UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

September 20, 2013

CAL 4-12-001
EA-13-083

Mr. Peter Dietrich
Senior Vice President and
Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

EA-13-083

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION – NRC CONFIRMATORY ACTION LETTER RESPONSE INSPECTION 05000361/2012009 AND 05000362/2012009

Dear Mr. Dietrich:

Following the June 7, 2013, announcement of Southern California Edison's decision to permanently shut down San Onofre Nuclear Generating Station, Units 2 and 3, the U.S. Nuclear Regulatory Commission (NRC) terminated our review of your Confirmatory Action Letter Response (ML12285A263) for Unit 2, dated October 3, 2012. The enclosed report documents the NRC assessment of your activities through June 7, 2013, in response to our March 27, 2012, Confirmatory Action Letter (ML12087A323). The NRC also reviewed the two remaining open unresolved items identified in Augmented Inspection Team Report 05000361/2012007 and 05000362/2012007 (ML12188A748). The two unresolved items were related to the mechanistic cause of the excessive and unexpected wear in both Units 2 and 3 steam generator tubes, which resulted in a steam generator tube leak on Unit 3 on January 31, 2012. The results of this inspection were discussed with you and other members of your staff on August 28, 2013.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures, documents, and records and interviewed personnel.

On June 12, 2013, Southern California Edison submitted a Certification of Permanent Cessation of Power Operations letter to the NRC, certifying that Units 2 and 3 have permanently ceased power operations. On June 28 and July 22, 2013, Southern California Edison certified that all fuel had been permanently removed from the Units 3 and 2 reactors, respectively (ML13183A391 and ML13204A304).

When Southern California Edison announced the permanent shutdown of both Units 2 and 3, the NRC halted its review of the operational assessments and other open issues related to the Unit 2 CAL response, and no final determination on the adequacy of that response was made. The NRC Office of Nuclear Reactor Regulation was also conducting a technical evaluation, which could not be completed because further information, including the adequacy of using squeeze film dampening for determining the effectiveness of the anti-vibration bars and subsequent vibration response, was required. Questions from NRC inspectors about the use of squeeze film dampening for the anti-vibration bar configuration at low frequency resulted in additional testing being conducted at the Atomic Energy Canada Limited facility in Chalk River, Canada, and at Mitsubishi Heavy Industries (Mitsubishi) in Kobe, Japan. Southern California Edison determined that revisions to their previously submitted operational assessments for Unit 2 were needed based on the testing results; however, those revisions were not completed before the permanent shutdown announcement was made.

The enclosed report documents two NRC-identified findings associated with the thermal-hydraulic unresolved item, one finding of very low safety significance (Green) for Unit 2 and one finding that was preliminarily determined to have a low to moderate safety significance (White) for Unit 3. The Mitsubishi FIT-III thermal-hydraulic computer model (FIT-III) output gap velocities were not appropriately modified for triangular pitch designed steam generators. There were opportunities to identify this error during the design of the replacement steam generators. Mitsubishi was the vendor selected by Southern California Edison to design and manufacture the replacement steam generators. On numerous occasions during the design process, Southern California Edison personnel questioned the results from and appropriateness of using FIT-III, but ultimately accepted the design as proposed by Mitsubishi. Mitsubishi hired consultants with expertise in designing large steam generators, but did not rigorously evaluate all concerns raised by the consultants about use of FIT-III and specific results obtained from that thermal-hydraulic model. As a result, replacement steam generators were installed at San Onofre with a significant design deficiency, resulting in rapid tube wear of a type never before seen in recirculating steam generators. The NRC assessed these findings based on the best available information using the applicable Significance Determination Process. For Unit 2, all the steam generator tubes were determined to meet the technical specification requirements for tube integrity; therefore, the design control violation for Unit 2 was determined to be of very low safety significance. For Unit 3, we conducted an independent risk analysis and determined that the risk was of low to moderate safety significance (White). The NRC's preliminary significance was based on the following conservative assumptions: an exposure time of 172 days; a steam generator tube rupture that results in core damage will always result in a large early release; degraded tubes resulted in an increased frequency of a steam generator tube rupture; and a main steam line break could have occurred during the exposure period, resulting in one or more tubes rupturing. The details of all primary assumptions associated with the preliminary significance determination are documented in Attachment 5 of the enclosed report.

As a corrective action, your staff revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance

The Unit 2 finding was determined to involve a violation of NRC requirements. The NRC is treating this violation as a noncited violation (NCV) consistent with Section 2.3.2.a of the

Enforcement Policy. The Unit 3 finding is an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The Enforcement Policy is included on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

In accordance with NRC Inspection Manual Chapter 0609, Significance Determination Process, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity: (1) to attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. The focus of the Regulatory Conference is to discuss the significance of the finding, not necessarily the root cause(s) or corrective action(s) associated with the finding. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final significance determination, in that, by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of Inspection Manual Chapter 0609.

Please contact Ryan Lantz at 817-200-1173 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public

Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Steven A. Reynolds
Acting Regional Administrator

Dockets: 50-361, 50-362
Licenses: NPF-10, NPF-15

Enclosure:
NRC Inspection Report 05000361/2012009
and 05000362/2012009

Attachments:
1. Supplemental Information

2. Independent Evaluation of San Onofre Nuclear Generating Station (SONGS) Steam Generator Tube Wear Problems

3. NRC International Travel Trip Report

4. Report to NRC, Submitted by V.K. Dhir

5. Preliminary Significance Determination Loss of Steam Generator Tube Integrity

cc w/enclosure:
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R4Enforcement

R:_REACTORS_SONGS\2012\SO IR 12-09
ADAMS ACCESSION NUMBER: ML13263A271

ADAMS: □ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: GEW
	☒ Publicly Available	☒ Nonsensitive
	☐ Nonpublicly Available	☐ Sensitive

RIV:RI:SPB	RI:SPB	SSP:NRR	SSP:NRO	SRA:DRS:EB1	I&AL:SPB:ORA	RSLO:ORA
MRBloodgood	JPReynoso	ELMurphy	CGThurstson	GDReplogle	GEWerner	WAMaier
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SPAO:ORA	SES:ACES	C:ACES/ORA	C:SPB/ORA	RC/ORA	TM:SSP	Acting RA
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000361, 05000362
License:	NPF-10, NPF-15
Report:	05000361/2012009 and 05000362/2012009
Licensee:	Southern California Edison Company
Facility:	San Onofre Nuclear Generating Station, Units 2 and 3
Location:	5000 S. Pacific Coast Highway San Clemente, California
Dates:	December 3, 2012, through June 7, 2013
Team Lead:	G. Werner, RIV, SONGS Project Branch, Inspection Lead
Inspectors:	R. Lantz, Chief, SONGS Project Branch J. Reynoso, RIV, Resident Inspector E. Murphy, NRR, Senior Materials Engineer C. Thurston, RES, Reactor Systems Engineer M. Bloodgood, RIV, Reactor Engineer
Accompanying Personnel:	J. Billoue, Steam Generator Design Engineer, Beckman and Associates V. Dhir, PhD, Thermal-Hydraulic and Vibration Contractor, Beckman and Associates G. Warrier, PhD, Laboratory Research Engineer Contractor, Beckman and Associates
Approved By:	A. Howell, Team Manager San Onofre Nuclear Generating Station Special Project

SUMMARY OF FINDINGS

IR 05000361/2012009; 05000362/2012009; 12/03/2012 – 06/07/2013; San Onofre Nuclear Generating Station; Confirmatory Action Letter Response Inspection.

This inspection team was comprised of one resident, two region-based, two headquarters-based, and three contractor inspectors. One apparent White violation of low to moderate safety significance and one Green noncited violation of very low safety significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the Unit 2 replacement steam generators, resulting in excessive and unexpected steam generator tube wear after one cycle of operation. The licensee initiated Nuclear Notification NN 202447268 to address this issue in the corrective action program. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance. Subsequently, on June 7, 2013, Southern California Edison announced that Units 2 and 3 would be permanently shut down.

The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Event Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that the finding was of very low safety significance because the finding did not involve a degraded steam generator tube that could not sustain three times the normal operating differential pressure and did not violate the accident leakage performance criterion. No cross-cutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and other safety culture issues. The NRC determined that the performance behavior that existed at that time is not indicative of current performance.

Apparent Violation. The inspectors identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the Unit 3 replacement steam generators, which resulted in significant and unexpected steam generator tube wear after 11 months of operation and an associated apparent violation of Technical Specification 5.5.2.11, “Steam Generator Program,” loss of tube integrity on Unit 3 Steam Generator 3E0-88. The licensee initiated Nuclear Notification NN 202447265 to address this issue in the corrective action program. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance. Subsequently, on June 7, 2013, Southern California Edison announced that Units 2 and 3 would be permanently shut down.

This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design resulted in excessive and rapid tube wear due to fluid elastic instability, which challenged the structural integrity of the tubes to perform their pressure boundary function. The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that this finding required evaluation in accordance with Inspection Manual Chapter 0609, Appendix J, because the finding involved a degraded steam generator tube condition where one tube could not sustain three times the differential pressure across a tube during normal full power, steady-state operation. In accordance with Inspection Manual Chapter 0609, Appendix J, this finding required a detailed risk analysis, since it involved two or more tubes that could not sustain three times the normal differential pressure and one or more steam generators that violated “accident-induced leakage” performance criterion. A Phase 3 analysis was completed using the San Onofre SPAR model, Revision 8.22, assuming average test and maintenance, and a truncation limit of 1.0E-11. Based on the best available information, the performance deficiency was preliminarily characterized as a finding of low to moderate safety significance (White). The final significance of this finding is to be determined. No cross-cutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and other safety culture issues. The NRC determined that the performance behavior that existed at that time is not indicative of current performance

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Prior to the Unit 3 steam generator tube leak, Unit 2 was shut down for a scheduled refueling outage and Unit 3 was operating at 100 percent rated thermal power with no plant evolutions in progress. On January 31, 2012, Unit 3 control room operators received an alarm that indicated a primary-to-secondary reactor coolant leak from Steam Generator 3E0-88. The alarm received was from the main condenser air ejector radiation monitors, which continuously sample from a vent line for the purpose of rapidly identifying steam generator tube leaks. Although the leak rate was small, it increased enough in a short period of time for the licensee to perform a rapid shutdown. The estimated leak rate was 75 gallons per day. The facility license allows full power operation with a steady-state leak rate of less than 150 gallons per day. On February 2, 2012, Unit 3 reached cold shutdown conditions. The licensee reviewed the amount of gaseous radioactivity released and estimated a dose of approximately 0.0000452 mrem to a member of the public. The annual regulatory limit to a member of the public is 100 mrem per year. At the time of the inspection, Units 2 and 3 continued to remain in a shutdown status, with Unit 3 defueled.

On June 7, 2013, Southern California Edison (SCE) announced that Units 2 and 3 would be permanently shut down. On June 12, SCE submitted a Certification of Permanent Cessation of Power Operations to the NRC, certifying that Units 2 and 3 have permanently ceased power operations. On June 28 and July 22, 2013, SCE certified that all fuel had been permanently removed from the Units 3 and 2 reactors, respectively (ML13183A391 and ML13204A304).

1. REACTOR SAFETY

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 Event Report Review

a. Inspection Scope

The inspectors reviewed the following Licensee Event Report and related documents to assess: (1) the accuracy of the Licensee Event Report; (2) the appropriateness of corrective actions; (3) violations of requirements; and (4) generic issues.

(Closed) Licensee Event Report 05000362/2012-002-00, "Unit 3 Steam Generator Tube Degradation Indicated by Failed In-Situ Pressure Testing"

As described above, Unit 3 was shut down on January 31, 2012, because of a small steam generator tube leak. Subsequent in-situ pressure testing was conducted in March 2012 on a total of 129 tubes in Steam Generators 3E0-88 and 3E0-89. Eight tubes failed in-situ pressure testing in Steam Generator 3E0-88, with no tube failures in Steam Generator 3E0-89.

The inspectors identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for not verifying the design of the replacement steam generators' thermal-hydraulic and vibration analysis, which resulted in multiple losses of tube integrity contrary to Technical Specification 5.5.2.11.

See Report Section 4OA5, Subsection 5, for a detailed description of the closure of the unresolved item associated with the Licensee Event Report.

4OA5 Other Activities

Inspection Procedure 92702, "Followup on Traditional Enforcement Actions Including Violations, Deviations, Confirmatory Action Letters, Confirmatory Orders, and Alternative Dispute Resolution Confirmatory Orders"

.1 Review of Root Cause Evaluations

a. Inspection Scope

The inspectors reviewed SCE Root Cause Evaluation Nuclear Notifications 201836126 and 201836127 and the Mitsubishi Heavy Industries (Mitsubishi) Technical Evaluation Report, Document L5-04GA564, to determine the following: (1) complete and accurate identification of the problem; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, and common cause; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; (7) completion of corrective actions on Unit 2, including corrective actions based on the Unit 3 cause evaluation; and (8) effectiveness reviews.

b. Observations and Findings

No findings were identified.

The inspectors reviewed the SCE root cause evaluations and Mitsubishi technical evaluation related to the SCE steam generator tube degradation issue. The licensee and Mitsubishi used multiple techniques, which included event and causal factor charting, barrier analysis, and Kepner-Tregoe analysis to assess the probable causes of the tube wear identified in both Units 2 and 3 steam generators. Each steam generator has 9727 U-tubes, which are supported by seven tube support plates and six sets of V-shaped anti-vibration bars. The inspectors identified that both SCE and Mitsubishi identified four types of wear in both Units 2 and 3 steam generators. The identified wear was assessed by both SCE and Mitsubishi and corrective action recommendations were identified. The four types of tube wear are discussed in the following sections of this report.

- Type 1 (tube-to-tube wear) - Wear in the tube free-span sections in the U-bend region. Most of the tubes with this type of wear also have wear indications at anti-vibration bars and tube support plates. In this case, it is considered that the entire tube, including the straight leg, was vibrating excessively.

- Type 2 (anti-vibration bar wear) - Wear at only the tube-to-anti-vibration bar intersections, with no wear indications in the tube free-span sections. Some of these tubes have wear indications at the tube support plates as well. In this case, it is considered that mainly the U-bend section of the tube was vibrating.

- Type 3 (tube support plate wear) - Wear at the tube-to-tube support plate intersections only in the straight section of the tubes. In this case, it is considered that only the straight section of the tube was vibrating.

- Type 4 (retainer bar wear) - Wear at the anti-vibration bar structure retainer bars in the tube U-bend section. These tubes have no wear indications in the free span, at anti-vibration bars or at tube support plates. In this case, it is considered that the retainer bar itself was vibrating and the tube was not vibrating.

(1) Tube-to-Tube Wear (Type 1)

The inspectors reviewed the licensee's extent of condition associated with the identified tube-to-tube wear, which affected 326 tubes in Unit 3 steam generators during eddy current testing following the steam generator leak identified on January 31, 2012. Unit 2 was in a current shutdown for the first refueling outage following the replacement of both of the steam generators. The licensee had completed an initial 100 percent bobbin coil eddy current inspection of the Unit 2 steam generators, which failed to identify any tube-to-tube wear. Following the identification of the tube-to-tube wear in the Unit 3 steam generators, the licensee performed eddy current testing using a more sensitive (+P) probe of approximately 1300 tubes in each of the Unit 2 steam generators from the same area as the identified Unit 3 tube-to-tube wear. The licensee identified two tubes with approximately 15 percent tube-to-tube wear that were not identified during the initial eddy current testing using the bobbin coil probe.

The inspectors reviewed the mechanisms that were determined to be contributors to the tube-to-tube wear by the licensee and Mitsubishi. The mechanical cause of the Unit 3 tube-to-tube wear was determined by SCE and Mitsubishi to be fluid-elastic instability associated with adverse secondary thermal-hydraulic conditions and lack of effective in-plane tube support for the tubes. Mitsubishi determined that the tube-to-anti-vibration bar contact forces used in the replacement steam generators was insufficient to prevent the in-plane motion given the thermal-hydraulic conditions in the secondary side of the steam generators. Mitsubishi identified, as part of their evaluation, that the contact forces in Unit 3 were less than the Unit 2 contact forces following the review of the manufacturing dimensional tolerances. The anti-vibration bar dimensional tolerances are further discussed in Section 6 of this report. Westinghouse's independent assessment

(SG-SGMP-12-10, "Operational Assessment of Wear Indications in the U-bend Region of San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators Supporting Restart," Revision 3) concluded that Unit 2's tube-to-tube wear may be a result of tube-to-tube proximity in conjunction with flow-induced vibration. This mechanism was determined by Westinghouse to be a probable cause due to the wear patterns on the anti-vibration bars being limited to the anti-vibration bar width and not exhibiting the longer wear patterns associated with in-plane movement. The inspectors identified that this extent of condition for the tube proximity wear mechanics was not evaluated as part of the root cause evaluation. The licensee issued Nuclear Notification NN 201836127 to update the tube-to-tube wear root cause analysis with the licensee's and Mitsubishi's analysis of the Westinghouse Operations Assessment manufacturing issues and any resulting corrective action to address these issues.

The inspectors reviewed the licensee's corrective actions associated with the tube-to-tube wear. The licensee initially plugged the two tubes that were identified as having tube-to-tube wear in Unit 2. In addition, the licensee preventively plugged 321 Unit 2 tubes using selective process information from the Unit 3 steam generator wear data. The preventive tube plugging selection processes used nine screening criteria, including the location of anti-vibration bars and tube support plate wear indications, length of anti-vibration bar wear indications, average void fraction over the length of the tube, location of the tube within the bundle, and coupling between adjacent susceptible tubes. Results of the assessments of each tube against the nine screening criteria were reviewed cumulatively to identify which tubes would be preventively plugged. No issues were identified.

(2) Tube to Anti-Vibration Bar Wear (Type 2)

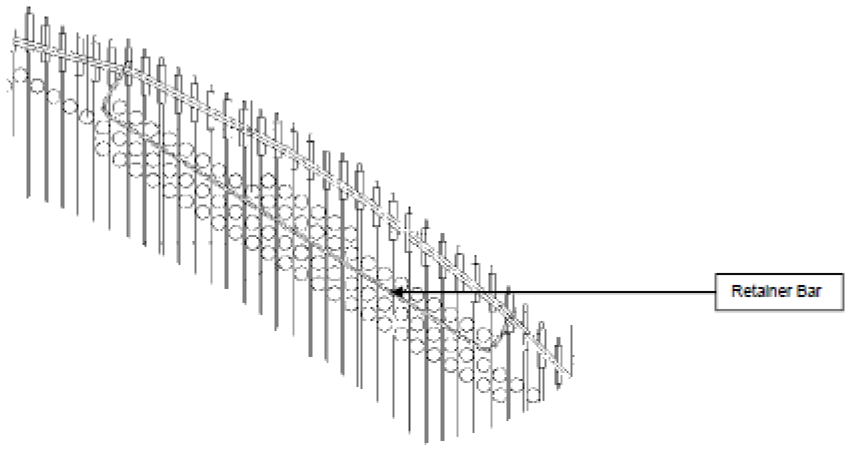
The inspectors reviewed the licensee's extent of condition associated with the identified tube-to-anti-vibration bar wear, which affected 1767 tubes in Unit 3 and 1399 tubes in Unit 2 steam generators. The licensee considered tube-to-anti-vibration bar wear patterns as part of their evaluation consisting of: (1) wear patterns in tubes that exhibited tube-to-tube wear and (2) wear patterns in tubes that did not exhibit tube-to-tube wear. The first pattern was determined to be the result of conditions resulting in fluid-elastic instabilities and subsequent in-plane motion of the tubes. These tube-to-anti-vibration bar wear patterns tended to extend beyond the edges of the anti-vibration bar, indicating an in-plane sliding motion of the U-bend that also led to the tube-to-tube wear in the affected tube. The second tube-to-anti-vibration bar wear pattern was considered to be due to turbulence-induced vibration in the out-of-plane direction. The length of these wear patterns was confined to the width dimension of the anti-vibration bars. Mitsubishi only discussed the second wear pattern described by SCE as part of their Type 2 wear pattern. The first wear pattern was addressed as part of the tube-to-tube wear (Type 1) discussion. Mitsubishi determined that the wear was due to random tube vibration. Mitsubishi describes random vibration as a phenomenon where the tubes vibrate due to forces created by turbulent flow as a result of fluid velocity and density fluctuations, which are smaller than those due to tube fluid-elastic instability. Mitsubishi's determination was consistent with SCE review of the second wear pattern. The adverse secondary thermal-hydraulic condition and the lack of effective anti-vibration bar supports contributed to the presence of fluid-elastic instabilities and turbulence induced vibration. The licensee stabilized and plugged four of these tubes in Unit 2 and one tube in Unit 3 in accordance with their Steam Generator Program due to the wear identified during the eddy current inspection.

