoid FEI. Corrective actions to fix the condition include stabilizing and plugging the eight tubes in SG E088 that did not meet leakage and structural performance criteria, and stabilizing and plugging other tubes with high wear indications as required by the SG Program. SONGS will also plug some tubes in the region of FEI susceptibility as a preventive measure, will implement a Mid-Cycle outage to monitor SG conditions and identify potential degradations for repair prior to consequence, and verify corrective actions have been effective. These actions will remain in place until monitoring and inspection results indicate FEI induced tube-to-tube wear is no longer an issue requiring enhanced monitoring and inspections. SONGS will update this RCE upon further MHI analysis of the programmatic and organizational issues underlying the mechanistic cause.

At the time of the Unit 3 SG tube leak, Unit 2 was in the first refueling outage after replacing its SGs and undergoing ECT testing per the SG Program. While initial 100% ECT results did not identify tube-to-tube wear in Unit 2, station management decided to delay return-to-service pending an evaluation of the susceptibility of the Unit 2 steam generators to tube-to-tube wear similar to Unit 3. Additional ECT examination on Unit 2 using more sensitive equipment identified minor tube-to-tube wear at one location involving two tubes, indicating the potential existence of FEI but to a much lesser extent than Unit 3. As a part of the extent of condition for this Unit 3 RCE, corrective actions for Unit 2 include stabilizing and plugging the two tubes with tube-to-tube wear, preventive plugging of tubes in areas of the Unit 2 SGs identified by analysis of the Unit 3 tube-to-tube wear locations, lowering power operations levels to reduce tube excitation forces and improve the ability to dampen vibration, and implementing a Mid-Cycle Outage to proactively monitor SG conditions and identify potential degradations for repair prior to consequence.

PROBLEM STATEMENT

Governing Requirements

SG tubes provide heat transfer capability to support power operations, allow removal of Reactor Coolant System (RCS) heat during accident conditions, and provide a physical barrier to control the release of radioactive fission products.

Deviation

On January 31, 2012, while operating at nominal 100% power and within the first operating cycle after SG replacements, Operators identified a primary to secondary system leak in Unit 3 SG E088. The tube leak was approximately 75 gallons per day and rising.

Consequences and Significance

SONGS Operators promptly shut down Unit 3 on January 31, 2012, before exceeding Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.13 which specifies a Reactor Coolant System operational leakage limit of 150 gallons per day. There were no public health or safety consequences. On March 29, 2012, SONGS submitted Licensee Event Report LER 2012-001, “Unit 3 Manual Reactor Trip due to Steam Generator Tube Leak.”

The leak was from a tube in Unit 3 SG E088 due to tube through wall wear. Subsequent inspections and in-situ pressure testing in the Unit 3 SGs identified a total of eight SG E088 tubes that did not meet the accident-induced leakage criterion and/or structural performance criterion as required by the Technical Specifications. In accordance with 10 CFR 50.72(b)(3)(ii)(A), SONGS made eight-hour reports to the NRC for the eight in-situ pressure test failures. SONGS is preparing LER 2012-002, “Unit 3 Steam Generator Tube Degradation Indicated by Failed In-Situ Pressure Testing.”

On March 27, 2012, the NRC issued a Confirmatory Action Letter (CAL) to SONGS. The CAL specified actions to be completed prior to Unit 2 entering Mode 2 and Unit 3 entering Mode 4. Completion of this cause evaluation fulfills a portion of one of those actions.

Because the RCS operational leakage limit was not exceeded and the radioactive release to the environment was well below allowable limits, there was no actual safety consequence relative to the as-found degraded condition of the Unit 3 SG tubes. In addition, both units remain shut down until reasonable assurance of tube integrity to support safe operation can be provided in accordance with the CAL. Unit 2 has been in a refueling outage since January 9, 2012. Unit 3 was shut down on January 31, 2012, due to the SG tube leak. The potential safety significance of the degraded condition of the Unit 3 SG tubes is discussed in the Safety Significance Section of this RCE.

The Maintenance Rule Plant Level Performance Criteria for Forced Loss Rate was exceeded as a result of the Unit 3 SG tube leak and subsequent extended shutdown. A Maintenance Rule Evaluation, in accordance with SO123-XV-5.3 (Maintenance Rule Program), is being performed under Notification 201961729 (Maintenance Rule Plant Level Performance Criteria Exceeded).

MHI issued Interim 10CFR Part 21 Notification on Steam Generator Tube Wear at SONGS on 4/23/2012 (Event 47833).

Regulatory Requirements

Technical Specification (TS) 3.4.17 requires that SG tube integrity be maintained and that all SG tubes meeting the tube repair criteria be plugged in accordance with the Steam Generator Program. TS 5.5.2.11 requires a Steam Generator Program to be established and implemented to ensure that SG tube integrity is maintained. TS 5.5.2.11.b specifies three performance criteria that must be met for SG tube integrity, as quoted below:

1. "Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and Design Basis Accidents (DBAs). This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads."

2. "Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any DBA, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs."

3. "The operational leakage performance criterion is specified in LCO 3.4.13, "RCS Operational Leakage." [This LCO is applicable in Modes 1-4 and states RCS operational leakage shall be limited to: (a) no pressure boundary leakage; (b) 1 gpm unidentified leakage; (c) 10 gpm identified leakage; and (d) 150 gallons per day (gpd) primary to secondary leakage through any one SG."]

IMMEDIATE ACTIONS

1) 1/31/2012: SONGS Operators performed a forced Unit 3 shut down to maintain plant safety margins in accordance with Abnormal Operation Instruction (AOI) SO23-13-14 (Reactor Coolant Leak), AOI SO23-13-28 (Rapid Power Reduction), Emergency Operating Instruction (EOI) SO23-12-4 (Steam Generator Tube Rupture) and SO23-12-1 (Standard Post Trip Actions). The shutdown was due to Operations receiving secondary system radiation alarms (RE 7818 and 7870).

2) Operators isolated the SG with the leak and cooled down and depressurized the reactor to minimize/stop any additional leakage.

3) 1/31/2012: SONGS Shift Manager notified the Nuclear Regulatory Commission (NRC) of the forced Unit 3 shutdown in accordance with 10CFR 50.72 (b) (2) RPS Actuation (manual scram) – a 4 hour report.

4) 2/1/2012: SONGS generated Notification 201836127 (Steam Generator Tube Leak) to identify and track resolution including Reportability (RPS Actuation) and Root Cause Evaluation assignments, in accordance with Corrective Action Program Procedures SO123-XV-50.CAP-1 (Writing Notifications) and SO123-XV-50.CAP-2 (Notification Screening). Reference Licensee Event Report Unit 3 LER-2012-001.

5) 3/14/2012: SONGS performed in-situ testing on 129 tubes and reported failures of eight tubes in accordance with 10 CFR 50.72(b)(3)(ii)(A) Serious SG Tube Degradation. SONGS generated Notification 201897717 (U3 Steam Generator E088 In-situ Pressure Test Failure) to identify and track resolution including Reportability. The event was reportable because performance criteria specified in the SG Program and TS 5.5.2.11 were not met. Reference Unit 3 LER-2012-002.

INTERIM ACTIONS

1) SONGS established the Units 2&3 Steam Generator Recovery Project, and obtained the services of industry experts in the fields of SG design, manufacturing, operation, and repair to conduct and/or independently review inspections, testing, modeling, failure analysis, repair plans, and corrective actions.

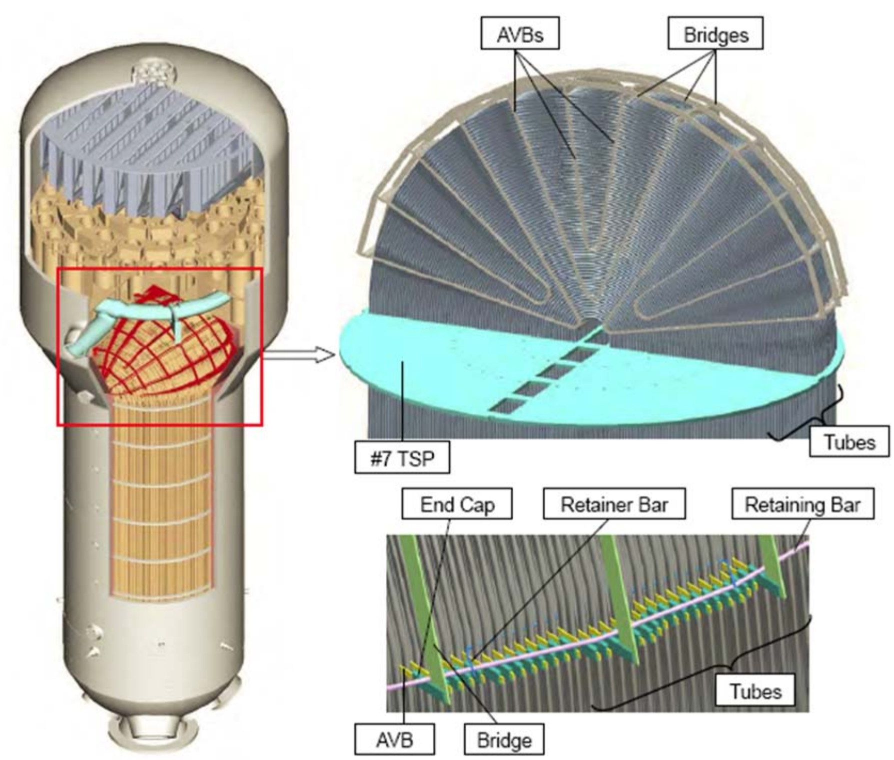
2) SONGS applied conservative decision making and delayed return-to-service of Unit 2 pending Unit 3 SG tube failure analysis. The concern was the potential susceptibility of the Unit 2 SGs to the unexpected tube-to-tube wear mechanism identified in Unit 3 which resulted in the tube leak.

3) Units 2 and 3 will remain shut down until the cause of the tube leak is thoroughly understood and actions to prevent additional tube failures resulting in a leak are completed.

BACKGROUND INFORMATION

SONGS Units 2 and 3 are two-loop Combustion Engineering Pressurized Water Reactor (PWR) plants, each employing two recirculating, U-tube SGs. The originally installed CE SGs were replaced during the previous refueling outages with new SGs designed and manufactured by MHI. The replacement SGs incorporate thermally treated Alloy 690 tubing which has demonstrated through laboratory testing and industry experience, superior resistance to stress corrosion cracking as compared with the mill-annealed Alloy 600 tubing used in the original SGs. Other design features include stainless steel Anti-Vibration Bar (AVB) support structure for the tube U-bends and seven stainless steel trefoil broached hole Tube Support Plates (TSPs) for the tube straight leg sections. These features were chosen primarily to minimize the potential for tube wear and corrosion.

There are 9727 tubes in each SG, arranged in a triangular pitch in 142 rows and 177 columns. The tubes in Rows 1-13 are thermally stress-relieved to further minimize the potential for in-service stress corrosion cracking in the U-bends. The tube bundle U-bend region is supported by a floating AVB structure consisting of three sets of two V-shaped AVBs between each tube column. The AVBs are made of Type 405 ferritic stainless steel and are equipped with two Alloy 690 end caps. Each AVB end cap is welded to an Alloy 690 retaining bar. Thirteen Alloy 690 bridges run perpendicular to the retaining bars and hold the entire structure together. A total of 24 Alloy 690, chrome-plated retainer bars welded to the retaining bars is provided to prevent AVB structure displacement during SG fabrication and during a limiting design basis accident. The retainer bars anchor the AVB structure to the tubes, but were designed such as to not contact the tubes under operating conditions. The AVB structure is not attached to any other SG component and under operating conditions is held in place by friction between the AVBs and the tubes.



In the SONGS plants, the closed-loop Reactor Coolant System (RCS) circulates primary system water in a closed cycle, removing heat from the reactor core and internals and transferring it to the plant secondary side. The SGs provide the interface between the RCS and the plant secondary side. Reactor coolant is separated from the secondary side fluid by the SG tubes and tube-sheet, which form a barrier preventing release of radioactive materials to the environment. The secondary side systems also circulate water, and in the process of doing so remove heat from the reactor coolant and generate steam. However, the secondary side systems are not totally closed systems and present several potential release paths to the environment in the event of a primary-to-secondary leak. Detection of primary coolant leakage is accomplished primarily by independent radiation monitoring systems including the Condenser Air Removal System monitors. TS 3.4.13 provides RCS operational leakage limits and required actions applicable in Modes 1 through 4.

The SG tubes have a number of important safety functions. As noted above, the SG tubes are an integral part of the Reactor Coolant Pressure Boundary (RCPB) and, as such, are relied on to maintain primary system pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary systems. The SG tubes also provide the heat transfer surface between the primary and secondary systems to remove heat from the primary system.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety and heat transfer functions consistent with the licensing basis, including applicable regulatory requirements. SG tubing is subject to a variety of degradation mechanisms related to corrosion phenomena, along with other mechanically induced phenomena such as wear. These degradation mechanisms can affect tube integrity if they are not eliminated or managed effectively.

The processes used to meet the TS SG performance criteria are defined in the Steam Generator Program Guidelines (NEI 97-06). The guidelines establish the following requirements:

> If operational leakage causes a forced outage, a root cause evaluation shall be performed and included as a part of the OA report for the forced outage. A forced outage can result from incorrect assumptions or errors in past analyses. During an inspection outage following operational leakage of greater than 5 GPD in any SG, the following steps shall be taken to establish information about the leak:
>
> 1. Determine which SG(s) are leaking: Monitor all SGs to determine which SG(s) are leaking.
>
> 2. If possible, determine the source of the leakage: This is typically performed by a hydrostatic test, bubble test, or helium leak test to identify suspect tube(s) locations on the tubesheet. Quantify the rate (for example, drops per minute or gallons per minute [liters per minute]) of leakage. Correlate the calculated leakage (pressure/temperature adjusted leakage) versus the operational leakage. Determine if results have accounted for the observed operational leakage, while recognizing that an accurate comparison of operating and shutdown leakage measurements is difficult. If the source of the leakage cannot be identified using the methods described above, 100% eddy current examination should be considered. If the eddy current examination locates the potential leakage, proceed with Step 4. If the leakage has not been identified, an evaluation of the actions within Step 6 should be considered.
>
> 3. Examine leaking location(s): This inspection is typically performed by bobbin coil eddy current examination to establish axial location within the SG.

4. Examine to determine extent, orientation, and morphology: This is typically performed by rotating coil or array coil technology. Refer to the SGMP PWR Steam Generator Examination Guidelines [1].

5. Review prior inspection history: Review the information contained in the database and the actual historical bobbin and rotating data to establish factual information about the data. If the leakage is originating from a plug or sleeve, review the installation records for that location. Evaluate if installation parameters were met and identify any inconsistencies or nonconforming conditions.

6. Perform a root cause evaluation that includes all SG program elements in accordance with the utility's program(s). This evaluation should address the need to perform eddy current and/or secondary-side visual inspections. Also consider supplementing the root cause team with industry peers. The root cause team shall identify immediate, short-term, and long-term actions to correct any process deficiencies.

7. Execute root cause corrective actions.

8. Update and revise the Degradation Assessment, Condition Monitoring, and Operational Assessment as necessary to address the unexpected leakage.

9. Perform required repairs.

TS 3.4.17 provides SG tube integrity requirements associated with SG tubes satisfying the tube repair criteria for plugging in accordance with the SG Program in TS 5.5.2.11. The SG Program contains provisions for condition monitoring, inspection, repair, and requires 100% inspection of tubes during the first refueling outage following SG replacement.

EXTENT OF CONDITION

The purpose of the extent of condition review is to evaluate the extent of tube wear in all 4 SGs (2E088, 2E089, 3E088 and 3E089).

Visual inspection of the tube sheet primary side of the SG 3E088, with the secondary side pressurized, identified the tube with the leak. Subsequent Eddy Current Testing (ECT) inspection identified extensive unexpected tube free-span wear at the leakage location, typically not seen in recirculating SGs, and tube-to-support wear. A full-length ECT inspection of each tube (100%) in all four SGs using a bobbin coil probe was performed and provided a comprehensive extent of condition evaluation for tube free-span and tube-to-support wear. The bobbin coil probe inspection was supplemented by Rotating Coil (+Point) probe inspection, which provided further confirmation of the extent of condition. This supplemental rotating probe examination covered the U-bend portion of approximately 1300 tubes in each SG. This inspection identified additional tube wear indications in tube free spans, at AVBs, at TSPs, at retainer bars, and due to a foreign object.

NOTE:
The tube free-span wear was in the early stage of evaluation postulated to be produced by tube-to-tube contact, so henceforth it will be called “tube-to-tube wear.”

Extent of Condition - Tube-to-Tube Wear

Tube Degradation	Unit 2- Tubes Affected	Unit 3 – Tubes Affected
Tube-to-Tube Wear	2	326

Extent of Condition - Tube-to-Support Wear

Tube Degradation	Unit 2- Tubes Affected	Unit 3 – Tubes Affected
AVB Wear	1399	1767
TSP Wear	299	463
Retainer Bar Wear	6	4

Extent of Condition – Foreign Object Wear

Tube Degradation	Unit 2- Tubes Affected	Unit 3 – Tubes Affected
Foreign Object	2	0

Note 1: Unit 2 AVB, TSP, Retainer Bar and Foreign Object wear was addressed under Unit 2 RCE 201843216 which focused on unexpected SG tube-to-retainer bar wear. The Unit 2 RCE also addressed the Unit 3 tube-to-retainer bar wear as a part of the extent of condition.

Note 2: Unit 3 Tube-to-Tube, AVB, and TSP wear is being addressed under this Unit 3 RCE 201836127 which focuses on the unexpected tube-to-tube wear. This RCE is also addressing Unit 2 tube-to-tube wear as a part of the extent of condition.

See Attachment 04 for a partial set of diagrams depicting tube wear locations in Unit 3.

Extent of Tube-to-Tube Wear (Unit 3)

The primary to secondary tube leak in 3E088 was due to through wall tube-to-tube wear. ECT inspection of the tubes in 3E088 and 3E089 identified a total of 326 tubes with tube-to-tube wear. In-situ pressure testing was performed on 73 tubes in 3E088 and 56 tubes in 3E089 due to higher levels of tube-to-tube wear (the wear indications in these tubes exceeded the condition monitoring limits of the SG Program). This testing resulted in eight tubes in 3E088 that did not meet leakage and/or structural performance criteria as required by Technical Specifications. All other tubes in-situ tested met the performance criteria.

In-Situ Pressure Test Failures

Date	Notification	SG	Tube		In-Situ Test	Max Pressure** (psig) / Criteria Not Met
3/14/12	201897717	3E088	R106	C78	Failed*	2874 / Main Steam Line Break
3/14/12	201897883	3E088	R102	C78	Failed	3268 / Main Steam Line Break
3/14/12	201898071	3E088	R104	C78	Failed	3180 / Main Steam Line Break
3/15/12	201899579	3E088	R100	C80	Failed	4732 / 3 x Diff Pressure at Full Power
3/15/12	201899965	3E088	R107	C77	Failed	5160 / 3 x Diff Pressure at Full Power
3/15/12	201900019	3E088	R101	C81	Failed	4889 / 3 x Diff Pressure at Full Power
3/15/12	201900244	3E088	R98	C80	Failed	4886 / 3 x Diff Pressure at Full Power
3/16/12	201901456	3E088	R99	C81	Failed	5026 / 3 x Diff Pressure at Full Power

***Note 1:** 3E088 Tube R106 C78 is the tube with the leak that resulted in the Unit 3 forced shutdown on 1/31/2012.

****Note 2:** Max pressures are based on the ambient temperature.

These eight tubes were stabilized and plugged in accordance with the SG Program (see the Corrective Action Matrix). All other tubes with ECT indications of tube-to-tube wear will be stabilized and plugged in accordance with the SG Program (see the CA Matrix).

SONGS and MHI developed and implemented a Tube Plugging Screening process to identify tubes which may be susceptible to future tube-to-tube wear. Application of this process resulted in identification of 292 additional tubes in 3E088 and 3E089 that will be stabilized and plugged (see the CA Matrix).

The tube leak, eight in-situ tube pressure failures and a total of 326 tubes with tube-to-tube wear was unexpected in the recirculating SGs, and especially new SGs, half-way through their first operating cycle. Therefore, the tube-to-tube wear is a focus of this RCE.

Extent of Tube-to-Tube Wear (Unit 2)

At the time of the Unit 3 SG tube leak, Unit 2 was in the first refueling outage after SG replacement and undergoing ECT inspections per the Steam Generator Program. While initial 100% bobbin coil results did not identify tube-to-tube wear, station management decided to delay Unit return to service pending an evaluation of the susceptibility of the Unit 2 SGs to tube-to-tube wear similar to that seen in the Unit 3 SGs.

ECT inspection of approximately 1300 tubes in each Unit 2 SGs using a +P probe (more sensitive than bobbin coil) identified tube-to-tube wear at one location involving two tubes. The number of tubes selected was based on the same area of the problem tubes on Unit 3 and additional tubes in the surrounding area to ensure conservatism. Therefore, the +P probe inspection program scope was similar to the inspection scope implemented in the Unit 3 SGs. The location of the tube-to-tube wear in the Unit 2 SG was in the same region of the tube bundle as in the Unit 3 SGs. This indicates the existence of causal factors similar to those resulting in tube-to-tube wear in the Unit 3 SGs. The two affected tubes were stabilized and plugged in accordance with the SG Program.

The same Tube Plugging Screening process used in Unit 3 to identify tubes which might be susceptible to future tube-to-tube wear was applied to the Unit 2 SGs. This will result in preventive stabilizing and plugging of 316 additional tubes in 2E088 and 2E089 (see the CA Matrix).

Extent of Tube-to-AVB Wear (Units 2 and 3)

In the Unit 2 SGs, there were 1399 tubes with tube-to-AVB wear and in the Unit 3 SGs, there were 1767 tubes. Of these tubes, four tubes in Unit 2 and one tube in Unit 3 were stabilized and plugged in accordance with the SG Program (see CA Matrix). This evaluation considered two distinct wear patterns, one associated with tubes that also have tube-to-tube wear and the other associated with out-of-plane vibration. For the tubes that show tube to AVB wear that are not associated with the tubes exhibiting FEI, the wear is caused by turbulence induced vibration. The wear rate from this mechanism is lower than that associated with FEI and, based on industry OE, decreases over time. Although there are a high number of indications, with the exception of the tubes stabilized and plugged, the indications can be dealt with in the monitoring section of the SG Program. These tubes do not require additional causal analysis

Extent of Tube-to-TSP Wear (Units 2 and 3)

In the Unit 2 SGs, there were 299 tubes with tube-to-TSP wear and, in the Unit 3, there were 463 tubes. The primary contributor to the large difference in the number of affected tubes between units is the TSP wear caused by the high displacement vibration of the tubes that also experienced the tube-to-tube wear. Many of these tubes were stabilized and plugged as a result of both tube-to-tube wear and tube-to-TSP wear. After accounting for the tubes with tube-to-tube wear, the remaining tube-to-TSP wear required no additional tubes in Unit 2 or in Unit 3 to be stabilized and plugged in accordance with the SG Program (see CA Matrix). For the tubes that show tube to TSP wear that are not associated with the tubes exhibiting FEI, the wear is caused by turbulence induced vibration. The wear rate from this mechanism is lower than that associated with FEI and, based on industry OE, decreases over time. No additional causal analysis will be performed for tube-to-TSP wear because of 1) the close correlation between tube-to-tube and tube-to-TSP wear, 2) the similarity in tube numbers between units regarding wear at the TSPs otherwise, and 3) the increased monitoring of tube wear during mid-cycle outages.

Extent of Tube-to-Retainer Bar Wear (Unit 3)

Tube-to-retainer bar wear was first identified in the Unit 2 SGs by ECT during the first refueling outage following SG replacement. The wear was unexpected and RCE 201843216 was initiated to address this phenomenon. The tube wear was adjacent to the small diameter retainer bars. The identified cause was inadequate bar design, and the corrective action was to plug all 94 tubes adjacent to the retainer bars to eliminate any potential for occurrence of a primary to secondary leak. As a part of the extent of condition for the Unit 2 RCE, a focused ECT inspection was performed in the Unit 3 SGs, which identified three tubes with wear at the retainer bars greater than 35%. The corrective actions for the Unit 3 SGs included plugging these three tubes and preventive plugging of a total of 94 tubes adjacent to small diameter retainer bars in each Unit 3 SG. Selective stabilizing of the tubes to be plugged was also used to ensure tube structural integrity. No further analysis or actions are necessary for this issue under this Unit 3 RCE. See

RCE 201843216, “Steam Generator Tube Wear,” for a more complete description of corrective actions for the retainer bar wear issue.

Extent of Foreign Object Wear (Unit 3)

No tubes with indications of foreign object wear were detected during ECT inspections of the Unit 3 SGs.

EVIDENCE AND FACTS (SEQUENCE OF EVENTS)

4/1984
SONGS Unit 3 started commercial operation at nominal 100% power with new Combustion Engineering (CE) Steam Generators.

4/1984 through 10/10/10
During this period, Unit 3 was operated for 15 power cycles at nominal 100% reactor power level, except for outages and periods of operation at reduced reactor power.

10/10/10
Unit 3 started the Cycle 16 Refueling and Steam Generator Replacement Outage. The original CE SGs (3E088 and 3E089) were replaced with MHI SGs.

2/18/11
Unit 3 completed the Cycle 16 Refueling and Steam Generator Replacement Outage and returned to operation at nominal 100% reactor power.

1/31/12
Operators received a high radiation alarm from the condenser air ejector line. The alarm indicated that there was a primary-to-secondary leak in Unit 3 SG E088, indicating tube leak. The calculated tube leak was approximately 75 gallons per day and rising.

Equipment Failure: SG 3E088 had a tube leak.

1/31/12 15:05
Unit 3 Operators entered Abnormal Operation Instruction (AOI) SO23-13-14 (Reactor Coolant Leak) which was required for a primary-to-secondary SG tube leak exceeding 5 gallons per day.

1/31/12 16:10
Operators commenced AOI SO23-13-28 (Rapid Power Reduction) for a leak rate greater than 75 gallons per day with an increasing rate of leakage exceeding 30 gallons per hour.