(3) Tube-to-Tube Support Plate Wear (Type 3)

The inspectors reviewed the licensee's extent of condition associated with the identified tube-to-tube support plate wear, which affected 463 tubes in Unit 3 and 299 tubes in Unit 2 steam generators. The licensee considered two categories of wear as part of their evaluation consisting of: (1) tube-to-tube support plate wear affecting tubes also exhibiting tube-to-tube wear, and (2) tube-to-tube support plate wear in tubes not exhibiting tube-to-tube wear. The licensee determined that the higher tube-to-tube support plate wear identified in the Unit 3 steam generator was due to in-plane fluid-elastic instability, resulting in higher displacement vibrations. This conclusion was based on the relationship between the identified tubes with tube-to-tube support plate and tube-to-tube wear. Mitsubishi considered Type 3 wear as only straight leg wear due to vibrations corresponding to the second category described by SCE. Mitsubishi concluded that the wear was caused by cross-flow induced random vibration in the region where secondary fluid cross-flow velocities are high. The licensee did not perform additional cause analysis for the tube-to-tube support plate wear due to the close correlation with the tube-to-tube wear and the corrective actions being inclusive in the tube-to-tube wear corrective actions.

(4) Tube-to-Retainer Bar Wear (Type 4)

The licensee requested Mitsubishi to conduct an evaluation of the tube-to-retainer bar wear as a result of the wear indications found in the Unit 2 steam generators. Mitsubishi's evaluation determined that the wear was the result of movement of the retainer bar located at anti-vibration bars 2 and 3 on the hot leg side of the U-bend and anti-vibration bars 10 and 11 on the cold leg side. A review of the retainer bars at these locations revealed that the replacement steam generator retainer bars were significantly longer (24.02 inches in length) when compared to other steam generator designs (7-13 inches in length). In addition, the natural frequency associated with the longer, thinner retainer bars was significantly (approximately 5 times) less than the other steam generator designs reviewed during the evaluation. The licensee determined that the mechanical root cause of the wear was the combination of the longer length and smaller diameter (0.187 inches) retainer bar natural frequency and the increased flow velocities, which contributed to a flow-induced vibration of the retainer bars, resulting in tube-to-retainer bar contact.



The licensee determined that the lack of tube-to-retainer bar wear at anti-vibration bars 1 and 12, which are the same dimensions as the four previously discussed, was attributed to the lower flow velocity (7.5 ft/s) at the anti-vibration bar 1 location compared to the flow velocity (9.8 ft/s) at the anti-vibration bar 2 location. Mitsubishi failed to evaluate the potential effects of flow-induced vibrations during the anti-vibration and retainer bar design, due to assumptions that the natural frequency of the retaining bar was high enough to preclude flow-induced vibration. Mitsubishi failed to consider the effects of using retainer bars that were longer and of smaller diameter than those they had previously used.

During the design of the replacement steam generators, SCE questioned Mitsubishi on the lack of vibration analysis of the retainer bars but failed to independently evaluate the Mitsubishi response and agreed with the determination that the retainer bars would not come into contact with the tubes. Mitsubishi provided additional chromium plating of the retainer bars to reduce the wear coefficient and minimize potential wear in response to the licensee's questions.

The licensee's corrective actions included plugging 94 tubes in each of the Units 2 and 3 steam generators in the vicinity of the retainer. The licensee also stabilized 14 tubes in the Unit 2 and 12 tubes in the Unit 3 steam generators. For details about the retainer bar design noncited violation, please refer to NRC Inspection Report 05000361/2012010 and 05000362/2012010 (ML12318A342).

(5) Return to Service Defense-in-Depth Actions

The inspectors reviewed the licensee's defense-in-depth actions, which were specified in Section 9 of the San Onofre Nuclear Generating Station Unit 2 Return to Service Report dated October 3, 2012. The inspectors reviewed the following defense-in-depth actions and determined them to be enhancements which assist the operator's identification and response in the event a steam generator tube leak occurs:

(a) Injection of Argon into the Reactor Coolant System

The inspectors reviewed the defense-in-depth action of adding argon to the reactor coolant system. The addition of argon was planned to be added to enhance/improve the primary-to-secondary leak rate detection level at the condenser air ejector radiation monitor. The inspectors reviewed the licensee's actions to add argon to improve the leak rate sensitivity. The inspectors reviewed Procedure SO123-III-2.22.23, "Unit 2/3 Steam Generator Tube Leakage Monitoring Program," Revision 25, which specified that the reactor coolant system activated argon activity should be maintained between 0.05 uCi/ml and 0.15 uCi/ml. A value of 0.10 uCi/ml of activated argon ensures that the condenser air ejector radiation monitor is capable of detecting a 5 gpd primary-to-secondary leak rate instead of the normal setpoint of 30 gpd. In addition, the inspectors reviewed argon injection and controls specified in Procedures SO23-3.2.1, "CVCS Operation," Revision 40, and SO23-3.2.1.1, "CVCS Alignment," Revision 20.

(b) Installation of Nitrogen-16 Radiation Detection System on the Main Steam Lines

The inspectors reviewed the licensee's implementation of a radiation detection system for detecting nitrogen-16. The licensee installed nitrogen-16 detectors, a more sensitive radiation detection system, adjacent to the main steam line to provide early operational responses to primary-to-secondary leaks in the steam generators in addition to current radiation detection systems. The nitrogen-16 detectors are located so that a leak would be detected seconds after it occurs. This is an improvement from the current condenser offgas radiation monitors and blowdown samples, which could take up to one hour. The inspectors reviewed Nuclear Notification NN 800905312, which described the implementation of the nitrogen-16 detectors and procedural guidance for the operation of the detecting system. The inspectors identified that Procedure SO23-3-2.24, "Radiation Monitoring System Guidelines and RDU Operation," Revision 14, was changed to direct actions to implement Procedure SO23-13-14, "Reactor Coolant Leak," Revision 21, primary-to-secondary operator actions in the event that the nitrogen-16 monitor alarms. This will provide for earlier operator actions in the event of a primary-to-secondary leak.

(c) Reduction of Administrative Limits for the Reactor Coolant System Activity Level

The inspectors reviewed changes to Procedure SO123-III-1.1.23, "Unit 2/3 Chemistry Control of Primary Plant and Related Systems," Revision 60, for defense-in-depth actions associated with administrative limits for reactor coolant system activity levels. The inspectors identified that Nuclear Notification NN 201836127, Task 39, lowered the dose equivalent iodine-131 normal range in Procedure SO123-III-1.1.23 from 1.0 µCi/gram to

≤0.5 μCi/gram. This provides an additional action level prior to reaching the technical specification limit of 1.0 μCi/gram. The inspectors determined that procedural guidance will require Operations personnel to make a determination of continued plant operation if the dose equivalent iodine-131 exceeds the new lower normal range (≤0.5 μCi/gram) instead of the technical specification limit.

(d) Enhanced Operator Response to Early Indication of Steam Generator Tube Leakage

The inspectors reviewed procedural changes, training material, and training records, in addition to interviewing licensee personnel associated with the enhanced operator response. The training consisted of classroom lectures; simulator scenarios; and just-in-time training associated with steam generator tube ruptures, the addition of argon and temporary nitrogen-16 monitors, and changes in plant procedures related to steam generator tube ruptures, primary leaks, and modifications. This training was performed as part of the operator requalification program. The inspectors reviewed the training completion records associated with the training related to the enhanced operator actions and identified that the training was being performed but that not all operators had completed all of the training at the time of the inspection. The licensee is tracking the training completion in their operator requalification program.

The inspectors determined that the defense-in-depth actions were determined to be appropriate and could be performed in accordance with procedural guidance.

.2 Operational Assessments

a. Inspection Scope

The inspectors reviewed the SCE operational assessment of steam generator tube integrity for Unit 2 for the period extending from Unit 2 restart from Refueling Outage 17 to Unit 2 shutdown for its next scheduled steam generator inspection. The inspectors reviewed the following specific documents pertaining to the operational assessment:

- SCE Confirmatory Action Letter response dated October 3, 2012, Enclosure 2, “San Onofre Nuclear Generating Station Unit 2 Return to Service Report”
- “San Onofre Nuclear Generating Station Unit 2 Return to Service Report,” Attachment 4, “MHI Document L5-04GA564, Tube Wear of Unit-3 RSG - Technical Evaluation Report,” Revision 9 (proprietary version), prepared by Mitsubishi
- “San Onofre Nuclear Generating Station Unit 2 Return to Service Report,” Attachment 6, “SONGS U2C17 Steam Generator Operational Assessment”:

- Appendix A, Document 1814-AU651-MO144, “SONGS U2C17 Outage - Steam Generator Operational Assessment,” Revision 0 (proprietary version), prepared by AREVA NP Inc. for degradation mechanisms other than tube-to-tube wear

- Appendix B, Document 1814-AU651-MO146, “SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear,” Revision 0 (proprietary version), prepared by AREVA NP Inc.

- Appendix C, Document 1814-AU651-MO145, “Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at the End of Cycle 16,” Revision 1, prepared by Intertec APTECH

- Appendix D, Document 1814-AA086-M0190, “Operational Assessment of Wear Indications in the U-Bend Region of San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators,” Revision 4, prepared by Westinghouse Electric Company, LLC.

The inspectors assessed the implementation of the operational assessments relative to Technical Specification 5.5.2.11, “Steam Generator (SG) Program,” and Electric Power Research Institute (EPRI) Report 1019038, “Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines,” Revision 3, as referenced in Procedure SO23-SG-1, “SONGS Steam Generator Program,” Revision 20. The inspectors’ review included the following items:

- Degradation mechanisms, growth rate calculations, and assumptions
- Tube plugging and stabilization
- Operating restrictions
- Appropriateness of 5-month inspection interval
- Application of Unit 3 extent-of-condition on Unit 2 operational assessments
- Midcycle inspection methodology

The inspectors also reviewed the thermal-hydraulic models and flow-induced vibration models used for the operational assessments. This part of the review is addressed in Section 4OA5.3 of this inspection report.

b. Observations and Findings

Based on SCE’s decision to retire both units, the reviews of the operational assessments were not completed. No conclusions were made as to the adequacy of each operational assessment.

.3 Thermal-Hydraulic and Vibration Models

a. Inspection Scope

The inspectors reviewed the Mitsubishi ATHOS thermal-hydraulic model as well as the FIVATS tube vibration models as specified in Specification SO23-617-1, "Specification for Design and Fabrication of the Replacement Steam Generators for Unit 2 and Unit 3," Revision 4. As part of the review, the inspectors compared the results of the Mitsubishi ATHOS thermal-hydraulic model results to results of AREVA CAFCA4 and Westinghouse ATHOS. The NRC independently ran the ATHOS model to verify consistency and appropriateness of the steam velocities and void fractions. The inspectors also had the assistance of a contractor with expertise in steam generator tube vibration who reviewed the technical basis for the revised Connors' equation used to calculate stability ratios in the in-plane direction.

b. Observations and Findings

No findings were identified.

ATHOS is an industry three-dimensional computational fluid dynamics code developed by EPRI to assess thermal-hydraulic conditions in steam generators. The code was developed in the 1980's and is still being used today by nuclear facilities and vendors. The code iteratively solves the conservation of the mass, momentum, and energy equations along with empirical correlations to calculate the thermal-hydraulic parameters for a given steam generator geometry design and set of plant operating conditions. The code has many other uses, including performance trending, deposit mapping, and sludge pile predictions. The code is not used for safety analysis and has not been reviewed or approved by the NRC.

Mitsubishi used the latest version, EPRI ATHOS version 3.1, of the code while Westinghouse used its ATHOS60 version 3.0. Westinghouse has the most extensive experience with the code, having developed several pre- and post-processor add-ons for its analysis methodology.

Westinghouse uses the ATHOS code for new designs as well as assessments for the various model series of Westinghouse steam generators. Westinghouse performed validation and verification of ATHOS by benchmarking the code against several full-sized and scaled model tests[1]. They concluded that the ATHOS calculated thermal-hydraulic parameters were in good agreement with measured test data.

Mitsubishi and Westinghouse each independently developed ATHOS models and ran cases to generate three-dimensional thermal-hydraulic performance data for various specific SCE steam generator boundary conditions. The NRC also developed an ATHOS model and ran limited cases for independent assessment of the vendor analyses.

[1] Westinghouse LTR-SGDA-12-50 dated 10/14/2012

Both Mitsubishi (L5-04GA566) and Westinghouse (LTR-SMP-12-36) ran cases on Units 2 and 3 for plugging scenarios and for power levels ranging from 50-100 percent in 10 percent increments. Velocity and density output from these cases were used as input to the flow induced vibration analysis. For the vibration analysis, Westinghouse uses FASTVIB and Mitsubishi uses FIVATS. Both analyses compute the stability ratios based on a form of the Connors' Equation, but there were considerable differences in the empirical constants used to characterize the tube excitation threshold. The empirical constants are the critical factor K and the overall tube damping ratio h. The FASTVIB code incorporates the analytical approaches with constants that were largely defined by the work of H. J. Connors while conducting research for Westinghouse at their research lab. The analytical approach of Mitsubishi was a best estimate and was principally based on more recent work by the Canadian researcher, M.J. Pettigrew (École Polytechnique de Montreal).

Prevention of excessive vibration and fretting wear is generally achieved by a combination of design, analysis, and testing as defined by each vendor and their own methodology. Each vendor methodology is required to have some level of validation, as noted for design of ASME components. This methodology is used to lay out and evaluate the many aspects of design before the steam generators are manufactured. The anti-vibration bars and tube support plates should be arranged to meet specified design limits established to prevent fluid instabilities and minimize tube wear. The risk of vibration is highest in the U-bend region, where velocity and void fraction are highest or there is significant unsupported span length which lowers the natural frequency of the tube making that span more susceptible to vibration.

The methodology, based on the Connors' Equation, is of the following general form:

$$\boldsymbol{Uc = K \left[parameters * \frac{h}{\rho}\right]^{0.5}} \qquad (1)$$

where:
Uc - Critical flow velocity
K - Critical factor
h - damping ratio
ρ – fluid density surrounding the tube

and where:

$$\boldsymbol{SR = \frac{U_{eff}}{Uc}} \qquad (2)$$

U_{eff} – effective gap velocity (computed from ATHOS)
SR – stability ratio

The computations for K and h vary for in- and out-of-plane stability ratio and are different in each vendor's methodology. The vibration analysis methods used by Mitsubishi were similar to that recommended by ASME Code Section III, Appendix N, originally designed for traditional out-of-plane stability analysis. The anti-vibration bars were designed primarily to prevent out-of-plane motion. The design thickness of the anti-vibration bars provides for a small gap that essentially closes when the plant heats up to normal operating temperature. The damping improves as the gaps close, providing stability for the tube in the out-of-plane direction. Although not specifically part of the design, anti-vibration bars are known to provide some limited stability and support for tube motion in the in-plane direction based on experimental and empirical information.

Historically, during the design of all steam generators, including the SCE replacement steam generators, in-plane vibration was not considered since it had not been seen in any operating steam generators and was believed bounded by (i.e., can occur only as a consequence of) out-of-plane vibration analysis.

Based on the steam generator tube-to-tube wear caused by in-plane motion, both vendors modified the Connors' Equation for applicability to in-plane fluid-elastic instability based on a mixture of internal experimental methods and academic research. The methodology is highly dependent upon the number of assumed continuous ineffective anti-vibration bars and other assumptions that have not been verified by experimental data.

In licensee calculations at 100 percent power with all anti-vibration bar supports effective, neither of the vendor models predicted stability ratios above 1.0. Both the Mitsubishi and Westinghouse analyses have evaluated a minimum of two continuous ineffective anti-vibration bars, and the calculations then advance in continuous increments of two ineffective supports until in-plane fluid-elastic instability is predicted, i.e., stability ratio greater than 1.0 with various levels of plant power.

Overall, the inspectors determined that the Mitsubishi vibration stability ratio results were higher than Westinghouse's independent analysis and, therefore, provided some confirmation that Mitsubishi's model was not under-predicting the conditions of steam generators.

The NRC did not develop independent vibration calculations; however, the vendor computations for vibration were independently reviewed and checked for correctness. The inspectors did not identify any issues with the vibration calculations.

.4 Design Modification Review

a. Inspection Scope

The inspectors reviewed the design changes, listed below, that were associated with the licensee's Confirmatory Action Letter Response for Unit 2 to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report, had been reviewed and documented in accordance with 10 CFR 50.59 requirements.

- Plant operation at 70 percent power level
- Argon (Ar-40) injection into the reactor coolant system
- Nitrogen (N-16) radiation detection system on the main steam lines

- Vibration and loose parts monitoring system
- Annunciator for 70 percent power operation
- Tube plugging and stabilization

The inspectors reviewed the various information used by SCE to make the changes to Unit 2 associated with their return to operation, including calculations, analyses, design change documentation, procedures, the Updated Final Safety Analysis Report, the Technical Specifications, and plant drawings. The inspectors interviewed plant personnel responsible for developing and evaluating the design changes. The inspectors compared the safety evaluations and supporting documents to the guidance and methods provided in NEI 96-07, "Guidelines for 10 CFR 50.59 implementation," Revision 1, as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the evaluations.

b. Observations and Findings

The inspectors determined that the 50.59 screen and evaluations were performed in accordance with the requirements of 10 CFR 50.59, “Changes, Tests, and Experiments," with the exception of the following discussions.

During the inspectors’ review of the 10 CFR 50.59 screenings for operation at 70 percent reactor power associated with Nuclear Engineering Change Package 800873488-0131 and vibration and loose parts monitoring system modification associated with Nuclear Engineering Change Package 800457837-0550, the following deficiencies were identified:

(1) Operation at 70 percent Reactor Power

(a) Technical Specification Surveillance Requirement 3.3.1.11 states, “Using the incore detectors, verify the shape annealing matrix elements to be used by the CPCs [core protection calculators],” with a specified frequency of “Once after each refueling prior to exceeding 85% RTP [rated thermal power].” However, the inspector identified that the 10 CFR 50.59 screen for 70 percent evaluation of this technical specification incorrectly stated that “SR [Surveillance Requirement] 3.3.1.11 is not required to be performed until 12 hrs after THERMAL POWER has reached or exceeded 85% RP [reactor power].”

(b) Technical Specification Surveillance Requirement 3.3.1.2 states, “Verify total Reactor Coolant System (RCS) flow rate as indicated by each CPC is less than or equal to the RCS total flow rate. If necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the RCS flow rate,” with a specified frequency of 12 hours. Technical Specification Surveillance Requirement 3.3.1.5 states, “Verify total RCS flow rate indicated by each CPC is less than or equal to the RCS flow determined by calorimetric calculations,” with a specified frequency of 31 days. Technical Specification Surveillance Requirement 3.3.1.2 and Surveillance Requirement 3.3.1.5 each contain a note stating that the Surveillance Requirement is “Not required to be performed until 12 hours after THERMAL POWER $\geq$ 85 percent RTP.” However, the 50.59 screen for 70 percent evaluation of the technical specification did not evaluate these surveillance requirements.

Nuclear Notification NN 202243314 was written to address these deficiencies. Corrective actions included revising the 10 CFR 50.59 screening to evaluate the actual wording of the Technical Specification Surveillance Requirement 3.3.1.11 frequency (“Once after each refueling prior to exceeding 85% RTP,”) which allows performing the surveillance at 70 percent power. The 10 CFR 50.59 screening change also added evaluation of Technical Specification Surveillance Requirements 3.3.1.2 and 3.3.1.5, which stated that, although the surveillances are not required to be performed until 12 hours after reaching 85 percent RTP, the associated surveillance procedures had been revised to direct these surveillance requirements be performed above 68 percent RTP. The surveillance procedures were changed so the procedure steps do not conflict with the technical specification wording. The licensee determined that changes to Technical Specification Surveillance Requirements 3.3.1.2, 3.3.1.5 and 3.3.1.11 were not required.

The inspectors noted that the original 10 CFR 50.59 written evaluation for this change did not adequately evaluate the effect of limiting reactor power operations to 70 percent on Technical Specification Surveillance Requirements 3.3.1.2, 3.3.1.5, and 3.3.1.11. Therefore the inspectors determined that the evaluation was not adequate, in that it did not provide an adequate basis for the determination that the change to limit reactor power operations to 70 percent did not require a license amendment prior to implementing the change. Title10 CFR 50.59(d)(1) requires that the licensee maintain records of changes in the facility that “include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment . . .” Contrary to the above, on September 14, 2012, the licensee’s written evaluation in the 10 CFR 50.59 screening for ECP 800873488-0131 did not provide an appropriate basis for the determination that the change to limit reactor power operation to 70 percent did not require a license amendment.

Because this violation impacted the regulatory process, the inspectors assessed it in accordance with the NRC Enforcement Policy, as directed by Inspection Manual Chapter 0612, Appendix B, “Issue Screening.” The NRC Enforcement Manual contains specific processes and guidance for implementing this Policy. NRC Enforcement Manual, Part II, Section 2.1.3E.6.b states, in part, that minor violations include the failure to meet 10 CFR 50.59 requirements that involve a change to the final safety analysis description where there was no reasonable likelihood that the change would ever require NRC approval per 10 CFR 50.59. As described above, the change to limit reactor power operations to 70 percent did not require changes to Technical Specification Surveillance Requirements

3.3.1.2, 3.3.1.5, and 3.3.1.11; thus, with respect to Part II, Section 2.1.3E.6.b of the NRC Enforcement Manual, there is no reasonable likelihood that this would ever require NRC approval. Therefore, in accordance with the NRC Enforcement Manual, the inspectors determined that the failure to provide an adequate written evaluation of the change to limit reactor power operation to 70 percent was a minor violation of 10 CFR 50.59(d)(1).