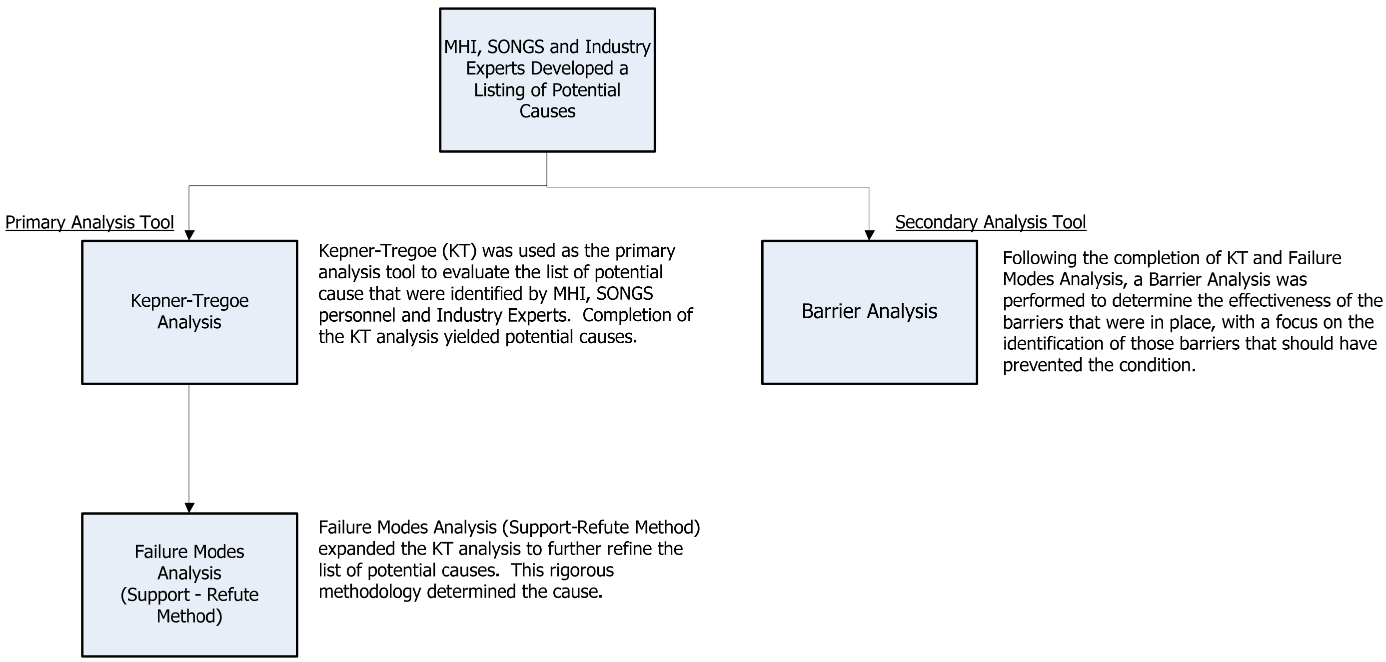
1/31/12 17:31
Operators manually tripped the reactor from 35% power.

1/31/12 17:38
Operators entered Emergency Operating Instruction (EOI) SO23-12-4 (Steam Generator Tube Rupture).

1/31/12 18:00
Operators isolated the affected SG.

ANALYSIS AND CAUSES

The following diagram provides a pictorial representation of the analysis methodology.



The initial part of the event analysis started with unit shutdown, cooldown, draindown of the primary side of the SG, and performance of ECT inspection to determine the location and type of wear. Once specific information regarding the wear became available, SONGS engineers, with the help of MHI and industry experts, developed a list of potential causes. In parallel with the engineering effort to understand the problem, the station established an RCE team. One of the primary functions of the RCE team is to use systematic processes to determine causes. The primary tool used by the RCE team to determine causes was a Kepner-Tregoe (K-T) problem analysis, expanded to include failure modes analysis using the support-refute methodology, and Barrier Analysis.

The K-T analysis started, using as input, the comprehensive list of (21) potential causes developed through the initial engineering assessment (see Attachment 03, K-T Analysis, for details). The list of potential causes was narrowed down, using facts, analysis, and expert input, to determine probable causes that warranted further technical evaluation to determine the mechanistic cause and contributors.

Once the eight probable causes were identified, a more rigorous analysis using both empirical and theoretical data and support-refute methodology was used to identify likely causes and to eliminate non-causes. Each area of concern was evaluated and documented by engineering analysis of the supporting and refuting data, and was approved by an SONGS quality review board (QRB). The QRB consisted of senior level engineers with SG experience and managers, including both SONGS personnel and independent reviewers (see Attachments 05 to 13 for details of analyses performed). The results of the analyses are documented in reports that are attached to this RCE with a summary provided below.

In addition to the engineering analyses performed as a part of this RCE, an expert panel comprised of experienced peers and industry experts in the areas related to SG design, fabrication, operation and inspection conducted an independent review of data, analyses, and industry experience to determine the cause of tube-to-tube wear in the SONGS SGs and define potential corrective actions. The expert panel included the representatives from Palo Verde Nuclear Generating Station, AREVA, Babcock & Wilcox Canada, EPRI, MPR, Polytechnique Montreal and INPO, as well as independent consultants. The represented expertise included SG design engineering, thermal-hydraulics analysis, flow-induced vibration and wear analysis, computer modeling, regulatory affairs and nuclear oversight. Results of the independent expert panel review of the wear issues are consistent with the findings presented in this RCE.

Independent of the K-T analysis, a barrier analysis was performed to assess technical and potential programmatic causes. This analysis was performed recognizing that MHI is performing an MHI specific root cause analysis that will address the MHI organizational and programmatic root causes, and that SONGS would be updating this RCE after receipt of MHI's Root Cause Analysis.

Independent of the K-T analysis, and after the KT analysis was completed, another potential cause was postulated. Although not part of the KT, once identified as a potential cause, a similar rigorous process was used to evaluate Secondary Side Acoustic Waves. This is included at Attachment 14. Because it was not part of the original KT, and it was not determined to be a significant contributor to tube vibration, it is not discussed below.

The specific areas analyzed using the processes described above included:

Wear Indications
This included analysis of the type, size, and location of the wear indications, measured or observed, and their interrelationship as relevant to determining the cause. This analysis was based on the results of the SG ECT and visual inspections and was a foundation for determination of the tube wear causes and corrective actions (Attachment 04 and 05).

Thermal Hydraulic (T/H) Conditions
This included analysis of what SG secondary side T/H conditions were necessary for the observed tube wear to occur, and whether those conditions could have existed in the SONGS SGs during operation. It also included analysis of tube response to these conditions, once they did occur. Part of the T/H Conditions, but included in separate analyses are TSP Distortion and Tube Bundle Distortion. The TSP distortion has both T/H aspects and fabrication aspects (see Attachment 07 for T/H Conditions, and Attachment 10 for TSP Distortion).

T/H and FIV Models
This included analysis of the T/H and FIV models used to predict the SG secondary side T/H conditions and tube responses to these conditions. The analysis includes actions taken to validate models that are being used to justify return to service of Units 2 and 3 (see Attachment 07 for details).

Manufacturing/Fabrication
This included analysis of the manufacturing (making of the parts) and fabrication (putting the parts together) to determine if either or both processes created mechanical conditions that directly

caused or contributed to the observed tube wear. Part of Manufacturing/Fabrication, but included in separate analyses are Divider Plate Weld Failure and TSP Distortion, which also includes effects of the T/H conditions (see Attachment 08 for analysis details).

Shipping
This included analysis of the shipping conditions to determine if shipping caused mechanical changes after fabrication was complete that could directly cause or contribute to the observed tube wear (see Attachment 11 for details).

Primary Side Flow Induced Vibration
This included analysis of primary system flow effect on the SG tubes, with emphasis on the reactor coolant pump impeller blade passing frequency, to determine if it could cause or contribute to the observed tube wear (see Attachment 12 for details).

Vibration/Loose Parts Monitoring System (VLPMS) Data
This included analysis of VLPMS data to determine if there was an opportunity for detection of tube vibration prior to the tube failure. This analysis is not part of the K-T analysis as it cannot point to a mechanistic cause; it can only identify a potential opportunity to take action to prevent tube failure (see Attachment 13 for details).

The order of the above list is not indicative of the order in which the analyses were performed. The focus of the causal analysis was broad based and was adjusted as information became available with much of the information feeding into each potential cause. The order provided in this report was selected based on the progress of the analysis and the knowledge of what the causes were ultimately determined to be. Therefore, the order represents that which provides the reader with the logical progression from the indications to the cause. This is followed by analysis of the potential causes that were eliminated. The sections below are summaries of the full analyses, which are included as attachments.

Wear Indications

The purpose of this analysis was to confirm that the wear in the free-span sections of the tubes identified in the Unit 3 SGs was produced by tube-to-tube contact.

SONGS conducted initial SG tube ECT inspections using the industry standard bobbin coil probe, which identified multiple tubes with wear indications. Once the extent of the wear was qualitatively identified, SONGS used a more sensitive +Point probe to accurately determine the type and quantify the magnitude of the indicated wear. Also, visual inspections were performed of the region where the majority of wear indications were found. The ECT and visual inspection results were then analyzed to determine the wear types and what produced these wear types. The key findings of the analysis of the ECT and visual inspection results are:

1. Significant wear occurred in the tube free-span sections of the U-bend region of the tube bundle.

2. A large number of tubes in the Unit 3 SGs have wear in the free spans and at AVBs. See Extent of Condition for details.

3. A large number of tubes in the Unit 3 SGs have wear in the free spans and at TSPs. See Extent of Condition Section for details.

4. A shiny wear surface was observed on the tubes with wear in the free spans.

5. The wear in tube free spans is at 90° from the tube-to-AVB contact points indicating that the tubes were moving and making contact in the in-plane direction.

6. In general, the ECT inspection results show that the wear in the free spans is on the same tubes in both the hot and cold leg side of the U-tube.

7. According to the ECT inspection report, all wear indications in the free spans have a facing match, which confirms tube-to-tube contact.

8. Most of the wear locations are in the vicinity of AVB B03 and B09. This means that the free-span wear occurred about half way between the top TSP and the center of the U-bend.

9. Post-test ECT inspection of the tubes which failed the in-situ pressure test revealed that the tubes leaked at the location of the observed wear in the free spans.

Based on the above, it is concluded that the tube wear in free spans was produced by tube-to-tube contact in the in-plane direction and that a resulting tube-to-tube wear ultimately led to the tube leak in Unit 3 SG E088 (see Attachment 05 for analysis details).

Thermal Hydraulic (T/H) Conditions

The purpose of this analysis was to determine what SG secondary side T/H conditions could result in movement of the tube bundle components (tubes and AVBs) such that it caused, or contributed to, tube-to-tube wear seen in the Unit 3 SGs.

The steam/water two-phase flow in SGs creates T/H conditions which cause vibration of the SG internals, including the tubes. There are three major types of vibration caused by such flow: (1) turbulence-induced vibration, (2) vortex-induced vibration (vortex shedding), and (3) fluid-elastic instability (FEI). Although FEI is a complex phenomenon, there are two major T/H parameters that may cause it - high fluid velocity and high void fraction. Fluid velocity is a function of the mass flow rate, local steam quality and local U-bend geometry (spacing between the tubes). Void fraction is a function of saturation temperature. Turbulence-induced vibration is by itself not capable to cause tube-to-tube wear of the magnitude seen in the Unit 3 SGs. In the enclosed geometry of the SGs and in the two-phase flow with a high void fraction, vortex shedding does not occur. Therefore, it is postulated that the mechanistic cause of tube-to-tube wear observed in the Unit 3 SG is FEI.

In tube arrays, such as in the SGs, FEI manifests itself by uncontrollable, high amplitude vibration of the tubes in synchronous oval orbits (in the in-plane and out-of-plane directions). FEI has a negative effect on tubes including premature wear at support points and wear in the free-span sections (between tube support points) due to tube-to-tube contact. Because of these negative effects, designers evaluate the parameters that cause FEI using T/H modeling, FIV analysis, and additional engineering evaluations to assure tubes are not subject to FEI. The primary indicators that FEI occurred in the SONGS SGs are:

1. Secondary side mass flow at 100% power is capable of producing high fluid velocities in the tube bundle region where the wear was observed.

2. Secondary side saturation temperature at 100% power is capable of producing high void fractions in the tube bundle region where the wear was observed.

3. The results of ECT and visual inspections confirm tube-to-tube free span wear.

4. The SONGS SGs are designed with the AVBs that prevent out-of-plane tube vibration, but do not have provisions to prevent in-plane tube vibration.

Based on the above, it is concluded that the T/H conditions are a very likely contributor to the mechanism causing tube-to-tube wear seen in the Unit 3 SGs (see Attachment 06 for analysis details).

Tube Bundle Distortion (an aspect of T/H conditions)

In the original design, the AVB support was considered active if there were no gaps between the tube and the AVB in the hot condition (during operation). The tube-to-AVB gaps which were expected in the cold condition were measured in order to meet the hot-no-gap requirement. Until discovery of significant tube-to-tube wear in the SONGS Unit 3 SGs, U-bend support provided by the AVB, if properly designed and installed, was deemed sufficient to prevent in-plane tube motion based on the state-of-knowledge at the time the SONGS SGs were designed and fabricated.

Following Unit 3 wear discovery, MHI postulated that a “zero” gap in the hot condition does not necessarily ensure that the support is active and that contact force between the tube and the AVB is required for the support to be considered active. The most likely cause of the observed tube-to-tube wear is multiple consecutive AVB supports becoming inactive during operation. This is attributed to redistribution of the tube-to-AVB-gaps under the fluid hydrodynamic pressure exerted on the tubes during operation. This phenomenon is called by MHI, “tube bundle flowering” and is postulated to result in a spreading of the tube U-bends in the out-of-plane direction to varying degrees based on their location in the tube bundle (the hydrodynamic pressure varies within the U-bend). This tube U-bend spreading causes an increase of the tube-to-AVB gap sizes and decrease of tube-to-AVB contact forces rendering the AVB supports inactive and potentially significantly contributing to tube FEI.

The flowering phenomenon has not been identified or analyzed for other SGs in the industry. MHI analyzed this postulated condition using finite element modeling and determined that the area where tube-to-AVB gap sizes are shown to increase under dynamic pressure correlates well with the area where tube-to-tube wear was observed in the Unit 3 SGs. SONGS and industry subject matter experts reviewed the approach and results of the MHI analysis and had several comments that require further investigation. For the purpose of this RCE, “flowering” is being considered as a potential contributor to the mechanism causing the wear, pending further evaluation by MHI and other industry subject matter experts.

Based on the above, it is concluded that at this time tube bundle distortion cannot be ruled out as contributing to the mechanism of tube-to-tube wear seen in the Unit 3 SGs.

Thermal Hydraulic (T/H) and Flow Induced Vibration (FIV) Models

During the original design of the SONGS SGs, MHI used a number of computer codes to analyze the design to substantiate that FEI would not occur. The analyses start with operating parameters, such as fluid temperatures and flow rates from the primary side of the reactor system, and SG dimensions as inputs to the computer codes.

The design and operating parameters were then input to an MHI developed computer code Steam Generator Steady State Performance Code (SSPC). This code is a 1-dimensional thermal-hydraulic calculation code that calculates global SG parameters, such as the tube bundle circulation ratio (a ratio of total bundle mass flow to downcomer flow), and other secondary side operating conditions.

The SSPC results and other design inputs were then used by MHI in a detailed thermal-hydraulic analysis of the SG tube bundle using FIT-III Version 1. FIT-III is a three-dimensional analytical code developed by MHI for PWR SG secondary side detailed thermal-hydraulic analyses. The model simulates the secondary side from the tube sheet to the exit of the moisture separators. Results of the FIT-III analysis for SONGS SGs provided a prediction of the maximum steam quality and void fraction in the U-bend region as well as their distributions.

The results of FIT-III were then used by MHI in a fluid-elastic analysis using the Fluid Induced Vibration Analysis of Tubular Structures (FIVATS). The code ultimately is used to determine if the SG tubes are subjected to thermal hydraulic and structural conditions that may result in their fluid elastic instability. This is demonstrated by the calculation of a stability ratio, consisting of the ratio of an effective fluid velocity passing a tube to the critical fluid-elastic velocity. A stability ratio that is less than 1.0 indicates the tube is in a stable region and is not subject to FEI.

During the SONGS SG design, MHI used FIVATS results to conclude that stability ratios in the SG tube straight leg sections were less than 1.0 and therefore judged to be a region of stable fluid elastic vibration. For the U-bend region, MHI assumed that one of the twelve AVB supports was not active, and confirmed that the stability ratios for all tubes remained less than 1.0 and concluded there is “negligible possibility of fluid elastic vibration”.

Based on the Unit 2 and 3 SG tube inspection results, MHI and SONGS (this RCE) now recognize that FEI did in fact exist in regions of the SGs, and that the combination of T/H conditions and lack of effective tube AVB supports resulted in FEI. This condition was not predicted by the MHI methods for the design of the SONGS SGs. The reason that the design codes and assumptions did not predict FEI is not yet understood and this will not be the subject of further evaluation within this RCE. Additional analysis will occur, after receipt of MHI’s organizational and programmatic RCE.

To develop the T/H operating conditions and to justify plant return to service, MHI, working with SONGS and other industry experts, are developing T/H models for analysis of the SONGS SG design. The analysis results will then be used in the Flow Induced Vibration analysis to set the operating conditions for restart and used in the operational assessment. To validate the ATHOS[1] model, SONGS commissioned independent T/H evaluations of the SONGS SGs to be performed by both Westinghouse (WEC) and AREVA. WEC maintains its own version of ATHOS. AREVA uses a French code called CAFCA. The intention is to perform two independent analyses to compare their results to the latest results obtained by MHI using the refined ATHOS model. Preliminary results of the independent analyses indicate very good agreement with the MHI results.

To support plant restart, the results from the latest MHI, WEC, and AREVA T/H models will be compared and judged based on a comprehensive list of criteria. Ultimately, the MHI ATHOS results will be used to establish operating plant conditions, such as reactor power, to be used by SONGS and AREVA for determining the final repairs and justification for plant operation.

[1] ATHOS is the SG thermal hydraulic model developed by EPRI and used extensively in the US.

Additionally, these results will be compared with FIT-III (see Attachment 07 for details and CA Matrix for action tracking).

Manufacturing/Fabrication

The purpose of this analysis was to determine if the SG manufacturing/fabrication processes could cause or contributed to tube-to-tube wear.

The history of manufacturing and fabrication of the SGs and the ECT data was reviewed to identify issues that might have resulted in tube-to-tube wear. As a part of this review, it was noted that there were manufacturing and fabrication differences between the Unit 2 and Unit 3 SGs. Investigation of these differences was performed because of the significant difference in tube wear in the Unit 3 and Unit 2 SGs. The potentially significant manufacturing and fabrication differences were:

- Tube and AVB dimensional dispersion
- Tube-to-tube spacing
- AVB thickness
- Tube-to-AVB gap sizes
- Divider plate weld failure and repair (addressed in a separate section below)

The results of this review were as follows:

1. The standard deviations of the outer diameter of tubes (G-values) are smaller for the Unit 3 SGs than for the Unit 2 SGs.
2. The number of adjustments of tube bending radii, to control the tube-to-tube spacing which had to be performed on the Unit 3 SGs was significantly less than adjustments on the Unit 2 SGs.
3. The AVBs were manufactured within design tolerances.
4. The ECT data suggests (through reduced voltage readings) that tube-to-AVB gaps in the Unit 3 SGs are slightly larger than in the Unit 2 SGs.

The as-built tube-to-AVB gaps were measured after each tube bundle was fully assembled, but due to measurement technique limitations, only the gaps between the outermost tubes and their respective AVBs could be measured. This means that the as-built gap sizes inside the tube bundle were not known; as it is undetermined if and how the peripheral gap sizes correlate with the gap sizes inside the tube bundle. However, larger gaps in the cold condition generally mean greater probability of the tube supports at AVBs becoming inactive during operation.

The facts identified in this analysis indicate that the Unit 3 tube bundle components (tubes and AVBs) might have been manufactured with greater precision, the as-built tube-to-AVB gaps might be indeed larger in the Unit 3 SGs as suggested by the ECT results and more uniform than in the Unit 2 SGs. This could result in reduction of the tube-to-AVB contact force and consequently in multiple consecutive AVB supports being inactive during operation, a condition necessary for FEI to occur.

Based on the above, it is concluded that differences in manufacturing/fabrication could contribute to the mechanism of tube-to-tube wear seen in the Unit 3 SGs (see Attachment 08 for analysis details).

Divider Plate Failure/Repair

The purpose of this analysis was to determine if the divider plate weld failure/repair could cause or contributed to tube-to-tube wear.

It is reasonable to assume that the failure of the divider plate-to-channel head weld during hydrostatic testing caused deformation of the tubesheet. Even though such deformation was most likely elastic, it had a potential to cause plastic deformation of the tubes which are anchored to the tubesheet. It also had the potential to cause plastic deformation of the TSPs which are connected to the tubesheet by means of the rigid stay rods, and are much thinner than the tubesheet. Such plastic deformation could have altered the tube bundle geometry by decreasing spacing between tubes and/or gaps between the tubes and AVBs, and lead to the tube-to-tube wear observed in the Unit 3 SGs.

The repair of the SG divider plate weld included: cutting the channel head from the tubesheet, re-welding the divider plate to the channel head, re-welding the channel head to the tubesheet, additional post weld heat treatment (PWHT), additional hydrostatic tests and an additional approximately 300 tube bundle rotations associated with the repairs. All of these repair activities had the potential to cause deformation/displacement of the tubesheet, TSP or tubes, potentially leading to the observed wear. However, the following indicates that they did not cause tube-to-tube wear:

1. No distortion of the tubesheet was observed during primary side visual inspections, recognizing the limited capability of visual inspections to detect distortion.

2. The calculated maximum displacement for the tubesheet (and thus the TSPs and tubes, excluding the U-bend region) is 0.189” in the vertical direction.

3. It is calculated that each tube bundle rotation could alter the center column tube-to-AVB gap by 2.0×10^{-6} mm. With the additional approximately 300 tube bundle rotations associated with the divider plate weld repair, this would equate to 6.0×10^{-4} mm, which is of no consequence to tube wear.

4. In general, the flame cutting, welding, and PWHT involved in the repair would only affect tubes in the periphery of the tube bundle. Additionally, temperature profiles for the tubesheet and tubes during PWHT were determined analytically, monitored, and evaluated, and it was concluded that the temperatures were not sufficient to produce plastic deformation. Multiple heat treatments would not produce temperatures required for plastic deformation.

5. Unit 2 SGs did not experience a divider plate weld failure/repair and the SG show indications of tube-to-tube wear. This indicates that tube-to-tube wear is not a direct result of divider plate weld failure/repair.

Based on the above, it is concluded that the likelihood of the divider plate weld failure and associated repairs contributing to the mechanism causing tube-to-tube wear seen in the Unit 3 SGs is very low (see Attachment 09 for analysis details).

TSP Distortion

The purpose of this analysis was to determine if TSP distortion could cause or contributed to tube-to-tube wear.

Tube bundle deformation was postulated as a potential cause of tube-to-tube wear in the Unit 3 SGs. Such deformation could be an effect of TSP distortion resulting in a reduction in the distance between the tubes in two adjacent rows of the same column, potentially leading to tube-to-tube wear. TSP distortion might also affect tube support conditions, which would make TSP distortion a contributor to flow induced vibration mechanisms. There are two mechanism that had a potential to affect the TSP geometry - one directly and one indirectly. The mechanism which could directly affect the TSP geometry is transient elastic distortion of the TSP during operation due to differential thermal growth between the wrapper and the stay rods, which would increase progressively in the direction of TSP #7. The mechanism which could indirectly affect the TSP geometry is permanent plastic distortion of the TSP due to divider plate-to-channel head weld failure as described in the preceding section. However, the primary indicators that TSP distortion did not occur or did not cause tube-to-tube wear are:

1. The tube-to-tube wear map indicates that the region with tube-to-tube wear is located asymmetrically in the tube bundle and is relatively small in size. The mechanical nature of TSP distortion caused by thermal expansion of the stay rods during operation should cause a uniform deflection of the tube sheet, and hence the region with wear should be located symmetrically and be larger in size.

2. The location tube-to-tube wear caused directly by TSP distortion would be within the first ~6” above TSP #7, which is the end of the tube straight leg section or Row 142, however there are no indications of wear at that location.

Based on the above, it is concluded that TSP distortion did not contribute to the mechanism causing tube-to-tube wear seen in the Unit 3 SGs (see Attachment 10 for analysis details).

Shipping

The purpose of this analysis was to determine if SG shipping configuration could cause or contributed to tube-to-tube wear.

All SGs were shipped in the horizontal position with the tube bundle U-bend oriented 45^o off the gravity neutral position and without any temporary tube bundle support. It was postulated that shipping the Unit 3 SGs in a horizontal position could have negatively impacted the geometry of the tube bundle U-bend region where significant wear was observed by causing plastic deformation of the selected tubes. Such deformation might have occurred because the tubes might have sagged excessively under their own weight and the weight of the AVB assembly. Additionally, it is possible that small transient dynamic loads during transport might have created sufficient stresses to cause plastic deformation of the tubes or movement of the AVB assembly in a way that could affect tube-to-tube and tube-to-AVB spacing.

MHI assessed the sagging of tube bundle by the deadweight. The test concluded that while the bundle could sag, it would not sag to the extent that the tubes could be pinched at TSP #7 and plastically deformed. From a tube support perspective, TSP #7 is the location where excessive stress during shipment would have manifested themselves. The ECT inspection performed during the outage did not reveal pinched tubes at the TSP #7, so it is reasonable to conclude that significant movement of the bundle/AVB assembly did not occur during shipping. Additionally, accelerometers were installed for shipping to monitor for motion (seismic or handling) that could possibly deform the tubes due to transient dynamic loading. Lastly, if sagging had caused some deformation, the wear should be similar in all four SGs in terms of type and magnitude since all four SGs were shipped in the horizontal position with the tube U-bend 45^{o} off the gravity neutral position.

Based on the above, shipment did not contribute to the mechanism causing tube-to-tube wear seen in the Unit 3 SGs (see Attachment 11 for analysis details).

Primary Side Flow Induced Vibration

The purpose of this analysis was to determine if the SG primary side flow could cause or contribute to tube-to-tube wear.

The Reactor Coolant Pumps (RCPs) impeller vane passing volute/discharge port creates primary fluid pressure pulsations with the frequency equal to that of the vane passing frequency. If the frequency of these pulsations is equal to or close to the natural frequency of the tube through which the fluid flows, tube excitation will occur. It was postulated that this excitation could result in tube vibration, possibly resulting in tube-to-tube wear.

The RCP impellers rotate at 1180 rpm (in three RCPs) and at 1194 rpm (in one RCP) as dictated by the prime mover (electric motor). This translates to 19.7 and 19.9 revolutions per second. The RCP impellers have 5 vanes, therefore, the vane passing frequency, and hence the pressure pulse frequency, is five times higher, i.e., is approximately 98.3 and 99.5 Hz. This is the basis for analyses that refer to “95-100 cps” as the pulsation frequency. The maximum amplitude of the pressure pulses generated at the vane passing frequency does not exceed 8psi (+/- 4 psi from the normal operating pressure). For comparison, the RCS normal operating pressure is 2250 psia. Based on MHI analysis, the natural frequency of the tubes with wear in the U-bend region (~100Hz) is close to that of the vane passing frequency when three or more AVB supports are assumed inactive. This suggests that the primary fluid flow pulsations could excite some tubes.

The RCPs are located in each of the RCS cold legs, downstream of the SG. The pumps discharge into the reactor vessel, so the pressure pulsations are dampened by the reactor vessel internal structures and fuel assemblies. The Pressurizer is attached to one of the RCS hot legs, which serves as a large damper to any fluctuations in RCS pressure. Then, these attenuated pulses travel to the SG channel head, where the tubesheet further dissipates their energy. The pressure pulsations which reach the SG tubes are judged to be of a negligible magnitude.

Based on the fact that all tubes in all SGs are subject to the same ~100 Hz forcing function, but only relatively few tubes in a concentrated area of the Unit 3 SGs experienced significant tube-to-tube wear, it is concluded that the primary flow did not contribute to the mechanism causing tube-to-tube wear seen in the Unit 3 SGs (see Attachment 12 for analysis details).

Evaluation of Ability to Detect Condition Prior to Failure
Vibration/Loose Parts Monitoring System (VLPMS)

The purpose of this analysis is to determine if the VLPMS might have provided indication of the tube-to-tube wear/failure on Unit 3 SGs. There are sixteen piezoelectric sensors and sixteen preamplifiers located inside Containment to provide inputs to the twelve Loose Parts and Four Vibration Channels. Two Accelerometers are mounted on each SG. They are mounted on the support skirt. The SG support skirt is a separate assembly welded to the bottom of the SG. Per the NRC Regulatory Guide 1.133, “Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors,” “the primary purpose of the loose-part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate safety-related damage to or malfunctions of primary system components.”