(2) Vibration and Loose Parts Monitoring System

(a) Updated Final Safety Analysis Report Section 5.4.1.5.5 states, "Motor vibration is sensed by the VLPM [vibration and loose parts monitoring] and pump shaft vibration systems attached to the pump driver mount (motor stand). Excessive vibration is alarmed in the control room."

(b) Updated Final Safety Analysis Report Table 7.6-4, "Safety and Nonsafety instrumentation Flooding Analysis," referenced the reactor coolant pump P002/P004 vibration and loose parts monitoring transmitters.

The inspectors identified that Updated Final Safety Analysis Report Section 5.4.1.5.5 and Table 7.6-4 were affected by the vibration and loose parts monitoring system modifications and not addressed as part of the 50.59 screening for Nuclear Engineering Change Package 800457837-0540. The licensee evaluated the replacement vibration and loose parts monitoring system against Regulatory Guide 1.133, "Loose-part Detection Program for the Primary System of Light Water Cooled Reactors," Revision 1, which specifies the monitoring locations for the vibration and loose parts monitoring system detectors and determined that the replacement of the Combustion Engineering system with the Westinghouse Digital Metal Impact Monitoring System met the guidance specified in Regulatory Guide 1.133. The licensee monitors reactor coolant pump vibration using the reactor coolant pump shaft vibration system, which provides an alarm indication in the control room, and the replacement of the vibration and loose parts monitoring system does not affect the operation of the system. In addition, there were no safety impacts associated with not updating Table 7.6-4, since the changes involved abandoning equipment in place.

The inspectors observed the initial testing setup of the new Unit 2 vibration and loose parts monitoring system. The inspectors reviewed the test results of the Westinghouse Digital Metal Impact Monitoring System (vibration and loose parts monitoring system) that included newly designed externally mounted sensors located at the 7th tube support plate level of the replacement steam generators. The test consisted of small impacts with a specially designed transducer hammer inside the steam generator near the 7th tube support plate. The test results demonstrated that the new system could detect the hammer impacts near the 7th tube support plate level. However, the inspectors noted that the test results could not be calibrated to detect tube-to-tube impact, since the test was limited to only impacts to the tube support plate. There was not a test that could simulate tube contact. Even with this limitation, the new external sensors were sensitive enough to detect internal impacts at the upper level of the steam generator, specifically associated with the 7th tube support plate.

Title 10 CFR 50.59(d)(1) requires that the licensee maintain records of changes in the facility that "include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment . . ." Contrary to the above, on December 20, 2012, the licensee's Nuclear Engineering Change Package 800457837-0540 did not provide an adequate basis for the determination that the change to the vibration and loose parts monitoring system did not require a license amendment. Specifically, the 10 CFR 50.59 screening did not evaluate the effect of the vibration and loose parts monitoring system modifications on Updated Final Safety Analysis Report Section 5.4.1.5.5 and Table 7.6-4.

Because this violation impacted the regulatory process, the inspectors assessed it in accordance with the NRC Enforcement Policy, as directed by Inspection Manual Chapter 0612, Appendix B, "Issue Screening." The NRC Enforcement Manual contains specific processes and guidance for implementing this Policy. The NRC Enforcement Manual, Part II, Section 2.1.3E.6.b, states, in part, that minor violations include the failure to meet 10 CFR 50.59 requirements that involve a change to the final safety analysis report where there was no reasonable likelihood that the change would ever require NRC approval per 10 CFR 50.59. As described above, the change to the vibration and loose parts monitoring system modifications did not require a license amendment prior to implementing the change so, with respect to Part II, Section 2.1.3E.6.b of the NRC Enforcement Manual, there is no reasonable likelihood that this change would ever require NRC approval. Therefore, in accordance with the NRC Enforcement Manual, the inspectors determined that the failure to provide an adequate written evaluation of the vibration and loose parts monitoring system modifications was a minor violation of 10 CFR 50.59(d)(1). This deficiency was entered into the licensee's corrective action program as Nuclear Notification NN 202258050.

.5 (Closed) Unresolved Item 05000362/2012007-08, "Non-Conservative Thermal-Hydraulic Model Results"

NRC Inspection Report 05000361/2012007 and 05000362/2012007described this unresolved item, in part, as follows:

> The team identified an unresolved item associated with the adequacy of Mitsubishi's FIT-III thermal-hydraulic code. The FIT-III code predicted nonconservative low velocity and low void fraction results which were used as inputs to the vibration code FIVATS. These non-conservative thermal-hydraulic results led Mitsubishi to conclude that margins to instability were significantly larger than they actually were.

During the original design, Mitsubishi used a number of computer codes to determine that the design of the steam generators was adequate in regard to potential for vibration and excessive wear. Based on key boundary operating parameters, primary temperature, and flow rates from the SCE replacement steam generator design specification, the Mitsubishi-developed steam generator steady-state performance code was used to provide a one-dimensional thermal-hydraulic calculation of the overall steam generator performance parameters, such as primary outlet temperature, steam pressure, steam flow, and tube bundle circulation ratio.

The steady-state performance calculation code results were then used in the Mitsubishi FIT-III thermal-hydraulic code to define velocity, density, and void fraction for input into the vibration calculation. The Mitsubishi-developed FIVATS code was then used by Mitsubishi in fluid-elastic stability analyses to determine if tubes were subjected to conditions that would exceed their stability ratio design limit of 1.0, assuming that 1 of the 12 anti-vibration bar support contacts was ineffective.

It was determined during the Augmented Inspection Team inspections in March 2012 that the SCE steam generators were under-designed in regard to margin to vibration and that the lack of margin was largely due to under-prediction of gap velocity and void fraction by the Mitsubishi FIT-III code analysis (Mitsubishi Document L5-04GA521, “Three-Dimensional Thermal and Hydraulic Analysis,” Revision 3). The local thermal-hydraulic analysis had a significant effect on the stability ratio results. It was concluded that the U-bend velocities were underpredicted by a factor of 2.5 to 3. Additionally, the FIT-III peak void fraction and quality were 0.95 and 0.67, respectively. The comparable values from the ATHOS results were 0.996 and 0.91 at 100 percent power, respectively, where fluid density is also 2.5 to 3 times smaller than the values computed by FIT-III. These two under-predicted factors produced stability ratios that were lower by 20 to 40 percent as compared to stability ratios based on ATHOS results. This resulted in no margin for a small number of tubes and for the majority of tubes much less margin to the onset of fluid-elastic instability than the designers or the licensee intended. Using Mitsubishi design criteria of one inactive anti-vibration bar, there were some tube stability ratios that exceeded 1.0.

a. Inspection Scope

The inspectors reviewed corrective action program documents and supporting engineering evaluations associated with this unresolved item to determine if a performance deficiency existed or if the issue constituted a violation of NRC requirements. The inspectors reviewed Nuclear Notification NN 201836127 and the associated cause evaluation performed by SCE to address the mechanistic cause of the nonconservative results of the FIT-III thermal-hydraulic model and flow-induced vibration analysis developed by Mitsubishi for the design of Units 2 and 3 replacement steam generators. The inspectors also reviewed the status of SCE’s and Mitsubishi’s cause evaluation to identify the organizational and programmatic factors leading to the nonconservative thermal-hydraulic model.

Additionally, the inspectors reviewed the replacement steam generator design specification for the replacement steam generators to identify the applicable design standards for thermal-hydraulic modeling and flow-induced vibration. The review of design information included design basis documents for the original steam generators to identify any design requirements for thermal-hydraulic modeling and flow-induced vibration, in order to determine if those requirements were properly translated into the replacement steam generator design specification. The inspectors reviewed the applicable design standards to identify design information that would have prompted the licensee to identify deficiencies in the thermal-hydraulic model and flow-induced vibration analysis. Particularly, the inspectors reviewed the technical justification for critical assumptions and design inputs.

The inspectors interviewed licensee staff and reviewed applicable quality assurance requirements and site procedures for the verification of supplier documents to assess whether the licensee had a reasonable opportunity to identify any deficiencies with the thermal-hydraulic modeling and the flow induced-vibration analysis based on the requirements and guidance in site procedures.

b. Findings

Introduction. For Unit 2, the NRC identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators.

For Unit 3, the NRC identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators, which also resulted in an associated apparent violation of Technical Specification 5.5.2.11, “Steam Generator Program,” because of a loss of tube integrity on Unit 3 Steam Generator 3E0-88.

Description. The effective construction code for SCE replacement steam generators was the 1998 Edition, with 2000 Addenda, of the ASME BPVC, Section III. Article NCA-3200 required that the owner shall prepare, review, and approve the replacement steam generator design specification, which contains the specific design requirements for the applicable code component. The replacement steam generator design specification for SCE replacement steam generators (Document SO23-617-01, “Specification for Design and Fabrication of replacement steam generators for Unit 2 and Unit 3,” Revision 4), Section 3.8.2, stated that: “The Supplier [MHI] shall prepare and submit for SCE’s approval a Performance Analysis Report documenting all thermal-hydraulic aspects of the replacement steam generators. The Report shall include all computer codes and modeling for the thermal-hydraulic performance of the replacement steam generators. The Report shall include detailed calculations, by region, showing that cross-flow velocities within the tube bundle shall be such as to minimize tube wear at the tube to tube-support interfaces. The calculations shall clearly identify the damping factor(s) used and margins to flow instability for steam flow rates of up to 120% of the design flow rate."

Additionally, Section 3.21.7 of the replacement steam generator design specification stated that, "The Supplier shall provide a new thermal-hydraulic analytical model, or update the existing plant EPRI ATHOS model for the replacement steam generators, and furnish all input parameters required to update the existing steam generator simulator model. The Supplier shall provide an executable version of the thermal-hydraulic computer codes used in the design of the replacement steam generators."

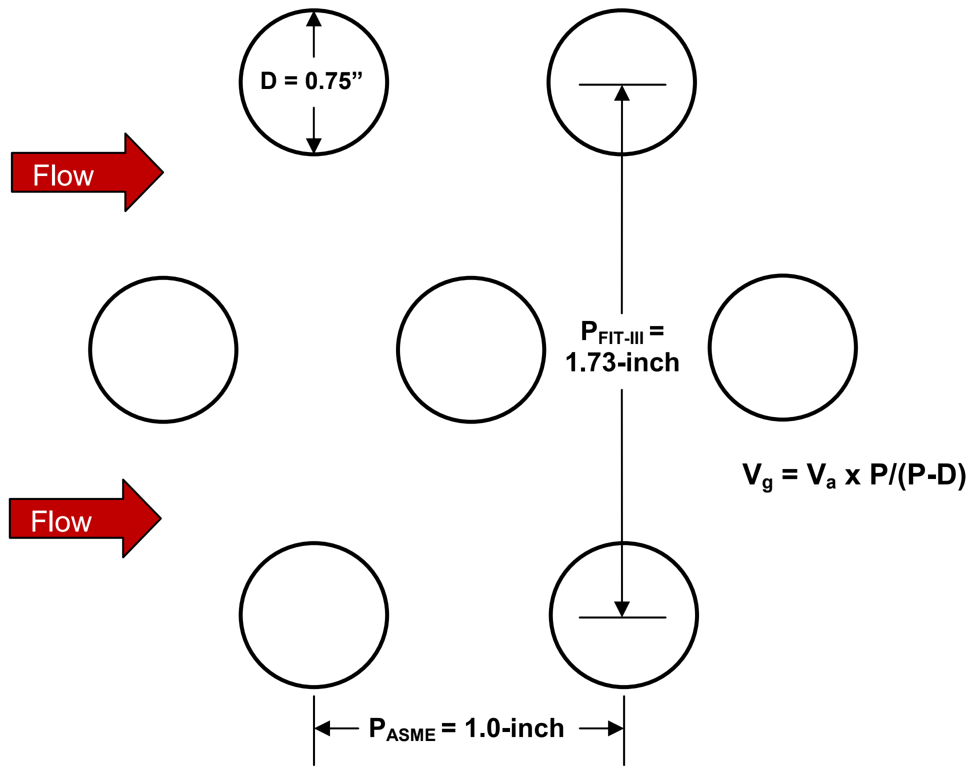
Furthermore, Section 3.5.1 of the replacement steam generator design specification stated that, "To the extent practical, the version and identity of all Codes, Standards, and other documents applicable to this Specification are shown in this Section." Following this statement, the replacement steam generator design specification listed ASME BPVC, Section III, Subsection NCA, and Division 1 Appendices as part of the applicable standards.

Mitsubishi Document L5-04GA504, "Evaluation of Tube Vibration," Revision 3, adopted the methodology in Non-Mandatory Appendix N to ASME BPVC, Section III, "Dynamic Analysis Methods," to evaluate flow-induced vibration in tube arrays exposed to cross flow. Specifically, Section 5 of this document stated that the vibration analysis was performed in accordance with the procedures and suggested inputs given in Appendix N-1330 to ASME Code Section III. This analysis was reviewed and approved by SCE on January 28, 2008, during the design stage of the replacement steam generators.

Paragraph N-1330 in Appendix N, to ASME Code Section III, provided recommendations and inputs for avoiding fluid-elastic instability of tube arrays. Paragraph N-1331.1, "Prediction of the Critical Velocity," stated that the onset of instability is governed, in part, by the flow velocity in the gaps between the tubes, which is determined by Vg = Va x P/(P-D), where "Vg" is the gap velocity, "Va" is the approach flow velocity that would occur if the tubes were not present, "P" is the tube array pitch as defined in Figure N-1331-3, and "D" is the outside diameter of a tube.

In response to this unresolved item, Mitsubishi identified that one of the factors responsible for the nonconservative flow velocities was that the flow area definition was not consistent with the recommendations in Appendix N (Mitsubishi Document L5-04GA591, "Validity of Use of the FIT-III Results during Design," Revision 1). Mitsubishi determined that the tube-to-tube gap used in the FIT-III thermal-hydraulic code, to determine the gap velocities, was larger than the recommended value in Appendix N, which resulted in lower calculated flow velocities. The difference in flow area definition is illustrated below. The tube array in the SCE replacement steam generators is a triangular array rotated 60 degrees, with a tube pitch of P = 1.0-inch. For that type of array, ASME BPVC, Section III, Appendix N, Figure N-1331-3, defines the tube pitch as the center-to-center distance between two tubes along the same column/row and in the longitudinal direction of the flow, which in this case would be

P = 1.0 inch. However, the flow area defined in FIT-III used the tube pitch in the transverse direction of the flow, which in SCE-rotated triangular array would be P = 1.73 inches. The use of a larger pitch in the FIT-III thermal-hydraulic analysis resulted in nonconservative calculated (lower) flow velocities.



Mitsubishi Document L5-04GA521, “Three-Dimensional Thermal and Hydraulic Analysis,” Revision 3, performed by Mitsubishi and approved by SCE during the design phase of the replacement steam generators, showed that the thermal-hydraulic model was built with two different pitch values. The report stated that the model was built with a 1.0-inch pitch in the longitudinal direction of flow and 1.73-inch pitch in the transverse direction. This analysis report was approved by SCE on April 2, 2008. Additionally, the “Evaluation of Tube Vibration” report by Mitsubishi stated that the thermal-hydraulic conditions for the FIT-III modeling were based on a 1.0-inch pitch in the longitudinal direction of flow and 1.73-inch pitch in the transverse direction. These two design calculations were supporting documents for the “Performance Analysis Report” required in the replacement steam generator design specification. As indicated above, FIT-III’s output for gap velocity results used the 1.73-inch distance instead of the 1.0-inch distance. The FIT-III code was developed by Takasago, MHI’s research and development center. Takasogo was responsible for conducting the thermal-hydraulic analysis, using FIT-III, for each steam generator design. The FIT-III results were then provided to the MHI Steam Generator Design

Department, which input the gap velocity information into the FIVATS vibration code. However, it was not recognized that the gap velocities input into the vibration code were incorrect.

The inspectors determined that the licensee did not ensure that the thermal-hydraulic modeling and flow-induced vibration analysis of the replacement steam generators were adequate with respect to the replacement steam generator design specification. Specifically, the licensee failed to ensure that the design calculations appropriately incorporated the methodology from the ASME BPVC, Section III, Appendix N, standard that was adopted by Mitsubishi for the flow-induced vibration analysis. There were opportunities to identify this error during the early design stage of the replacement steam generators. Licensee personnel questioned the analysis results of FIT-III during design review meetings, but ultimately accepted the model results and resultant design. From shortly after the contract was awarded until 2006, there were letters, e-mails, meeting minutes, action item lists, and internal memoranda that suggested concerns with all three of the elements that cause fluid-elastic instability, which is void fraction, gap velocity, and adequacy of anti-vibration bar tube supports. Regarding concerns raised about FIT-III gap velocities, Mitsubishi compared the velocities to other Mitsubishi designed triangular pitch steam generators that also used FIT-III, but did not compare the results to other similar-sized steam generators.

As a result of the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design, both Unit 3 replacement steam generators experienced fluid-elastic instability in a localized area of the tube bundle leading to rapid, significant, unexpected tube-to-tube wear. The tube degradation progressed to the point of causing a primary-to-secondary leak in Steam Generator 3E0-88 through Tube R106C78. Additionally, from March 13-21, 2012, the licensee conducted in-situ pressure testing of the suspect tubes in both Unit 3 steam generators and identified a total of eight tubes (including the leaking tube) that failed to meet the performance criteria in plant Technical Specifications. The in-situ pressure testing identified that Tubes R106C78, R102C78, R104C78, R100C80, R107C77, R101C81, R98C80, and R99C81 in Steam Generator 3E0-88 failed to meet the structural integrity criterion in Technical Specification 5.5.2.11. In addition to failing the structural integrity criterion, Tubes R106C78, R102C78, and R104C78 also failed to meet the accident-induced leakage criterion in Technical Specification 5.5.2.11.

Southern California Edison completed a review of the tube failures, including conducting a deterministic root cause, an organization and programmatic root cause (still ongoing), three different operational assessments, modification testing, and submittal of a response dated October 3, 2012 (ML12285A263) to the NRC's March 27, 2012, Confirmatory Action Letter (ML 12087A323). The organizational and programmatic root cause evaluation has not been completed as of the issuance of this report, in order to identify the causes of the breakdown in design control such that comprehensive corrective actions can be taken to not only prevent recurrence, but prevent the failures of other important structures, systems, and components that may be subject to the same or similar design problems.

Unit 2:

Analysis. The inspectors determined that the licensee's failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design of the replacement steam generators was a performance deficiency. Criterion III specifies that design control measures shall provide for verifying or checking the adequacy of design, in particular, thermal and hydraulic analyses. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Initiating Event Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that the finding is of very low safety significance because the finding did not involve a degraded steam generator tube that could not sustain three times the normal operating differential pressure and did not violate the accident leakage performance criterion.

The licensee initiated Nuclear Notification NN 202447268 to address this issue in the corrective action program and implement corrective actions to prevent recurrence. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance.

No crosscutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and various crosscutting themes. The NRC determined that the performance behavior that existed at that time is not indicative of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall be established to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods or by the performance of a suitable testing program.

Contrary to the above, on January 28, 2008, and April 2, 2008, SCE failed to verify or check the adequacy of Mitsubishi's developed design Documents L5-04GA504 (SO23-617-1-C157), "Evaluation of Tube Vibration," Revision 3, and L5-04GA521 (SO23-617-1-C683), "Three-Dimensional Thermal and Hydraulic Analysis," Revision 3, respectively, for the flow-induced vibration and thermal-hydraulic designs. Specifically, the output of the thermal-hydraulic code and input to the vibration code were not verified or checked to be in accordance with ASME Section III, Appendix N, "Dynamic Analysis Methods." Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Nuclear Notification NN 202447268, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000361/2012009-01, "Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 2 Replacement Steam Generators."

Unit 3:

Analysis. Regarding Unit 3, this failure also constitutes a performance deficiency for the same reason previously discussed for Unit 2. Specifically, the failure to verify the adequacy of the thermal-hydraulic and flow-induced vibration design resulted in significant and unexpected steam generator tube wear due to fluid-elastic instability, which challenged the structural integrity of the steam generator tubes to perform their pressure boundary function.

The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4 and Appendix A, to evaluate the significance of this finding. In accordance with Exhibit 1 of Inspection Manual Chapter 0609, Appendix A, the inspectors determined that this finding required evaluation in accordance with Inspection Manual Chapter 0609, Appendix J, because the finding involved a degraded steam generator tube condition, where one tube cannot sustain three times the differential pressure across a tube during normal full power, steady-state operation. In accordance with Inspection Manual Chapter 0609, Appendix J, this finding required a detailed risk analysis, since it involved two or more tubes that could not sustain three times the normal differential pressure and one or more steam generators that violated "accident-induced leakage" performance criterion. A Phase 3 analysis was completed using the San Onofre SPAR model, Revision 8.22, assuming average test and maintenance, and a truncation limit of 1.0E-11. Based on the best available information, the performance deficiency was preliminarily characterized as a finding of low to moderate safety significance (White). Refer to Attachment 4 for the detailed Phase 3 analysis.

The licensee initiated Nuclear Notification NN 202447265 to address this issue in the corrective action program. Southern California Edison revised the thermal-hydraulic code of record and ensured that the code was in accordance with ASME guidance.

No crosscutting aspect was assigned because this performance deficiency occurred in the 2005 to 2008 timeframe. Substantial management and personnel changes have occurred, including taking actions to address a chilled work environment and other safety culture issues. The NRC determined that the performance behavior that existed at that time is not indicative of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that design control measures shall be established to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods or by the performance of a suitable testing program.

Technical Specification 5.5.2.11, "Steam Generator Program," Section b, "Performance criteria for SG tube integrity," states, in part, that steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity and accident induced leakage. Technical Specification 5.5.2.11 b.1, “Structural integrity performance criterion,” states, in part, that this includes retaining a safety factor of 3.0 against burst under normal steady-state full power primary-to-secondary differential pressure. Technical Specification 5.5.2.11 b.2, “Accident induced leakage performance criterion,” states, in part, that leakage shall not exceed 0.5 gallons per minute per steam generator for a main steam line break accident.

Contrary to the above, on January 28 and April 2, 2008, SCE failed to verify or check the adequacy of Mitsubishi’s developed design Documents L5-04GA504 (SO23-617-1-C157), “Evaluation of Tube Vibration,” Revision 3, and L5-04GA521 (SO23-617-1-C683), “Three-Dimensional Thermal and Hydraulic Analysis,” Revision 3, respectively, for the flow-induced vibration and thermal-hydraulic designs. Specifically, the output of the thermal-hydraulic code and input to the vibration code were not verified or checked to be in accordance with ASME Section III, Appendix N, “Dynamic Analysis Methods.”

Consequently, the inadequate thermal-hydraulic and flow-induced vibration design resulted in adverse flow conditions, along with insufficient tube support, which caused fluid-elastic instability of a group of tubes in both Unit 3 replacement steam generators. This resulted in one tube leaking and required operator response to rapidly shut down Unit 3 on January 31, 2012. In March 2012, in-situ pressure testing on Unit 3 Steam Generator 3E0-88 confirmed that eight steam generator tubes failed to meet the performance criterion for structural integrity and three of those tubes also failed to meet the accident-induced leakage criterion. During in-situ pressure testing, Tubes R106C78, R102C78, R104C78, R100C80, R107C77, R101C81, R98C80, and R99C81 in Steam Generator 3E0-88 failed to meet the structural integrity criterion limit of three times the normal steady-state primary-to-secondary differential pressure of 5250 psig (room temperature equivalent to 4290 psi under hot 100 percent power conditions), with the tubes failing at test pressures ranging from 2874 psig to 5026 psig (at room temperature). In addition, Tubes R106C78, R102C78, and R104C78 failed to meet the accident-induced leakage criterion of not exceeding 0.5 gpm leakage per steam generator at a main steam line break test pressure of 3200 psig (room temperature equivalent to 2560 psig differential pressure during main steam line break), with each tube having leakage rates of approximately 4.5 gpm, prior to exceeding 3200 psig. Because this finding has been preliminarily determined to be of low-to-moderate safety significance (White), it will be treated as an apparent violation and tracked as AV 05000362/2012009-02, “Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 3 Replacement Steam Generators.”

.6 (Closed) 05000362/2012007-04: "Evaluation of Changes in Dimensional Controls during the Fabrication of Unit 2 and Unit 3 Replacement Steam Generators"

a. Inspection Scope

NRC Inspection Report 05000361/2012007 and 05000362/2012007 described this unresolved item, in part, as follows:

> Based on the information gathered by the team on the differences in dimensional controls of critical parameters in Unit 2 and Unit 3 replacement steam generators, the team determined that Mitsubishi did not consider the potential impact of improving dimensional controls for tube roundness and anti-vibration bars on the final tube bundle clearances at normal operating conditions.

The inspectors reviewed the following reports, which assessed the differences in dimensional controls between SCE Units 2 and 3, and the impact of these differences on the tube-to-anti-vibration bar gap distributions and tube-to-anti-vibration bar contact force distributions throughout the U-bend region and their effect on fluid-elastic instability performance at Units 2 and 3.

- Mitsubishi Document L5-04GA564, "Tube Wear of Unit-3 RSG - Technical Evaluation Report" (Attachment 4 to the "San Onofre Nuclear Generating Station Unit 2 Return to Service Report")
- AREVA Document1814-AU651-MO160, "SCE Unit 2 Cycle 17 Steam Generator Operational Assessment for Tube-to-Tube Wear," Revision 0

The Mitsubishi/AREVA assessment of the dimensional control differences was performed as part of the SCE operational assessment. The inspectors also met with cognizant Mitsubishi and AREVA personnel to discuss this assessment.

b. Observations and Findings

No findings were identified.

The inspectors determined that Mitsubishi relied on industry standards and guidance during the fabrication and design of the replacement steam generators. Dimensional controls for tube roundness and anti-vibration bars were identified by Mitsubishi as conservative on the final tube bundle clearances at normal operating conditions. The inspectors assessed whether the reported changes to the dimensional controls of the upper bundle structure were properly evaluated during the fabrication of the Units 2 and 3 replacement steam generators. These changes were within the design specifications and included manufacturing changes to the anti-vibration bars and steam generator tube, which were meant to reduce the wear between anti-vibration bar and tubes. As documented in NRC Augmented Inspection Team Report 05000361/2012007;362/2012007, the replacement steam generators were required to be designed, fabricated, and tested in accordance with the "Conformed Specification for Design and Fabrication of the Replacement Steam Generators," also known as the design specification, and contained identical technical requirements for Units 2 and 3 steam generators. All replacement steam generators were required to be designed, fabricated, and tested in accordance with the 1998 edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, with the 2000 Addenda, industry standards, and NRC endorsed methods described in applicable regulatory guides. The inspectors determined that the design specifications contained the same requirements for both the Units 2 and 3 replacement steam generators.

Mitsubishi's Document L5-04GA564, "Tube wear of Unit-3 RSG – Technical Evaluation Report," Revision 5, determined that improvement of dimensional controls associated with anti-vibration bar flatness was the key contributing cause in determination of the failure mechanism leading to tube-to-tube wear in Unit 3. The inspectors noted that the specifications also referenced inputs from industry guidance found in Tubular Exchanger Manufacturers Association (TEMA), 8th Edition, 1999. The inspectors noted that the TEMA guidance permitted practical use of engineering principles and field experience in the manufacturing and design of the tube heat exchangers. The general scope of the TEMA standard clearly indicates that damaging tube vibration can occur under certain conditions, including "unsupported tube spans." The replacement steam generator design specification input used the TEMA guidance in determining the maximum unsupported tube spans. In cases where engineering analyses showed the probability of destructive vibration, the designer was directed to analyze key design conditions, including the thermal-hydraulic limitations, baffle design, and tube span. This was necessary to account for and prevent flow-induced vibration damage.

Mitsubishi organizational and programmatic root cause analysis Report UES-20120254, "Root Cause Analysis Report for Tube Wear Identified In The Unit 2 and Unit 3 Steam Generators of San Onofre Nuclear Generating Station," Revision 0, concluded that the replacement steam generators' thermal-hydraulic conditions (void fractions) were high but designers performed feasibility studies to improve them. The report stated that several design adjustments were considered to reduce the adverse thermal-dynamic conditions, but the effects were small. Therefore, the designers concluded that the final design was adequate. The inspectors noted that the Mitsubishi cause evaluation stated that in-plane fluid-elastic instability was a phenomenon that had not been experienced in nuclear U-tube steam generators and that the upper bundle tube-to-anti-vibration bar gaps were within design specifications. However, the root cause identified that tube-to-tube wear was caused by large displacements of tubes in the in-plane direction because of ineffective anti-vibration bar supports.

As identified by the cause evaluation, prior to the design phase of SCE replacement steam generators, the accepted industry U-bend design practice applied flat bar (anti-vibration bar) supports because of significant advantages such as decreased tube-to-anti-vibration bar wear rates. Prior to 2005, it had been an established practice to consider anti-vibration bar or flat bar supports very effective in U-Bend steam generators. However, the inspectors noted that earlier industry studies had documented that flow-induced vibration characteristics were not well understood. In fact, in a 1983 study referenced in the SCE Westinghouse operational assessment titled, "The Effect of Flat Bar Supports on the Cross flow Induced Response of Heat Exchanger U-Tubes," in the Journal of Engineering for Power, October 1983, page 27, it stated the need for additional research, since supports (flat bars) may not adequately restrict in-plane tube motions.

The NRC reviewed the Mitsubishi technical evaluation report on Unit 3 replacement steam generator tube wear and noted that dimensional controls were reportedly improved. This improvement was in standard deviation of Unit 3 anti-vibration bar dimensions. The inspectors evaluated the information, which shows a small improvement in standard deviations associated with only the bending portion of anti-vibration bar thickness change from nominal thickness. The straight bar section standard deviation was not notably better between Units 2 and 3 replacement steam generators. It was noted that differences in anti-vibration bar twist between Units 2 and 3 were determined to be the biggest contributor to differences in the tube bundles. The inspectors also reviewed the results of another comparative analysis based on visual inspection and review of Document L5-04GA564, Appendix 7, "Visual Inspection Results for U-Bend Region for Unit-2/3." The licensee concluded that, even though Unit 3 operated for a much shorter time compared to Unit 2, there were no obvious visual differences, but there were differences in the wear patterns. For example, the visual observations from Units 2 and 3 did not show large gaps between the anti-vibration bars and tube. The anti-vibration bars appeared to be straight, with no detectable abnormalities with weld caps or upper structure orientation. Wear patterns in Unit 2, which had operated the longest, did not have the wear pattern seen in Unit 3 steam generators, which showed evidence of extended tube wear scaring attributed to in-plane motion or vibration, with evidence of orbital tube movement relative to the anti-vibration bar and tube.

The inspectors concluded that Mitsubishi followed its design specifications as required.

.7 Resolution of Independent Technical Review Findings

a. Inspection Scope

The inspectors reviewed the findings prepared by NRC consultants in a report, "Independent Evaluation of San Onofre Nuclear Generating Station (SCE) Steam Generator Tube Wear Problems," dated July 13, 2012 (see Attachment 2 of this report). The consultant's findings were based on their review of documents available at that time, including a draft copy of NRC Augmented Inspection Team Report 05000361/2012007 and 05000362/2012007 that was issued on July 18, 2012.

b. Observations and Findings

No findings were identified.

The NRC consultants made a number of observations during their independent review of the scope and effort of the Augmented Inspection Team. This inspection addressed each observation as follows:

- Consultant observation: Review and assess the inconsistencies between replacement steam generator Design L5-04GA510, “Thermal and Hydraulic Parametric Calculations,” Revision 5, and the revised calculated thermal-hydraulic performance.

  NRC inspection result: The inspectors determined that the steady-state performance code was built based on the design inputs and other requirements provided by the design specification and on the geometrical characteristics and internal components of the Mitsubishi replacement steam generators. The steady-state performance code was run first and it provided all the basic operating parameters and performance information. The analysis provided the main operating parameters, including saturation pressure, circulation ratio, steam flowrate, tube side pressure drop, feedwater pressure at the feedwater inlet nozzle, all individual component pressure losses in the circulation loop, global heat transfer coefficient, and the water and steam inventories. The calculation results are shown in Document L5-04GA510, Table 6.1.5-1. The data at beginning-of-life operating conditions for a T_{hot} of 598^{o}F were used as Cycle 16 operating conditions that were input into the original FIT-III analysis and recent ATHOS base case analysis. NRC staff noted that all the ATHOS models, i.e., the Mitsubishi model, the Westinghouse model, and the NRC independent model, used these steady-state performance code results as input.

  Since the purpose of the steady-state performance code was only to determine the global parameters, it was not necessary that local phenomena, such as subcooled boiling height in the bundle, match with the ATHOS analysis. The performance code results were used as boundary conditions so that the ATHOS analysis could define the local node-by-node thermal-hydraulic conditions that were needed for the vibration analysis. Additionally, in the ATHOS analysis, the subcooled boiling height varied by tube row based on tube temperature and local flow conditions. In conclusion, there were no major issues with this document or the performance results provided.

  The key results of the FIT-III analysis were provided in Document L5-04GA521, Figure 8.1-2a, showing a predicted maximum steam quality of 0.67 and maximum void fraction of 0.95. The conditions postulated considerably underestimated the steam quality and, therefore, also underestimated steam velocities and void fraction.

  The outputs of the FIT-III analysis were used in the vibration analysis of essential components, primarily the tubes. The vibration design was contained in Document L5-04GA504, “Evaluation of Tube Vibration,” Revision 3. In this analysis, the Mitsubishi FIVATS code was used to evaluate the design for tube vibration to justify the number and layout of the tube support plates and anti-vibration bars proposed. The Mitsubishi design methodology used the Connors' Equation to evaluate out-of-plane flow instability using typical ASME suggested design values for K of 2.4 and h of 1.5 percent in the U-bend region.

Table 2-4 of Document L5-04GA504 showed the limiting stability ratio results, assuming one anti-vibration bar support point was inactive, indicating that the maximum expected stability ratio was 0.54. These stability ratio results were considerably low with respect to the ASME acceptance criteria of less than 1.0.

- Consultant observation: The observed difference in standard deviation values for tube diameter on the U-bend flanks (sometimes referred to as "G-value" or "ovality" in the consultant's and Mitsubishi's reports) as reported in Document L5-04GA564, "Tube wear of Unit-3 RSG – Technical Evaluation Report," Revision 2, was considered by the consultants to have minimal effect on any difference in the contact forces at the tube-to-anti-vibration bar supports between Units 2 and 3, particularly if one takes into consideration that the low radius U-bends are the biggest contributor to tube ovality.

  NRC inspection result: The inspectors reviewed the tube diameter data in Document L5-04GA564, Revisions 2 and 9, and concur with this observation for the same reasons cited by the consultants. The inspectors noted that this observation was consistent with the results of the Mitsubishi manufacturing dimensional dispersion analysis in Document L5-04GA564, Revision 9, Appendix 9, which shows that the mean differences in G-values from nominal and differences in G-value standard deviations from the mean between Units 2 and 3 are small compared to the differences in anti-vibration bar twist between Units 2 and 3, which are the dominant contributor to the higher contact forces (tighter tube bundle) being calculated by Mitsubishi for Unit 2 versus Unit 3.

- Consultant observation: Absent the existence of additional information, there is no apparent basis to believe that the number of local radius adjustments during manufacture of the U-bends, as reported in Mitsubishi Document L5-04GA564, Revision 2, has any relevance to the observed steam generator tube degradation.

  NRC inspection result: The inspectors reviewed the U-bend radius adjustment data in Mitsubishi Document L5-04GA564, Revisions 2 and 9, and concur with this observation. Local radius adjustments are sometimes needed to bring the U-bend profiles to within the required specifications. However, the inspectors found that U-bend radius variability did not directly impact tube-to-anti-vibration bar gaps, provided the profile requirements were met. In addition, the inspectors noted that this observation was consistent with the results of the Mitsubishi manufacturing dimensional dispersion analysis in Document L5-04GA564, Appendix 9, Revision 9, which did not consider U-bend radius variability a relevant parameter affecting the tube-to-anti-vibration bar gap and contact force distributions.

- Consultant observation: The average of the gaps between the outermost tubes and the central columns was found to be essentially the same between the Units 2 and 3 steam generators, which does not support a premise that more uniform manufacturing practices for the Unit 3 tube bundles resulted in less contact force between the tubes and anti-vibration bars. In the absence of more dimensional information for the steam generator tube bundles, it is not believed possible to explicitly define the number of active supports in the Units 2 and 3 steam generators.

  NRC inspection result: The inspectors reviewed the gap information (Mitsubishi Document L5-04GA564, Revision 2) cited by the consultants in addition to Revision 9 of the same report. The inspectors concur that the measured gaps between the outermost tubes and the anti-vibration bars in the central columns do not in-and-of-themselves support a premise that more uniform manufacturing practices for the Unit 3 tube bundles resulted in less contact force between the tubes and anti-vibration bars. The inspectors noted, however, that conclusions by SCE relating to the number of active supports in the Units 2 and 3 steam generators are based on analyses documented in Mitsubishi Document L5-04GA564, Revision 9, Appendix 9, and in SCE Document 1814-AU651-MO160, “SONGS Unit 2 Cycle 17 Steam Generator Operational Assessment for Tube-to-Tube Wear,” Revision 0, prepared by AREVA. These analyses were not reviewed by the consultants, but were reviewed as part of the NRR technical evaluation and by the inspectors; however, the NRC review was not completed because of the decision by SCE to permanently cease operation of Units 2 and 3.

- Consultant observation: Eddy current test inspection measurements of the tube-to-anti-vibration bar gap were determined to be of questionable value in an assessment of likely tube wear behavior. Review of Figure 4.1.2-1 in Mitsubishi Document L5-04 GA564, Revision 2, indicated the potential fallacy in projecting differences in average contact forces (at tube-to-anti-vibration bar intersections) between Units 2 and 3. Specifically, Figure 4.1.2-1 shows virtually identical average absolute signal amplitudes at anti-vibration bar locations for Steam Generators 2E0-88 and 3E0-89 that have shown significant differences in operational tube wear behavior.

  NRC inspection result: The inspectors reviewed the Mitsubishi data in both Revisions 2 and 9 of L5-04GA564 and concur that this data lends little insight as to the significantly different amounts of tube-to-tube wear observed between Units 2 and 3. The Mitsubishi signal amplitude data is based on bobbin probe data. Since the time of the consultant review, AREVA conducted extensive gap measurements using eddy current pancake and ultrasonic techniques expected by AREVA to provide a more accurate and comprehensive gap assessment. The purpose of these measurements was to determine if there were highly heterogeneous spatial distributions of large gaps among the steam generators that might affect the results of the AREVA operability assessment.

- Consultant observation: Review the potential cause for low flow velocities. There is a region of almost stagnant flow, possibly caused by higher flow resistance for the cross-flow from the wrapper inlet ports into the tube bundle due to the smaller pitch-to-diameter ratio of the replacement steam generators than in the original steam generators. A comparison of the replacement steam generator thermal-hydraulics with that of the original steam generator was not found in either the Augmented Inspection Team or SCE root cause reports, which could aid in the determination of the cause for the flow abnormalities in the replacement steam generator.

  NRC inspection result: The inspectors determined that flow in the downcomer was restricted by protrusions of inspection ports, wrapper supports, and tube support plate anti-rotation blocks. At the bottom of the downcomer (top of tubesheet), flow is forced to make a 90-degree turn and is directed into the tube bundle. This flow is affected by the height of the wrapper opening, size of the open lane, and pitch-to-diameter ratio.

  As the flow continues in the bundle, it slows considerably as it begins upflow in the bundle. There are some places on top of the tubesheet that should be evaluated for low flow and for potential sludge accumulation. It is not uncommon to see areas of low flow, so modern replacement steam generators include features for 1-2 percent blowdown flow and enhanced sludge lancing. There also are slightly more tubes in the replacement steam generator, but not enough to have any effect on sludge potential or any flow concerns in the first span.

  There is no thermal-hydraulic model associated with the original steam generators, since these were designed prior to the development of the ATHOS model in 1985; therefore, the inspectors were not able to compare the flows.

- Consultant observation: The correlation between boiling in a small region at the bottom of the tube bundle, based on Mitsubishi document calculations, and the observed region of tube wear increasing from tube support Plate 1 levels upward into the U-bend region, has apparently not been addressed. Some mechanism is moving the tubes and causing the tube-to-tube support plate wear in a small region.

  NRC inspection result: There is potential of subcooled nucleate boiling in lower tube support plates in lower rows on the hot side of the steam generators. The effects of this nucleation are usually not significant, so it is quite often not considered during design of steam generators. Steam generators with lower recirculation ratios, like the replacement steam generators, are more susceptible because the tube bundle contains a larger percentage of steam than other steam generator designs; however, the likelihood of this causing tube movement would be considered inconsequential. It should be noted that the pitch-to-diameter ratio in the vertical section of the bundle is the same (1.33) in the replacement steam generator and original steam generator. However, in the U-bend area, incrementation (indexing) was used after row 72, which resulted in an increased pitch-to-diameter ratio of up to a maximum of approximately 1.53 for the outermost tube row.

- Consultant observation: Review the simplified scenario of Section 6.3 and make a determination as to the validity. From the report, “More evaluations would be required to substantiate the postulated scenario as the source of the high void fraction and velocity in a specific U-bend region.”

  NRC inspection result: The inspectors developed their own independent ATHOS thermal-hydraulic model and reviewed three other thermal-hydraulic models, all yielding relatively similar results. The codes used include the EPRI ATHOS code, Westinghouse modified ATHOS code, and French CAFCA4 code. Each of the codes used homogeneous methods with empirical models for heat and mass transfer and drift flux models to compute two-phase flow conditions. The code methods date to the 1970’s and 1980’s, but they have been widely used and benchmarked to available scaled and full-plant data. Additionally, the ATHOS and CAFCA4 code results have been successfully used to design many replacement steam generators.