There were four factors associated with the VLPMS that indicate it may have been capable of detecting the tube-to-tube vibration that was occurring during the cycle:

1. Multiple alarms on various VLPMS channels after new SGs were installed in R3C16.

2. Primary side of the SG were inspected in U3C16, and no indication of loose parts were found on 3E088 (NMO 800842826) or 3E089 (NMO 800842830). This indicates that the alarms could have been initiated by secondary side noise.

3. Westinghouse Impact Analysis of Unit 3 determined impacts to be metal to metal (Attachment 3 Westinghouse, “SONGS Unit 3 Impact Analysis” ITS3206 Rev. 0)

4. Valid alarms were seen in Unit 3 and not in Unit 2 during the C16 operating cycle. The valid alarms could have been caused by the tube to tube contact.

Engineering analyses of the various alarms received determined the source of the alarms to be SG motion. Engineering responses are documented in NNs 201790804-1 (2/23/12) and 201818719-2 (3/15/12). WEC could not conclusively differentiate between the noise signatures of the valid alarms during temperature changes and steady state. Per the Westinghouse Impact Analysis of Unit 3, "the events on both Unit 3 SGs are the result of true metallic impacts and not false indications from electrical noise or fluctuations in background noise. WEC found that the events that occurred prior to the forced outage were similar to the events that occur when the SGs shift during RCS temperature transients. However, WEC cannot conclusively state that the events are from the same source without additional data for comparison and evaluation. Even with additional data, determination of the source of the impacts could be hindered by the location of the sensors."

Based on the analysis and the fact that the alarms were discrete events, it is concluded that they were not indicative of on-going vibration and therefore did not provide an indication of tube-to-tube contact. (See Attachment 13 for analysis details)

Analysis Summary

The preponderance of evidence shows that the tubes in the adjacent rows within the same column contact each other in the U-bend region experiencing wear. This means that the tubes move in the in-plane direction. Tube movement in this direction is possible, as the SONGS SGs by design have provisions to restrict tube movement in the out-of-plane direction only. The results of the ECT and visual inspections confirm that the tubes moved in the in-plane direction causing wear.

In order for a tube to move in the in-plane direction, the tube has to become fluid-elastic unstable, i.e., the velocity of the fluid flowing around this tube has to be equal to, or greater than, the critical velocity for this particular tube. The critical velocity is a direct function of the tube natural frequency, tube damping ratio in vacuum, tube mass per unit length and tube outside diameter (OD) and indirect function of the tube pitch-to-OD ratio in the U-bend region, which is determined by the design. Another essential contributor to the tube becoming unstable is inadequate condition of its supports. If the tube becomes fluid-elastic unstable, it starts to vibrate uncontrollably (in the in-plane and out-of-plane directions) with a high amplitude. If not restrained, the tube may degrade and subsequently fail.

As described earlier, the tube bundles in the SONGS SGs are exposed to a high velocity fluid flow field (due to high mass flow), damping ratios are low (in the high void fraction region) and tube natural frequencies are low (lack of adequate tube support). The fluid velocity and tube damping ratio are a function of the SG secondary side T/H conditions during operation. The tube natural frequency is a function of the tube span length between the supports and support condition (the number of consecutive supports that may be inactive) during operation. The combination of these factors creates conditions conducive to tube becoming fluid-elastic unstable. The results of SGs inspections and analyses confirm that the tubes showing tube-to-tube wear were fluid-elastic unstable.

Mechanistic Cause: The mechanistic cause of the observed tube-to-tube wear in Unit 3 is tube FEI due to a combination of secondary side T/H conditions and tube support conditions.

All tubes with tube-to-tube wear will be stabilized and plugged; all tubes with any type of wear exceeding the TS limit of 35% will be also stabilized and plugged. The tubes determined susceptible to tube-to-tube wear in the future will be preventively plugged.

In addition, the corrective actions include refinement of the existing, and possibly development of new, T/H and FIV models, which will allow for more accurate prediction of tube bundle and individual tube behavior under operating conditions. The models will be used to establish operational T/H limits (lower reactor power) such as to minimize the tube susceptibility to FEI. The time permitted for operation will be limited such that any additional wear, should it occur, will not result in the loss of tube integrity. After a limited operating period, additional inspections will be performed to determine effectiveness of currently implemented corrective actions and establish new corrective actions, if necessary, based on the results of the inspection (see CA Matrix). The Steam Generator program (SO23-SG-1) includes a degradation assessment section that requires: identification of actual and potential degradation mechanisms, monitoring limits for that mechanism, and measuring techniques for detection and sizing. Because tube to tube wear has been identified as a new degradation mechanism, it will be programmatically captured within the program and must be addressed in the condition monitoring and operational assessment section.

EXTENT OF CAUSE

The mechanistic cause of the tube-to-tube wear in Unit 3 SG E088 and E089 was identified to be tube FEI. The occurrence of tube FEI was caused by the combination of localized high water/steam mixture flow velocities (high excitation forces), high void fractions (low vibration damping ratios) and inadequate condition of multiple AVB supports in the U-bend region due to low, or lack of, tube-to-AVB contact forces (low tube natural frequencies). The potential extent of cause involves the Unit 2 SG E088 and E089.

At the time of the Unit 3 SG tube leak, Unit 2 was in the first refueling outage after the SG replacement outage, and undergoing ECT inspection per the Steam Generator Program. While initial 100% ECT results did not identify tube-to-tube wear, station management decided to delay Unit 2 return-to-service pending evaluation of its susceptibility SGs to tube-to-tube wear seen in Unit 3. Additional ECT inspection in Unit 2, using a more sensitive probe identified tube-to-tube wear involving two adjacent tubes in one location, indicating Unit 2 susceptibility to the same wear mechanism as, but to a much lesser extent, than seen in Unit 3. Based on available information, this is due to the greater variation in tube diameters and AVB thicknesses in the Unit 2 SGs as compared to the Unit 3 SGs (manufacturing tolerances improved from Unit 2 to Unit 3 resulting in less contact force between tubes and AVBs).

The same Tube Plugging Screening process used for Unit 3 to identify tubes which may be susceptible to future tube-to-tube wear was applied to the Unit 2 SGs. This review will result in preventive stabilizing and plugging of 316 additional tubes in 2E088 and 2E089 (see the CA Matrix).

OPERATING EXPERIENCE

A search for related internal and external Operating Experience (OE) was performed by the RCE Team using the SONGS OE - Operating Experience (TOPIC Information Server) search database. Other search databases that were used for the OE search included: SONGS SAP/ActionWay, the INPO (IERs and SOERs) website, and the NRC website. The search encompassed a review of events over the past eight (8) years using the following key words and combinations of key words, such as: “steam generator, replacement steam generator, new steam generator, tube wear, tube leak, tube-to-tube, retainer bar, retaining bar, anti-vibration bar, wear, fluid elastic instability and flow induced vibration.” The events identified in the search were reviewed and those most relevant are discussed in Attachment 16, Operating Experience.

The review did not identify a missed-opportunity for SONGS to use industry or site OE in developing design specifications for the new SGs. There were no events identified at other stations involving tube-to-retainer bar wear, and there were only three events involving tube-to-tube wear. Two of these events involved Once-Through Steam Generators (OTSGs) and only one recirculating SGs, such as at SONGS.

Three Mile Island (TMI), equipped with OTSGs, reported tube-to-tube wear in November 2011. The cause of the tube-to-tube contact wear was under evaluation by the component manufacturer at the time of OE issue. Arkansas Nuclear One (ANO) also equipped with OTSGs, reported tube-to-tube wear in December 2011. The discovery of tube-to-tube wear came after review of TMI’s November 2011 OE, review of previous ANO SG inspection data, and after recognition of tube-to-tube wear being mischaracterized by ANO personnel. There was no discussion in the ANO Apparent Cause Evaluation as to the cause of the tube-to-tube wear. These two OEs are not considered to be missed opportunities by SONGS due to the differences in SG design and the timing of the events (November and December 2011). SONGS SGs were designed, fabricated and installed prior to these events, and discovery of SONGS tube-to-tube wear was the result of ECT inspection following the January 2012 SG tube leak shutdown.

A third OE at Palisades involved one tube-to-tube wear indication in a recirculating SG with a square bend tube bundle U-bend design. A formal RCE was not performed, but Palisades noted that the likely cause was due to “manufacturing tolerances associated with tube bending for the square bend region,” and a possible cause due to a “square bend with bend angle not equal to 90 degrees.” Additional information was identified discussing tube-to-tube wear in original SGs (since replaced) at Palo Verde, however the information was not readily available as industry operating experience within the INPO (IER, SER, etc.). These OEs were not considered to be missed opportunities due to the lack of readily available information.

The review of site OE going back approximately eight years did not identify previous problems with the SONGS original SGs with respect to tube-to-retainer bar wear, tube-to-tube wear, or FEI in general. Thus, there was no missed opportunity for SONGS to identify and address the potential for these types of wear in the new SGs.

In summary, although tube-to-tube wear had occurred in the industry in the past, it was not comparable in magnitude to that experienced at SONGS. SONGS believes that SONGS’ experience would be a benefit for other stations and will distribute an Outgoing OE to the industry to inform others of the potential for tube-to-tube wear with FEI (see CA Matrix).

SAFETY SIGNIFICANCE

There was no actual safety significance relative to the as-found degraded condition of the Unit 3 SG tubes, and both units remain shutdown. The Unit 3 shutdown on January 31, 2012, due to a SG tube leak, resulted in small, monitored radioactive releases to the environment, well below allowable limits. The potential safety significance of the degraded condition of the Unit 3 SG tubes is discussed below.

The SONGS Updated Final Safety Analysis Report (UFSAR) Section 15.10.1.3.1.2 presents the current licensing basis steam line break post-trip return-to-power (post-trip SLB) event. The post-trip SLB RCS activity concentration limits are equivalent to 1.0 microcuries/gram Dose Equivalent Iodine-131 (DEI) and 725 microcuries/gram Dose Equivalent Xenon-133 (DEX). The post-trip SLB also considers an accident-induced (concurrent) iodine spiking factor of 500. The post-trip SLB is evaluated at the TS 5.5.2.11.b.2 limit for primary-to-secondary SG tube leakage of 0.5 gpm into the affected SG and 0.5 gpm into the unaffected SG. UFSAR Section 15.10.1.3.1.2 concludes that the post-trip SLB event Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses are each less than 0.1 Rem Total Effective Dose Equivalent (TEDE).

The January 25, 2012, RCS chemistry sample (the most recent sample prior to the January 31, 2012 tube leak/manual reactor trip) was evaluated and determined to be equivalent to RCS activity concentrations of 3.4E-04 microcuries/gram DEI and 0.325 microcuries/gram DEX. These actual RCS activity concentrations are a factor of at least 2230 times less severe than the concentrations modeled in the post-trip SLB dose presented in the UFSAR. It is estimated that had the post-trip SLB occurred, then the actual primary-to-secondary leakage rate through the ruptured (affected) SG tubes would be no more than 1500 gpm. Based on the actual plant RCS chemistry data, the accident-induced iodine spiking factor of 500, and the estimated SG tube rupture leakage rate, the calculated dose would have been at least 32 percent lower than the dose consequences reported in the UFSAR for the post-trip SLB event with a concurrent iodine spike. The postulated post-trip SLB with tube rupture and concurrent iodine spike Exclusion Area Boundary, Low Population Zone, and Control Room doses would be less than 0.068 Rem TEDE, which is well below the post-trip SLB CR limit of 5 Rem TEDE, and the EAB and LPZ limit of 2.5 Rem TEDE.

SAFETY CULTURE ASSESSMENT

This SONGS RCE is focused on determination of the mechanistic cause for steam generator tube-to-tube wear. No safety culture components (human performance, programmatic, and organizational issues) were identified involving SONGS as defined in SONGS Procedure SO123-XV-50.CAP-3 (Evaluations and Action Plans).

The newly installed SGs were purchased from MHI and, in accordance with their Appendix B program, MHI will be performing a RCE to understand the human performance, programmatic and organizational issues involved in SG design/fabrication and the mechanistic cause of FEI. MHI's RCE should provide insights into safety culture issues. SONGS RCE will be revised to include a Safety Culture Review after receipt of MHI's RCE (See CA6 in the CA Matrix).

CORRECTIVE ACTION MATRIX

Cause Evaluation Element	Corrective Action Description	Owner and Tracking
Problem: Unit 3 SG E088 tube leak.	**CA1 (for condition):** SONGS stabilized and plugged the tube with the leak in E088 (R106 C78). In addition, SONGS plugged and stabilized the 7 other tubes that did not pass in-situ pressure testing (R102 C78, R104 C78, R100 C80, R107 C77, R101 C81, R98 C80, and R99 C81).	Owner: A. Matheny CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. Documentation of this completed action will be included in the closure of CAPR1.*
Interim Action: .	**IA1:** SONGS established a Units 2 and 3 Steam Generator Recovery Project Organization. SONGS also obtained the services of industry steam generator designers, manufacturers and consultants to conduct and/or independently review inspections, testing, modeling, failure analysis, repair plans, and corrective actions.	Owner: J. Brabec CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation entry only.*
	IA2: SONGS applied conservative decision making and delayed return-to-service of Unit 2 following the Cycle 17 Refueling Outage pending Unit 3 Steam Generator failure analysis. The concern was the potential susceptibility of Unit 2 Steam Generators to the unexpected tube-to-tube wear mechanism identified in Unit 3 that resulted in a tube leak.	Owner: SLT (Brabec) CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation entry only.*
	IA3: Maintain U2 and U3 shutdown until the cause of the tube leak is thoroughly understood and actions to prevent additional	Owner: J. Brabec CR: 201836127 Due Date: 7/15/2012

	tube failures resulting in a leak are completed.	
Extent of Condition Unit 3: Tube-to-Tube Wear in 3E088 and 3E089, and potential wear. Also includes AVB & TSP wear.	**CA2:** SONGS to establish criteria, review tube wear indications, and document a 3E088 and 3E089 tube stabilization and plugging list. This is to include tube degradation and preventive plugging. *Ref. SO23-617-1-M1519*	Owner: D. Calhoun CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation entry only.*
Extent of Condition Unit 3: Tube-to-Tube Wear in 3E088 and 3E089, and potential wear.	**CAPR1:** SONGS to implement 3E088 and 3E089 tube stabilization and plugging list (as identified in CA2, above), and document completion	Owner: A. Matheny CR: 201836127 Due Date: 06/30/12
Extent of Condition Unit 2: Tube-to-Tube Wear in 2E088 and 2E089, and potential wear.	**CA3:** SONGS to establish criteria, review tube wear indications, and document a 2E088 and 2E089 tube stabilization and plugging list. This is to include tube degradation and preventive plugging. *Ref. SO23-617-1-M1519*	Owner: D. Calhoun CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation entry only.*
Extent of Condition Unit 2: Tube-to-Tube Wear in 2E088 and 2E089, and potential wear.	**CAPR2:** SONGS to implement 2E088 and 2E089 tube stabilization and plugging list (as identified in CA3, above).	Owner: A. Matheny Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012*

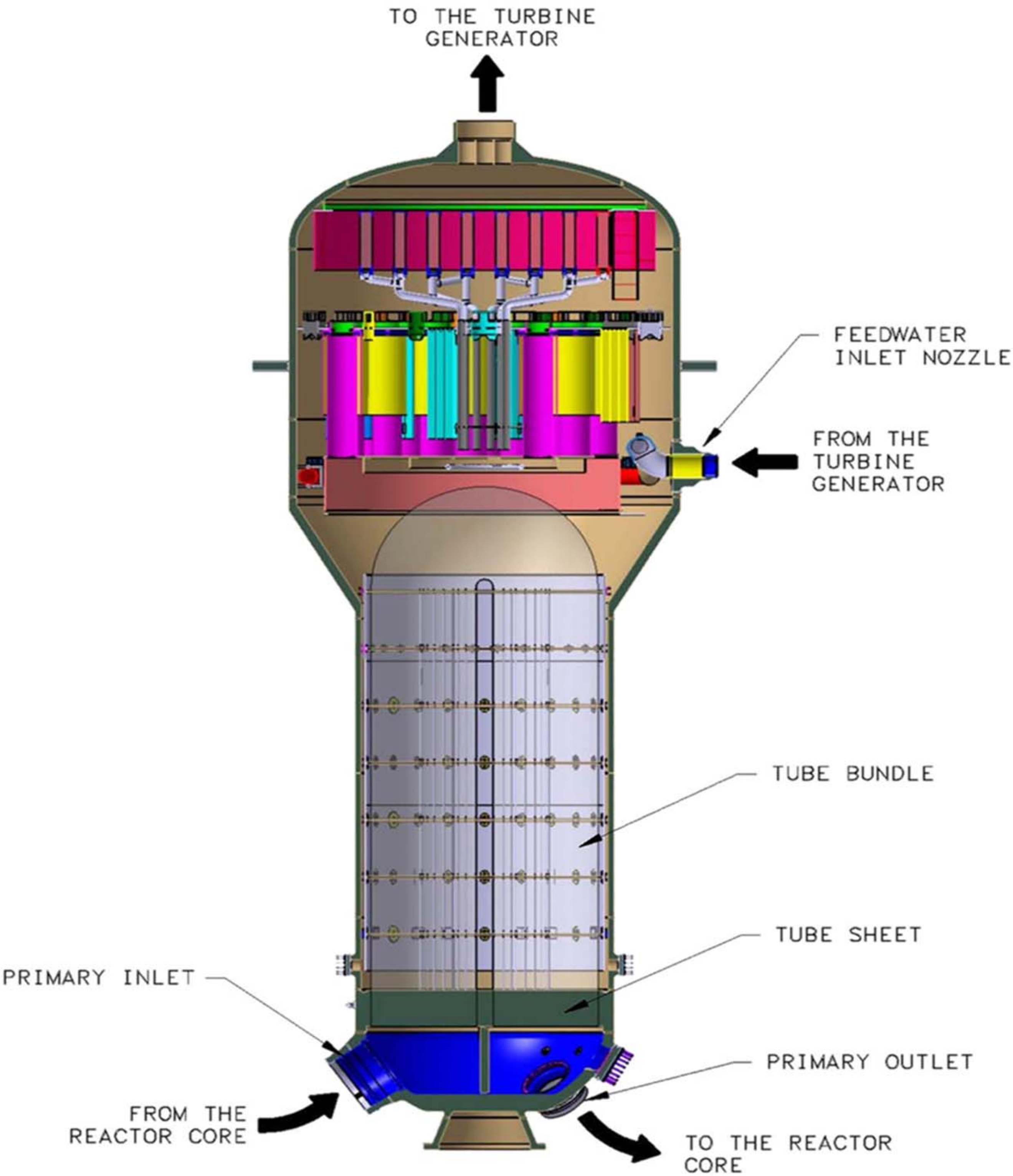
		is for documentation only.
Mechanistic Causes Unit 3: Combination of thermal hydraulic and support conditions existed that allowed FEI to occur.	**CAPR3:** Establish and document the schedule and scope for the initial Unit 3 mid-cycle outage and SG inspections.	Owner: J. Brabec CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation only.*
	CAPR4: Develop and use T/H Models and FIV Models capable of predicting SG velocities and void fractions within tolerances to determine operational limits to avoid FEI. Note: The Models need to be verified and validated against test or operational data, or verified and validated against alternative code before restart of either Unit.	Owner: J. Brabec CR: 201836127 Due Date: 6/22/12
	CAPR5: Identify and implement Unit 3 plant operation limits, based on the models developed in CAPR 4, to minimize the probability of FEI to drive tube-to-tube wear between start-up and the initial mid-cycle outage. This is a one-time CAPR. Future operational limit changes will be addressed through the Operational Assessment section of the SG Program.	Owner: E. Torres CR: 201836127 Due Date: 05/31/12
	CAPR6: Implement the initial Unit 3 mid-cycle SG inspections. The results are summarized in the Condition Monitoring Report. Document that report.	Owner: A. Matheny CR: 201836127 Due Date: 11/30/12
	CA4: Intermediate and Long Term: Establish and implement long term fixes to improve tube contact forces or compensate for current contact force.	Owner: B. Olech CR: 201836127 Due Date: 10/29/12 (for Intermediate repairs) Due Date: 05/31/15 (for Long Term repairs)
	CA5: Review and validate acceptability of the FIT III model, including review by Expert Panel. Document the results of the Expert Panel review including any deficiencies for use in the planned revision of this RCE under CA6. Note: The review of	Owner: A. Bates CR: 201836127 Due Date: 06/15/12

	the FIT III model must occur before restart of either Unit.	
Extent of Cause Unit 2: Unanticipated combination of thermal hydraulic and support conditions existed that allowed FEI to occur.	**CAPR7:** Establish and document the schedule and scope for the initial Unit 2 mid-cycle outage and SG inspections. *This CAPR serves as both a corrective action and an effectiveness review. By establishing the schedule for the mid-cycle inspection, the amount of time for wear to occur is reduced to a point prior to which sufficient tube degradation could occur to cause a tube leak or structural criterion failure. By performing the inspection, the effectiveness of the other actions will be determined.*	Owner: J. Brabec CR: 201836127 Due Date: **Complete** *Note: This corrective action is complete. This assignment is being entered into ActionWay to ensure that adequate documentation (objective evidence) of performance is provided. The due date of June 18, 2012 is for documentation only.*
Extent of Cause Unit 2: Unanticipated combination of thermal hydraulic and support conditions existed that allowed FEI to occur.	**CAPR8:** Identify and implement Unit 2 plant operation limits, based on the models developed in CAPR 4, to minimize the probability of FEI to drive tube-to-tube wear between start-up and the initial mid-cycle outage. This is a one-time CAPR. Future operational limit changes will be addressed through the Operational Assessment section of the SG Program.	Owner: E. Torres CR: 201836127 Due Date: 05/31/12
Extent of Cause Unit 2: Unanticipated combination of thermal hydraulic and support conditions existed that allowed FEI to occur.	**CAPR9:** Implement the initial Unit 2 mid-cycle SG inspections. The results are summarized in the Condition Monitoring Report. Document that report.	Owner: A. Matheny CR: 201836127 Due Date: 12/28/12
Industry Operating Experience: SG Tube-to-Tube (Free Span Wear)	**OA1:** Issue an Outgoing OE to inform the industry of the potential for Tube-to-Tube wear in light of findings of this SONGS RCE.	Owner: W. Lippitt CR: 201836127 Due Date: 06/25/12
Effectiveness Review Unit 3: Implement Unit 3 mid-cycle outage	**EFR01 (addresses U3):** Review the results of the initial Unit 3 mid-cycle outage and SG inspections, as documented in the Condition Monitoring Report and determine effectiveness of corrective actions. • No evidence of additional FEI Tube-to-Tube wear. • No evidence of additional TSP/Retainer Bar wear caused by FEI.	Owner: A. Bates CR: 201836127 Due Date: 11/30/12
Effectiveness Review Unit 2: Implement Unit 2 mid-cycle outage	**EFR02 (addresses U2):** Review the results of the initial Unit 2 mid-cycle outage and SG inspections, as documented in the Condition Monitoring Report and determine effectiveness of corrective actions. • No evidence of additional FEI Tube-to-Tube wear. • No evidence of additional TSP/Retainer Bar wear caused by FEI.	Owner: A. Bates CR: 201836127 Due Date: 12/28/12
Change Management Plan	**OA2:** Management sponsor for RCE to submit an	Owner: Gary Kline

	approved Change Management Plan in accordance with SO123-XV-50.7 within 15 days after CARB approval of the RCE.	CR: 201836127 Due Date: 05/22/12
RCE Revision	**CA6:** Revise this RCE, including performance of safety culture assessment, after review of MHI Unit 3 Cause Analysis (based on MHI technical and programmatic, causes and conclusions). Present MHI Report to CARB for their information, and SONGS RCE revision for review and approval.	Owner: J. Osborne CR: 201836127 Due Date: 07/27/12
Defense in Depth: Improve primary-to-secondary leak monitoring.	**OA3:** Revise AOI SO23-13-14 (Reactor Coolant Leak) steps to require formal Management evaluation using the ODM process for any valid indication of SG tube leakage and to direct preparation for an orderly plant shutdown. Reference NN 201969741	Owner: David Ford CR: 201836127 Due Date: 6/1/12
Defense in Depth: Evaluate Operator response to the Unit 3 SG Tube Leak shutdown and implement changes or improvements.	**OA4:** Using the lessons learned from the Unit 3 forced outage (due to a steam generator tube leak), develop and implement actions to enhance Operations performance. This includes improving procedures and Operations training. These actions are being tracked under NN #201839732, in which Dennis Brill is the responsible owner of the NN.	Owner: D. Brill NN: 201839732 Due Date: 6/7/2012 *Note: D. Brill is the documented "owner" of NN #201839732.*
Defense in Depth: Analyze Operator response to Steam Line Break with concurrent SGTR on same steam generator.	**OA5:** NTD to run Operator response to steam line break with concurrent SGTR scenario with licensed Operators and document crew performance, define and track to completion additional actions.	Owner: David Ford CR: 201836127 Due Date: 6/1/12
Defense in Depth: Improve primary to secondary leak detection.	**OA6:** Develop a portable, temporary, N-16 monitor for use on Units 2 & 3 until a modification can be developed and installed.	Owner: R. Ewing CR: 201836127 Due Date: 6/29/12
	OA7: Develop an NECP for a permanent N-16 monitor for Unit 2.	Owner: R. Ewing CR: 201836127 Due Date: 5/31/13
	OA8: Develop an NECP for a permanent N-16 monitor for Unit 3.	Owner: R. Ewing CR: 201836127 Due Date: 5/31/13
	OA9: Install N-16 monitor on Unit 2.	Owner: S. Noonan CR: 201836127 Due Date: 5/31/13
	OA10: Install N-16 monitor on Unit 3.	Owner: S. Noonan CR: 201836127 Due Date: 5/31/13
Defense in Depth: Improve primary to secondary leak detection.	**OA11:** Use Argon injection to enhance leak detection. Develop and issue NECPs to increase the ability to detect primary-to-secondary leaks by introducing Ar-40 into the RCS.	Owner: K. Johnson CR: 201836127 Due Date: 05/18/12
MHI Report Validation Conduct a SONGS	**OA12:** Following delivery of MHI organizational/programmatic RCE, conduct an acceptance review to verify analysis is of sufficient	Owner: J. Osborne CR: 201836127 Due Date: 08/27/12

acceptance review of the MHI organizational/programmatic RCE	depth and can be used to perform a safety culture assessment. Verify the report contains an analysis of the design change between the old SGs and the new SGs that led to the FEI.	This review requires the involvement of both Engineering and Performance Improvement.