  Modeling two-phase flows is very complex since it exhibits various flow regimes, or flow patterns, depending on the void fraction of the two-phase fluid and the flow rate. Additionally, flow patterns can be irregular or chaotic. However, averaged behavior based on conservation equations can be used to model one-dimensional steady-state and transient two-phase flow in reactors and steam generators. Some simple transients are relatively easy to model and can be validated with data while others are more difficult and data to validate the results are scarce. Three-dimensional analysis with axial, radial, and tangential control volumes adds additional complexities of momentum equations for the added directions. The three-dimensional code uses a porous media approach to represent local geometries and requires customized pre-processors to model modern designs of anti-vibration bars and tube support plates.

  Westinghouse maintains its own version of ATHOS and has completed the most comprehensive validation of the code. To support plant restart, Mitsubishi was directed by SCE to use the ATHOS, with their analyses being reviewed by AREVA and other SCE consultants. The NRC reviewed the benchmarking and validation of the ATHOS code to actual steam generator conditions as follows:

  The Westinghouse validation suite includes:

  (1) Model Boiler Number 2 (MB-2) one percent power-scaled model of the Westinghouse Model F steam generator, designed to be geometrically and thermal-hydraulically similar to the Model F, and capable of generating 10 MWt of power. The model was able to produce dry saturated steam at 1000 psia, the same as with Model F.

(2) Full-scale steam generator data measured at Electricity of France (EDF) nuclear power plants Bugey 4 and Tricastin 1 (Westinghouse Model 51A and 51M) operated at full and reduced power levels.

(3) EPRI full-scale steam generator test data collected from operating Westinghouse Model F and D4 steam generators.

(4) French Alternative Energies and Atomic Energy Commission (CEA) Clotaire scaled test program with the main focus on thermal-hydraulic data collected on the secondary side fluid void fraction and axial vapor velocity throughout the tube bundle. This data had not been previously verified. Nine organizations from six countries participated in this program to verify six different three-dimensional thermal-hydraulic codes developed for pressurized steam generator analysis. Westinghouse participated to verify the ATHOS code.

(5) Leonard Cold Flow Test using a scaled down model of the U-bend region of a bundle similar to the Westinghouse Model 51 design with three test configurations: (a) no anti-vibration bars, (b) two sets of anti-vibration bars, and (c) three sets of anti-vibration bars. Experiments were conducted in each configuration to obtain velocity distributions along the U-bends.

From these test cases, Westinghouse concluded that the ATHOS code calculates thermal-hydraulic parameters and behavior in good agreement with measured data.

These experimental data and test parameter matrices appear to adequately encompass the regions and conditions of concern for Cycle 16 operation of SCE replacement steam generators. The code has wide use and acceptance in the nuclear industry for analysis of recirculating-type steam generators. As with any thermal-hydraulic code, the accuracy of the results is a function of the code and the ability of the user to correctly model and interpret the output. The inspectors believe that ATHOS code modeling is sufficiently capable of representing these thermal-hydraulic conditions, and the users were sufficiently proficient in building the input models.

The results of several models were reviewed to establish a level of confidence in the key ATHOS output for vibration analysis for two tubes in the tube-to-tube wear affected region of the bundle. Results reviewed include the NRC independent ATHOS thermal-hydraulic analysis, the Mitsubishi ATHOS thermal-hydraulic analysis, the Westinghouse in-house ATHOS60 thermal-hydraulic analysis, and the AREVA CAFCA4 thermal-hydraulic analysis. Comparisons are shown for gap velocity, void fraction, and fluid density along the U-bend of the subject tubes at 100 percent power operation. As indicated, the major differences in the results are located on the hot side of the U-bend, where the tubes are hotter and there is more bulk boiling. Additionally, the wrapper transition opening is located in this region so there is an increase flow area and, consequently, increases in fluid velocities along the entire periphery.

The NRC and Mitsubishi results show the higher peak void fractions, which also peak at a lower location along the U-bend as compared to the Westinghouse and AREVA results. At higher angles along the U-bend, the void fraction predicted by each code tends to merge, and the remainder along the cold side tends to show very similar trends. As expected, comparisons of fluid density show similar trends as the void fraction. Dry saturated steam at about 850 psia has a density of about 2.0 lb_m/ft^3, and the NRC, Mitsubishi, and Westinghouse results predict that there are significant areas in the U-bend, in the affected region, where velocities are high and steam is nearly dry. Peak velocities and void fraction support the tube-to-tube wear patterns found in the Unit 3 steam generators. The inspectors concluded that the ATHOS code models are adequate to predict the approximate location and magnitude of the high void fraction and velocity, which are the key contributors to tube-to-tube wear found in the U-bend region.

Figure 4: U-Bend Gap Velocities

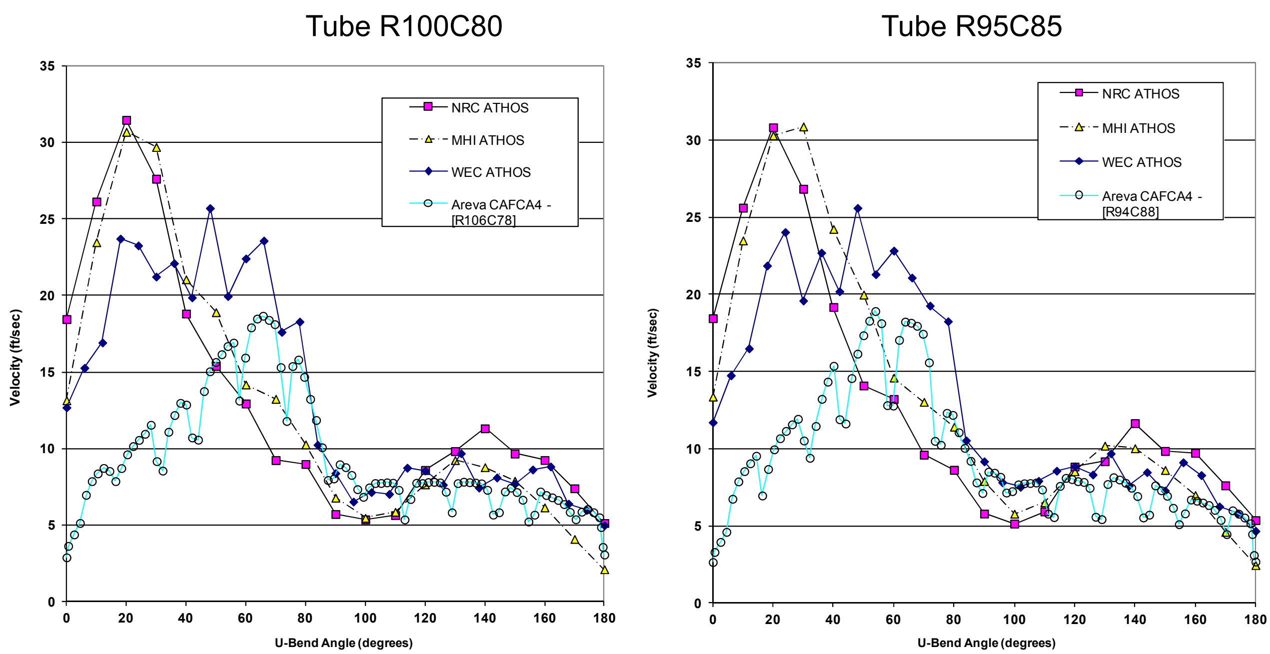
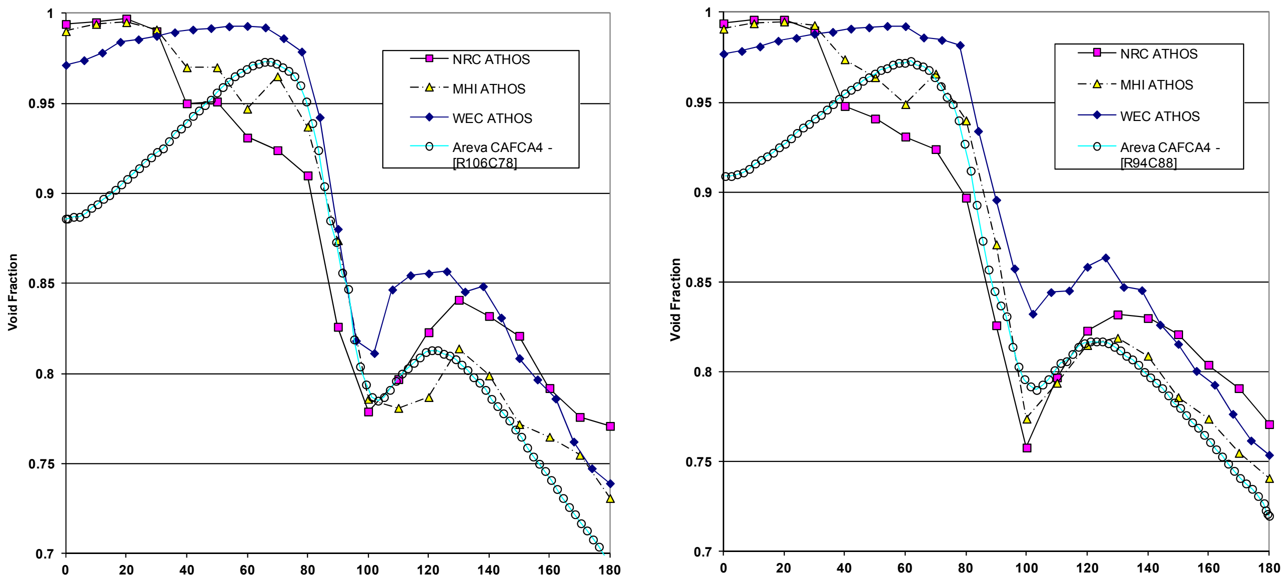


Figure 5: U-Bend Void Fraction

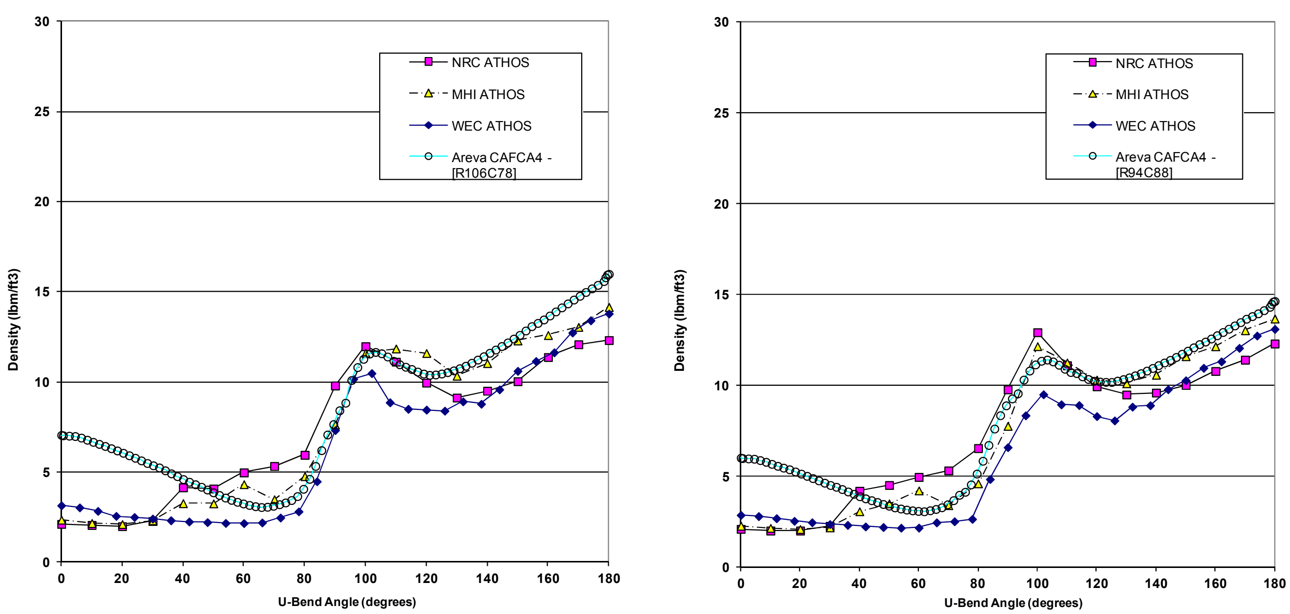
Tube R100C80 Tube R95C85



F

Figure 6: U-Bend Density

Tube R100C80 Tube R95C85



.8 Chalk River Testing

The purpose of the Chalk River testing was to support the SCE Unit 2 return-to-service steam generator operational assessment analyses for Unit 2 as related to the Unit 3 tube leak and loss of tube integrity due to tube-to-tube wear in the upper bundle. One of the analytical methods contained in a number of the operational assessments used an empirical damping correlation, called squeeze film damping, based on testing done in 1988 for a tube and drilled support plate arrangement. During the NRC's review of the operational assessments and their application of squeeze film damping, the inspectors had a number of questions related to the applicability of the data since the 1988 testing configuration was significantly different from the flat-bars (anti-vibration bars), and the test range did not include lower frequencies where this information was being applied.

Atomic Energy Canada Limited performed a series of tests with a straight steam generator tube under a variety of support and excitation conditions, including the type of support, tube-to-support gaps, excitation levels, tube-to-support impacts, and tube-to-support preload. The tests were performed in air and still water at four different tube lengths corresponding to vibration frequencies in water between 4.5 Hz and 32 Hz. The primary objective of the tests were to obtain damping data at low frequency (<20 Hz) with flat bar (anti-vibration bar type) supports.

a. Inspection Scope

The inspectors reviewed design specifications of the test rig, test rig setup, calculations, procedures, and comparison of original test rig data to revised test rig data. In some instances, the inspectors performed independent calculations to verify the testing results. The inspectors verified that the condition of the components was consistent with the design; verified that the test measuring devices were appropriately calibrated and of the correct range; reviewed equipment dedication; observed a number of actual tests, with varying configurations; and independently verified that the test results were properly documented in accordance with the test procedures. In addition, the inspectors reviewed maintenance work records and corrective action documents associated with the test rig and data acquisition system.

b. Description

From March 11-14, 2013, two NRC inspectors, along with one contractor, observed a portion of Phase 1 and Phase 2 testing for squeeze film damping effects on a pair of flat anti-vibration bars. Tests were conducted in air and water, with tube frequency being varied. The test rig did allow for the adjustment of the simulated anti-vibration bar. The tube was excited by a pair of electromagnetic coils with three eddy current proximity probes used to measure tube vibration.

For a more detailed description of the inspection and testing performed at Chalk River, refer to Attachments 3 and 4.

c. Findings

No findings were identified.

The inspectors did concur with the testing results that showed squeeze film damping was negligible and did not contribute to the damping values assumed in the operational assessments. At the time of the plant permanent shutdown announcement, Southern California Edison had not completed revision of their return to service plan operational assessments to account for these results. Therefore, the inspectors did not complete an inspection of the affect of these results on the return to service plan.

4OA6 Meetings

Exit Meeting Summary

On August 28, 2013, the inspectors presented the inspection results to Mr. P. Dietrich, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. Proprietary information was provided to the inspectors.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Dietrich, Senior Vice President and Chief Nuclear Officer
D. Bauder, Site Vice President
T. Palmisano, Vice President of Engineering, Projects and Site Support
B. Sholler, Director, Maintenance
O. Flores, Director, Nuclear Oversight
R. St. Onge, Director, Regulatory Affairs/Emergency Planning
E. Avella, Director, Project Management Organization
R. Davis, Director, Training
J. Madigan, Director, Safety Culture
C. McAndrews, Director, Special Projects
R.Treadway, Manager, Nuclear Regulatory Affairs
A. Martinez, Manager, Chemistry
L. Mosher, Manager, Communications
K. Yhip, Technical Advisor
M. Brown, Project Manager, Communications
R. Swanson, Consultant
K. Gallion, Manager, Focus Assessments/Performance Improvements
M. Pawlaczyk, Technical Specialist, Regulatory

NRC Personnel

D. Beaulieu, NRR, Project Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000361/2012009-01	NCV	Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 2 Replacement Steam Generators

Opened

05000362/2012009-02	AV	Failure to Verify Adequacy of Thermal-Hydraulic and Flow-Induced Vibration Design for the Unit 3 Replacement Steam Generators

Closed

05000362/2012-002	LER	Unit 3 Steam Generator Tube Degradation Indicated by Failed In-Situ Pressure Testing

05000362/2012007-04	URI	Evaluation of Changes in Dimensional Controls during the Fabrication of Units 2 and 3 Replacement Steam Generators
05000362/2012007-08	URI	Nonconservative Thermal-Hydraulic Model Results

LIST OF DOCUMENTS REVIEWED

DESIGN BASIS DOCUMENT	TITLE	REVISION
DBD-SO23-360	Reactor Coolant System	10, 11

DRAWING	TITLE	REVISION
L5-04FU001	Design Drawing – Component and Outline Drawing	6
L5-04FU011	Design Drawing – Channel Head 1/4	14
L5-04FU016	Design Drawing – Divider Plate 1/2	12
L5-04FU021	Design Drawing – Tubesheet and Extension Ring 1/3	16
L5-04FU031	Design Drawing – Lower Shell, Middle Shell, Transition Cone, and Upper Shell ¼	6
L5-04FU041	Design Drawing – Upper Shell, Upper Head Ring, and Upper Head Top ¼	6
L5-04FU051	Design Drawing - Tube Bundle 1/3	1
L5-04FU052	Design Drawing -Tube Bundle 2/3	1
L5-04FU053	Design Drawing - Tube Bundle 3/3	3
L5-04FU054	Design Drawing -Tubing Expansion and Seal Welding	2
L5-04FU101	Design Drawing - Wrapper Assembly 1/5	5
L5-04FU106	Design Drawing -Tube Support Plate Assembly 1/3	3
L5-04FU107	Design Drawing -Tube Support Plate Assembly 2/3	3
L5-04FU108	Design Drawing - Tube Support Plate Assembly 3/3	3
L5-04FX001	Fabrication Drawing, General Shipping Arrangement [SON-2A (2E089), SON-2B (2E088)]	4
L5-04FX002	Fabrication Drawing, General Shipping Arrangement [SON-3A (3E089), SON-3B (3E088)]	5

DOCUMENT	TITLE	REVISION/DATE
UGNR-SON2-RSG-067(7)	Nonconformance Report: 2563901/G101, “Unacceptable gaps between Tubes and AVBs.”	January 23, 2007
UGNR-SON3-RSG-030	Mitsubishi Nonconformance Report – “Some Gaps between Tubes and AVBs are larger than the criterion”	0
SON-3A(3E089)-1, -2, -3, -4	[Sumitomo] G-Values	September and October 2007
SON-3B(3E088)-1, -2	[Sumitomo] G-Values	December 2007
SON-2A(2E089)-1, -2, -3,	[Sumitomo] G-Values	October and November 2006
SON-2B(2E088)	[Sumitomo] G-Values	December 2006
KAS-20130179	SONGS U-tube damping measurement test for comparing with the test result of AECL: Test plan	0
KAS-20040251	FIVATS (Fluid Elastic Vibration Analysis Code) Code Description Note (User’s Manual)	1
KAS-20040252	FIVATS Code Validation and Qualification Plan	3
KAS-20040253	FIVATS Code Validation and Qualification Report	3,4
N/A	Damping Measurement Test Procedure	1
N/A	Sample Calculation of Stability Ratio	February 1, 2013
N/A	Evaluation of Liquid Film Thickness of Tube at AVB Support Point	January 23, 2013
N/A	The procedure how frequencies are computed by FIVATS	January 17, 2013
L5-04GA571	Screening Criteria for Susceptibility to In-Plane Tube Motion	4
L5-04GA564	Tube Wear of Unit-3 RSG – Technical Evaluation Report	2
L5-04GA428	Design of Anti-Vibration Bar	5
L5-04GA504	Evaluation of Tube Vibration	3,4
L5-04GA102	Nitrogen Plenum / Accelerometer Data Report for Unit 3	1
L5-04GA224	Material Selection Report for Anti-Vibration Bar	2
L5-04GA521	Three-Dimensional Thermal and Hydraulic Analysis (FIT-III Code Analysis)	3

DOCUMENT	TITLE	REVISION/DATE
L5-04GA591	Validity of Use of the FIT-III Results during Design	1,3
UES-2010254	Root Cause Analysis Report for tube wear identified in the Units 2 and 3 Steam Generators of San Onofre Nuclear Generating Station	0
N/A	SGTL Unit 2 Mode 4 Entry JITT [S/G Tube Leak Unit 2 Mode 4 Entry Just in Time Training]	December 2012
N/A	JITT for SGTR QUIZ [Just in Time Training for S/G Tube Rupture Quiz]	December 2012
N/A	Plant Changes - Unit 2 Return to Service information package, Lesson Plan 2RP548, Attachment 9.2	December 2012
N/A	Scenario Title: Mode 1 and 2 SGTL and SGTR	May 18, 2012
N/A	SCE Purchase Order 4500555142 with CANDU Energy, Inc.	January 28, 2013
N/A	SCE Change Order 1 to PO 4500555142	March 7, 2013
N/A	SONGS NOD memo to file, SUBJECT: LEAD AUDITOR ANNUAL EVALUATION	December 11, 2012
N/A	Commercial grade item survey report, Survey Report AECL-CS1-13	February 18-22, 2013
N/A	Letter to Elmo E. Collins from Southern California Edison; Docket 50-361, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generation Station Unit 2	October 3, 2012
N/A	San Onofre Nuclear Generating Station Unit 2 Return to Service Report	October 3, 2012
N/A	SONGS U2C17 Steam Generator Operational Assessment	October 3, 2012
1814-AU651-MO157	SONGS Unit 2 Cycle 17 Steam Generator Operational Assessment (AREVA)	0
1814-AU651-MO160	SONGS Unit 2 Cycle 17 Steam Generator Operational Assessment for Tube-to-Tube Wear (AREVA)	0
1814-AU651-MO145	Operational Assessment for SONGS Unit 2 SG for Tube-to-Tube Wear Degradation at the End of Cycle 16 (Intertek)	1

DOCUMENT	TITLE	REVISION/DATE
1814-AA086-M0190	Operational Assessment of Wear Indications in the U-Bend Region of San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators (Westinghouse)	4
SO23-617-01	Specification for Design and Fabrication of the Replacement Steam Generators for Unit 2 and Unit 3	4
1814-AV651-M0165	SONGS Unit 2, Steam Generator Internal Impact Test Results	0
1814-AU651-MO151	SONGS U2C17 and U3F16B AVB Gap and Tube-to-Tube Proximity Measurement Program	0
LTR-SGDA-12-36	Flow-induced Vibration and Tube Wear Analysis of the San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators Supporting Restart	1
SG-SGMP-12-10	Operational Assessment of Wear Indications in the U-bend Region of San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators	3
NECP 800873488-132	50.59 Screen 70% Power Operation	December 19, 2012
NECP 800873488-124	50.59 Evaluation 70% Power Operation	October 15, 2012
NECP 800901029-0041	50.59 Screen 70% Power Annunciator	September 14, 2012
NECP 800867185-0021	50.59 Screen Argon Injection into RCS	November 19, 2012
NECP 800905312-0270	50.59 Screen Nitrogen-16 Monitors	December 3, 2012
NECP 800698429-0140	50.59 Screen U2C17 Reload ECP	August 30, 2012
NECP 800698429-0150	50.59 Screen for TR-PL Correction	July 30, 2012
NECP 800873488-0130	50.59 Screen Plugging and Stabilization	December 6, 2012
NECP 800457837-0550	50.59 Screen Vibration and Loose Parts Monitoring System	January 14, 2013

DOCUMENT	TITLE	REVISION/DATE
NN 201843216	RCE Steam Generator Tube Wear, San Onofre Nuclear Generating Station, Unit 2	April 23, 2012
NECP 800905312-0270	50.59 Screen Nitrogen-16 Monitors	December 3, 2012
NECP 800698429-0140	50.59 Screen U2C17 Reload ECP	August 30, 2012

MISCELLANEOUS DOCUMENT	TITLE	REVISION/DATE
TEMA	Standards of The Tubular Exchanger Manufacturer Associations (TEMA), 8th Edition	1999
N/A	Steam Generator RTS Schedule AECL Testing	February 19, 2013
1814-AD799-M0001	AECL EACL Test Plan, Measurement of Steam-Generator Damping Due to Anti-Vibration Bar Supports	1
51-9198780-000	[AREVA] Distribution of Indications for SONGS 2 and 3 – 2012 Inspections	February 27, 2013
51-9182205-000	[AREVA] SONGS Unit 3 2012 Forced Outage (U3F16B) Technical Summary Steam Generator Eddy Current Inspection	May 23, 2012
51- 9188725-001	[AREVA] SONGS U2C17 and U3F16B AVB Gap and Tube-to-Tube Proximity Measurement Program	November 16, 2012
N/A	AECL Damping Tests for SONGS (PowerPoint Presentation)	March 8, 2013
Proceedings of the Institution of Mechanical Engineers, Volume 184, Part 1, No. 36	Void Fractions In Two-Phase Flow: A Correlation Based Upon An Equal Velocity Head Model	1969
Journal of Mechanical Design, Volume 100	Fluidelastic Vibration of Heat Exchanger Tube Arrays	April 1978
10th International Conference on Flow-Induced Vibration (& Flow-Induced Noise)	Study on In-flow Fluid-elastic Instability of Circular Cylinder Arrays	2012

MISCELLANEOUS DOCUMENT	TITLE	REVISION/DATE
Journal of Vibration, Acoustics, Stress, and Reliability in Design, Volume 105	The Effect of Approach of Flow Direction on the Flow-Induced Vibration of a Triangular Tube Array	January 1983
Journal of Pressure Vessel Technology, Volume 126	Damping of Heat Exchanger Tubes in Two-Phase Flow: Review and Design Guidelines	November 2004
Journal of Pressure Vessel Technology, Volume 128	Fluidelastic Instability of an Array of Tubes Preferentially Flexible in the Flow Direction Subjected to Two-Phase Cross Flow	February 2006
Journal of Pressure Vessel Technology, Volume 127	Fluidelastic Instability and Work-Rate Measurements of Steam-Generator U-Tubes in Air–Water Cross-Flow	February 2005
IMECE2002-32707	Vibration Analysis of Steam Generators and Heat Exchangers: An Overview Part 1: Flow, Damping, Fluidelastic Instability	November 2002
Journal of Applied Mechanics, Volume 47	Fluid Forces on Rods Vibrating in Finite Length Annular Regions	June 1980
Journal of Pressure Vessel Technology, Volume 133	Damping of Heat Exchanger Tubes in Liquids: Review and Design Guidelines	February 2011
Journal of Engineering for Power, Volume 105	The Effect of Flat Bar Supports on the Crossflow Induced Response of Heat Exchanger U-Tubes	October 1983
Nuclear Technology, Volume 55	Flow-Induced Vibration and Wear of Steam Generator Tubes	November 1981
Journal of Pressure Vessel Technology, Volume 117	Vibration of a Tube Bundle in Two-Phase Freon Cross-Flow	November 1995
Nuclear Insights	Fluid Elastic Instability Causing Tube Damage in Main Steam Condensers of Nuclear Power Plants	Spring 2009
Nuclear Engineering and Technology, Volume 38, No. 1	Fluid-Elastic Instability of Rotated Square Tube Array in an Air-Water Two-Phase Cross-Flow	February 2006
R&D 13-2259	Rust on AVB material	March 12, 2013
R&D 13-2279	Free span length 2.993 m vs. 3.000 m in test plan	March 12, 2013

MISCELLANEOUS DOCUMENT	TITLE	REVISION/DATE
R&D 13-2278	Temperature recording – air temperature not recorded	March 12, 2013
R&D 13-2277	Circular hole support plate thickness	March 12, 2013
R&D 13-2372	Force transducer	March 14, 2013
R&D 153-127370-TP-001	AECL/EACL Test Plan, “Measurement of Steam Generator Tube Dampening Due to Antivibration Bar Supports”	D1, D2, and O
CW-510200-PRO-344	AECL/EACL procedure: “Assessment and Identification of Personnel Qualifications and Training Needs”	2
CW-510100-FM-164	AECL/EACL Position Description, Senior R&D Engineer	4
N/A	AECL/EACL student training history, TRAK printout (4 students/employees)	1999 to 2013
CW-510200-PRO-498	AECL/EACL employee training, “Approving, Attending, and Reading”	1
CW-510200-FM-240	AECL/EACL Personnel Qualifications and Training Needs (4 students)	4
N/A	AECL/EACL Technical Document 0987-00, “Test Instruction for Straight-Tube Dampening Tests for Southern California Edison”	0, 5
N/A	AECL/EACL Technical Document 0987-00, “Test Preparation Instructions for Straight-Tube	0, 5
153-127370-TR-002	[AECL] Measurement of Steam-Generator Tube Damping Due to Anti-Vibration Bar Supports	0
1022832	Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines	4
TR-016743-V2R1	EPRI Guidelines for PWR Steam Generator Tubing Specifications and Repair, Volume 2	1
TR-103824s-V1R1	EPRI Steam Generator Reference Book, Volume 1	1
LTR-SGMP-12-70	Distributions of Gap Velocities, Void Fractions and Fluid Densities along Selected Tubes of the SONGS Replacement Steam Generators	December 4, 2012
12 -9176741-002	[AREVA] Technical Data Record for Stabilizer Design	April 27, 2012

MISCELLANEOUS DOCUMENT	TITLE	REVISION/DATE
12 -9176741-005	[AREVA] Technical Data Record for Stabilizer Design	October 16, 2012
N/A	Advisory Committee on Reactor Safeguards Materials, Metallurgy and Reactor Fuels Steam Generator Action Plan	September 24, 2009
A-SONGS-9416-1168	Evaluation of Southern California Edison Songs Unit 3 Steam Generators with Degraded Eggcrates	1
LTR-SGDA-12-50	ATHOS Computer Code Verification & Validation Summary Report for SONGS Unit 2 RSG Recovery Project	September 14, 2012
90202	Comparison Of FIV Structural Models	0

NUCLEAR NOTIFICATIONS

200525719	201937413	202243314	201836127	201969741	201836127
201279476	201969131	201907105	201843216	201979105	201843216

PROCEDURE	TITLE	REVISION
SO23-12-4	Steam Generator Tube Rupture	23, 24
SO23-5-1.7	Power Operations	54
SO23-3-2.21	Core Operating Limits Supervisory Limits (COLSS)	29
SO23-15-50.A2	ARP Annunciator Panel 50A, PZR/CEA Window 31-60	19
SO23-XXXVII-1.20	RCS Calorimetric Flow Measurements	6
SO23-3-3.25	Once a Shift Surveillance (Mode 1-4)	37
SO23-3-2.1	Operation of Pressurizer Degas System	39
SO23-15-50.A1	ARP Annunciator Panel 50A	13
SO123-III-2.22.23	Unit 2/3 Steam Generator Tube Leakage Monitoring Program	25
SO23-13-14	Reactor Coolant Leak	16, 17, 21
SO23-3-2.1.1	CVCS Alignment	16, 18, 20
SO23-3-2.24 ISS 2	Radiation Monitoring System Guidelines and RDU Operation	14
SO23-3-2.1	CVCS Operation	40

PROCEDURE	TITLE	REVISION
SO123-III-1.1.23	Unit 2/3 Chemical Control of Primary Plant and Related Systems	59, 60
SO23-3-3.37	Reactor Coolant System Water Inventory Balance	35
SO123-XII-2.19	Qualification and Certification of Auditing Personnel	8
SO123-XII-2.20	Certification Documentation Processing for Personnel Involved in Audits, Inspections, and Examinations	10

Independent Evaluation Of San Onofre Nuclear Generating Station (SONGS) Steam Generator Tube Wear Problems

Ian Barnes, Beckman & Associates
Hans Garkisch, Beckman & Associates
July 13, 2012

1. INTRODUCTION

An Augmented Inspection Team (AIT) was approved by the Nuclear Regulatory Commission (NRC) on March 16, 2012, to assess the facts and circumstances surrounding the occurrence of a tube leak and detection of unexpected tube wear in SCE Unit 3 steam generators (SGs). The AIT was established in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," and implemented using Inspection Procedure 93800, "Augmented Inspection Team."

In addition to performance of the AIT inspection, the NRC requested performance of this independent evaluation, with a primary focus to be identification of any perceived gaps in the response actions taken by the NRC or licensee.

2. BACKGROUND, EVENT DESCRIPTION, AND INSPECTION RESPONSE

Replacement steam generators for SCE Units 2 and 3 were designed and fabricated by Mitsubishi (MHI), with Unit 2 return to power after installation occurring on April 13, 2010 and Unit 3 on February 18, 2011. The replacement steam generators were designed to be like-for-like replacements for the original steam generators, but included major improvements such as: (a) replacement of carbon steel egg crate design tube support plates (tube support plates) with Type 405 ferritic stainless steel tube support plates using trefoil hole configurations, and (b) use of the significantly more corrosion resistant thermally treated Alloy 690 tubing in place of the original mill annealed Alloy 600 tubing.

Unit 2 was shut down for a scheduled refueling outage on January 10, 2012. During the required inservice inspection (ISI) activities, unexpected steam generator tube wear was detected. Specifically, pluggable tube wear was found in: (a) two steam generator 2E0-89 tubes that had been caused by retainer bars, and (b) four steam generator 2E0-88 tubes that had been caused by anti-vibration bars (anti-vibration bars) for two tubes and by retainer bars for two tubes. As a result of the unexpected wear, the licensee preventatively plugged an additional 92 tubes in steam generator 2E0-89 and an additional 94 tubes in steam generator 2E0-88.

During the Unit 2 refueling outage, the Unit 3 control room received an alarm on January 31, 2012 from the main condenser air ejector radiation monitors that indicated a primary-to-secondary tube leak in steam generator 3E0-88. The initial estimated leak rate was 75 gallons per day (gpd) and was increasing, versus a facility license requirement of less than 150 gpd steady state leak rate. Shutdown was initiated with cold shutdown reached on February 2, 2012. Subsequent inspection identified that the location of the leak in steam generator 3E0-88 was the tube in the Row 106 Column 78 (R106C78) location, with no other tubes found to be leaking. Subsequent eddy current inspections of all of the tubes in the Unit 3 steam generators discovered unexpected wear in both replacement steam generators, including significant tube-to-tube wear in the freespan areas of the U-bends and tube wear at retainer bars similar to that identified in the Unit 2 steam generators. Condition monitoring requirements were not met for 129 tubes (73 tubes in steam generator 3E0-88, 56 tubes in steam generator 3E0-89), which required in-situ pressure testing of the tubes. Eight steam generator 3E0-88 tubes failed the in-situ pressure test. As a result of the wear findings in the Unit 3 steam generators, additional eddy current testing inspections were performed of the Unit 2 steam generators, which included use of the more sensitive +Point rotating probe. The area included all rows above R79 between columns 70-110.

Table 1 shows summary information regarding the extent of measurable tube wear that was detected by eddy current testing inspections of the Unit 2 and Unit 3 steam generators. The most significant difference between the units is the extent of detected tube-to-tube wear (tube-to-tube wear) in the Unit 3 steam generators versus that found in the Unit 2 steam generators. Of the 326 affected Unit 3 steam generator tubes exhibiting tube-to-tube wear, eddy current testing inspection identified a total of 202 to have pluggable ≥ 35 percent through-wall depth (TWD). Neither of the two affected Unit 2 steam generator tubes showing tube-to-tube wear were found to have ≥ 35 percent TWD, but were preventatively plugged and stabilized. A significant incidence of tube wear was detected in both units at the anti-vibration bars and tube support plates. From review of Root Cause Evaluation (RCE) 201843216, it was ascertained that eddy current testing inspection identified at tube support plate locations a total of 230 tubes in the Unit 3 steam generators with pluggable ≥ 35 percent TWD versus 0 tubes with ≥ 35percent TWD in the Unit 2 steam generators. The Unit 3 steam generator tube degradation occurred in slightly less than one year of power operations compared to a full cycle for Unit 2.

Table 1

Steam Generator Extent of Condition		
Tube Degradation Type/Location	Number of Unit 2 Affected Tubes	Number of Unit 3 Affected Tubes
Tube-to-Tube Wear	2	326
Wear at Anti-Vibration Bars	1399	1767
Wear at Tube Support Plates	299	463
Wear at Retainer Bars	6	4
Wear at Foreign Objects	2	0

Unit 2 and Unit 3 steam generators were found by eddy current testing inspection to both contain a total of two tubes with pluggable ≥35 percent TWD at anti-vibration bar locations. Eddy current testing inspection wear depth measurements with reasonable credibility (i.e., 20 percent TWD or greater) did not indicate any particular difference between Unit 2 and Unit 3 steam generators regarding tube wear at anti-vibration bars. Specifically, the respective totals for tubes with an eddy current testing measured depth of 21-30 percent TWD at anti-vibration bar locations were 52 for steam generator 2E0-88, 51 for steam generator 2E0-89, 19 for steam generator 3E0-88, and 31 for steam generator 3E0-89. Taking into consideration the difference in operating cycle time between Units 2 and Unit 3, these numbers suggest little difference in tube wear behavior between Units 2 and 3 steam generators at anti-vibration bar locations. This behavior was considered somewhat surprising, considering the extent of Unit 3 steam generator freespan tube-to-tube wear and the apparent resulting enhanced tube wear at upper tube support plates that occurred from transfer of this vibration energy.

What is clear from reviewing inspection data for the steam generators is that wear for both Units 2 and 3 steam generators occurred in the same localized region of the tube bundle, which suggests a common thermal-hydraulic root cause.

3. SUMMARY OBSERVATIONS

- No evidence was found in the reviewed documentation that addressed the cause for the abnormal and localized high void fraction in the replacement steam generator design, which historical degradation information would indicate was absent in the original steam generator design. It would also appear that the existence of the localized high void fraction and flow velocities, as calculated by ATHOS, was not questioned as a replacement steam generator design feature or compared against the replacement steam generator design basis. Rather, the thermal-hydraulic analysis results were accepted, initially by design, and again during the extensive review process.

- The conditions leading to the wear were calculated with the EPRI ATHOS code, a standard steam generator design tool. The calculated high void fraction region included the smaller region of tubes with the observed wear. It was not obvious that a small region with conditions which caused tube wear can be extracted from the ATHOS results.

- Any conclusions to be drawn from contribution of tube ovality to tube wear differences between Unit 2 and Unit 3 steam generators should specifically consider the Unit 2 and Unit 3 G-Values of tubes in the affected Unit 3 sub-population wear region. The tube ovality decreases gaps between tubes and anti-vibration bars, but also increases the propensity for in-plane vibration due to the decrease of cross-section stiffness.

- Absent the existence of additional information, there is no apparent basis to believe that the number of local radius adjustments during manufacture of U-bends has any relevance to observed steam generator tube degradation.

- The average of the gaps between the outermost tubes and the central columns was found to be essentially the same between the Unit 2 and Unit 3 steam generators, which does not support a premise that more uniform manufacturing practices for Unit 3 steam generator tube bundles resulted in less contact force between anti-vibration bars and tubes. In the absence of more dimensional information for the steam generator tube bundles, it is not believed possible to explicitly define the number of active supports in the Unit 2 and Unit 3 steam generators.

- Eddy current testing inspection measurements of tube-to-anti-vibration bar gap were determined to be of questionable value in an assessment of likely tube wear behavior.

- The performance conditions calculated with the thermal-hydraulic codes ATHOS and FIT-III are inconsistent with the thermal-hydraulic design described in L5-04GA510, Rev. 5, “Thermal and Hydraulic Parametric Calculations.” The consequences of this inconsistency between the replacement steam generator system design and calculated thermal-hydraulic performance has not been addressed.

- No evidence was found that the adoption of a smaller tube pitch/tube diameter ratio of 1.33 compared with the original steam generator, with a potential choking effect on inlet secondary flow (from the wrapper ports), was evaluated in design or addressed in reviews.

4. PROBABLE CAUSE EVALUATION

Condition Report NN 201836127, Root Cause Evaluation: Unit 3 Steam Generator Tube Leak and Tube-to-Tube Wear Condition, Revision 0, identified the mechanistic cause of tube-to-tube wear in SGs 3E0-88 and 3E0-89 to be fluid elastic instability (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-anti-vibration bar contact forces to overcome the excitation forces. The results of and visual inspections strongly support FEI being the applicable tube-to-tube wear damage mechanism. Section 8 of the draft AIT report concluded from independent NRC ATHOS code thermal-hydraulic analysis that the SCE replacement steam generators were not designed with adequate margin to preclude onset of FEI. The AIT inspection also concluded that the deficiencies appear to be related to the Mitsubishi FTI-III thermal-hydraulic code having predicted non-conservative low velocity results. Use by Mitsubishi of the ATHOS code and independent thermal-hydraulic analyses by Westinghouse (using a company version of the EPRI ATHOS code) and by AREVA (using a proprietary code) arrived at similar velocity conclusions as reached by the independent NRC ATHOS code review.

It was concluded from this review that the fluid condition alone did not explain the abnormal tube-to-tube wear based on the standard FEI criteria of ASME Boiler and Pressure Vessel Code, Section III, Division1, Appendix N, Section N-1330, 1998 . As noted in both the AIT draft report and the SCE RCE, no tube wear of the type detected in the Unit 3 replacement steam generators has been previously observed in other recirculating SGs in the domestic fleet. The observed severe tube-to-tube wear was also restricted to a small region of the tube bundle cross section. Based on the wear indications, the U-bend tube support by the anti-vibration bars was ineffective in a small region of the tube bundle for in-plane tube vibrations. Various assumptions were made to rationalize the mechanism, which caused the observed tube-to-tube wear including: (a) insufficient tube support at multiple anti-vibration bar locations, (b) vibration in an in-plane mode, (c) gaps between tubes and anti-vibration bars, and (d) spreading of the upper U-bend structure due to fluid dynamic forces and thermal effects.