Attachment 01: Steam Generator Design/Operations/Physical Features



Attachment 01: Steam Generator Design/Operations/Physical Features (Continued)

Parameter	**OSG**	**RSG**
General		
Thermal rating, MWt	1729	1729
Number of Tubes	9350	9727
Heat Transfer Area, ft^2	105,000	116,100
UA, Btu/hr ^{o} F	1.5E8	1.49E8
Tubes Outside Diameter, in.	0.750	0.750
Tube Wall Thickness, in.	0.048	0.0429
Tube Pitch, in.	1.0 triangular	1.0 triangular
Tube Plugging Margin, %	8	8
Primary Side		
Design Pressure, psia	2500	2500
Design Temperature, °F	650	650
Operating Pressure, psia	2250	2250
Operating Temperature (T_{hot}), °F	611.2	598.0
Operating Temperature (T_{cold}), °F	553.0	541.3
Reactor Coolant Flow (at cold leg temperature), gpm	198,000	209,880
Reactor Coolant Volume, ft^3	1895	2003
Secondary Side		
Design Pressure, psia	1100	1100
Design Temperature, °F	560	560
Operating Pressure (@100% power), psia	900	833
Operating Temperature (@100% power), °F	532	523
Steam Flow, lb/hr	7,414,000	7,588,000
Steam Moisture Content, %	<0.20	<0.10
Feedwater Temperature, °F	445	442
Blowdown Flow, lb/hr	151,000	154,860
Dimensions		
Top of the Tube Bundle, in.	381.0	388.2
Overall Height (including support skirt), in.	786	785.6
Upper Shell OD, in.	264.125	264.125
Lower Shell OD, in.	172.375	174.65
Dry Weight, lbm	1,242,366	1,286,200
Flooded Weight, lbm	1,971,840	2,041,300
Operating Weight, lbm	1,505,437	1,548,700

Attachment 02: Unit 3 Plant Operations

An analysis of Unit 3 operational history with the Replacement Steam Generations (RSG) indicates operations were conducted within the requirements of approved, applicable operating instructions. Parameters directly related to RSG performance were reviewed from the startup of Unit 3 on February 16, 2011 to its shutdown on January 31, 2012. Historical data for the parameters representing RSG performance was taken from R*Time (Real Time Viewer) and included the following:

- CV9005 – Secondary Calometic Power
- L1113Avg – Channel 1 Steam Generator No. 1 Level (E089) – Narrow Range
- L1123Avg – Channel 1 Steam Generator No. 2 Level (E088) – Narrow Range
- TCold – Average Reactor Coolant System Cold Leg Temperature
- P1013Avg – Steam Generator #1 Average Pressure
- P1023Avg – Steam Generator #2 Average Pressure
- R7818GC – Condenser Air Ejector Low Range Radiation Monitor

Acceptable RSG performance was based on operations within the procedurally approved operating bands during at power, transient and shutdown conditions. These conditions are identified in the following operating instructions:

Document Number	SONGS Title	SONGS Rev Number
SO23-12-1	Standard Post Trip Actions	26
SO23-12-4	Steam Generator Tube Rupture	23
SO23-13-14	Reactor Coolant System Leak	17
SO23-13-28	Rapid Power Reduction (RPR)	6
SO23-14-1	Standard Post Trip Actions – Bases And Deviations Justification	9
SO23-14-4	Steam Generator Tube Rupture – Bases And Deviations Justification	8
SO23-5-1.3	Plant Startup From Cold Shutdown To Hot Standby	42
SO23-5-1.3.1	Plant Startup From Hot Standby To Minimum Load	33
SO23-5-1.4	Plant Shutdown To Hot Standby	22
SO23-5-1.5	Plant Shutdown From Hot Standby To Cold Shutdown	38
SO23-5-1.7	Power Operations	52
SO23-9-6	Feedwater Control System Operation	30

The review of Unit 3 power history also included several instances of operations at a reduced power condition:

Year 2011

- May 22 to May 27 – reduced power to approximately 95% to support Heater Drain Pump 3P058 removal from service

- August 7 to August 15 – reduced power to approximately 65% to support repair to Main Feedwater Pump Turbine K006
- September 8 – Reactor trip on both Units due to system disturbance
- September 11 – Unit 3 return to full power
- December 4 to December 6 – reduced power to approximately 90% due to dropped Control Element Assembly.

Year 2012

- January 8 to January 10 – reduced power to 85% to remove Circulating Water Pump P115 due to salt water leak in the Southwest Condenser Waterbox. January 31 – unit removed from service due to Steam Generator E088 tube leak.

Analysis of data indicates during normal full power conditions, reduced power conditions, and reactor trips; operators controlled Steam Generator E088 and E089 levels and pressures in accordance with procedural requirements. The review also indicates the Feedwater and Steam Generator Digital Control Systems functioned as designed to maintain levels within prescribed operating bands.

Attachment 03: Kepner Tregoe Analysis

SONGS Unit 3 Steam Generator Kepner-Tregoe (KT) Problem Analysis

The KT process is a rational process for finding the cause of a deviation.
The KT Problem Analysis Technique is divided into the following activities:

- State the Problem
- Specify the Problem
- Develop Possible Causes from knowledge, experience or from distinctions and changes
- Evaluate possible causes by comparing them to the symptoms
- Determine the most probable cause
- Verify the most probable

KT Team

- San Onofre Steam Generator Engineering Team
  - Rick Coe
  - Bob Olech
  - Mike Short (Consultant/Technical Reviewer)
  - Gary Johnson (SONGS Qualified KT Facilitator)
- MPR Associates
- RJ Anderson and Assoc. (KT Facilitator)

On January 31, 2012 SONGS Unit 3 developed a tube leak in Steam Generator (SG) E088. The Unit was removed from service to perform inspections to assist in determining the cause of the tube leak. The ensuing inspections determined that there was unusual tube to tube wear within Unit SG E088 and SG E089. This KT problem analysis was performed to specifically address the tube to tube wear in SONGS Units 2 and 3 SGs.

KT Problem Analysis

Problem Statement

Unexpected tube to tube wear in Unit 3 Steam Generators 3E088, 3E089 and Unit 2 Steam Generator 2E089. In Unit 3 SG 088, eight tubes did not meet structural integrity performance criteria and three tubes did not meet accident induced leakage performance criteria. (Reference 1)

Problem Specifications

This portion of the process organizes the symptom facts into WHAT, WHERE, WHEN and the EXTENT of the problem. The initial phase of the process uses an IS-IS NOT approach to set the limits and provide comparative facts.

	Is	Is-Not
What	Tube-tube wear has been detected in SONGS Unit 3, SG E088, E089 and Unit 2 SG E089 Tube to tube wear has been detected in replacement Once Through SG's (OTSG) at TMI Unit 1 (Reference 2) and ANO (Reference 3).	Tube to tube wear has not been detected in U2 SG E088 Tube to tube wear has not been detected in replacement U-tube SG's prior to this event. Another US plant replacement steam generator has not detected tube to tube wear (Built by same OEM)
What object?	SG U-tubes (free span curved sections)	SG tubes straight section
What deviation?	Tube to tube wear resulted in SG performance criteria not meeting industry and technical specification criteria	Sudden failures or out of plane wear leading to performance criteria not being met

We then look at *What is different or has changed about an IS compared to an IS NOT?*

	Distinctions	Changes
What is different or has changed when comparing SONGS Replacement SGs to Another US plant's Replacement SG	1) SONGS 1180 MWe Another US plant 478 WWe 2) SONGS # tubes 9727 Another US plant # tubes 5200 3) SONGs Heat Transfer Area 116,089 ft^2 Another US plant 48,980 ft 4) SONGS Secondary Pressure- 838 psia Another US plant- 852 psia 5) SONGs Maximum void fraction- 0.95 Another US plant- 0.93 6) Gaps between tubes to AVB much tighter, smaller gaps than Another US plant; and mfg. tolerances stricter 7) Nominal Gap between tube and AVB" Songs - 0.002" Another US plant 0.0031"	

What is different or has changed when comparing SONGS Replacement SG to other replacement U-tube SGs of comparable size	Westinghouse Design Replacement SG for ANO-2 SONGS operates at 13.6% higher thermal power <u>SONGS</u> <u>ANO-2</u> Pri Pwr 3458 MWt 3044 MWt Different size tube diameter (d) and pitch (P) Tube Index= P/d <u>SONGS</u> <u>ANO-2</u> Tube Index 1.33-1.433 1.518-1.672 SONGs has less flow area: - Smaller wrapper area and less TSP flow area - Tighter U-bend Area AVB Configuration is different Reference 4 (Design Comparison SONGs vs ANO-2 RSG's)	

What is different or has changed when comparing the SG U-tubes section to the SG straight tube section	1) U Tube area (free span) has less support in the in-plane direction 2) AVB's and Retainer Bars support U tube section vs. TSP's's supporting straight section; 3) Cross flow in U tube area and axial flow in straight except the first span(lower span between the tube sheet and first TSP) 4) U-bend area of the bundle was assembled using a process called indexing. Ratio of the distance between the tubes and the tube diameter (Pitch/diameter, P/d). U2 and U3 SG P/d ratio is smaller than comparable sized nuclear units (ANO-2 . (Ref. 4) 5) U bend area has higher velocities, higher quality steam and less dampening, (high void fraction). (Ref. 7) 6) High wear areas were located in the area where peak void fraction occurs (Ref 8)	

We then use knowledge and experience as well as distinctions and changes to develop possible causes

Possible causes from the **What** specification:

Thermal Hydraulic conditions at 100% power may not have been accurately predicted.
U3 SG manufacturing process used more accurate and tighter tolerances which improved alignment such that tubes have less contact with AVB's.

We now use the WHERE specification to develop possible causes.

	Is	Is-Not
Where	SONGS Unit 3, SG 088, 089 and SONGs U2 SG 089	SONGS U2 E088
Where geographically?	SG U-tubes (free span curved sections). See above	SG tubes straight section
Where on the object?	Tubes with wear clustered closely in the tube sheet bundle: U3 SG E088: Row 90 - 122; Columns 74 - 90 U3SG089: Row 84 - 109 Columns 76 – 93 U2 SG E089 Row 111 and 113 Col 81	Randomly throughout the bundle Row 1-89 and 91 and higher; Columns 1-73 and 91 and higher Rows 1 -83; 110 and higher; Column 1-75; 94 and higher Rows 1-110, 112 or 114 and higher Not any other columns

We then look at changes and distinctions for the WHERE specifications *about an IS compared to an IS NOT?*

	Distinctions	Changes
What is different or has changed when comparing SONGS U3 Replacement SGs to SONGS U2 Replacement SGs	1) Divider plate weld cracked during pressure testing and required weld repair on U3 SG 88 & 89 2) U3 SGs had a total of 800 rotations for manufacturing and divider plate repairs, U2 SGs had 500 rotations during manufacturing 3) U3 primary side hot leg inlet flow orifice 1” smaller than the U2 Hot leg orifice 4) U3 channel head 1.1” thinner than U2 5) Retainer bars were tied to the tubes before helium leak test and tube expansion on U3. Retainer bars were tied after helium leak testing and tube expansion on U2 6) MHI and ASME Hydro tests on the secondary side of the U3 SGs were performed 6 times for SG 3E089 and 4 times for 3E088. They were performed only two (2) times for U2 SGs E088 and E089. (Ref. 7 App. 5) 7) During shipping the U3 SGs were in a different orientation then the U2 SGs. U2 and U3 SG were both shipped in the horizontal position with the tube bundle at a 45°	U3 tube sheet to channel head welding and post weld heat treat (PWHT) was performed twice on U3 due to the divider plate weld repair During design, flow orifice was added to maintain original design flow conditions

	angle, however the U3 SGs were shipped with the Hot Leg upwards vs U2 Hot Leg downwards (Ref. 7, App. 4) 8) The U3 SGs were also shipped without the nitrogen cover pressure on the secondary side (prevents moisture/oxygen intrusion)	
What is different or has changed in the tubes with wear clustered closely in the tube sheet bundle: U3 SG E088: Row 90 - 122; Columns 74 - 90 U3SG089: Row 84 - 109 Columns 76 – 93 U2 SG E088 Row 111 and 113 Column 81	1) These tubes also show high levels of TSP wear (Ref 1, Sec. 6.2) 2) The tube that leaked in-service and the tubes that failed the insitu pressure test were in U3 SG E088 3) Tube to tube wear is concentrated in the same regions in both U3 SG tube bundles and the two tubes identified in U2. 4) The affected U3 tubes have reduced wear at AVBs 6 and 7. 5) This region of the bundle is consistent with the location of high void fraction (determined by MHI analysis). (Ref. 7 Sec. 4.3.2)	

Possible causes from the **Where** specification:

U3 SG cracked divider plate during pressure testing, leading to changes in gaps in the U- bend structure
Additional U3 SG rotations when repairing the divider plate weld resulted in changes in AVB to tube gaps
Primary side thermal hydraulic conditions result in flow induced vibration
Tube bundle sag during shipping led to changes in gaps resulting in less tube support
TSP distortion resulted in gap changes within the tube bundle

We now use the WHEN (Time) specification to develop possible causes.

	Is	Is-Not
When	ECT indicates wear has been occurring since initial operation of the U3 Replacement SGs (Feb 2011)	No wear observed during pre-operational ECT
When since	N/A	N/A
When in the life cycle	On 1/31/2012 U3 SG E088 was shut down for a primary to secondary tube leak. Subsequent inspections in Feb. 2012, identified the leaking tube as well as evidence of tube to tube wear.. This was 11 months, approximately 338 EFPD from initial installation	Wear in U2 SGs that lead to an in-service primary to secondary leak

We then look at changes and distinctions for the WHEN specifications *about an IS compared to an IS NOT?*

	Distinctions	Changes
What is different or has changed when comparing in-service wear to pre-operational wear	High number of tube to tube wear indications vs industry experience and mfg. Design expectations	In-service pressure, temperatures and flows are at their maximum during 100% power operations
What is different or has changed when comparing the life cycle of U3 SGs to U2 SGs	ECT testing shows tube to tube wear in U3 steam generators greater in numbers and in depth then U2 SGs. Ref. ECT results U2 ran for approximately 620 EFPD with no primary to secondary leaks.	

Possible causes from the **WHEN** specification:

Distortion of the tube bundle (flowering) during operation results in some gaps between AVBs and tubes

We now use the EXTENT specification to develop possible causes

	Is	Is-Not
How many objects?	One tube U3 SG 088 R106 C78 leaked in-service Tubes with tube to tune wear: U3 SG E088 and E89- 326 tubes with tube to tube wear SG3E088 had: 3 tubes fail MSLB pressure insitu pressure test 5 tubes fail 3 times differential pressure test (Ref. 6 SONGS U3 Tube-Tube Wear Orientation Summary, AREVA Inc, date 3/1/12) U2 SG 089- 2 tubes with tube to tube wear	No leaks during operation of U2 Multiple leaks in operation
What is the size?	N/A	N/A
How many deviations?	1 leaking tube out of 19,545 tubes in U3	Multiple leaks in operation
What is the trend?	Tube wear in U3 faster than can be tolerated	Normal wear rate

No new possible causes were developed from the extent specification.

Determination of the most probable cause used a Support/Refute Methodology as outlined in the SONGS Cause Evaluation Manual. This is an expansion of the KT processes to incorporate further rigorous engineering analysis for determining the most probable cause.

KT analysis indicates the most likely cause of tube to tube wear is a result of reduced tube support in conjunction with thermal hydraulic conditions leading to Fluid Elastic Instability (FEI)

See Causal Analysis Supporting/Refuting evidence matrix for further evaluation of the causes identified below from the KT process.

1	Thermal Hydraulic conditons at 100% power may not have been accurately predicted	Likely
2	Distortion of the tube bundle (flowering) during operation results in some gaps between AVBs and tubes	Likely
3	U3 SG manufacturing process used more accurate and tighter tolerances which improved alignment such that tubes have less contact with AVB's. (Ineffective tube to AVB gap size control during tube bundle assembly)	Likely
4	U3 SG cracked divider plate during pressure testing, leading to changes in gaps in the U- bend structure	Not Likely
5	Additional U3 SG rotations when repairing the divider plate weld resulted in changes in AVB to tube gaps	Not Likely
6	Primary side thermal hydraulic conditions result in flow induced vibration	Not Likely
7	Tube bundle sag during shipping led to changes in gaps resulting in less tube support	Not Likely
8	TSP distortion resulted in gap changes within the tube bundle	Not Likely

The causes below were eliminated during the KT process

Possible Causes	Reason
SG out of plumb	SONGS completed the review of the SG installation records and concluded that all installation parameters, including plumbness, were within the specified tolerances.
Flow orifice in the primary side hot leg	Eliminated based on the fact that the orifices are present in the U2 and U3 SGs, and that their bore is only slightly smaller than the pipe ID and that the orifice is profiled such as to prevent disruption of flow (formation of eddies).
Large number of SG rotations during fabrication (Original 500 rotations for U2 and U3)	The tubes calculated to lose contact with the AVBs due to rotation were primarily in the peripheral tubes, with little change calculated in the region that the significant free span wear was observed. (Ref. 7, App. 5).
Tube sheet displacement due to divider plate weld failure “during operation”	SONGS performed UT examination of the divider plate weld which confirmed the integrity of the weld in both U3 SGs.
Out-of-tolerance tube straight leg length	MHI reviewed tube fabrication QC/QA records and concluded that the relevant tube dimensions were within design tolerances for both Unit and U3 SGs.
Uneven insertion of selected tubes	MHI reviewed tube installation QC/QA records and concluded that the dimensional parameters during tube bundle assembly were within fabrication tolerances for both U2 and U3 SGs.
SG snubber (s) locked	SONGS confirmed that the U3 SG snubbers were tested before U3C16 run and visual inspection per NMO 800853211 during the SG outage determined the snubbers to be operating properly

Tube U-bend sagging during tube bundle assembly	Tube bundle assembly and tube bundle sagging by itself cannot produce AVB distortion (plastic deformation) which could result in tube-to-tube wear. Ref SO23-617-01R3
Thermal transients during divider plate weld repair (flame cutting and PWHT)	The flame cutting, welding, and PWHT involved in the repair would only affect tubes in the periphery of the tube bundle. Additionally, temperature profiles for the tubesheet and tubes during PWHT were determined analytically, monitored and evaluated; concluding that the temperatures were not sufficient to produce plastic deformation of any sort. Consequently, multiple heat treatments would not produce temperatures required for plastic deformation.
AVB structure too flexible as designed	MHI evaluation of AVB structure stiffness and its propensity to excite the tubes in itself does not allow for the tube bundle to "flower."
Departure from the OSG design in terms of tube U-bend configuration and U-bend support configuration	Changing design from the original SG to the Replacement SG, is not causal factor in itself for tube to tube wear.
Departure from the OSG design in terms of replacing the stay cylinder with the divider plate and separator configuration	Changing design from the original SG to the Replacement SG, is not causal in itself for tube to tube wear.
Departure from the OSG design in terms of tube straight leg support onfiguration	Changing design from the original SG to the Replacement SG, is not causal in itself for tube to tube wear.

References:

1) SONGS Unit 3 2012 Steam Generator Condition Monitoring Assessment, AREVA, DOC. # 51-9180143-000
2) Three Mile Island, Unit 1 Summary of Tube – Tube Wear Identified during TIR19 (Fall 2011)
3) NRC Presentation Arkansas Nuclear One Steam Generators Tube-Tube Wear, Entergy/AREVA January 2012
4) Design Comparison SONGS vs. ANO-2 RSGs Lower Shell Assembly Tube Bundle Region
5) Another US plant- EPRI SGDD 03-20-2012
6) SONGS U3 Tube-Tube Wear Orientation Summary, AREVA NP Inc. 3/1/2012
7) Tube Wear of SONGS U3 RSG, Technical Evaluation Report , MHI
   3/30/12*
   SO23-617-M1520 Rev. 1
8) MHI's Thermal & Hydraulic Analysis using FIT-III SO23-617-1-C-683

*Still under review

Causal Analysis Supporting/Refuting Evidence Matrix

Cause	Supporting	Refuting	Analysis/ Conclusions	Cause (Y/N)
Thermal Hydraulic conditons at 100% power may not have been accurately predicted	1) Tube to tube wear in 3 of 4 replacement SGs (U2 ME089, U3 ME088 and ME089) 2) Tube wear is the same in terms of kind and very similar in terms of magnitude in both U3 SGs 3) Tube wear is concentrated in the same region of the tube bundle in both U2 and U3 SGs 4) Location of the free-span wear region within the tube bundle is consistent with the location of the high void fraction region in Unit 2 and U3 SGs 5) Tube wear at AVBs is similar in the U2 and U3 SGs.	1) Tube wear is generally more severe in the U3 SGs than in the U2 SGs. Only 2 tubes in U2 SG E089 show tube to tube wear. Tube to tube wear in U2 E088 was not detected. 2) Tube wear at AVBs extending past the AVB width is seen only in the U3 SGs.	See Attachment 07	Likely Contributing Cause
Distortion of the tube bundle (flowering) during operation results in some gaps between AVBs and tubes	1) Tube wear is the same in terms of kind and very similar in terms of magnitude in both U3 SGs 2) Tube wear is concentrated in the same region of the tube bundle in both U2 and U3 SGs 3) Tube wear at AVBs is similar in the U2 and U3 SG 4) Gap between AVBs and tubes in the center columns is around 0.06 mm due to dynamic pressure (Ref 7 Appendix 8 Steam Generator Tube Flowering Analysis SONGS Units 2 and 3) 5) The area where gaps are generated is correlated with the area where free span wear has occurred (Ref 7 Appendix 8 Steam Generator Tube Flowering Analysis SONGS Units 2 and 3)	1) Tube wear is generally more severe in the U3 SGs than in the U2 SGs. Only 2 tubes in U2 SG E089 show tube to tube wear. Tube to tube wear in U2 E088 was not detected. 2) Secondary side operating parameters were similar in the U3 and U2 SGs, and well within their design limits 3) Tube wear is concentrated in the same region of the tube bundle in both U2 and U3 SGs	See Attachment 07	Likely Contributing Cause

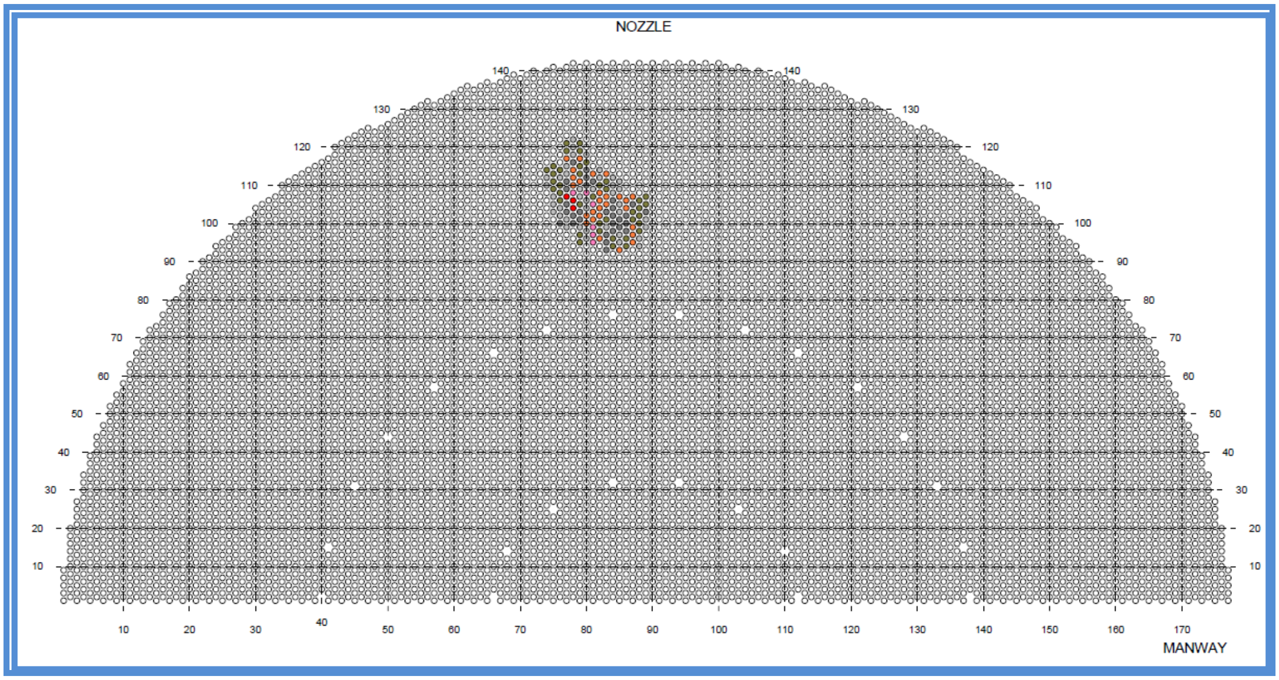
Cause	Supporting	Refuting	Analysis/ Conclusions	Cause (Y/N)
U3 SG manufacturing process used more accurate and tighter tolerances which improved alignment such that tubes have less contact with AVB's. (Ineffective tube to AVB gap size control during tube bundle assembly)	1) Contact forces between tubes and AVBs in U3 are smaller when compared to U2 (Ref 7 Appendix 9 Simulation of Manufacturing Dispersion for SONGs Units 2 and 3) 2) Difference in contact force between Unit 2 and 3 will be even greater if the "flowering" effect due to hydraulic dynamic pressure is taken into consideration. (Ref 7 Appendix 9 Simulation of Manufacturing Dispersion for SONGs Units 2 and 3) 3) SGs in U3 are more likely to have inactive AVB support points during the operating condition which makes them more susceptible to in plane vibration. (Ref 7 Appendix 9 Simulation of Manufacturing Dispersion for SONGs Units 2 and 3) 4) Tube wear is generally more severe in U3 then U2 5) Tube wear is the same in terms of kind and very similar in terms of magnitude in both U3 SGs	1) Tube wear at the AVBs is similar in the U3 and U2 SGs 2) Tube wear is concentrated in the same region of the tube bundle in both U2 and U3 SGs 3) Tube wear at AVBs extending past the AVB width is seen only in the U3 SGs.	See Attachment 08	Likely Contributing Cause
U3 SG cracked divider plate during pressure testing, leading to changes in gaps in the U- bend structure	1) Tube wear is the same in terms of kind and very similar in terms of magnitude in both U3 SGs. 2) Tube wear is generally more severe in the U3 SGs than in the U2 SGs. Only 2 tubes in U2 SG E089 show tube to tube wear. Tube to tube wear in U2 E088 was not detected. 3) Both U3 SGs experienced failure of the divider plate weld during the shop hydro test and were subsequently repaired. 4) The significant free span wear was found only in the Unit 3 RSGs (however, 2 instances of minor free span wear were discovered in one of the Unit 2 RSGs). Both of the Unit 2 RSGs successfully passed hydrostatic testing without divider plate weld failures. Prior to installation, the Unit 2 RSGs underwent NDE (VT, PT and UT); no indications were found of weld failures similar to those found in the Unit 3 RSGs.	1) Tube to tube wear in 3 of 4 replacement SGs (U2 ME089, U3 ME088 and ME089). 2) The maximum displacement for the tubesheet (and thus the TSPs and tubes, excluding the U bend region) was calculated to be 0.003 inch in the X direction, 0.067 inch in the Y direction, and 0.189 inch in the Z direction. MHI concluded that all tubes were displaced in approximately the same direction and by equal distance; therefore the tube-to-tube spacing was not altered by the weld failure and subsequent postulated tubesheet/TSP/tube displacement. Horizontal displacements (X and Y directions) were calculated to be negligible in the U bend region of the tubes, with displacement in the Z direction equal to the elastic displacement of the tubesheet, TSPs, and straight section of the tubes, at 0.189 inch. Based on negligible change in the tube-to-tube gap, MHI concluded that the calculated displacement attributed to the divider plate weld failure was not sufficient to cause plastic tube deformation, and thus was not related to the free span wear. 3) The change in tube-to-AVB gap due to hydro testing was calculated to be 10-6mm. MHI concluded that this difference was of no consequence to tube wear, given the elastic nature of the displacement.	See Attachment 09	Not Likely

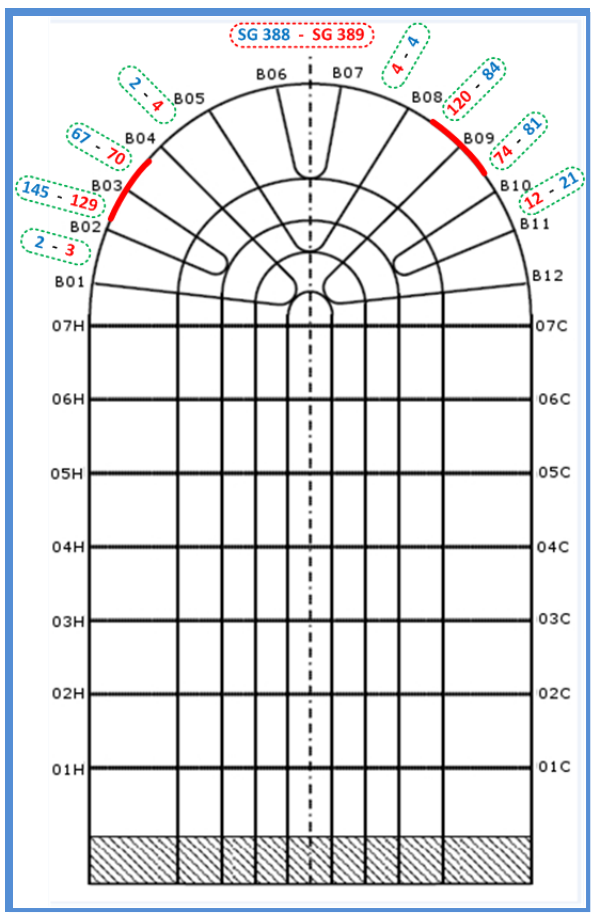
Cause	Supporting	Refuting	Analysis/ Conclusions	Cause (Y/N)
Additional U3 SG rotations when repairing the divider plate weld resulted in changes in AVB to tube gaps	1) Tube wear is the same in terms of kind and very similar in terms of magnitude in both U3 SGs. 2) Tube wear is generally more severe in the U3 SGs than in the U2 SGs. Only 2 tubes in U2 SG E089 show tube to tube wear. Tube to tube wear in U2 E088 was not detected. 3) Both U3 SGs experienced failure of the divider plate weld during the shop hydro test and were subsequently repaired 4) The additional approximately 300 tube bundle rotations associated with the divider plate repair may have increased tube-to-AVB gap size in peripheral tubes in cold conditions, which would result in a decrease of the contact force during operation. The increase in tube-to-AVB gaps in the tube bundle perimeter region could be redistributed to the center region due to flowering (hydrodynamic pressure during operation increasing the tube-to-AVB gaps). It is possible, with enough reduction of contact force between tubes-to-AVBs, that the AVB supports could become inactive, lowering the natural frequencies of the tubes, and bringing the critical velocity closer to the secondary side gap velocities, resulting in the observed wear of the tubes	1) Difference of the tube to AVB gaps for SONGs Units 2 and 3 SGs due to the number of SG rotations is very small (Ref 7 Appendix 5 SG Tube Bundle Rotation and Hydrortatic Test Analysis for SONGs Units 2 and 3) 2) Tube wear is concentrated in the same region of the tube bundle in both Units 2 and 3 SGs. 3) Tube to tube wear in 3 of 4 replacement SGs (U2 ME089, U3 ME088 and ME089) 4) The tubes calculated to lose contact with the AVBs due to rotation were primarily in the peripheral tubes, with little change calculated in the region that the significant free span wear was observed. 5) Each tube bundle rotation was calculated to alter the center column tube-to-AVB gap by 2.0x10-6mm. With the additional 300 tube bundle rotations associated with the divider plate weld repair, this would equate to 6.0x10-4mm, which MHI concluded was of no consequence to tube wear. 6) MHI calculated the outer region tubes would see a tube-to-AVB gap change of 10-2mm. MHI postulated that the gap change would be shifted to the center column region, due to the flowering phenomenon. MHI concluded the gap per column would be small enough that the difference in tube-to-AVB gap between Unit 2 and Unit 3 RSGs due to the additional tube bundle rotations would be very small.		Not Likely

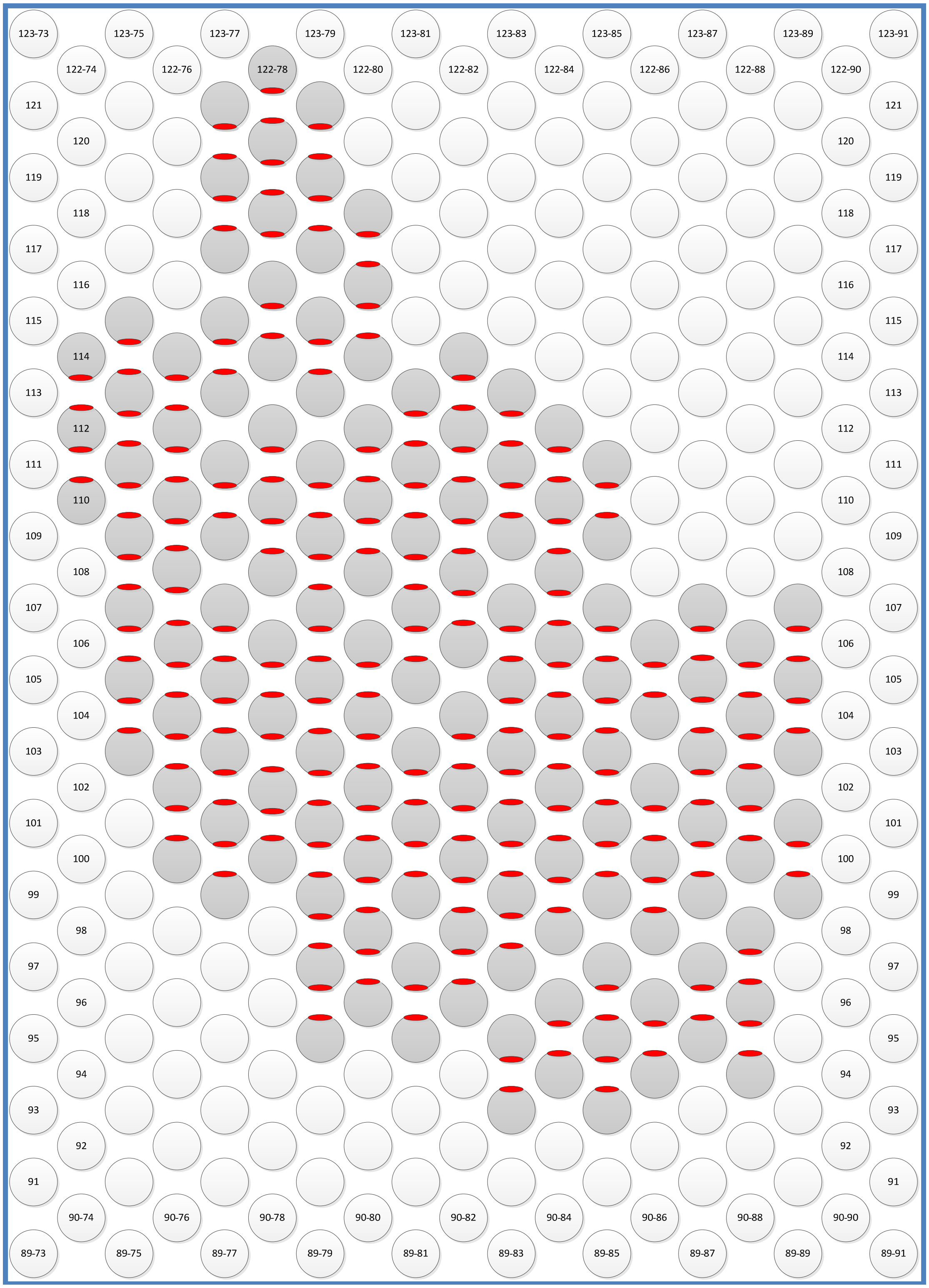
Cause	Supporting	Refuting	Analysis/ Conclusions	Cause (Y/N)
Primary side thermal hydraulic conditions result in flow induced vibration	The natural frequency of the tubes with wear in the U-bend region is close to that of the blade passing frequency (~100 Hz) when three or more supports are assumed inactive. An inactive U-bend support is thought to occur when the tube-to-AVB clearance in the hot condition is not as designed.	The Reactor Coolant System (RCS) piping configuration limits propagation of pressure pulsations induced by the reactor coolant pump impeller vanes. Therefore, primary side fluid excitation cannot be by itself the cause of the observed tube wear. Analysis by MHI for the worn tubes shows that if three or more consecutive Anti-Vibration Bars (AVBs) are inactive, the natural frequency of the unsupported span of tube could approach 100 Hz, or equal to the RCP vane passing frequency. However, U-tube shape distortion due to vibration in the natural mode is not consistent with wear locations observed in the tubes that have experienced tube-to-tube wear. The natural mode vibration at 100 Hz with three inactive AVBs causes a high displacement only in the tube sections where the supports are inactive. Conversely, tube-to-AVB wear in the tubes that have experienced tube-to-tube wear is typically observed at many of the AVB locations, rather than just the three inactive supports. This observed wear is indicative of more than three inactive AVBs supports, which would lower the natural frequency of the tube to below the RCP vane passing frequency. The observed tube-to-tube wear is highly localized. Other tubes in the same row, with very similar dimensions and surrounding secondary flow conditions did not exhibit any tube-to-tube wear. Since all tubes are equally subjected to the RCP vane passing frequency, these adjacent tubes would have also seen tube-to-tube wear if that mechanism was independently capable of creating that type of wear A report from Continuum Dynamics Inc. (Bilanin)~~,~~ concludes that the external forcing by the RCP vane passing frequency does not constitute a sole cause of the wear because: (i) free span wear is highly localized and (ii) the acoustic wavelengths are sufficiently long that adjacent tubes would respond closely in phase, thus rendering tube-to-tube contact unlikely. However, it does not rule out the possibility that other factors, such as manufacturing tolerances, tube bundle deformation, etc., could envelope the localized damage observed in the plant as a result of long wavelength forcing functions on the tubes (ie, RCPs).	See Attachment 12	Not Likely

Cause	Supporting	Refuting	Analysis/ Conclusions	Cause (Y/N)
Tube bundle sag during shipping led to changes in gaps resulting in less tube support	1. U3 RSGs were shipped in a horizontal position with the tube bundle 45° off the tube U-bend gravity neutral position. The U3 RSGs were shipped with the hot leg facing up, while U2 RSGs were shipped with the hot leg facing down. 2. The tube bundles in U3 RSGs were not supported by any temporary means during shipping to prevent sagging. 3. During U3 RSGs transportation, there were 188 accelerometer recordings indicating accelerations over 0.5G; 137 recordings on RSG 3E088 and 51 on RSG 3E089 (Ref 4). In contrast, there were 99 recordings over 0.5G during U2 RSGs transportation 4. Monitoring and maintaining of the dew point, oxygen concentration or nitrogen blanket pressure was not done on the U3 RSG during transportation.	1. Both U2 and U3 RSGs were shipped with the tube bundle 45° off the U-bend gravity neutral position~~.~~ 2. The ECT results show the wear pattern on the RSG 2E089, RSG 3E088 and RSG 3E089 is biased towards Column 1, while on the RSG 2E088 there is no bias. 3. Neither U2 nor U3 RSGs were shipped with a temporary tube bundle support fixture. 4. Similar to U3 RSGs, the accelerometers on U2 RSGs also experienced accelerations greater than 0.5G; there were 99 recordings of accelerations over 0.5G during the transportation of the U2 RSGs. 5. Although there were 137 recordings on RSG 3E088 and only 51 on RSG 3E089, the tube-to- tube wear on both RSGs is almost the same.	See Attachment 11	Not Likely
TSP distortion resulted in gap changes within the tube bundle	1) ECT data reviewed for a selected set of FSW tubes indicates that there are tapered wear indications at the 6th and 7th TSPs.5 The wear marks align with the orientation of the broached-hole lands for an upward "bent" TSP, and indicate some TSP distortion for both Unit 3 Steam Generators 2) Each TSP has 48 stay rods of .71" diameter distributed in two circular patterns concentric to the TSPs.1 These internal reinforcing structures are suspected to cause TSP distortion due to thermal expansion, as the 1.38" thick TSPs are relatively thin and flexible compared to the 27.95" thick tube sheet to which the stay rods are attached	1) Tube-tube map indicates that the wear is only occurring on one half of the hot and cold legs of the tube bundle. The mechanical nature of the hydrostatic TSP distortion caused by thermal expansion of the stay rods during operation deflection should cause a uniform deflection of would have deflected the tube sheet and TSPs symmetrically from the center lines of the Steam Generator. However, there is no matching wear patterns on the other half of the tube bundle for both Unit 3 Steam Generators	See Attachment 10	Not Likely

Attachment 04: SG Tube Wear Indications (FIGURES)



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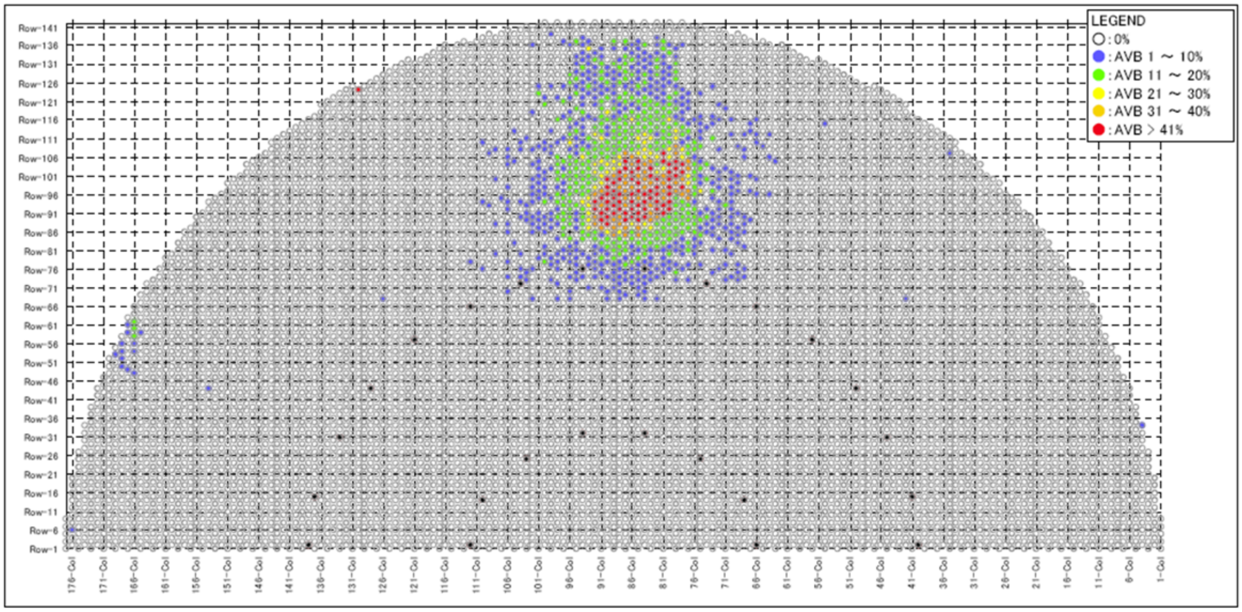
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: AVB 1 ~ 10%
: AVB 11 ~ 20%
: AVB 21 ~ 30%
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Attachment 05: Analysis of Tube Wear Indications

Analysis Title: FEI and Not Other Mechanisms
Purpose: Based on physical observations, investigate the possibility that mechanisms other than Fluid Elastic Instability (FEI) caused the tube wear/failure in SONGS Steam Generators.
Description of why this is a potential cause: It is postulated that a mechanism other than Fluid Elastic Instability in causing the wear and failure of tubes in the SONGS Steam Generators. It is also postulated that the tube bundle support system could be causing some vibrations and thus the tube-to-tube wear.
Facts to support as a cause/contributor (in descending importance): 1. Eddy Current Testing results and visual observations show measurable wear at many locations throughout the tube bundle at tube-to-support interfaces, at both tube support plates and Anti-Vibration Bars (AVBs). The vibrations that caused this wear could be due to a number of vibration types; random turbulence, vortex shedding, flow induced vibration, etc.
Facts to refute as a cause/contributor (in descending importance): 1. SONGS has observed tube-to-tube wear in the U-bend region of the tube bundle. This type of wear would require a large amplitude in-plane movement of the tubes. Fluid Elastic Instability can result in very large amplitude tube vibrations often limited only by impacting another tube (Ref. 1). 2. Eddy Current Testing results of SONGS Unit 3 Steam Generators show a large number of tubes which have both tube-to-tube wear and tube-to-AVB wear (Ref. 4). A tube undergoing FEI can vibrate in an orbital motion (Ref. 1, 2 & 3). The two-dimensional wear patterns reflect this orbital motion. 3. In-plane tube vibration would manifest itself as wear on the intrados and extrados surfaces of the tube. An AREVA report on SONGS' tube-to-tube wear shows, in general, that the observed wear was on the same tubes in both the hot leg and cold leg side of the Steam Generator. Any variance in these observations is likely due to the detection threshold of the eddy current probe. According to the report, all wear indications have a facing match. The wear locations were in the vicinity of AVB B03 and B09. This means the tube-to-tube wear occurred about half way between the top and sides of the U-bend (Ref. 5). It is reasonable to conclude that the area of greatest tube motion would result in the greatest tube wear. Based on the observed wear, the in-plane tube motion was predominantly side-to-side rather than up-and-down.

4. Post-failure Eddy Current Testing of the tubes that failed the in-situ pressure testing revealed that the tube leaked at the location of the observed tube-to-tube wear (Ref. 6). This evidence proves that tube-to-tube wear is what ultimately led to the leaking tubes in SONGS' steam generators, and not another failure mode. FEI is the only mechanism known to be capable of causing tube-to-tube wear (Ref. 1).

Analysis of facts:

If the vibration of an AVB or other structural member was causing the wear in the SONGS' Steam Generators, there would be patterns in the observed wear indicating a particular problematic component. For example, if an AVB was vibrating, all of the tubes in the two adjoining columns that are touching that AVB would show tube-to-AVB wear. This type of pattern was not observed in SONGS' Steam Generators (Ref. 7). Further, if an overarching structural problem was causing the tube wear, we would have observed widespread wear throughout the tube bundle. The observed wear is highly localized and is not in close proximity to any single piece of the tube support structure.

FEI is the only mechanism known to be capable of causing tube-to-tube wear over the as-designed tube clearances in the tube bundle. However, if during the tube-insertion process the tube-to-tube clearances turned out to be much lower, random low-amplitude vibrations due to turbulence could cause tube-to-tube wear. The smallest tube-to-tube clearances would be at the top of the tube bundle, and the tube-to-tube wear would be observed at that location. Since the tube-to-tube wear is not at that location, as described above, FEI is still the only known cause of the tube-to-tube wear in the Steam Generators (Ref. 1).

Conclusions:

The physical observations of tube wear in SONGS Steam Generators overwhelmingly supports FEI being the primary cause of the wear. While there is a possibility that other mechanisms of flow induced vibration contributed to the overall vibration and wear of the tube bundle, these mechanisms are not contributors to tube-to-tube wear, which was the ultimate cause of the tube failures (leaks).

Recommendations (if significant contributor or cause):

None.

References:

1. (Attached) ASME Section III, Appendix N, Paragraphs 1320, 1330 & 1340.

2. (Attached) FEI Industry Paper – “Fluid-Elastic Instability of Rotate Square Tube Array in an Air-Water Two-Phase Crossflow”, Chung and Chu, 2005
3. (Attached) FEI Industry Paper – “Fluid Elastic Instability Causing Tube Damage in Main Steam Condensers of Nuclear Power Plants”, Conzen, 2009
4. * SO23-617-1-M1520, Mitsubishi Heavy Industries, “Tube wear of Unit-3 RSG – Technical Evaluation Report”
5. (Attached) AREVA, “SONGS Unit 3 Tube-Tube Wear Orientation Summary”
6. (Attached) AREVA Eddy Current Testing results, post in-situ test.
7. ** (Attached) AREVA, “SONGS Unit 3 February 2012 Leaker Outage – Steam Generator Condition Monitoring Assessment”

*Preliminary report. SO23-617-1-M1520 has not been approved by SONGS.
**Preliminary report. AREVA document # 51-9180143-000 is still a draft. The pertinent information is unlikely to change.

Attachment 06: Analysis of Fluid Elastic Instability

Analysis Title: Fluid Elastic Instability (FEI)

Purpose:

The purpose of this analysis is to determine if the conditions in the Steam Generators allowed for tubes to reach fluid-elastic-instability (FEI).

Description:

FEI is a common phenomenon that occurs when certain conditions exist in mechanical structures that have fluid flow. FEI can have a negative effect on equipment including premature wear and equipment failure. Because of these negative effects, design engineers evaluate the parameters that allow for FEI through Thermal/Hydraulic modeling, Flow-Induced-Vibration (FIV) analysis, and additional engineering evaluations to assure their structures are not exposed to FEI. Subsequent to the tube leak in Unit 3's E088 Steam Generator, a recovery team was formed to identify the causes and provide corrective actions. This team included the Steam Generator manufacturer.

MHI, SONGS subject matter experts, and Steam Generator design industry subject matter experts. During collection of evidence many factors that allow for FEI were identified.

Facts to support as a cause/contributor (in descending importance):

1. SONGS has observed through Eddy Current testing and visual inspections, tube-to-tube, free span wear in the steam generators indicating high amplitudes of in-plane vibration indicative of FEI (Ref. 52).

2. SONGS has observed through visual inspections two-dimensional wear patterns. The two-dimensional wear patterns reflect orbital motion which is a primary wear pattern of FEI as described in industry papers (Reference 5 & 6).

3. MHI's cause evaluation attributed loss of contact force between the AVBs and tubes to the conditions that allowed for FEI. MHI (substantiated by Ref. 54 & 55) determined "…the free span wear in the U-bend region was caused (by) in-plane fluid elastic vibration due to (the)reduction of AVB-to-tube contact force.." (Reference 37, pgs. 15, 23, 25 & 43). When AVB-to-tube contact force is reduced, this could cause loss of support at certain locations of the tube during plant operation, causing the natural frequency of the tube to be lowered, a critical parameter to cause FEI.

4. Substantial industry papers and research performed identifying FEI as a prominent flow induced vibration mode in Steam Generator tube bundles subject to cross flow (reference section on pg. 576 of Ref 54 provides a compilation of several industry

studies of FEI and its impact on nuclear steam generators).

5. ASME Section III (Rules for Construction of Nuclear Power Plant Components) recommends a damping ratio of 0.5% in gas, or 1.5% in "wet" steam or liquid (Ref. 50). The fluid characteristics in the area where tube damage has occurred is above 90% void fraction, at that void fraction percent, steam is much closer to gas characteristics than "wet" steam. During original design, MHI used a damping ratio of 1.5% as per ASME code (Ref. 3). Assuming a higher damping coefficient during design would predict the steam generators were stable against FEI in scenarios when they were not.

Facts to refute as a cause/contributor (in descending importance):

1. Original MHI Evaluation of Tube Vibration (Ref. 3) determined FEI would not occur for the original design assumptions for the Steam Generators. When the stability ratio is less than one (SR<1), the system is stable against Fluid Elastic Instability. The maximum stability ratio MHI predicted was 0.24 for SG tubes, with all or all but one support active. MHI evaluated an "extreme conservative case" and had one location with a stability ratio above 1, but considered this negligible (Ref. 3).

2. Preliminary results from a recently revised MHI thermal-hydraulic and flow–induced-vibration model show increased velocities and void fraction but still show tubes stable against FEI when all supports are active (final results will be incorporated in a final MHI document).

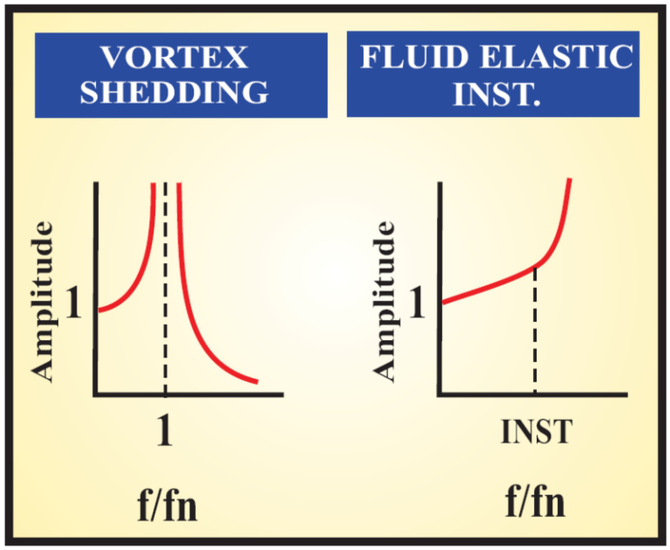
Analysis of facts:

During review of the original design documentation and review of industry research, Fluid Elastic Instability is a design consideration for steam generator tubes for nuclear power plants. The primary equation to evaluate if steam generator tubes are susceptible to Fluid Elastic Instability is;

$$V_c/f_n D = C(m_t/\rho D^2)^a(2\pi\xi_n)^b$$

Where:
V_c = critical velocity
f_n = natural frequency of *n*th vibration mode
D = tube diameter
$\frac{m_t}{\rho D^2}$ = mass ratio
ξ_n = damping ratio
$C, a \ \& \ b =$ functions of the tube array geometry

Computer programs such as MHI's FIVATS, EPRI's SGFW, and Areva's GERBOISE have been developed to analyze potential Flow-Induced-Vibration mechanisms, including FEI. Unlike other Flow-Induced-Vibration mechanisms such as vortex shedding, FEI grows exponentially once the critical velocity is reached as shown in the graph below (Ref. 53);



This exponential growth can create high amplitudes of vibrations causing substantial wear and damage to steam generator tubes and associated equipment, similar to what has been seen at SONGSs. Small changes in one variable from the FEI equation can have considerable impact on a system, and its stability against FEI. Three of the primary variables being evaluated are; natural frequency of tubes, and how it changes with loss of active supports, the damping coefficient assumed for FIV modeling, and tube cross flow velocities.

Contact forces at tube supports impact the tube's natural frequency. MHI's design assumes no gap, and no pressure at the tube-to-AVB supports (Ref. 4). Actual contact conditions during operation are unknown and difficult to predict. These contact forces affect the systems response to flow induced vibration by impacting the tubes natural frequency. During original design, MHI ran two different scenarios, one with all supports active, and another scenario with one support inactive. SONGS is currently working to validate this model and to run a model with additional supports becoming inactive. The number of active or inactive supports is critical to the natural frequency and susceptibility of the tubes to FEI, so the assumptions made in MHI's original analysis may be invalid.

MHI's original design assumed 1.5% damping during FIV analysis. The ASME Code recommends 1.5% for fluid conditions, and 0.5% in gas. During investigation of the thermal/hydraulic parameters, MHI's recent review has predicted the potential for void fractions higher than the originally predicted 95%, and has determined potential void fractions in the wear areas as high as 99.5%. When void fraction is increased, the ability for the system to dampen the forces (absorb vibrations) is affected.

During the recent reviews of the thermal/hydraulic model outputs, flow velocities have been identified as changing considerably, generally trending to a higher velocity which negatively impacts system stability against FEI. Currently the full evaluation of this potential impact is underway and may provide additional potential contributors to FEI (Ref. 3).

During non-destructive examination (eddy current) and visual inspection of the Steam Generator tubes, physical characteristics of wear supported the presence of FEI. This included orbital wear identified at the tube-to-AVB interfaces, and tube-to-tube contact, indicating a high amplitude vibration mechanism. During the recent evaluations, additional research was done to identify if other potential causes could be causing FEI. In MHI's cause evaluation (Ref. 3), they performed analysis and determined that not only would loss of supports contribute to FEI by lowering the natural frequency of the tube, but by simply losing contact force between the AVBs and tubes could create a condition that allows for FEI by allowing unrestricted in-plane vibration.