No evidence was found in the reviewed documentation that was pertinent to the following: (a) the cause for the abnormal and localized high void fraction in the replacement steam generator design, which historical degradation information would indicate was absent in the original steam generator design; and (b) the existence of the localized high void fraction and flow velocities, as calculated by ATHOS, and originally by FIT-III with lower flow velocities, that has apparently been accepted without question as a replacement steam generator design feature. No evidence was found that the flow conditions, void fractions, flow velocities and temperatures of the tube bundle were compared against the replacement steam generator design basis. Rather, the thermal-hydraulic analysis results were accepted, initially by design, and again during the extensive review process.

The conditions leading to the wear were calculated with the EPRI ATHOS code, a standard steam generator design tool. The calculated high void fraction region included the smaller region of tubes with the observed wear. It was not obvious that a small region with conditions which caused tube wear can be extracted from the ATHOS results.

The U-bend tube support conditions were evaluated in detail by Mitsubishi and in the root cause evaluation, and a preliminary status of these evaluations is described in the draft AIT inspection report. The anti-vibration bar and tube tolerances were considered in estimating the tube support; however, the component and design tolerances, as described in Document L5-04GA428, “Design of Anti-Vibration Bar,” Revision 5, were not stacked across anti-vibration bars, tubes, and the tube bundle to accumulate maximum possible gaps or interferences, and determine the potential extremes of the tube support conditions that would be appropriate for a loose bundle.

5. DESIGN AND MANUFACTURING DIFFERENCES

The AIT inspection did not identify any significant differences in the design requirements of the Unit 2 and Unit 3 replacement steam generators. This evaluation also did not note any significant differences in design requirements between the Unit 2 and Unit 3 replacement steam generators.

The AIT inspection identified two unresolved items during its review pertaining to: (a) retainer bar-to-tube wear in the Unit 2 replacement steam generators, and (b) consideration of the potential impact of improving dimensional controls for tube roundness and anti-vibration bars. It was concluded from review of these unresolved items that the retainer bar-to- tube wear issue warranted no additional comment. Review of the subject material pertaining to dimensional controls of tube roundness and anti-vibration bars led to the following observations:

- The standard deviation for tube O.D. in the Unit 2 and Unit 3 replacement steam generators was calculated by Mitsubishi to be: steam generator 2E0-88, 0.71 mils; steam generator 2E0-89, 0.71 mils; steam generator 3E0-88, 0.63 mils; steam generator 3E0-89, 0.55 mils. Mitsubishi postulated in Report L5-04GA564, Rev. 2 that improved dimensional controls for Unit 3 replacement steam generators such as anti-vibration bar thickness, tube roundness, and gaps between tubes and anti-vibration bars probably resulted in less contact force between the tubes and the anti-vibration bars. This difference in standard deviation values for the O.D. of the tube populations in the individual SGs is considered by this review to have minimal effect, particularly if one takes into consideration that the low radius U-bends are the biggest contributor to tube ovality and higher G-values. The localized region of tube wear, however, is located in high row number tubes where variations in G-values would not be expected in the large radius U-bends during ongoing production. It is believed that any conclusions to be drawn from contribution of tube ovality to wear differences between Unit 2 and Unit 3 SGs should specifically consider the Unit 2 and Unit 3 G-Values of tubes in the Unit 3 sub-population wear region.

- Mitsubishi Report L5-04GA564, Revision 2, noted that the number of adjustments to tube bending radius was smaller for the Unit 3 SGs than for the Unit 2 SGs. Specifically, the reported values were: steam generator 2E0-88, 265; steam generator 2E0-89, 390; steam generator 3E0-88, 132; steam generator 3E0-89, 149. The inference drawn was in the context of promoting greater uniformity in tube to anti-vibration bar gaps. The required profile for U-bends is established on an inspection layout table and needed local, minor radius adjustments are made to assure conformance to the required profile. Absent the existence of additional information, there is no apparent basis to believe that the number of local adjustments to U-bends has any relevance to observed steam generator tube degradation.

- The average gap between outermost tubes and anti-vibration bars for the Unit 2 and Unit 3 SGs was reported by Mitsubishi Report L5-04GA564, Rev. 2 to be: steam generator 2E0-88, 0.59 mils; steam generator 2E0-89, 0.76 mils; steam generator 3E0-88, 0.15 mils; steam generator 3E0-89, 0.21 mils. This report additionally stated that the average of the gaps between the outermost tubes and the central columns is essentially the same between the Unit 2 and Unit 3 SGs. This data obviously does not support the premise that more uniform manufacturing practices for Unit 3 steam generator tube bundles resulted in less contact force between anti-vibration bars and tubes. In the absence of more dimensional information for the steam generator tube bundles, it is not believed possible to explicitly define the number of active supports in the Unit 2 and Unit 3 SGs.

- Data was not specifically searched for during this review to allow formal assessment of the technical credibility of eddy current testing inspection for measurement of gaps between tubes and anti-vibration bars; i.e., it is currently unknown whether a qualified Examination Technique Specification Sheet (ETSS) exists for this measurement. Difficulties in use of the CERTREC system for information retrieval negatively affected conduct of this review. Paragraph 4.1.2 of Mitsubishi Report L5-04GA564, Rev. 2 states, in part, with respect to comparison of bobbin probe signals in the Unit 2 and Unit 3 SGs for estimating tube-to-anti-vibration bar gap sizes "…This data did not reveal significant differences and indicates that the gaps in the affected region of the tube bundle are below 20 mils (0.5 mm). However, the average voltage signal in the Unit 3 SGs is slightly lower than the average signal in the Unit 2 SGs, indicating that the average gap size in the Unit 3 SGs is slightly larger than in the Unit 2 SGs, and indicating that the average contact force between the tubes and anti-vibration bars during operation may be lower in the Unit 3 SGs." These comments are believed to be speculative, and rely on a global number of unknown technical credibility for predicting values in a bundle sub-population. Review of Figure 4.1.2-1 in Report L5-04GA564, Rev. 2 indicates the potential fallacy in making these projections. Specifically, Figure 4.1.2-1shows virtually identical average absolute signal amplitude signals at anti-vibration bar locations for SGs 2E0-88 and 3E0-89, SGs that have shown significant differences in operational tube wear behavior. Accordingly, this review concluded that eddy current testing inspection measurements of tube-to-anti-vibration bar gap were of questionable value in assessment of likely tube wear behavior.

The most significant fabrication difference between the Unit 2 and Unit 3 replacement steam generators relates to the cracking indications that were identified in both Unit 3 replacement steam generators in the weld between the divider plate and the channel head subsequent to completion of the ASME Section III Code primary side hydrostatic test. Repair of these defects necessitated removal of the channel head from each of the Unit 3 steam generators. As a result, the tubesheet-to-channel head circumferential weld had to be repeated and further post weld heat treatment (PWHT) and hydrostatic tests performed. Review of Section 12.0 of the draft AIT report found that the team had comprehensively reviewed divider plate repair activities. An unresolved item was identified by the team pertaining to the adequacy of Mitsubishi evaluation and controls for the divider plate weld repairs. Subject areas in question included: (a) the approximate 300 additional rotations of each Unit 3 replacement steam generator that resulted from additional welding of the channel head to tubesheet, and the lack of consideration by Mitsubishi of the potential impact of these rotations on tube bundle configuration in terms of anti-vibration bar gaps or distortion; (b) the lack of a full assessment by Mitsubishi of the impact of heat input activities such as local PWHT, grinding and flame cutting on steam generator configuration in terms of tubesheet thermal expansion or distortion; and (c) the non-performance by Mitsubishi of dimensional checks after repair to confirm that critical secondary side dimensions were not affected by the repairs.

This review concluded that local post weld heat treatment of the channel head to tubesheet weld had the highest potential (of the AIT noted activities) for affecting the tube bundle. This view derives from the possible effects of the temperature gradient across the tubesheet that is created by the local post weld heat treatment cycle applied to the channel head-to-tubesheet weld. The gradient creates progressively lower thermal expansion in the legs of U-bends as the distance increases from the periphery to the center of the tubesheet. This variation in U-bend thermal expansion has resulted in the past in the detection of tube ding (DNG) eddy current testing indications at the upper tube support plate location in replacement steam generators. A DNG sort was requested from SCE. Limited review of the supplied information did not, however, identify any correlation of DNG signals with performance of local post weld heat treatment cycles, or any noted incidence at tube support plate locations in the replacement steam generators.

6. REVIEW OF REPLACEMENT STEAM GENERATOR THERMAL-HYDRAULIC DOCUMENTATION

6.1 Scope

This independent review of the thermal /hydraulic related aspects of the SCE steam generator condition was primarily limited to available documentation in the CERTREC system and focused on the features and operating conditions, which caused or could have contributed to tube damage. The review focused on the replacement steam generator tube bundle, anti-vibration bar and retainer bar designs, and the tube bundle flow condition. No specific difference was noted with respect to the basic findings of the AIT and the RCE reports. The high void fraction in the U-bends was, however, viewed as abnormal and not intended by design. Accordingly, a review was performed, which took into consideration: (a) inspection evidence, tube wear in the U-bends and at the tube support plates; (b) results from the ATHOS, FIT-III, and FEI calculations; (c) the replacement steam generator system thermal-hydraulic design; and (d) a comparison of the replacement steam generator changes from original steam generator design features. This review led to a sequence of conclusions, which trace the origin of the abnormal event conditions to a potential source not addressed by either the AIT inspection or the RCE reports.

6.2 Thermal-Hydraulic Overview

6.2.1 Sub-cooled Height Above the Tubesheet

Document L5-04GA510, Rev. 5, "Thermal and Hydraulic Parametric Calculations," defined generic parameters for the replacement steam generator design and operation. The calculations were noted to be easy to follow and provided replacement steam generator system flows, pressures and temperatures for a range of operating conditions, with input parameters listed for the thermal-hydraulic analysis. Design requirement are specified in SO23-617-01, Rev. 4, "Specification for Design and Fabrication of replacement steam generators for Units 2 & 3." Specific requirements and acceptance criteria are listed by reference to other documents. A spot-check of the design requirements and acceptance criteria showed that they were satisfied by the calculated system parameters. The intended tube bundle flow management is indicated in Figure 1 of Appendixes 10 and 11 to Document L5-04GA510, Rev. 5. These figures identified a sub-cooled height, $H_{PH,}$ at the bottom of the tube bundle, which was calculated with "SG Steady State Performance Calculation Code" (SSPC Code) to be H_{PH} 1364.965 mm for a T_{hot} temperature of 598 ^{0}F and H_{PH} 1145.173 mm for a T_{hot} temperature of 611 ^{0}F. Accordingly, boiling was not intended to start at the tube sheet level, or below tube support plate 1, as calculated by the thermal-hydraulic codes in L5-04GA521, Rev. 3 and confirmed by independent calculations with ATHOS. The performance conditions calculated with the thermal-hydraulic codes are thus inconsistent with the thermal-hydraulic design described in L5-04GA510, Rev. 5.

The consequences of this inconsistency have not been addressed. Additionally, no requirement was found that this non-boiling level should be maintained.

6.2.2 Flow from the Wrapper Inlet Ports to the Tube Bundle

No documentation was found, which detailed the flow from the wrapper inlet ports to the bottom of the tube bundle, other than the result of the thermal-hydraulic calculations that modeled the flow conditions. The only apparent change of the secondary flow condition from the original steam generator to the replacement steam generator design is the smaller pitch-to-diameter ratio of the tube bundle in the replacement steam generator (i.e., P/D = 1.33 in the replacement steam generator, P/D = 1.433 in the original steam generator) and 327 more tubes. All other parameters for the like-for-like steam generator replacements are essentially unchanged. No documentation was found for comparing the design of the wrapper inlet ports between the replacement steam generator and the original steam generator due to CERTREC administration problems. The smaller pitch-to-diameter ratio of the replacement steam generator tube bundle increases the cross-flow resistance in the tube bundle. As a result, the penetration of the flow from the wrapper ports into the tube bundle and the flow distribution in the bundle changes when compared with the original steam generator. The inlet flow stagnated in a region outside the outer row of stay rods, visible on the flow velocity plots of ATHOS and FIT-III. No evidence was found that this change of the effect of tube bundle inlet flow distribution was evaluated in design or addressed in reviews.

6.2.3 Thermal-hydraulic Analysis

6.2.3.1 Mitsubishi FIT III Code Analysis

Document L5-04GA521, Three Dimensional Thermal and Hydraulic Analysis (FIT III Code Analysis) calculated the void fraction and flow velocities of the tube bundle in some detail. The void fraction, flow velocities and temperatures throughout the tube bundle are presented and are the input to the vibration calculations. Issues with this code were addressed in the draft AIT inspection report. The AIT inspection noted that the Mitsubishi reported flow velocities and void fractions appeared to be low and, as a result, performed independent calculations using the EPRI ATHOS code. These calculations found much higher velocities in the tube bundle. The cause for the FIT-III discrepancy has currently not been resolved. The AIT inspection also noted that the validation and verification of the FIT-III code did not provide sufficient evidence that the code had been adequately benchmarked. The AIT inspection concluded, without performing vibration analysis, that the higher flow velocities and void fractions were the cause of the observed FEI and tube wear.

6.2.3.2 ATHOS and AREVA CAFCA Analyses

After the identification of FIT III code issues, an ATHOS thermal-hydraulic analysis was performed by the replacement steam generator supplier, Mitsubishi, with results compared against the independent analysis results from the NRC (ATHOS), Westinghouse (Company ATHOS version) and AREVA (CAFCA4 Code). The results of these calculations appeared to be in reasonable agreement, with some variation of results due to modeling differences, and all showed a local region of high void fractions starting below tube support plate 7 and extending into the U-bend region. In normal steam generator design, one would expect a relatively flat distribution of the void fraction level increasing from the sub-cooled region. The void fraction and fluid velocities were noted to be high in a local U-bend tube region of the bundle where the tubes exhibited significant wear.

With ATHOS calculated flow conditions as input, the critical velocities and stability ratios obtained with the Connors' equation in the ASME Boiler and Pressure Vessel Code Section III, Division1, Appendix N Section N-1330 still yielded stability ratios (ratio of effective flow to critical FEI flow) <1 for tubes with effective restraint at tube support plate and anti-vibration bar support locations. Accordingly, for FEI to occur, critical cross-flow velocities have been assumed to occur for U-bend tube sections not restrained from in-plane motion by insufficient tube-to-anti-vibration bar contact. No information was noted during documentation review regarding why U-bend tubes with larger U-bend radius, outside the small tube-to-tube wear damage region with high void fractions, did not experience the FEI tube movement, because with a larger curvature they were more susceptible to tube vibration unless the restraints were more effective.

One observation from examining these calculation results was that the nucleate boiling started in a region outside the outer row of stay rods. Figure 8.3-3 of Document L5-04GA521, Rev. 3 showed local void fractions, and Figures 8.3-1 and 8.3-2 showed the flow velocities above the tubesheet. Nucleate boiling at the tubesheet surface is an anomaly not intended to occur in the intended design as described in Mitsubishi Document L5-04GA510, Rev.5. The consequences of this inconsistency between the replacement steam generator system design and calculated thermal-hydraulic performance has not been addressed. It was also noted that Westinghouse in their independent ATHOS analysis indicated that their design approach precluded boiling at the tubesheet surface. The graphs of flow pattern above the tubesheet indicate a region of low flow velocities where higher void fractions than in the surrounding fluid are indicated. A potential cause for these low flow velocities, a region of almost stagnant flow, is the higher flow resistance for the cross-flow from the wrapper inlet ports into the tube bundle due to the smaller pitch-to-diameter ratio of the replacement steam generators than in the original steam generators. A comparison of the replacement steam generator thermal-hydraulics with that of the original steam generator was not found in either the AIT or the RCE reports, which could aid in the determination of the cause for the flow abnormalities in the replacement steam generator.

6.2.4 Tube Wear at tube support plates above the Tube Sheet Hot-Spot

Mitsubishi Report L5-04GA564, "Tube Wear of Unit-3 RSG – Technical Evaluation Report," Revision 2, includes figures showing the distribution of tube wear at tube support plate 1 to 7 levels and in the U-bend sections. The region at the tube support plate 1 level with TWD tube wear is immediately above the location where nucleate boiling at the tube sheet level started. A reasonable supposition is that flow caused this local wear. The correlation between the tube wear increasing upward from tube support plate to tube support plate and the location of the hot spot appeared to be systematic in the replacement steam generators.

The coolant flow between tubes in the straight tube sections is predominantly axial, upward with low cross-flow velocities. Mitsubishi postulated in L5-04GA564, Rev. 2, that turbulent excitation was the potential cause for wear at tube support plates. The evaluation did not specifically address the small region of tube wear shown in Figure 2-6 that was observed at tube support plates 1 through 7.

The correlation between boiling in a small region at the bottom of the tube bundle, based on Mitsubishi Document calculations, and the observed region of tube wear increasing from tube support plate 1 levels upward into the U-bend region has apparently not been addressed. Some mechanism is moving the tubes and causing the tube-to-tube support plate wear in a small region. Standard steam generator evaluation procedures may not model in sufficient detail the unusual flow pattern evolving from the tubesheet.

6.2.5 Tube Plugging

It has been proposed in Document L5-04GA571, "Screening Criteria for Susceptibility to In-Plane Tube Motion," Revision 4, to plug the tubes exhibiting wear and surrounding tubes that have a susceptibility for damaging tube motion and freespan wear. The draft AIT inspection report that was available during this review addressed tube plugging to contain tube damage, but does not address the specific plugging strategy more recently proposed by Mitsubishi.

Tubes proposed to be plugged in Unit 3 and Unit 3 SGs are selected based on pre-damage based scoring system. The identified tubes are located in a relatively small contiguous cross-section area of the tube bundle. These tubes include tubes shown in Figure 2-6 of Document L5-04GA564, Revision 2 with TWD wear at the tube support plate 1 level. The region with severe wear in the U-bend tubes originates in the straight sections above the location where nucleate boiling starts. The implication of this correlation between the hot spot and tube wear above the spot has not been evaluated.

Plugging of the selected tubes, identified in L5-04GA571, is intended to permit reactor operation without failures for a time to be defined. Plugging these tubes also eliminates the specific hot spot at the tube sheet. Locating a cold region above the tube sheet should reduce fluid temperatures in the region surrounding the previous hotspot, and prevent boiling at the tube sheet level, which needs to be confirmed by analysis.

6.3 Summary

In summary, the simplified, postulated scenario leading to the damaging tube vibrations of the U-bend tubes is as follows:

- The secondary flow from the wrapper inlet port to the tube bundle does not penetrate the bundle because the flow resistance of the replacement steam generator bundle with a smaller pitch-to-diameter ratios higher than in the original steam generator. No evidence was found that this change was considered in design.

- Nucleate boiling occurs at the tube sheet level with low cross flow, a hot spot location, which is inconsistent with the replacement steam generator system design parameters.

- Above the hot spot, undefined flow conditions cause a small group of tubes to vibrate starting at the tube support plate1 level. Tube wear progressively increases to the upper tube support plate 7.

- The localized high velocity flow with high void fraction causes the U-bend tube bundle to vibrate violently in a small region above the TWD wear at the tube support plate levels.

More evaluations would be required to substantiate the postulated scenario as the source of the high void fraction and velocity in a specific U-bend region.

Mitsubishi has selected groups of tubing to be plugged based on damage screening criteria. The selected tubes include the tubes with TWD wear at the tube support plate 1 level. Plugging these tubes also eliminates the specific hot spot at the tube sheet. Locating a cold region above the tube sheet should reduce fluid temperatures in regions surrounding the previous hotspot and prevent boiling at the tube sheet level.

7. OPERATIONAL IMPACTS

The AIT identified an unresolved item requiring further review pertaining to whether the licensee appropriately reviewed and dispositioned numerous steam generator loose parts alarms during Unit 3 operation. Similar steam generator loose parts alarms did not occur during Unit 2 operations in Cycle 16, raising the question of whether the Unit 3 alarms were potentially indicating steam generator tube-to-tube contact during power operations. It was noted from review of the licensee RCE report that Westinghouse had performed an analysis of the various alarms for the licensee. Westinghouse concluded that the vibration and loose parts monitoring system events for both SGs were the result of true metallic impacts and not false indications from electrical noise or fluctuations in background noise. The alarm events were noted to be similar to events that occur when SGs shift during reactor coolant system temperature transients, but it could not be conclusively stated without additional data that the events were from the same source. The licensee noted that “even with additional data, determination of the source of impacts could be hindered by the location of the sensors.” This comment is related to the fact that accelerometers were mounted on the support skirt for each replacement steam generator, a remote location with respect to monitoring internal replacement steam generator conditions.