Conclusion:

The industry has developed tools and these tools have shown analytically FEI is possible in SONGS steam generators. All the physical evidence seen through non-destructive examination and visual inspections supports that FEI is occurring in SONGS steam generators. Based on these results, FEI is the most probable cause.

Recommendations (if significant contributor or cause):

1. Modify plant operating parameters to prevent FEI - Short Term

2. Modify physical conditions in Steam Generator to limit loss of active supports provided by AVBs – Long Term

References:

1. Original MHI Evaluation of Tube Vibration SO23-617-1-C157 Rev. 3 (will be revised when additional calculations are complete)
4. SO23-617-1-C491 Rev. 5 RSG Design of Anti-Vibration Bar
5. SO23-617-1-C683 Rev. 3 RSG 3D Thermal & Hydraulic Analysis using FIT-III
37. MHI Tube Wear on Unit 3 RSG Root Cause Evaluation Report SO23-617-1-M1520 (pending SONGS review & approval)
50. ASME Section III, Appendix N, Paragraphs N-1331.3
52. FEI Industry Paper – "Fluid-Elastic Instability of Rotate Square Tube Array in an Air-Water Two-Phase Crossflow," Chung and Chu, 2005
53. FEI Industry Paper – "Fluid Elastic Instability Causing Tube Damage in Main Steam Condensers of Nuclear Power Plants," Conzen, 2009
54. FEI Industry Paper – "Vibration Analysis of Steam Generators and Heat Exchangers: An Overview," Pettigrew & Taylor, 2002

55. FEI Industry Paper – “Vibration of Tube Bundles in Two-Phase Freon Cross Flow,” M.J.Pettigrew and C.E.Taylor
56. Continuum Dynamics, Inc. Letter from A. Bilanin to M. Short, 4/12/12

Attachment 07: Analysis of Thermal/Hydraulic and Fluid Induced Vibration Models

During the original design of the SONGS RSGs, MHI used a number of computer codes to analyze the design to substantiate that fluid elastic instability will not occur. The analyses starts with operating parameters, such as fluid temperatures and flow rates from the primary side of the reactor system, and SG dimensions as inputs to the computer codes.

The design and operating parameters were then input to an MHI developed computer code Steam Generator Steady State Performance Code (SSPC) (Reference 1). This code is a 1-dimensional thermal-hydraulic calculation code that calculates global SG parameters, such as the tube bundle circulation ratio (a ratio of total bundle mass flow to downcomer flow), and other secondary side operating conditions.

The SSPC results and other design inputs were then used by MHI in a detailed thermal-hydraulic analysis of the RSG tube bundle using FIT-III Version 1 (Reference 2). FIT-III is a three-dimensional analytic code developed by MHI for PWR steam generator secondary side detailed thermal-hydraulic conditions. The model simulates the secondary side from the tube sheet to the exit of the moisture separators. Attachment 2 of the FIT-III code analysis includes a MHI comparison of predicted model results with experimental data. The results of the FIT-III analysis for SONGS SGs provided a prediction of the maximum steam quality and void fraction in the U-bend region as well as contours (distribution plots). An example result is shown in Figure 1, which illustrates the variation in steam quality in the vertical direction, and Figure 2 which shows a distribution of steam quality in the U-bend at the elevation plane where the maximum occurs. Steam quality is defined as the mass fraction of vapor in the two-phase mixture. The volume fraction of vapor is commonly referred to as void fraction.

The results of FIT-III were then used by MHI in a fluid-elastic analysis using the Fluid Induced Vibration Analysis of Tubular Structures (FIVATS) (Reference 3). This MHI code was developed to determine fluid elastic vibration, 3-dimensional analysis, for PWR steam generators. The code calculates individual steam generator tube vibrational conditions, such as natural frequencies and velocities critical to fluid elastic instability. The code ultimately is used to determine if the steam generator tubes are subjected to thermal hydraulic and structural conditions resulting in fluid elastic instability. This is demonstrated by the calculation of a stability ratio, consisting of the ratio of an effective fluid velocity passing a tube to the critical fluid-elastic velocity. A stability ratio that is less than 1.0 indicates the tube is in a stable region and not subject to FEI.

During the SONGS RSG design, MHI used FIVATS results to conclude that stability ratios in the RSG straight tube were less than 1.0 and therefore judged to be a region of stable fluid elastic vibration. For the U-bend region MHI assumed that one of the twelve AVB supports was not effective, or engaged, and confirmed that the stability ratios for all tubes remain less than 1.0 and concluded there is “negligible possibility of fluid elastic vibration”.

Based on Unit 2 and 3 SG tube inspection results, MHI (Reference 4) and SONGS (this RCE) now recognize that fluid elastic instability did in fact exist in regions of the SGs. The combination of thermal-hydraulic conditions and lack of effective tube AVB supports resulted in FEI. This condition was not predicted by the MHI methods for the design of the SONGS steam generators. The reason that the design codes and assumptions did not predict FEI is not yet understood and this will be the subject of future corrective actions within this RCE.

Restart Design Models

To develop the thermal-hydraulic operating conditions and to justify plant restart, MHI working with SONGS and other industry experts are developing thermal-hydraulic models for the SONGs design. The analysis results will then be used in the vibrational analysis to set the operating conditions for restart and used in the operational assessment.

The effort to confirm the cause of the vibration mechanism and wear cause(s) necessitated a reanalysis of the thermal hydraulic conditions within the SONGS RSGs. MHI decided to adopt the ATHOS code for the root cause investigation and restart analyses of the SONGS units.

MHI developed a plant-specific ATHOS model for the SONGS RSGs. The code is designed for 3-dimensional, steady-state and transient analysis of PWR steam generators developed for use by EPRI (Reference 5). MHI developed the application of ATHOS to the SONGS project by selecting and defining the input parameters pertinent to SONGS RSGs. There are many input files that define the model; geometry, tube information, and source of T/H boundary/input conditions.

ATHOS computes steady-state and time-dependent behavior of thermal-hydraulic parameters in steam generators. The code calculates numerous key parameters important in evaluating the SONGS RSG performance. These principally include:

- The prediction of T/H parameters under different operational conditions,
- Determination of flow behavior to identify problems areas such as high velocity areas (for vibration and sludge accumulation),
- Output along tube length for input to tube vibration analysis,
- Steam quality and void fraction,
- Velocity component of steam and water phases.

The code has many other features that can be used to evaluate RSG design alternatives. It is an industry developed computer code that has been adopted by the industry for design of SGs. MHI began using the ATHOS model to better understand the conditions that caused the Unit 3 RSG tube wear. The ATHOS model was used extensively in the development of the “Tube Wear of Unit-3 RSG Technical Evaluation Report” (Reference 4). The study calculated dynamic pressure and velocity from the 3-dimentional thermal and hydraulic analysis provided by ATHOS. These loadings were used in the vibration and wear analysis to evaluate stability ratios and to examine gap changes between adjacent tubes in-plane and between AVBs and tubes. The MHI developed ATHOS model was independently reviewed by AREVA and comments were provided to MHI to refine the model.

To validate the ATHOS model, SONGS management recognizes the critical importance of the accuracy of the performance analysis being done by MHI using ATHOS thermal-hydraulic evaluations. SONGS commissioned independent thermal-hydraulic evaluations of the SONGS RSGs to be performed by both Westinghouse and AREVA. Westinghouse maintains its own version of ATHOS. AREVA uses a French code called CAFCA. The intention is to run two independent models to compare the results to the latest MHI ATHOS model.

Westinghouse, as a designer of replacement steam generators and new steam generators for the AP-1000 reactor accordingly, has extensive experience with thermal-hydraulic modeling. The Westinghouse version of ATHOS is largely the same as that provided by EPRI with enhancements integrated in the pre-processing (geometry) and post-processing (visualization) modules. The Westinghouse version of ATHOS uses the same core solver as the MHI version of ATHOS. The Westinghouse independent modeling results are not yet complete. Preliminary results from the model indicate very good agreement with the refined MHI model.

AREVA has a significant capability to design replacement steam generators in the US and Europe. AREVA has been tasked to recommend which tubes should be plugged and stabilized in Unit 2 and 3 to address the FEI concerns. To perform the FEI task AREVA is:

1. Developing a T/H model using CAFCA,
2. Performing a FIV analysis,
3. Benchmarking FIV results from St. Lucie tube wear results.

The CAFCA code principally follows the same methodology as FIT-III and ATHOS, in the sense that it relies on finite cells to encompass several tubes that account for porosity (fraction of control volume filled with fluid) and flow restrictions. Empirical correlations for heat transfer and velocity slip between phases are applied and those correlations are different from the other two codes (MHI and Westinghouse ATHOS).

The development of the CAFCA T/H model provides the additional opportunity for an independent T/H model of the SONGS RSGs.

To support plant restart, the results from the latest MHI, WEC, and AREVA thermal-hydraulic models will be compared and judged based on a comprehensive list of criteria. Ultimately, the MHI ATHOS results will be used to establish operating plant conditions, such as reactor power, to be used by SONGS and AREVA in the final repairs and justification for plant power. Additionally, these results will be compared with FIT-III.

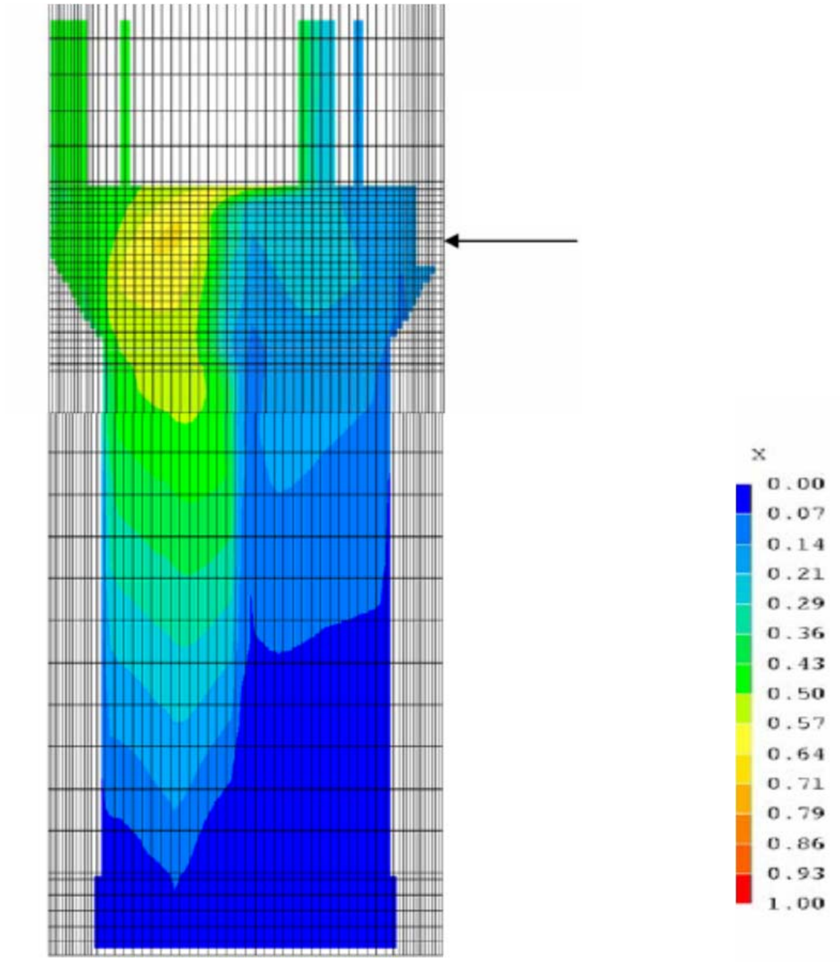


Figure 1: Contour of steam quality in the vertical cross-section for T_{hot} = 598°F (Figure 8.1-1 (a) in Reference [2])

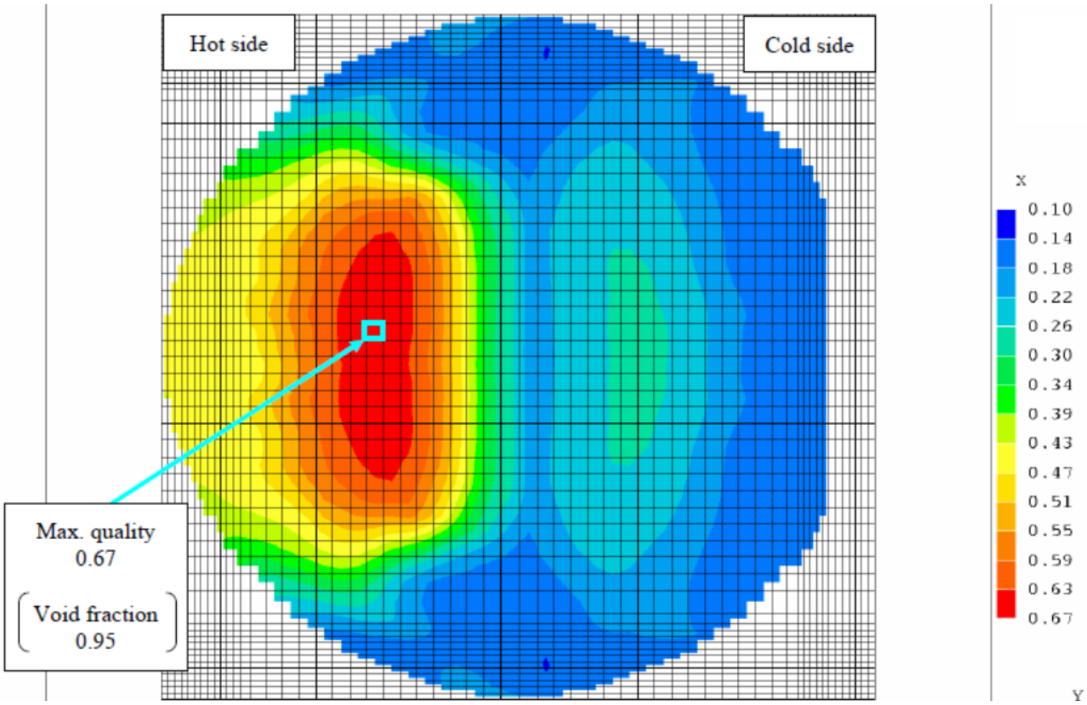


Figure 2: Contour of steam quality at the height of the maximum quality in U-bend region for T_{hot} = 598°F
(Figure 8.1-2 (a) in Reference [2])

References:

1. SO23-617-1-C1106, Thermal and Hydraulic Parametric Calculation, Latest revision.
2. SO23-617-1-C683, "Three-Dimensional Thermal and Hydraulic Analysis (FIT-III Code Analysis)", Revision 3.
3. SO-23-617-1-C157, "Evaluation of Tube Vibration, Revision 3.
4. SO23-617-1-M1520 (Reference document), "Tube Wear of Unit-3 RSG-Technical Evaluation", Revision 1.
5. The ATHOS code is a CFD Research Corporation product provided by EPRI; ATHOS/SGAP ATHOS (Analysis of Thermal Hydraulics of Steam Generators, Steam Generator Analysis Package) 3.1

Attachment 08: Analysis of Manufacturing/Fabrication

Analysis Title: Manufacturing/Fabrication

Purpose: The purpose of this analysis is to determine if tube-to-AVB gap control changes/enhancements in fabrication between Unit 2 and Unit 3 Replacement Steam Generators (RSGs), affecting the U-bend region of the tube bundle caused directly or contributed to the free span (tube-to-tube) wear on the Unit 3 Replacement Steam Generators (RSGs).

Description of why this is a potential cause:
It is postulated by MHI that dispersion (variation) of the tube and anti-vibration bar (AVB) dimensions may be required for the AVB supports to be active during plant operation (Ref. 37). Review of fabrication records identified some changes in fabrication of the Unit 2 and Unit 3 RSGs, which affected this dispersion.

Note:
The following items are also part of this topic, however these have been addressed by Item 059- "Divider Plate Weld Failure and Repair" (Ref. 40) and are not within the scope of this analysis.

- Hydrostatic tests
- Additional rotations due to divider plate weld repair.
- Additional post weld heat treat following divider plate weld repair.

Facts to support as a cause/contributor (In order of significance):

1. The standard deviations of the outer diameter of tubes (G-value) are smaller for the Unit 3 RSGs than for the Unit 2 RSGs (Ref. 37).
2. The number of adjustments of tube bending radii, to control the tube-to-tube gap which had to be performed on the Unit 3 RSGs was significantly less than adjustments on the Unit 2 RSGs. The number of tubes requiring adjustments were as follows (Ref. 37):

2A (2E089)	2B (2E088)	3A (3E089)	3B (3E088)
265 tubes	390 tubes	132 tubes	149 tubes

3. Eddy Current Testing (ECT) data suggest that tube-to-AVB gaps in the Unit 3 RSGs are slightly larger than in the Unit 2 RSGs.

Facts to refute as a cause/contributor (In order of significance):

1. The number of unacceptable tube-to-AVB gap sizes was less in the Unit 3 RSGs than the Unit 2 RSGs (Ref. 42, 43, 44, 45).
2. The number of tubes with consecutive inactive supports was greater in the Unit 2 RSGs than in the Unit 3 RSGs (Ref. 42, 43, 44, 45).
3. Majority of unacceptable gap sizes were recorded outside of the tube-to-tube wear regions in both Unit 3 RSGs (Ref. 42, 43, 44, 45).

Analysis of facts:
The review of fabrication records indicates that as the fabrication moved forward from the Unit 2 to Unit 3 RSGs, there was an overall decrease of tube and AVB dimensional dispersion (Ref. 37). In general fabrication acceptance limits were the same for all 4 RSGs, with execution improvements made as the project progressed. Issues that were not fabricated as specified were evaluated for Non-Conformances.

The results of bobbin coil ECT inspections during the outage indicate that the average signal voltage was lower on the U3 RSGs than on the U2 RSGs. This suggests that the tube-to-AVB gaps in the cold condition in the U3 RSGs were larger than in the U2 RSGs (Ref. 37).

The tube-to-AVB gaps were measured after each tube bundle was fully assembled, but due to measurement technique limitations, only the gaps between the outermost tubes and their respective AVBs were measured. This means that the as-built gap sizes inside the tube bundle were not known, as it is undetermined if and how the peripheral gap seizes correlate with the gap sizes inside the tube bundle. However, larger gaps in the cold condition generally mean greater probability of the tube supports at AVBs becoming inactive during operation (in the hot condition). The MHI inspection procedure (Ref. 24), states the following:

> “The acceptance criteria for dimensional inspection of the AVB-to-tube gaps shall be as follows; (1) For each location, the outermost gap on each side of the tube shall be no greater than 1.97 mil (0.05 mm) (2) If the gap on either side of the tube exceeds 1.97 mils (0.05 mm), the gap shall be considered unacceptable and the Tube Column No. and AVB Leg Location corresponding to that gap shall be recorded. After all gap measurements are performed, a Non-conformance Report (NCR) shall be generated."

Because not all measurements were within the acceptance criteria, Non-Conformance Reports (NCRs) were generated following the inspections (Ref. 42, 43, 44, 45). The recorded numbers of Non-Conforming gap sizes were as follows:

RSG	Number of unacceptable gaps	Number of tubes with 2 or more consecutive inactive supports
2A (2E089)	218	3
2B (2EO88)	218	6
3A (3E089)	153	1
3B (3E088)	184	0

Each of these unacceptable gaps was evaluated by use of the “Gap size evaluation Flow Chart” (Ref. 24). For those tubes with no more than one consecutive inactive support, the Non-Conforming gap was accepted "as-is". For those tubes with 2 or more consecutive inactive supports, the FIV stability calculation was performed, and if the tube was found stable, the gap was accepted "as-is". If the tube was found unstable, the gap had to be further evaluated by

engineering. All tubes with multiple supports inactive in the NCRs were shown to be stable and accepted "as-is".
The majority of the tubes identified with Non-Conforming gaps were outside of the tube-to-tube wear region of the Unit 3 RSGs.

Conclusions:
The facts identified in this analysis indicate that even though the Unit 3 tube bundle components (tubes and AVBs) might have been fabricated and assembled better, the tube-to-AVB as-built gaps might have been in fact larger in the Unit 3 RSGs as suggested by the ECT results. Based on this, it cannot be ruled out that the tube-to-AVB gaps are larger and more uniform in the Unit 3 RSGs than the Unit 2 RSGs. This might have resulted in reduction of the tube-to-AVB contact force and consequently in multiple consecutive AVB supports being inactive. Inactive tube supports might have resulted in tube-to-tube wear.

Recommended Actions (If significant contributor or cause):
1. Re-examine the justifications for the NCRs pertaining to unacceptable AVB to tube gaps. (NN-201954959)

References:

24. SO23-617-1-M821 Rev. 7 Anti-Vibration Bar Inspection Procedure (after assembling)N-SPT 201836127-026-"Item 059-Divider Plate Weld Failure and Repair" analysis.
25. SO23-617-1-M822 Rev. 8 Inspection Procedure for Tube and Anti-Vibration Bar Insertion
37. SO23-617-1-M1520 Rev 0 Tube Wear of Unit-3 RSG Root Cause Evaluation Report *Pending SONGS Review & Approval
40. N-SPT 201836127-026-"Item 059-Divider Plate Weld Failure and Repair" analysis
42. UGNR-SON2-RSG-067, Rev 7- "Non Conformance Report-Unacceptable Gaps between Tubes and AVBs." *Complete NCR available as an attachment to NN: 201836127 Task 25
43. UGNR-SON2-RSG-075, Rev 1- "Non Conformance Report-Unacceptable Gaps between Tubes and AVBs." *Complete NCR available as an attachment to NN: 201836127 Task 25
44. UGNR-SON3-RSG-024, Rev 1- "Non Conformance Report-Some Gaps between Tubes and AVBs are Larger than the Criterion." *Complete NCR available as an attachment to NN: 201836127 Task 25
45. UGNR-SON3-RSG-030, Rev 0- "Non Conformance Report-Some Gaps between Tubes and AVBs are Larger than the Criterion." *Complete NCR available as an attachment to NN: 201836127 Task 25

Attachment 09: Analysis of Divider Plate Failure During Manufacturing

Analysis Title: Divider Plate Weld Failure and Repair

Purpose: The purpose of this analysis is to determine if the divider plate weld failure and repair caused directly or contributed to the free span wear (tube-to-tube) on Unit 3 Replacement Steam Generators (RSGs).

Description of why this is a potential cause:

1. The failure of the divider plate to channel head weld during hydrostatic testing had the potential to cause tubesheet, tube support plate (TSP) and tube displacement/ deformation. This displacement/deformation could have altered tube bundle geometry (through the decrease in spacing between tubes and/or gap between the tubes and anti-vibration bars [AVBs]), leading to the free span wear observed in the Unit 3 RSGs. (Ref. 61)

2. The repair of the Steam Generator divider plate weld included: cutting the channel head from the tubesheet, re-welding the divider plate to the channel head, re-welding the channel head to the tubesheet, additional post weld heat treatment (PWHT), additional hydrostatic tests and an additional approximately 300 tube bundle rotations associated with the repairs. All of these repair activities had the potential to cause deformation/displacement of the tubesheet, TSP, or tubes, leading to the observed wear. (Ref. 33, 61)

Facts to support as a cause/contributor (In order of significance):

1. The significant free span wear was found only in the Unit 3 RSGs (however, 2 instances of minor free span wear were discovered in one of the Unit 2 RSGs). Both of the Unit 2 RSGs successfully passed hydrostatic testing without divider plate weld failures. Prior to installation, the Unit 2 RSGs underwent NDE (VT, PT and UT); no indications were found of weld failures similar to those found in the Unit 3 RSGs. (Ref. 34)

2. The additional approximately 300 tube bundle rotations associated with the divider plate repair may have increased tube-to-AVB gap size in peripheral tubes in cold conditions, which would result in a decrease of the contact force during operation. The increase in tube-to-AVB gaps in the tube bundle perimeter region could be redistributed to the center region due to flowering (hydrodynamic pressure during operation increasing the tube-to-AVB gaps). (Ref. 37) It is possible, with enough reduction of contact force between tubes-to-AVBs, that the AVB supports could become inactive, lowering the natural frequencies of the tubes, and bringing the critical velocity closer to the secondary side gap velocities, resulting in the observed wear of the tubes.

Facts to refute as a cause/contributor (In order of significance):

1. The maximum displacement for the tubesheet (and thus the TSPs and tubes, excluding the U bend region) was calculated to be 0.003 inch in the X direction, 0.067 inch in the Y direction, and 0.189 inch in the Z direction. MHI concluded that all tubes were displaced in approximately the same direction and by equal distance; therefore the tube-to-tube spacing was not altered by the weld failure and subsequent postulated tubesheet/TSP/tube displacement. Horizontal displacements (X and Y directions) were calculated to be negligible in the U bend region of the tubes, with displacement in the Z direction equal to the elastic displacement of the tubesheet, TSPs, and straight section of the tubes, at 0.189 inch. Based on negligible change in the tube-to-tube gap, MHI concluded that the calculated displacement attributed to the divider plate weld failure was not sufficient to cause plastic tube deformation, and thus was not related to the free span wear. (61)

2. The change in tube-to-AVB gap due to hydro testing was calculated to be 10^{-6}mm. MHI concluded that this difference was of no consequence to tube wear, given the elastic nature of the displacement. (Ref. 37, App. 5)

3. RSG 3A (3E088) underwent twice the number of hydro tests as RSG 3B (3E089), yet on RSG 3A less severe cracking of the divider plate weld toe was visible. Based on this, it is concluded that given the elastic nature of the displacement, the number of hydrostatic tests performed would have no bearing on the consequence of the weld failure. (Ref 37, Section 4.1.1)

4. The tubes calculated to lose contact with the AVBs due to rotation were primarily in the peripheral tubes, with little change calculated in the region that the significant free span wear was observed. (Ref. 37, App. 5)

   a. Each tube bundle rotation was calculated to alter the center column tube-to-AVB gap by 2.0×10^{-6}mm. With the additional approximately 300 tube bundle rotations associated with the divider plate weld repair, this would equate to 6.0×10^{-4}mm, which MHI concluded was of no consequence to tube wear. (Ref. 37, App. 5)

   b. MHI calculated the outer region tubes would see a tube-to-AVB gap change of 10^{-2}mm. MHI postulated that the gap change would be shifted to the center column region, due to the flowering phenomenon. MHI concluded the gap per column would be small enough that the difference in tube-to-AVB gap between Unit 2 and Unit 3 RSGs due to the additional tube bundle rotations would be very small. (Ref. 37, App. 5)

5. In general, the flame cutting, welding, and PWHT involved in the repair would only affect tubes in the periphery of the tube bundle. Additionally, temperature profiles for the tubesheet and tubes during PWHT were determined analytically, monitored and evaluated; concluding that the temperatures were not sufficient to produce plastic

deformation of any sort. Consequently, multiple heat treatments would not produce temperatures required for plastic deformation. (Ref. 28, 29, 30)

6. Channel head to tubesheet welding and PWHT processes involved in the repair were the same for all 4 RSGs, while only the Unit 3 RSGs exhibited the significant free span wear. (Ref. 34, Section 2.8) Additionally, these processes would affect only the peripheral tubes, which were not observed to have free span wear.

Analysis of facts:
The maximum calculated displacements of the tubesheet, tube support plates, and tubes due to the divider plate weld failure during hydrostatic testing were determined by MHI to not be of consequence to the observed free span wear. The PWHT and hydrostatic testing associated with the repair plan were also determined by MHI to not be of consequence to the observed free span wear.

The supporting facts presented through the referenced calculations and computer simulations indicate that while the divider plate repair plan processes would have decreased tube-to-AVB contact force or possibly slightly increased tube-to-AVB gaps, these increased gaps would occur primarily in the perimeter regions of the tube bundle. However, with the presence of flowering due to hydrodynamic pressure during operation, it is possible that the increased tube-to-AVB gaps at the tube bundle periphery could be redistributed to the center columns of the tube bundle, causing AVBs to become inactive, lowering the natural frequencies of the tubes, likely contributing to the observed free span wear.

Conclusions: The likelihood of the divider plate weld failure and associated repairs being the cause of the free span wear is judged to be of a very low level. It is possible, however, that the weld failure and activities associated with the divider plate repair contributed to the conditions causing the observed free span wear by possibly altering the geometry of the tube bundle.

Recommended Actions (If significant contributor or cause):
None

References:

28. SO23-617-1-M1260 Rev. 1 PWHT Report
29. SO23-617-1-M1309 Rev. 2 Thermal Analysis under PWHT
30. SO23-617-1-M1310 Rev. 0 RSG Evaluation of PWHT
33. SO23-617-1-M1398 Rev. 12 Divider Plate Weld Joint Repair Plan
34. SO23-617-1-M1414 Rev. 1 Divider Plate Weld Joint Separation Root Cause Evaluation Report
37. SO23-617-1-M1520 Rev 0 Tube Wear of Unit-3 RSG Root Cause Evaluation Report *Pending SONGS Review & Approval

61. "Analytical Evaluation of TS Deflection During Divider Plate Weld Failure Transient and its Impact on the TSPs and Tube Bundle" *MHI Analysis will be incorporated into Reference 37

Attachment 10: Analysis of Tube Support Plate Distortion

Analysis Title: Tube Support Plate distortion

Purpose: The purpose of this analysis is to determine if Tube Support Plate (TSP) distortion could cause directly or contribute to the tube-to-tube wear on the Unit 3 Steam Generators.

Description of why this is a potential cause:

Tube bundle deformation is postulated as a potential cause of tube-to-tube wear (hereto referred to as Free Span Wear(FSW)) in the Unit 3 Steam Generators. Such deformation could be an effect of TSP distortion resulting in a reduction in the distance between two adjacent rows of tubes in the same column, potentially leading to FSW. TSP distortion might also affect tube-support conditions, which would make TSP distortion a contributor to flow induced vibration mechanisms.

Two primary mechanisms are suspected as the initiators of TSP distortion. The first mechanism is transient distortion during operation due to differential thermal growth between the wrapper and the stay rods that would increase progressively in the direction of the 7th TSP. The second mechanism is permanent distortion caused by tube sheet deflection during the divider plate weld failure transient that occurred in both Unit 3 Steam Generators. These failures occurred during the hydrostatic pressure tests during Steam Generator manufacturing.

Facts to support as a cause/contributor (in descending importance):

1. Only Unit 3 Steam Generators have significant FSW. Both Unit 3 Steam Generators exhibit approximately the same type and magnitude of tube wear (this fact supports the weld failure effect only).
2. An informal MHI calculation indicates that the tube sheet (TS) moved by approximately 0.2” upward during the divider plate weld failure transient (Ref. 68). This suggests that such deflection could cause plastic deformation of tubes and/or TSPs. The tubes because they are connected to the TS; the TSPs because they are connected to the TS via the stay rods.
3. TSPs are much more flexible than the TS as a result of a difference in thickness (1.38” TSP thickness vs. 27.95” TS thickness) (Ref. 6-8, 13-15). This difference in thickness could cause the TSP to deform plastically even though the TS deformation might have been elastic.
4. The pattern of tube wear at the TSPs suggests that the TSPs are not perpendicular to the tubes. The AREVA ECT data reviewed for a selected set of tubes with FSW indicates that there are tapered wear indications at the 6th and 7th TSPs suggesting upward dishing of

these TSPs (Ref. 67).

5. There is a difference in thermal expansion under operating conditions between the wrapper and stay rods due to the fact that the wrapper is at a lower temperature than the stay rods, and they have different material thermal expansion coefficients. This can cause TSP deflection because the TSPs are rigidly attached to the wrapper and anchored to the TS via the stay rods (Ref. 13-15, 16, 23).
6. The effect of TSP distortion would be pitch changes throughout the U-bend portion, and the support condition between TSP #7 and the first AVB would be changed from pin-pin to fixed-pin. This could affect tube stability ratios, potentially making TSP distortion a contributor to FSW through another wear mechanism (i.e. Fluid Elastic Instability) (Ref. 3).

Facts to refute as a cause/contributor (in descending importance):

1. Both Unit 2 Steam Generators have only a single instance of free span wear affecting two tubes (this fact refutes the weld failure effect only).
2. No distortion of the TS was observed during visual inspections (Ref. 66).
3. The FSW map indicates that the concentrated tube wear region is located asymmetrically in the tube bundle (Ref. 37). The mechanical nature of TSP distortion caused by thermal expansion of the stay rods during operation should cause a uniform deflection of the tube sheet, and hence the wear pattern being located symmetrically.
4. TSP distortion from both mechanisms is postulated to primarily cause tube in-plane deflection due to the TS reinforcement preventing its deflection in the direction causing out-of-plane tube distortion provided by the divider plate. Therefore, the only plausible location for free span wear directly caused by TSP distortion would be within the first ~6" above TSP #7, which is the end of the tube straight leg for row 142 (Ref. 21,22). This would be the only location where tubes would be brought closer together, as in all other sections along the U-bend the effect of bending the tubes outward in-plane would be an increase in tube-to-tube distance. However, there were no FSW indications within this section of tubes in the reviewed ECT data (Ref. 67).

Analysis of facts:

While there are two primary mechanisms suspected to cause TSP distortion, the effects of both related to FSW would be the same. The effects would be a reduction in the tube-to-tube distance near TSP #7, and possibly a change in tube support conditions which would affect tube vibration characteristics. As no FSW was identified within the straight leg portion of the tube above TSP #7, TSP distortion could not have directly caused FSW. However, tapered wear patterns were identified from ECT data on tubes at the 6th and 7th TSP, indicating possible TSP distortion. Because TSP distortion could affect other flow induced vibration wear mechanisms,

it cannot be dismissed as a potential low-level contributor to FSW.
Conclusions: Based on the above analysis of facts, the conclusion reached is that TSP distortion cannot be the direct cause of FSW. However, it cannot be dismissed that TSP distortion could affect tube geometric and support conditions, contributing to FSW.
Recommendations (if significant contributor or cause): No applicable short/long term corrections or recommendations.

References:

3. SO23-617-1-C157 Rev 3 RSG Evaluation of Tube Vibration *Will be revised when additional calculations are complete
6. SO23-617-1-D106 Rev. 16 RSG Design Drawing Tubesheet and Extension Ring 1/3
7. SO23-617-1-D107 Rev. 6 RSG Design Drawing Tubesheet and Extension Ring 2/3
8. SO23-617-1-D108 Rev. 9 RSG Design Drawing Tubesheet and Extension Ring 3/3
13. SO23-617-1-D294 Rev. 4 Tube Support Plate Assembly Drawings
14. SO23-617-1-D295 Rev. 5 Tube Support Plate Assembly Drawings
15. SO23-617-1-D296 Rev. 6 Tube Support Plate Assembly Drawings
16. SO23-617-1-D391 Rev. 6 Design Drawing Wrapper Assembly 1/5
17. SO23-617-1-D411 Rev. 1 Tube Support Plate Fabrication Drawings 1/4
18. SO23-617-1-D412 Rev. 0 Tube Support Plate Fabrication Drawings 2/4
19. SO23-617-1-D413 Rev. 0 Tube Support Plate Fabrication Drawings 3/4
20. SO23-617-1-D414 Rev. 0 Tube Support Plate Fabrication Drawings 4/4
21. SO23-617-1-D507 Rev. 5 Anti-Vibration Bar Assembly Drawings 1/9
22. SO23-617-1-D508 Rev. 3 Anti-Vibration Bar Assembly Drawings 2/9
23. SO23-617-1-M346 Rev. 1 Material Selection Report for Tube Support Plate
37. SO23-617-1-M1520 Rev 0 Tube Wear of Unit-3 RSG Root Cause Evaluation Report *Pending SONGS Review & Approval
60. AREVA "SONGS Unit 3 Tube-Tube Wear Orientation Summary"
66. Visual Inspection DVDs of tube sheet for 3E088 and 3E089 performed 3/9/2012 (Total of 4)
67. Eddy Current Testing data for select set of tubes from 3E088 & 3E089
68. "Informal Design Calculation for Divider Plate Weld Failure Deflection" *MHI Doc will be incorporated into Reference 37

Attachment 11: Analysis of Shipping

Analysis Title: Shipping

Purpose: The purpose of this analysis is to determine if shipping conditions caused directly or contributed to the tube-to-tube wear on the Unit 3 RSGs.

Description of why this is a potential cause:
Shipping the U3 RSGs in horizontal position could have negatively impacted the geometry of the tube bundle U-bend region (plastic deformation of related tubes) where significant wear was observed.

Facts to support as a cause/contributor (in descending importance):

1. U3 RSGs were shipped in a horizontal position with the tube bundle 45° off the tube U-bend gravity neutral position. The U3 RSGs were shipped with the hot leg facing up, while U2 RSGs were shipped with the hot leg facing down (Ref. 26, 27).

2. The tube bundles in U3 RSGs were not supported by any temporary means during shipping to prevent sagging (Ref. 31).

3. During U3 RSGs transportation, there were 188 accelerometer recordings indicating accelerations over 0.5G; 137 recordings on RSG 3E088 and 51 on RSG 3E089 (Ref. 36). In contrast, there were 99 recordings over 0.5G during U2 RSGs transportation (Ref. 32).

4. Monitoring and maintaining of the dew point, oxygen concentration or nitrogen blanket pressure was not done on the U3 RSG during transportation (Ref. 35).

Facts to refute as a cause/contributor (in descending importance):

1. Both U2 and U3 RSGs were shipped with the tube bundle 45° off the U-bend gravity neutral position (Ref. 26, 27).

2. The ECT results show the wear pattern on the RSG 2E089, RSG 3E088 and RSG 3E089 is biased towards Column 1, while on the RSG 2E088 there is no bias (Ref. 69).

3. Neither U2 nor U3 RSGs were shipped with a temporary tube bundle support fixture (Ref. 31).

4. Similar to U3 RSGs, the accelerometers on U2 RSGs also experienced accelerations greater than 0.5G; there were 99 recordings of accelerations over 0.5G during the transportation of the U2 RSGs (Ref. 32).

5. Although there were 137 recordings on RSG 3E088 and only 51 on RSG 3E089, the tube-

to- tube wear on both RSGs is almost the same (Ref. 69).

Analysis of facts:

Shipping Orientation

It was necessary to orient U3 RSGs differently to facilitate handling and rigging.

The U3 RSGs were oriented differently than the U2 RSGs. Consequently, the U3 RSGs were shipped with Column 1 facing down, which suggests that wear locations should be biased towards Column 1. On the other hand, wear location on the U2 RSGs should be biased towards Column 177 because they were shipped with Column 177 down. However, the ECT results show that wear locations on both U3 RSGs as well RSG 2E089 are biased towards column 1. RSG 2E088 wear location shows no bias in either direction. This proves that the tube wear experienced in U2 and U3 RSGs tubes is not related to the shipping orientation.

Tube Bundle Support

The RSGs tube bundle design does not allow enough room for access to install or accommodate a temporary tube bundle support fixture. Additionally, it is not feasible to design and install a temporary support fixture which provides equal support to all tubes yet does not prove detrimental to the tubes during its installation and removal (Ref. 9-11). Therefore, a temporary tube bundle fixture was not used. However, MHI assessed the sagging of tube bundle by the deadweight. MHI concluded that while the bundle could sag, it would not sag to the extent that the tubes could be pinched at TSP #7 and plastically deformed (Ref. 30). Because pinching of the tubes is not possible, sagging during shipping would not lead to the observed tube deformation. Nevertheless, accelerometers were installed for shipping to indirectly monitor possible damage to the tubes due to impact loading. Lastly, if sagging had caused some deformation, the wear should be similar in all four RSGs in terms of type and magnitude.

Accelerometer Recordings

The largest accelerometer recordings were investigated by SONGS. SONGS investigated the recordings that occurred on land and MHI investigated the recordings that occurred at sea. In SONGSs investigations, the transportation log confirms that no credible events occurred at the time of the accelerometer recordings (in fact, the RSGs were stationary when the recordings were occurring). All these are attributable to lashing of the RSGs or other work activities adjacent to the accelerometers (Ref. 70).

Similar to SONGS results, MHI found that for four of the nine recordings, only 1 of the 3 installed accelerometers recorded. Two of those recordings happened while lashing of the RSGs and the other two as the RSGs were moved back from the barge to the MHI shop which was necessitated by the delay caused by crane malfunction. MHI analyzed the 5 recordings that occurred within a 90 second time frame on August 17th, 2010 between 11:25:00 and 11:26:32 AM PDT on RSG 3E089; max recorded acceleration was 1.23G (Ref. 36). RSG 3E088 did not record any accelerations above 0.5G during the same 90 seconds (Ref. 32), when both RSGs were still on the heavy lift ship.

Since the 16 investigated (4 by MHI, 12 by SONGS) recordings occurred during controlled and expected movements on and around the RSGs and under close visual surveillance, it is very unlikely that the recordings identified an event that could have caused tube deformation. Although 100% of the recordings were not investigated, single recordings from only one accelerometer can be attributed to local conditions affecting the accelerometer alone and not the RSG. Additionally, if the number of recordings was indicative of the damage to each RSG tube bundle, RSG 3E088 would show significantly more wear than the other RSGs. Because the facts do not support this trend, the recorded accelerations are not indicative of conditions that could have resulted in tube-to-tube wear.

Environmental Monitoring

The U3 RSGs were purged and filled with pure nitrogen prior to shipping for corrosion protection. Environmental monitoring, specifically monitoring the dew point, oxygen concentration or nitrogen blanket pressure, did not present a significant technical advantage in protecting RSGs from corrosion. Considering that the RSGs were sealed during shipment, it is very unlikely that the nitrogen would leak out during shipment. In the worst case, the nitrogen pressure in the RSGs would decrease to ambient pressure, but would continue to prevent marine environment from entering the RSGs and possibly causing corrosion of the internals. Consequently, lack of environmental monitoring could not be a cause of or contributor to tube-to-tube wear.

Conclusions:
Based on the analysis above, it is judged that conditions during RSG shipment could not cause or contribute to the tube-to-tube wear observed in the U3 RSGs.

Recommendations (if significant contributor or cause):
None - shipping is not a primary cause

Long Term Actions (if significant contributor or cause):
None - shipping is not a primary cause

References:

26. SO23-617-1-D1099 Rev. 4 RSG General Shipping (U2)
27. SO23-617-1-D1100 Rev. 5 RSG General Shipping (U3)
9. SO23-617-1-D116 Rev. 2 RSG Design Drawing Tube Bundle 1/3
10. SO23-617-1-D117 Rev. 2 RSG Design Drawing Tube Bundle 2/3
11. SO23-617-1-D118 Rev. 4 RSG Design Drawing Tube Bundle 3/3
26. SO23-617-1-D1099 Rev. 4 RSG General Shipping (U2)
27. SO23-617-1-D1100 Rev. 5 RSG General Shipping (U3)
30. SO23-617-1-M1310 Rev. 0 RSG Evaluation of PWHT
31. SO23-617-1-M1350 Rev. 6 RSG Shipping and Handling Procedure
32. SO23-617-1-M1385 Rev. 0 RSG Accelerometer Data Reports for U2

35. SO23-617-1-M1490 Rev. 4 RSG Shipping Plan (U3)
36. SO23-617-1-M1508 Rev. 0 RSG Accelerometer Data Reports for U3
69. Eddy Current Test Results, U-Bend Wear Types
70. "SCE Accelerometer Data Review" *Associated File to SO23-617-1-M1508 Rev 0

Attachment 12: Analysis of Primary Side Flow Induced Vibration

Blocks can be expanded as necessary for data entry.

Analysis Title: Primary Side Initiated Flow Induced Vibration
Purpose: The purpose of this analysis is to determine if tube excitation by Primary Side flow could be a potential cause of the tube-to-tube wear on the Unit 3 Steam Generators.
Description of why this is a potential cause: The Reactor Coolant Pumps (RCPs) impeller vane passing creates primary fluid pressure pulsations with the frequency equal to that of the vane passing frequency. If the frequency of these pulsations is equal to or close to the natural frequency of the tube through which the fluid flows, tube excitation will occur. This excitation will result in tube vibration, possibly resulting in tube-to-tube wear.
Facts to support as a cause/contributor (in descending importance): 1. The RCPs create a pressure pulsation which is generated by the impeller vane passing. The RCP impeller spins at a speed of 1180 RPM (3 RCPs) and 1194 RPM (1 RCP) as dictated by the prime mover (electric motor) (Ref 3). This translates to 19.7 and 19.9 revolutions per second. Because the impellers of the RCPs have 5 vanes, the vane passing frequency, and pressure pulse frequency, is 5 times the revolutions per second, or approximately 98.3 and 99.5 Hz. This is the basis for MHI analyses that refer to “95-100 cps” as the pulsation frequency. The maximum amplitude of the pressure pulse generated at vane passing frequency does not exceed 8psi (+/- 4 psi from nominal) (Ref 5). For comparison, the normal RCS operating pressure is 2250 psia. 2. Based on MHI analysis (Ref 2), the natural frequency of the tubes with wear in the U-bend region (~100Hz) is close to that of the vane passing frequency when three or more supports are assumed inactive. The vane passing frequency could provide a vibration forcing function in the range of the tube natural frequency for some configurations of unsupported tubes. Other frequencies present in the Reactor Coolant System measure more than 10dB less than the dominant vane passing frequency, making them a full order of magnitude lower in power and even less likely to cause significant vibration. The analysis also shows that the vane passing frequency will only induce in-plane vibration; since the excitation force is a pressure pulse and the surface it acts upon is circular (inside of tube), the opposing forces will cancel out and the tube will not see any vibration in the out-of-plane direction which results in tube-to-AVB wear. 3. Analysis from Continuum Dynamics, Inc. shows that the in-plane modal frequencies of tubes in the affected region of the SG are in the range of the vane passing frequency when several AVBs are inactive. The analysis also shows that required pressure pulsation amplitude to produce approximately 3 mm of deflection (the tube-to-tube gap is .25” or 6.35mm) and cause tube-to-tube wear in two adjacent tubes is within the range of the RCP vane passing frequency pressure pulse (Ref 1).

Facts to refute as a cause/contributor (in descending importance):

1. The RCPs are located in each of the RCS cold legs, downstream of the Steam Generators (Ref 5). The pumps discharge into the reactor vessel and the pressure pulsations are damped by the reactor vessel internal structures and fuel assemblies. Then these attenuated pulses are transmitted via RCS hot legs to the steam generator channel head. The Pressurizer is attached to one of the RCS hot legs, which serves as a large damper to any fluctuations in RCS pressure. At the SG channel head, the tube sheet further dissipates the energy. The pressure pulsations which propagate into the Steam Generator tubes are judged to be of a negligible magnitude, although a quantitative evaluation cannot be performed.
2. Additional analysis by MHI (Ref. 2) for the worn tubes shows that if three or more consecutive Anti-Vibration Bars (AVBs) are inactive, the natural frequency of the unsupported span of tube could approach 100 Hz, or equal to the RCP vane passing frequency. However, U-tube shape distortion due to vibration in the natural mode is not consistent with wear locations observed in the tubes that have experienced tube-to-tube wear. The natural mode vibration at 100 Hz with three inactive AVBs causes a high displacement only in the tube sections where the supports are inactive. Conversely, tube-to-AVB wear in the tubes that have experienced tube-to-tube wear is typically observed at many of the AVB locations, rather than just the three inactive supports. This observed wear is indicative of more than three inactive AVBs supports, which would lower the natural frequency of the tube to below the RCP vane passing frequency.
3. AREVA has provided a report (Ref. 6) which shows the observed tube-to-tube wear is highly localized. Other tubes in the same row, with very similar dimensions and secondary flow conditions did not exhibit any tube-to-tube wear. The eddy current testing (ECT) results show that these same tubes did have tube-to-support wear (e.g. 3E088 Row 108, Column 86). MHI analysis (Ref. 7) indicates that high tube-to-AVB wear occurs when there is low friction between the AVB, allowing in-plane vibration to occur (this fact has not been confirmed by SONGS). Since all tubes are equally subject to the 100 Hz primary side flow excitation, these adjacent tubes would also show tube-to-tube wear if this mechanism were by itself capable of creating that type of wear.

Analysis of facts:

The supporting and refuting facts do not rule out the possibility that primary side flow could contribute to the tube-to-tube wear in the steam generators. However, the fact that all tubes in all Steam Generators are subject to the same 100 Hz forcing function, and only relatively few tubes in a concentrated area experienced significant tube-to-tube wear makes this cause unlikely to be a significant contributor.

Conclusions: It is highly unlikely that tube excitation by RCP pressure pulsations was the primary cause of in-plane tube vibration and tube-to-tube wear in SONGS Units 2 and 3.
Recommendations (if significant contributor or cause): Engineering to consider long term project to measure RCS pressure pulsations near the Steam Generator to determine the real forcing function on the tubes and perform a quantitative analysis of the impact.

References:

1. (Attached) Continuum Dynamics, Inc. Letter from A. Bilanin to M. Short, 4/24/12
2. * SO23-617-1-1520, Mitsubishi Heavy Industries, "Tube wear of Unit-3 RSG – Technical Evaluation Report"
3. Full Load RPM data from SAP Functional Locations. RCP motors 3P001, 3 & 4 are made by Allis-Chalmers and have a Full Load RPM of 1180. RCP motor 3P002 is made by ABB and has a Full Load RPM of 1194.
4. SO23-922-2, "General Engineering Specification for Reactor Coolant Pumps"
5. 40111A, P&ID Reactor Coolant System
6. (Attached) AREVA, "SONGS Unit 3 Tube-Tube Wear Orientation Summary"
7. SO23-617-1-M1519, Mitsubishi Heavy Industries, "Screening Criteria for Susceptibility to In-Plane Tube Motion"

* Preliminary report. MHI Technical Evaluation L5-04GA564 (SO23-617-1-M1520), has not been approved by SONGS.

Attachment 13: Analysis of Vibration Loose Parts Monitoring System

Analysis Title: Vibration and Loose Parts Monitoring System

Purpose: The purpose of this analysis is to determine if the VLPMS could have provided indication for the tube-to-tube wear/failure on Unit 3 S/Gs.

Description of why this is a potential topic area/indicator: Per the NRC Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," "the primary purpose of the loose-part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate safety-related damage to or malfunctions of primary system components."

The VLPMS valid alarms will be addressed by this analysis.

Facts to support as a topic area/indicator (in order of significance):

1. Multiple alarms on various VLPMS channels after new S/G's installed in R3C16 (Ref. 64).

2. Primary side of the S/G's were inspected in F3C16, and no indication of loose parts were found on E088 (NMO 800842826) or E089 (NMO 800842830). This indicates that the alarms could have been initiated by secondary side noise.

3. Westinghouse Impact Analysis of Unit 3 determined impacts to be metal to metal (Ref. 63)

4. Valid Alarms were seen on Unit 3 and not on Unit 2 for the C16 operating cycle.

Facts to refute as a topic area/indicator (in order of significance):

1. Engineering requested external analysis be performed by Westinghouse to analyze data (documented in 201818719-SPT-2, closed 3/15/12), which is located in Reference 63. Per the SPT, conclusions from the analysis determined that "for the analyzed waveforms, the noise signatures of the valid alarms during temperature changes are essentially identical to the waveforms generated during steady state operation." The number of valid alarms received between February 18, 2011 and January 31, 2012 was 378 (E088 – 269, E089 - 109). Engineering filtered the number using Engineering Judgment to 30 (removed all alarms not associated with clear temperature changes due to S/G motion). Of those filtered, E088 had 9 alarms at steady state, and E089 had 21 (Ref. 65) Acoustic noise signatures for loose parts are typically characterized by multiple impacts. S/G motion alarms are typically characterized by a single large amplitude. The alarms analyzed have been single and double impact alarms, not multiple or continuous impacts which would be indicative of loose parts in the primary system.

2. The VLPMS system is designed in accordance with RG 1.133 for RCS primary side loose parts only. It is not designed to provide information for noise in the secondary side of

the S/G's. VLPMS is calibrated for the primary side, not the secondary side.

3. The VLPMS accelerometers are mounted on the Steam Generator support skirt (Ref. 62), which further isolates them from any S/G noise. Secondary side noise would be further away from the accelerometers and shielded by the tubesheet. Approximate length between the sensor location and the top of the U-tubes is 42.5’ (Ref. 62). The location of the S/G accelerometers (S/G Support Skirt) makes the SONGS VLPMS particularly ineffective in identifying secondary side noise.

4. Data recorded via the VLPMS records impacts, and does not record noise, so there is no “noise” data at the time of tube leak.

Analysis of facts:
There are sixteen piezoelectric sensors and sixteen preamplifiers located inside Containment to provide inputs to the twelve Loose Parts and Four Vibration Channels. Two Accelerometers are mounted on each steam generator. They are mounted on the support skirt. The S/G support skirt is a separate assembly welded to the bottom of the S/G. Any vibrations transmitted from the S/G to the support skirt will be attenuated and the directional focus will be severely restricted.

The two sensors from each Steam Generator are connected to channels five through eight of the vibration and loose parts channel cards. The Vibration and Loose Parts Monitoring (V&LPM) sensors detect acoustic signal generated by loose parts and flow. The signals from these sensors are amplified, filtered and the loose parts component of the filtered signal is compared with preset fixed and floating loose parts alarm setpoints to generate first tier loose part alarms when abnormal conditions are detected. Any alarm condition which exceeds the fixed or floating point setpoint (Ref. 2) is held in pending until the Loose Part Event Analysis Computer (LPEAC) tests it with 1 to 6 possible test to verify validity. Valid alarms are determined by the VLPMS computer when both accelerometers register a specified amplitude (state) within a specified time differential (state). Once the alarm is validated, it is released to the control room, provided the LPEAC is operating otherwise the alarm is sent directly to the Control Room on annunciator 50A51, "Vib & Loose Parts Monitor System Trouble."

Engineering analyses of the various alarms received determined the source of the alarms to be S/G motion. Engineering response are documented in 201790804-1 (2/23/12) and 201818719-2 (3/15/12). Westinghouse could not conclusively differentiate between the noise signatures of the valid alarms during temperature changes and steady state. Per the Westinghouse Impact Analysis of Unit 3 (Ref. 63), "the events on both S/Gs are the result of true metallic impacts and not false indications from electrical noise or fluctuations in background noise. Westinghouse found that the events that occurred prior to the forced outage were similar to the events that occur when the S/Gs shift during RCS temperature transients. However, Westinghouse cannot conclusively state that the events are from the same source without additional data for comparison and evaluation. Even with additional data, determination of the source of the

impacts could be hindered by the location of the sensors."
Conclusions: The Vibration and Loose Parts Monitoring System is intended to detect loose metallic parts in the primary system (Ref. 49). VLPMS was not designed, capable of, nor expected to indicate secondary tube contact. Data analyzed by industry experts and site personnel for the time of the event and valid alarm responses during C16 Operation could not conclusively determine that the VLPMS was indicating tube contact.
Recommended Actions (if significant contributor or cause): None.

References:

1. SD-SO23-616 Rev 2, "Seismic Monitoring, Vibration and Loose Parts (V&LP) Monitoring and Essential Instrumentation and Control Panel Systems"
2. SO23-II-1.14 Surveillance Requirement Vibration and Loose Parts Monitoring Calibration
11. SO23-617-1-D118 Rev. 4 RSG Design Drawing Tube Bundle 3/3
49. NRC RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors"
62. Modified SO23-617-1-D103 Rev. 8 RSG Design Drawing Component and Outline Drawing 1/3
63. "SONGS U3 Impact Analysis" MHI Doc ITS3206 Rev. 0 *Also Attached to NN 201818719
64. Table for Notifications written against S3.VLPM.3L194, "Notifications Dating February 1, 2011 to January 31, 2012"
65. San Onofre U3 Steam Generator Valid Alarms

Attachment 14 – Secondary Acoustic Wave Resonance

Analysis Title: Secondary Loop Flow-induced Acoustic Resonance
Purpose: The purpose of this analysis is to determine if there is an acoustic wave generated on the secondary side steam piping downstream of the Steam Generator (SG) and if it could be a potential cause of the tube-to-tube wear on the Units 2 and 3 Steam Generators.
Description of why this is a potential cause: The configuration of the steam piping downstream of the Steam Generators could be causing a flow-induced acoustic resonance to travel back into the Steam Generators and excite the tubes, causing them to vibrate.
Facts to support as a cause/contributor (in descending importance): 4. Analysis from Continuum Dynamics, Inc. concludes that frequencies associated with vortex shedding over the inlets to the main steam branch lines (leading to the safety relief valves) can overlap the in-plane modal frequencies of tubes in the affected region of the SG depending on the support conditions. Additionally, the Strouhal number calculated from the configuration of SONGS steam piping and branches is near the peak pressure response, which suggests that acoustic resonance may be responsible for vibration in the Steam Generator tubes (Ref 1). 5. Prior analysis for SONGS has shown the most plausible explanation for damaging vibrations in the main steam safety relief valves was vortex shedding at the inlets to the main steam branch lines (Ref 2). 6. Flow-induced acoustic resonance in safety relief valve stub pipes has caused damage to the steam dryers inside the reactor vessel in BWRs (Ref 3). 7. System acoustic noises which can propagate long distances are one source of tube excitation in tubes with axial flow (Ref 4).
Facts to refute as a cause/contributor (in descending importance): 4. All tubes in the Steam Generator are subjected to same excitation frequency, but only relatively few tubes in a concentrated area experienced significant tube-to-tube wear. The Steam Generator also has a significant dryer and moisture separator assembly above the tube bundle, which has not been observed to be damaged during the tube bundle inspections. 5. Flow-induced acoustic resonance is not explained as a wear mechanism in PWR Steam Generators by industry papers (Ref 5).

6. Based on the operating experience for Steam Generators accumulated to date, acoustic load has not been identified to be significant with regard to flow-induced vibration and tube wear (Ref 6).

Analysis of facts:

The supporting and refuting facts do not rule out the possibility that flow-induced acoustic resonances in the downstream secondary loop piping could contribute to the tube-to-tube wear in the Steam Generators. However, previous design and operating experience of PWR Steam Generators suggest that this contribution is small compared with the contribution of other flow-induced vibration sources.

Conclusions:

It is unlikely that tube excitation by flow-induced acoustic resonance was the primary cause of in-plane tube vibration and tube-to-tube wear in SONGS Units 2 and 3.

Recommendations (if significant contributor or cause):

Engineering to consider long term project to measure secondary loop steam line pressure pulsations between the Steam Generator and the branch lines to the safety relief valves to determine the frequency and magnitude of the acoustic wave and perform a quantitative analysis of the impact. NN 201972757 was written to track this recommendation.

References:

8. (Attached) Continuum Dynamics, Inc. Letter from A. Bilanin to M. Short (4/24/12).
9. (Attached) Bechtel Western Power Corporation, “Main Steam Safety Valves (MSSV’s) Vibration”, Meeting Minutes (01/29/87).
10. (Attached) Takahashi, Ohtsuka, et. al., *Experimental Study of Acoustic and Flow-induced Vibrations in BWR Main Steam Lines and Steam Dryers*, Proceedings of ASME Pressure Vessels and Piping Division Conference (2008).
11. (Attached) ASME Section III, Appendix N, Paragraphs 1340
12. (Attached) EPRI, “Steam Generator Vibration and Wear Prediction”
13. (Attached) MPR Associates, “Comments on the Contributions of External Acoustic Sources to the SONGS Steam Generator Flow-Induced Vibration Loads and Tube Wear Rates”, (05/03/12).

Attachment 15: Barrier Analysis

Consequences (What Happened?)	Barrier That Should Have Precluded the Event	Outcome (Was the Barrier Effective?)	Barrier Assessment (What was Wrong with the Barrier?)
Tube Leak in SG 3E088 and Unexpected Tube-to-Tube Wear Identified in Steam Generators 3E088 and 3E089			
a. The SG Design Did Not Account for Fluid Elastic Instability (FEI) in the in-plane direction	MHI Design Modeling for Thermal Hydraulics and Flow Induced Vibration	Not Effective	The design addressed Fluid Elastic Instability (FEI) based on usual method / industrial standard and ASME Code, and did not address in-plane FEI since there was no Code requirement or OE. The tube bundle behavior was not adequately identified by Thermal Hydraulics and Flow Induced Vibration models used in the up-scaled SG design. See Corrective Action Matrix for actions being taken to address this condition.
	MHI Design Review	Not Effective	The design review addressed Fluid Elastic Instability (FEI) based on usual method / industrial standard and ASME Code, and did not address in-plane FEI since there was no Code requirement or OE. FEI is a known phenomenon associated with heat exchangers and steam generators. The associated cause will be determined by the MHI programmatic cause analysis.
	Edison Document Reviews	Not Applicable	There was no document to review specifically addressing the effects of FEI.
	Manufacturing and Fabrication Procedures	Potentially Not Effective	The U3 SG manufacturing process used more accurate and tighter tolerances, which improved alignment such that the tubes have less contact with the AVBs. The associated cause

			will be determined by the MHI programmatic cause analysis.
	Edison – Nuclear Oversight Division (NOD) Audits	Not Applicable	The function to ensure that the Supplier is following 10 CFR50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants; baseline and 2 three-year follow-ups conducted. Not intended to detect proprietary design flaws.

Consequences (What Happened?)	**Barrier That Should Have Precluded the Event**	**Outcome (Was the Barrier Effective?)**	**Barrier Assessment (What was Wrong with the Barrier?)**
	Use of Industry Operational Experience	Not Applicable	There is industry (non-nuclear and nuclear) operational experience available that is related to the impact of Fluid Elastic Instability (FEI) on heat exchanger and steam generator tubes. With respect to the nuclear industry, information on FEI also exists on the NRC and INPO websites. However, this OE is generally associated with out-of-plane vibration and not the in-plane experienced at SONGS. This was not a Missed Opportunity in that OE was not readily retrievable, nor did the OE require an industry response.
b. Transportation	Accelerometers Installed on Shipping Skids	Effective	Records show that there were no evaluated drops or abnormal accelerations to cause significant movement of the tube bundle.
c. Installation at SONGS	Work Orders and Installation Procedures	Effective	There was no abnormal bump during installation to adversely affect SG internal components.
	Post-Installation Inspection	Not Applicable	No practical method to inspect. During installation there was nothing identified to indicate localized indications of FEI.
	Post-Installation Testing	Not Applicable	There was no known method to determine internal vibration prior to plant power operation.
d. Operational Events	Operating Procedures	Effective	U3 operated within design limits.

e. Foreign Object / Loose Parts	MHI Manufacturing Procedures	Effective	There were no loose parts found and no indications of tube wear that can be attributed to Foreign Material Exclusion (FME).
	SONGS SG Installation Procedures	Effective	There were no loose parts found and no indications of tube wear that can be attributed to Foreign Material Exclusion (FME).
f. Tube Material or Tube Manufacturing Defects	MHI Quality Monitoring	Effective	MHI provided in-shop inspection of the tube fabrication and review of test reports. No material defects were noted to date based on ECT testing results.
g. Indications of Vibration Prior to Component Failure	No Barrier in Place	Not Effective	The VLPMS alarms received were determined to be discrete events. No other methods in place to monitor vibration associated with SG TTW.

Consequences (What Happened?)	**Barrier That Should Have Precluded the Event**	**Outcome (Was the barrier Effective?)**	**Barrier Assessment (What was Wrong with the Barrier?)**
Wear Occurred in Tubes adjacent to Smaller Diameter Retainer Bars	*Note: The Retainer Bar length increased without a corresponding increase in thickness. Unit 2 Root Cause Evaluation (RCE) No. 201843216 addressed Unit 3 retainer bars as part of the extent of condition evaluation.*		

Attachment 16: Operating Experience

A search for related internal and external Operating Experience (OE) was performed by the RCE Team using the SONGS OE - Operating Experience (TOPIC Information Server) search database. Other search databases that were used for the OE search included: SONGS SAP/ActionWay, INPO (IERs and SOERs) website, and the NRC website. The search encompassed a review of events over the past eight (8) years using the following key words and combinations of key words, such as: "steam generator, replacement steam generator, new steam generator, tube wear, tube leak, tube-to-tube, retainer bar, retaining bar, anti-vibration bar, wear, fluid elastic instability and flow induced vibration." The events identified in the search were reviewed and the most relevant are discussed below. The information in the OE reports was used to validate and enhance this SONGS RCE 201836127.

Industry Operating Experience

The industry review did not identify OE involving retainer bar vibration and interaction with tubes causing tube wear. There were four OEs documented related to tube-to-tube wear. Two were associated with a different type of SG design (a once through SG) at TMI and ANO, one was associated with an older version SG manufactured by Combustion Engineering at Palisades, and one involving original SG (since replaced) at Palo Verde. Based on this review, there was no missed opportunity for SONGS to identify and address the potential for retainer bar vibration induced tube wear or tube-to-tube wear in our new MHI steam generators. The following industry OE was found to be relevant to tube-to-tube wear:

Document Number: OE34946
Title: Tube-to-Tube Contact Wear Identified in Steam Generators
Event Date: November 04, 2011
Plant/Facility: TMI Unit 1

Applicability to Event: This industry OE event was released in November 2011. Although a different SG design, it is similar to the event under evaluation in that tube-to-tube wear indications were found on SG tubes.

During the first refueling outage following SG replacement, ECT identified two different wear related damage mechanisms; tube-to-TSP wear and tube-to-tube contact wear. The OE notes that the tube-to-tube contact wear "is a previously unreported degradation mechanism for in service tubes in once through steam generators." Following the completion of 100% ECT, a total of 37 tubes were removed from service by plugging; the plugged tubes were stabilized utilizing full-length stabilizers. The cause of the tube-to-tube contact wear was under evaluation by the component manufacturer at the time that the OE was issued. The lesson learned for the industry was noted as, "the tube-to-tube wear indications were initially screened as absolute drift indications (ADIs) using the bobbin coil probe and were initially thought to be non-relevant, as tube-to-tube contact wear was not considered a potential degradation mechanism."

This OE is not considered to be a missed opportunity due the differences in SG design and the timing of the event (November 2011) with respect to the SONGS new SGs. SONGS SGs were designed, manufactured and installed prior to the OE, and discovery of SONGS tube-to-tube wear was the result of ECT following the January 2012 SG tube leak shutdown. Similar to TMI SONGSs is plugging and stabilizing tubes subject to tube-to-tube wear.

Document Number: Apparent Cause Evaluation (ACE) No. CR-ANO-1-2011-2609
Title: ANO Steam Generator Tube-to-Tube Wear
Event Date: December 2011
Plant/Facility: Arkansas Nuclear One (ANO) Unit 1

Applicability to Event: This industry OE event was released in December 2011. Although a different SG design, it is similar to the event under evaluation in that ANO identified tube-to-tube wear as a potential new damage mechanism.

During the previous discussion related to the tube-to-tube wear experienced at TMI Unit 1, it was noted that the tube-to-tube contact wear can be masked or mischaracterized and, therefore, not considered as a potential damage mechanism. Following the issuance of TMI Unit 1 OE34946, ANO reviewed their SG examination data and confirmed that ANO did have indications similar to that observed at TMI Unit 1. A review of 1R23 outage results found bobbin ADI indications that indicated tube-to-tube wear were mischaracterized by ANO personnel. There was no discussion in the ACE as to the cause of the tube-to-tube wear.

This OE is not considered to be a missed opportunity due the differences in SG design and the timing of the event (December 2011) with respect to the SONGS new SGs. SONGS SGs were designed, manufactured and installed prior to the OE, and discovery of SONGS tube-to-tube wear was the result of ECT following the January 2012 SG tube leak shutdown.

Document Number: Docket No. 50-255
Title: 2007 Steam Generator Tube Inspection Report
Event Date: April 07, 2008
Plant/Facility: Palisades Nuclear Plant

Applicability to Event: This industry OE event was released in April 2008. It is similar to the event under evaluation in that Palisades identified tube-to-tube wear as a damage mechanism. The replacement SGs at Palisades are Combustion Engineering (CE) Model 2530 and were installed in the fall of 1990. The Palisades 2007 Steam Generator Report was retrieved for our review from the NRC website, and additional information was gained by reviewing the subsequent response from Palisades to the NRC's request for additional information.

A review of the available information found that in their 2007 SG Report, Palisades noted that they found an indication that was associated with tube-to-tube wear. The effected tube was subsequently plugged and stabilized. Palisades indicated that the tubes did not move into the condition to allow tube-to-tube wear and that the condition had existed since the SGs. A formal root cause evaluation was not performed. In a response to the NRC, Palisades noted that the likely cause was due to "manufacturing tolerances associated with tube bending for the square bend region," and a possible cause due to a "square bend with bend angle not equal to 90 degrees."

This OE is not considered to be a missed opportunity due to the lack of readily available information related to the Palisades tube-to-tube wear. This industry OE was found by performing a review of the NRC website and was not identified through a routine search of INPO Operating Experience. In OE34946 (tube-to-tube wear at TMI), which was issued in November 2011, it was noted "No previous evidence of tube-to-tube contact wear was identified during a search of industry OE."

Document Number: Various
Title: Tube-to-Tube Wear

Event Date: 2004 to 2006
Plant/Facility: Pala Verde Nuclear Generating Station
In addition to the more recent OE related to SG tube-to-tube wear noted above, a review of the NRC website found that the Palo Verde Nuclear Generating Station (PVNGS) had previously experienced tube-to-tube wear in their original Unit 1, 2 and 3 SGs. Various documents were reviewed (Steam Generator Tube Inspection Summary Reports, NRC Safety Evaluations, and other miscellaneous NRC communications) in the timeframe of 2004 to 2006. For example, there numbers of tube-to-tube wear indications noted in the documents ranged from 4 to 11.

Additional Steam Generator Tube Related OE

In addition to the industry OE examples discussed above, the OE search also provided insights that SG tube wear and tube leaks do occur in the industry, including leaking tubes associated with replacement SGs. Examples of OE found in this category include:

OE19705 (Beaver Valley Unit 1) – August/September 2004: Replacement SG tube manufacturing issues due to the tubes not being manufactured to the specification requirements. A cause was identified as a "lack of oversight by the tube manufacturer's management and quality organizations to assure procedure adherence and quality program effectiveness during tube fabrication."

OE35359 (St. Lucie) – April 2009: There were a large number of Anti-Vibration (AVB) wear indications identified in SGs. The cause was determined to be "non-homogeneous gap distributions along U-bends, combined with side loads pushing the AVBs against the tubes are resulting in the wear indications."

OE35375 (Cook Unit 1) – October 2011: A large number of wear indications were identified in the SG fan bar region. A system "transient was propagated by operating conditions which established a resonance behavior in the tubing leading to increased vibration and tube wear." Prior to SG replacement, Cook implemented a reduced temperature and pressure (RTP) program in the RCS. "The RTP had the effect of lowering RCS temperature by approximately 20 degrees F. and reducing secondary side steam pressure from approximately 805 psig to 670 psig."

OE12223 (McGuire Unit 1) – March 2001: The calculated wear growth rate of replacement SG (at the Fan Bar and Lattice Grid) was large enough to delay transition into the Long Range Inspection Plan.

OE13700 (Oconee) – April 2002: A severed tube was found during ECT with the severed tube causing wear on adjacent tubes. The cause of the severed tube was determined to be intergranular attack (IGA).

OE20410 (Byron Unit 1) – March 2005: Eddy Current Testing found that the fan bar/collector bar did not fully engage a row of tubes. The cause was attributed to a "vessel fabrication anomaly" that occurred as a result of repair modifications.

Docket No. 50-250 (Turkey Point Unit 3) – April 2008: A "Steam Generator Tube Inspection Report" documented the steam generator inspection results at Turkey Point following Unit 3 End-of-Cycle 22. The inspections identified wear degradation at broached tube supports, Anti-Vibration Bars, and baffle plates.

Docket No. 50-313 (Arkansas Nuclear One) – July 2010: A "Steam Generator Tube Inspection Report" documented the steam generator inspection results at Arkansas Nuclear One, Unit 1 following completion of refueling outage inspection 1R22. The inspections identified wear degradation at the Tube Support Plate.

Document No. T2001-0830 (TMI Unit 1) – October 2001: A SG tube became swollen, the severed. The severed tube caused wear damage to the adjacent tubes.

Event Number 400-010601-1 (Harris Unit 1) – June 2001: Three tubes in the replacement SG were not hydraulically expanded to full-depth within the tubesheet.

Event Number 270-980330-1 (Oconee Unit 2) – March 1998: Five SG tubes were found to be rotated along their axial length due to fabrication error. This condition caused the effected tubes to be out of alignment with their respective holes in the tubesheet, resulting in two tubes remaining in service that should have been removed from service and plugged.

Site Operating Experience

A review of site OE going back approximately 8 years did not identify previous problems in SONGS original SGs with respect to retainer bar vibration induced tube wear, tube-to-tube wear, or fluid elastic instability (FEI). Thus, there was no missed opportunity for SONGS to identify and address the potential for retainer bar vibration induced tube wear and tube-to-tube wear exhibited in our new SG. It should be noted that prior to commercial operation of SONGS Unit 2 and Unit 3, SG tube damage at the point of contact with anti-vibration bars was identified in the now decommissioned SONGS Unit 1; this type of damage was also identified at Connecticut Yankee and Indian Point 1 during the same time period (discussed in INPO SER-43-81, Steam Generator Degradation at the Anti-vibration Bars; issued in March 1981). However, due to the age of the Unit 1 OE, it is not considered as a repeat event or missed opportunity.

Attachment 17: Reference Documents

1. SD-SO23-616 Rev 2, "Seismic Monitoring, Vibration and Loose Parts (V&LP) Monitoring and Essential Instrumentation and Control Panel Systems"
2. SO23-II-1.14 Surveillance Requirement Vibration and Loose Parts Monitoring Calibration
3. SO23-617-1-C157 Rev 3 RSG Evaluation of Tube Vibration *Will be revised when additional calculations are complete
4. SO23-617-1-C491 Rev. 5 RSG Design of Anti-Vibration Bar
5. SO23-617-1-C683 Rev. 3 RSG 3D Thermal & Hydraulic Analysis using FIT-III
6. SO23-617-1-D106 Rev. 16 RSG Design Drawing Tubesheet and Extension Ring 1/3
7. SO23-617-1-D107 Rev. 6 RSG Design Drawing Tubesheet and Extension Ring 2/3
8. SO23-617-1-D108 Rev. 9 RSG Design Drawing Tubesheet and Extension Ring 3/3
9. SO23-617-1-D116 Rev. 2 RSG Design Drawing Tube Bundle 1/3
10. SO23-617-1-D117 Rev. 2 RSG Design Drawing Tube Bundle 2/3
11. SO23-617-1-D118 Rev. 4 RSG Design Drawing Tube Bundle 3/3
12. SO23-617-1-C157 Rev. 3 RSG Evaluation of Tube Vibration
13. SO23-617-1-D294 Rev. 4 Tube Support Plate Assembly Drawings
14. SO23-617-1-D295 Rev. 5 Tube Support Plate Assembly Drawings
15. SO23-617-1-D296 Rev. 6 Tube Support Plate Assembly Drawings
16. SO23-617-1-D391 Rev. 6 Design Drawing Wrapper Assembly 1/5
17. SO23-617-1-D411 Rev. 1 Tube Support Plate Fabrication Drawings 1/4
18. SO23-617-1-D412 Rev. 0 Tube Support Plate Fabrication Drawings 2/4
19. SO23-617-1-D413 Rev. 0 Tube Support Plate Fabrication Drawings 3/4
20. SO23-617-1-D414 Rev. 0 Tube Support Plate Fabrication Drawings 4/4
21. SO23-617-1-D507 Rev. 5 Anti-Vibration Bar Assembly Drawings 1/9
22. SO23-617-1-D508 Rev. 3 Anti-Vibration Bar Assembly Drawings 2/9
23. SO23-617-1-M346 Rev. 1 Material Selection Report for Tube Support Plate
24. SO23-617-1-M821 Rev. 7 Anti-Vibration Bar Inspection Procedure (after assembling)
25. SO23-617-1-M822 Rev. 8 Inspection Procedure for Tube and Anti-Vibration Bar Insertion
26. SO23-617-1-D1099 Rev. 4 RSG General Shipping (U2)
27. SO23-617-1-D1100 Rev. 5 RSG General Shipping (U3)
28. SO23-617-1-M1260 Rev. 1 PWHT Report
29. SO23-617-1-M1309 Rev. 2 Thermal Analysis under PWHT
30. SO23-617-1-M1310 Rev. 0 RSG Evaluation of PWHT
31. SO23-617-1-M1350 Rev. 6 RSG Shipping and Handling Procedure
32. SO23-617-1-M1385 Rev. 0 RSG Accelerometer Data Reports for U2
33. SO23-617-1-M1398 Rev. 12 Divider Plate Weld Joint Repair Plan
34. SO23-617-1-M1414 Rev. 1 Divider Plate Weld Joint Separation Root Cause Evaluation Report
35. SO23-617-1-M1490 Rev. 4 RSG Shipping Plan (U3)
36. SO23-617-1-M1508 Rev. 0 RSG Accelerometer Data Reports for U3
37. SO23-617-1-M1520 Rev 0 Tube Wear of Unit-3 RSG Root Cause Evaluation Report *Pending SONGS Review & Approval
38. SO23-922-2 Rev. 0 General Engineering Specification for Reactor Coolant Pumps
39. 40111A Rev. 44 P&ID Reactor Coolant System
40. N-SPT 201836127-026-“Item 059-Divider Plate Weld Failure and Repair” analysis
41. Full Load RPM data from SAP Functional Locations. RCP motors 3P001, 3 & 4 are made by Allis-Chalmers and have a Full Load RPM of 1180. RCP motor 3P002 is made by ABB and has a Full Load RPM of 1194.

42. UGNR-SON2-RSG-067, Rev 7- “Non Conformance Report-Unacceptable Gaps between Tubes and AVBs.” *Complete NCR available as an attachment to NN: 201836127 Task 25
43. UGNR-SON2-RSG-075, Rev 1- “Non Conformance Report-Unacceptable Gaps between Tubes and AVBs.” *Complete NCR available as an attachment to NN: 201836127 Task 25
44. UGNR-SON3-RSG-024, Rev 1- “Non Conformance Report-Some Gaps between Tubes and AVBs are Larger than the Criterion.” *Complete NCR available as an attachment to NN: 201836127 Task 25
45. UGNR-SON3-RSG-030, Rev 0- “Non Conformance Report-Some Gaps between Tubes and AVBs are Larger than the Criterion.” *Complete NCR available as an attachment to NN: 201836127 Task 25
46. "FIT-III Code Validation Report," MHI Doc No: KAS-20050201 *Provided to NRC AIT item # 180 and available as an attachment to NN: 201836127 Task 22
47. "CLOTAIRE Benchmarking Report," MHI Doc No: WLS 09305 *Provided to NRC AIT item #190 and available as an attachment to NN: 201836127 Task 22
48. "ATHOS Thermal-Hydraulic Analysis of the SONGS U2&3 RSG" *Westinghouse Doc is pending SONGS review & approval. Available as an attachment to NN: 201836127 Task 22
49. NRC RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors"
50. ASME Section III, Appendix N, Paragraphs N-1331.3 *Attachment to NN: 201836127-23
51. ASME Section III, Appendix N, Paragraphs 1320, 1330 & 1340 *Attachment to NN: 201836127-31
52. FEI Industry Paper – “Fluid-Elastic Instability of Rotate Square Tube Array in an Air-Water Two-Phase Crossflow,” Chung and Chu, 2005 *Attachment to NN: 201836127-23 & 31
53. FEI Industry Paper – “Fluid Elastic Instability Causing Tube Damage in Main Steam Condensers of Nuclear Power Plants,” Conzen, 2009 *Attachment to NN: 201836127-23 & 31
54. FEI Industry Paper – "Vibration Analysis of Steam Generators and Heat Exchangers: An Overview," Pettigrew & Taylor, 2002 *Attachment to NN: 201836127-23
55. FEI Industry Paper – “Vibration of Tube Bundles in Two-Phase Freon Cross Flow,” M.J.Pettigrew and C.E.Taylor *Attachment to NN: 201836127-23
56. Continuum Dynamics, Inc. Letter from A. Bilanin to M. Short, 4/12/12 *Attachment to NN: 201836127-23 & 24
57. (Attached Q #3) “Possibility of Tube Wear Caused by RCP Pressure Pulsation” *MHI Doc will be incorporated into Reference 37
58. "7 Questions and Answers" *MHI Doc will be incorporated into Reference 37, currently in Attachment to NN: 201836127-24
59. “Screening Criteria for Susceptibility to In-Plane Tube Motion” - *MHI Doc will be incorporated into SO23-617-1-M1519, currently in Attachment to NN: 201836127-24
60. AREVA "SONGS Unit 3 Tube-Tube Wear Orientation Summary" *Attachment to NN: 201836127-24, 28, 31
61. "Analytical Evaluation of TS Deflection During Divider Plate Weld Failure Transient and its Impact on the TSPs and Tube Bundle" *MHI Analysis will be incorporated into Reference 37, currently in Attachment to NN: 201836127-26
62. Modified SO23-617-1-D103 Rev. 8 RSG Design Drawing Component and Outline Drawing 1/3 *Attachment to NN: 201836127-27
63. “SONGS U3 Impact Analysis” MHI Doc ITS3206 Rev. 0 *Also Attached to NN 201818719 and NN: 201836127-27
64. Table for Notifications written against S3.VLPM.3L194, "Notifications Dating February 1, 2011 to January 31, 2012" *Attachment to NN: 201836127-27
65. San Onofre U3 Steam Generator Valid Alarms *Attachment to NN: 201836127-27
66. Visual Inspection DVDs of tube sheet for 3E088 and 3E089 performed 3/9/2012 (Total of 4) *Attachment to NN: 201836127-28
67. Eddy Current Testing data for select set of tubes from 3E088 & 3E089 *Attachment to NN: 201836127-28
68. "Informal Design Calculation for Divider Plate Weld Failure Deflection" *Will be incorporated into Reference 37, currently in Attachment to NN: 201836127-28

69. Eddy Current Test Results, U-Bend Wear Types *Attachment to NN: 201836127-29
70. “SCE Accelerometer Data Review” *Associated File to SO23-617-1-M1508 Rev 0, also in Attachment to NN: 201836127-29
71. Eddy Current Testing Results, post in-situ test *Attachment to NN: 201836127-31
72. “SONGS Unit 3 February 2012 Leaker Outage – Steam Generator Condition Monitoring Assessment” *AREVA Doc # 51-9180143-000 is pending SONGS review & approval, currently in Attachment to NN: 201836127-31