During this independent review, it was ascertained that the accelerometer skirt location did not appear to comply with the requirements of the Design Specification SO23-617-01, “Specification for Design and Fabrication of Replacement Steam Generators for Unit 2 and Unit 3,” Revision 4. Specifically, Section 3.9.3.19, Loose Parts Monitoring Provisions, required mounting pads for sensors to be installed on the external surface of the inlet side of the channel head and on the lower shell. One pair of mounting pads (one active and one reserve) was required to be located with a vertical alignment above the tubesheet, and one pair with a vertical alignment below the tubesheet. Revision 4 of Design Specification SO23-617-01 was approved on July 28, 2010, which post dates the Unit 2 return to power on April 13, 2010 after replacement steam generator installation. The circumstances pertaining to relocation of sensors to a lower sensitivity measurement location, approval of this change, and the continuing conflict with current design specification requirements were not available for review.

8. REFERENCES

Document	Title	Revision
Condition Report NN 201836127	Root Cause Evaluation: Unit 3 Steam Generator Tube Leak and Tube-to-Tube Wear Condition	0
Condition Report NN 201843216	Root Cause Evaluation: [Unit 2] Steam Generator [Retainer Bar] Tube Wear	0
NRC 05000362/2012007	Augmented Inspection Team Report	Draft
L5-04GA564	Tube Wear of Unit 3RSG - Technical Evaluation Report	2
---------	ASME Boiler and Pressure Vessel Code, Section III, Division1, Appendix N, Section N-1330	1992
L5-04GA510	Thermal and Hydraulic Parametric Calculations	5
L5-04GA521	Three Dimensional Thermal and Hydraulic Analysis, FIT III Code Analysis	3
SO23-617-01	Specification for Design and Fabrication of RSGs for Units 2 & 3	4
L5-04GA504	Evaluation of Tube Vibration	3
L5-04GA428	Design of Anti-Vibration Bar	5
MHI Report KAS-20040233	SSPC Code Validation and Qualification Report	3
L5-04GA571	Screening Criteria for Susceptibility to In-Plane Tube Motion	4
KAS-20040233	SSPC Code Validation and Qualification Report	3
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NRC INTERNATIONAL TRAVEL TRIP REPORT

Traveler, Office, Division
Carl Thurston
Office of Research, Divisions of Systems Analysis, Reactor Systems Code Development Branch

Subject:
Low-Frequency Squeeze Film (SF) Damping Tests

Dates of Travel and Countries/Organizations Visited:
March 11th – 14th, 2012, Ontario, Canada/ AECL Chalk River Testing Laboratory

Summary of Trip
From March 11th – 14th, Mr. Ryan Lantz (R-IV), Mr. Carl Thurston (RES), and Dr. Gopinath Warrier (Contractor – UCLA Adjunct Professor) inspected and observed Low-Frequency Squeeze Film (SF) Damping Tests at Atomic Energy Canada Limited (AECL) Chalk River Testing Laboratory in Ontario, Canada. The purpose of the tests was to support the San Onofre Unit 2 return-to-service steam generator operational assessment (OA) analyses for San Onofre Unit 2 as related to Unit 3 tube leak and loss of tube integrity due to tube-to-tube wear in the upper bundle.[1] The analytical method used in the Southern California Edison (SCE) operational assessment uses an empirical damping correlation based on work by Dr. Michel J. Pettigrew. Dr. Pettigrew has worked as an AECL senior staff member and is currently Chair of Fluid and Structure Interaction at the Ecole Polytechnique in Montreal.

The licensee's methodology for the OA was developed by their replacement steam generator contractor, Mitsubishi Heavy Industries (MHI), in conjunction with SCE staff and industry experts including Dr. Pettigrew. During the NRC review of the licensee submittals, the NRC questioned the MHI application of the SF damping correlation used to compute tube stability margins for flow induced vibration. Specifically, the NRC review indicated that the correlation was being non-conservatively extrapolated to lower frequencies beyond published data, and that the correlation was being non-conservatively applied in regard to structural geometry. The SF damping correlation was developed for tube support plates where the geometry provides, to a certain degree, a confined fluid layer around the tube, i.e., for a support plate with drilled or broached holes. The NRC maintained that this confined layer is not present at flat bar anti-vibration bar supports and that the correlation should not be applied to anti-vibration bar intersections.

The SCE-sponsored AECL testing plan[2] was designed to provide justification for (1) SF model use at lower frequencies, in the range of 5-20 Hz, and (2) SF application to flat bar anti-vibration bar support structures.

AECL's existing straight single-tube test rig (from previous tests in 1988[3]) had been kept in storage at the lab and was used for all of the tests. The test rig can hold a straight tube

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[1] San Onofre Nuclear Generating Station – NRC Augmented Inspection Team Report 05000362/2012007 (ML12188A748)
[2] AECL EACL Test Plan, "Measurement of Steam-Generator Tube Damping Due to Anti-Vibration Bar Supports," 153-127370-TP-001 Revision D2.
[3] B.S. Kim, M.J. Pettigrew and J.H. Tromp, "Vibration Damping of Heat Exchanger Tubes In Liquids: Effects of Support Parameters," Journal of Fluids and Structures, Vol. 2, pp 593-614, 1988.

extending to approximately 4.26 m in length between fixed supports, either mounted in air or immersed in water. The tube-vibration frequency can be varied by selecting the tube length/diameter and type of tube support mount. The test rig rigidly allows for adjusting the type and clearance of tube supports. The test rig provides hardware to excite vibrations in the tube and instruments to measure and record the local damping response.

The purpose of the inspection was to (1) determine if the testing was adequate to support SCE's use of the squeeze film damping correlation in their OA, and (2) confirm that if the testing met the requirements of Appendix B (commercial grade dedication).

A. Test Setup

The rig consists of an inner trough attached to an outer trough. The inner trough is slightly shorter than the outer trough and both are made from ¼"-thick stainless steel plate. The vertical walls of the inner trough are reinforced by angled buttresses for maximum rigidity. The ends of the inner trough are closed so that it may be filled with water. All of the fixtures used to mount to the tube, instrumentation, and the tube supports are mounted from the top edge of the inner trough and are secured with set screws. These fixtures span the inner trough so that the tube can be approximately 75 mm below the surface when the rig is filled with water.

As indicated in Figure 1 (inner trough only), the supports are located in the middle of the span. In order to qualify the test rig and benchmark the 1988 results, the drilled-hole geometry shown was employed first and then the flat bar anti-vibration bar geometry (Phase 2) was inserted. In each case, the support was clamped to the mounting fixture so that it could be adjusted to vary the tube eccentrically within the drilled hole in attempts to replicate selected 1988 tests. The majority of the tests were performed with the two opposing flat anti-vibration bars oriented vertically, and perpendicular to the tube axis. This paired configuration of flat bars reasonably represents a typical anti-vibration bar intersection in a steam generator U-bend.

The mounting jig of the anti-vibration bar fixture allowed for varied lateral positioning of the anti-vibration bar in relation to the tube. This adjustment allowed for fine sizing of the gap with the anti-vibration bars including bringing the tube in contact with one side of the anti-vibration bar, with the contact ranging from slightly touching up to an applied normal-direction contact force as high as a 3 Newtons.

The test rig is equipped with two electromagnetic exciters ("coil" exciters) that are used to produce excite tube vibrations in both the vertical and horizontal directions, respectively, with no physical contact between exciter and tube. Both coil exciters are mounted on the same mounting fixture, at a location that is located approximately one quarter of the distance between the tube support and the tube end support clamps.

Three eddy current proximity probes were used to measure vibrations of the tube. The probes are capable of use in air or in water. Two of the probes were used to measure the vertical and lateral positions of the tube relative to the support surface. The third proximity probe was used to measure the vertical motion of the tube at the excitation location.

Alloy 800 SG tubing with an outside diameter of 0.625" (15.9 mm) and a wall thickness of 0.044" (1.12 mm) was used for these tests. The anti-vibration bar flat bars used were materials remaining at MHI from the RSG fabrication. The work was performed in accordance with the AECL Nuclear Laboratories Quality Assurance Program, which is compliant with ISO 9001:2008 requirements.

Although the test rig had been kept intact, much of the test setup and instrumentation was updated from what was used in the 1988 tests in order to take advantage of advancements in technology since the original tests were performed. The updated equipment included the excitation devices, the proximity probes, and the data acquisition systems. The NRC inspection examined the setup, calibration and operator training and also checked the “Commercial Grade Dedication” to confirm that the test results met the quality standards required for safety-related applications. The goals of the inspection team were to confirm that the tests were setup in accordance with the test plans, confirm adequacy of the test rig to reproduce the 1988 data, and independent verification of selected test results.

The testing was performed in three phases. Phase 1 tests were used to verify the set up and procedures, to check instrumentation, and to develop/check the test and analysis procedures. This was accomplished by using the drilled-hole setup and comparing the data to the 1988 tests. Phase 2 tests were performed for four different tube support configurations (via position of the clamping fixtures with active lengths of the tube set at 1.50, 2.0, 3.0, and 4.25 m) to produce expected fundamental frequencies of approximately 35.4, 20.0, 8.6, and 4.4 Hz, respectively, when the tube was filled with water. All tests were conducted at room temperature. Trapped air bubbles were removed from the tube prior to testing in water. Likewise, water droplets were wiped off the tube and supports and dried before tests in air. Phase 3 tests were not pre-defined during the initial testing and were intended to be follow-on tests based on findings/difficulties from the Phase 2 tests.

For each tube configuration series, tests were completed under the following conditions:

- without support with the trough empty of water to determine the inherent baseline damping of the configuration,
- without support with trough full of water to determine the viscous damping due to the presence of water,
- tube centered in the anti-vibration bar support with the trough full of water with vertical excitation,
- tube centered in the anti-vibration bar support with the trough full of water with lateral excitation,
- small one-sided tube-to-anti-vibration bar gap with the trough empty and full of water, and
- tube deflected at anti-vibration bar by amounts preload forces.

Damping results were determined using two different methods. The first method was the vibration–decay or log decrement method which measures the vibration decay after the sinusoidal excitation is ended. The “log-decrement” method fits the vibration amplitude decline with time to a logarithmically decaying sinusoidal function.

The second method was the “half-power spectral” method which is based on a standard vibration response distribution fit to the spectral response peak of the lowest vibration mode excited by a continuous band-limited random excitation. The continuous sinusoidal and/or random excitation power inputs by the shaker, used to generate tube vibration, can then be calculated and used to determine total modal damping.

Once the tube and support have been placed in the desired configuration, the tube is excited to the prescribed resonant vibration and the subsequent damping is measured and recorded via the data acquisition system. In each case, the coil exciter is used to build up the excitation amplitude of the fundamental vibration mode to the maximum amplitude allowed using sinusoidal excitation at the natural frequency. The excitation is then shut off and the vibration is allowed to decay. The rate of decay in amplitude is then used to determine damping. In order to provide more realistic conditions, lateral excitation of the tube was used in combination with vertical for a few select configurations.

B. Verification of Test Setup and Instrumentation

The setup of the test rig was reviewed and found to conform to the SCE specifications.[2] All documents related to setup and testing activities were reviewed. The documents adequately described the pre-testing, testing, and post-testing activities. Note that the fluid (air or water) temperature in the test rig was monitored to ensure that all testing was carried out at room temperature.

AECL's calibration records for the above test equipment were reviewed. The records indicate that the equipment was calibrated during the period Feb. 11-19, 2013 and that the calibration was performed as per ASTM standards. The range and accuracy of the test equipment used was appropriate for the tests being performed. Minor problems were noted with demineralized water in the trough causing rust on the anti-vibration bar surface, on the excitator, and on the proximity devices. The rust was photographed and cleaned and considered to have no impact on the results. AECL then began water change-outs on a more frequent basis and cleaned off the rust with each water change.

No problems were found with AECL tests processes or equipment setups or implementation of the test procedures.

C. Verification of Test Results

The raw data (vibration amplitude vs. time data from three proximity probes) measured during three of the tests (with drilled plate) was analyzed independently by the team. The log-decrement method was used to calculate the damping ratio. Table 1 shows a comparison of the results we obtained with those obtained by AECL/SCE and Kim et al. (1988).[3] The data reduction procedure used by AECL/SCE appeared to be correct.

Table 1. Comparison of damping ratios for drilled plate

Damping Ratio (%)				
Test No.	AECL/SCE*	Kim et al.	Warrier & Dhir*	Comments
Comm PH1-01 ExcL Rep1	0.017 (10.52 Hz)	0.012 (10.6 Hz)	0.019 (10.6 Hz)	Air w/o tube support
Comm PH1-02 ExcL Rep1	0.76 (8.65 Hz)	0.86 (8.6 Hz)	0.80 (8.6 Hz)	Water w/o tube support
PH1-T01_Sine_Lrg_1(PF)	1.23 (8.59 Hz)	1.44 (8.6 Hz)	1.25 (8.6 Hz)	Water with tube support

* using log-decrement method

The results of the Phase 1 qualification tests were examined for consistency with the previous 1988 test, where a value of 0.59 was found for L=19mm, H=1.5mm, and eccentricity ratio=0

(Table 2[3]). Our preliminary computations indicated SF damping in the range of 0.40 for this benchmark case. The team considered this result to be reasonably close and adequate.

The raw data from one of the tests using flat plate anti-vibration bar (with only vertical excitation) was also analyzed, and the log-decrement method was used to calculate the damping ratio. The natural frequency was also calculated. Table 2 shows the comparison of the damping ratio (and natural frequency) calculated to those obtained by AECL/SCE. The results in Table 2 also confirm that the data reduction procedure used by AECL/SCE is reasonable.

Table 2. Comparison of damping ratios for flat plate anti-vibration bar

Test No.	AECL/SCE*	Warrier & Dhir*	Comments
PH2-1_T12_Sine_Lrg_1	0.764 (8.72 Hz)	0.77 (8.7 Hz)	Water with anti-vibration bar tube support

* using log-decrement method

A comparison of the results given in Table 1 (for water w/o tube support) and Table 2 (water with anti-vibration bar tube support) shows that when flat plate anti-vibration bars are used and the tube oscillations are in-plane (vertical, i.e., along the length of anti-vibration bar) any additional damping due to the supports (anti-vibration bar) is negligible.

Based on periodic briefs with Dr. Vijay Dhir (Contractor – UCLA Dean Mechanical & Aerospace Engineering), the team requested AECL/SCE to take photos of droplet interface of the test water to tubing material and to the anti-vibration bar surfaces. The static contact angle was measured to be 55° ± 5°, for both tube and anti-vibration bar material. This indicates that water was only partially wetting the solid surfaces.

In conclusion, the anti-vibration bar test results examined by the inspection team showed that squeeze film damping is negligible when only vertical excitation (in-plane) is present. However, further Phase 2 testing provided results (note that Phase 2 was only about 25% completed when we exited the site on 3/14) that showed non-negligible squeeze film damping. The increased experimental damping observed in the later tests may be due to (1) addition of mixed lateral motions or (2) where there is a preload added to tube-to-anti-vibration bar contact. Regardless, based on this testing it is clear that the squeeze film damping correlation[3] does not apply directly to the anti-vibration bar geometry.

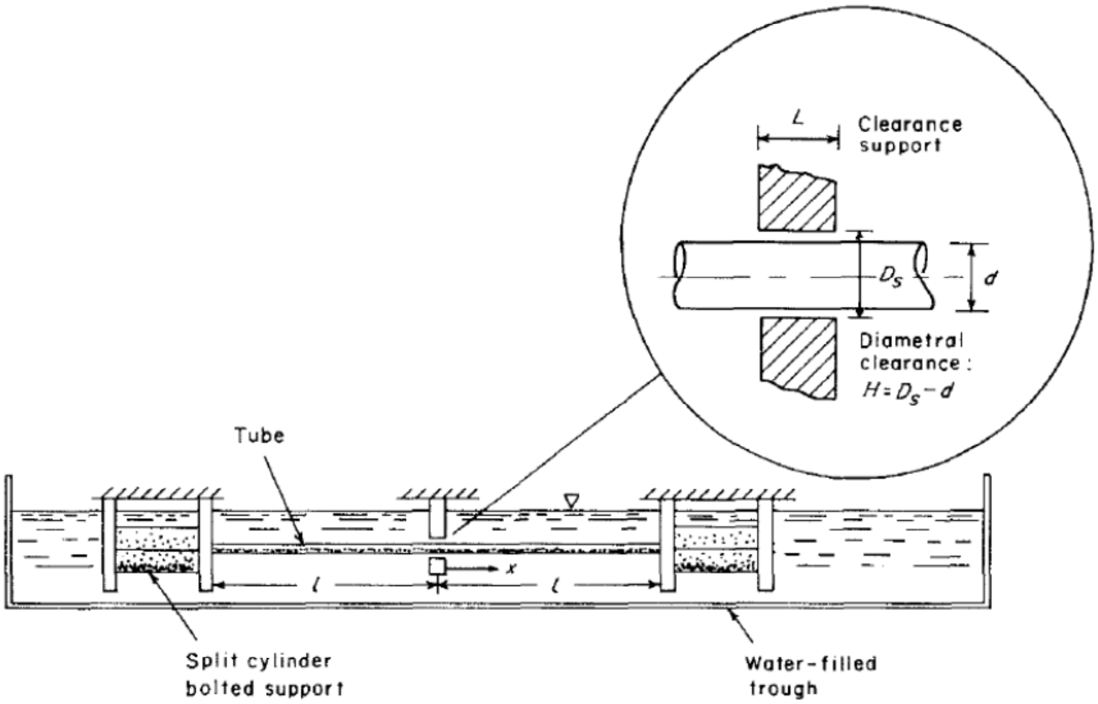
Additionally, the inspection team found no apparent problems with the commercial dedication.

The other participates included:

AECL
Dr. Victor P. Janzen
Bruce A. W. Smith (Test Eng)
Nigel J. Fisher
Dr. Paul Feenstra (Test Eng)

SCE
Mike Liu
Tom Yackle
Myles Pawlaczyk
Mike Jasurda

Figure 1: Sketch of 1988 Two-span Tube Rig inner Trough with Intermediate Support[3]



Report to NRC

Submitted by V. K. Dhir

May 31, 2013

The report is organized in order of the questions listed in the work scope.

A. Review and comment on the strength and weakness of existing methodology that is based on Connors' equation. Note any features of the correlation that are most prone to error and/or that may be limited in applicability. Comment on potential improvements based on current industry data and trends.

Studies of tube vibrations induced by fluid flowing parallel and across tube bundles have been reported in the literature since 1970. Most of the reported studies have been experimental in nature and only a few theoretical efforts have been carried out to understand the mechanisms of vibrations. Experimental studies have included both single and two phase adiabatic and diabatic flows over flexible tubes in a tube bundle. During flow along and across tubes, the coupling of flow induced force with the time varying displacement of tubes causes the tubes to vibrate. Under normal flow conditions, the amplitude of these vibrations is small and the vibrations do not impose a threat to the integrity of tubes. However, at certain flow conditions coupling between fluid induced forces and structural response may be such that a large jump in the amplitude of the vibrations occurs. Such a condition is referred to as onset of Fluid-Elastic Instability (FEI). Persistence of FEI can lead to damage to the tubes and loss of integrity. The flow velocity at which FEI occurs is termed as the critical velocity. In practice, heat exchangers are designed to avoid onset of FEI with a considerable margin.

In axial single and two phase flows along the tube bundle, the FEI can occur because of fluid flow inside or outside of the tubes. The dimensionless velocity ($\boldsymbol{U}$), required for onset of FEI in axial flow is given by Pettigrew and Taylor[i] as

$$\boldsymbol{U} = \boldsymbol{V}\boldsymbol{L}\sqrt{\boldsymbol{m_h}/(\boldsymbol{E}\boldsymbol{I})} \tag{1}$$

where, $\boldsymbol{V}$, is the axial flow velocity, $\boldsymbol{L}$, is the tube length, $\boldsymbol{m_h}$, is the hydrodynamic mass of the tube, $\boldsymbol{E}$, is Young's modulus, and, $\boldsymbol{I}$, is the moment of inertia of the area of the tube about the tube axis. Hardly any FEI data leading to large amplitude vibrations under axial flow conditions has been reported in the literature. Tube vibrations under axial flow conditions can also occur as a result of nucleate boiling on the surface (small pressure fluctuations on the surface) and as a result of pressure fluctuations and turbulence in the bulk flow. However, magnitude of vibrations resulting from these mechanisms is relatively small.

Mechanisms of vibrations of tubes subjected to cross flow have also been reviewed by Pettigrew and Taylor[1] and Blevins[ii]. The mechanisms identified for flow induced vibrations include coupling of the flow induced forces with the structural response, turbulence in the flow, and periodic vortex shedding. In most heat exchanger applications especially under two phase flow conditions, the latter two mechanisms are of lesser importance. Under certain flow and support conditions, the fluid forces can lead to onset of FEI. Generally, it is believed that FEI occurs when energy input by fluid forces acting on tubes exceeds the energy lost by the tubes to the fluid by damping. Connors[iii] was perhaps the first one to report data on FEI of a row of tubes held by piano wires and subjected to cross flow of air. Connors observed that at the onset of FEI, not only the amplitude of vibrations increases substantially, the tubes also develop a whirling motion and vibrate in oval orbits. Based on his single phase data and using dimensional analysis, Connors arrived at the criteria for FEI as

$$\frac{\boldsymbol{U_{cr}}}{\boldsymbol{f_n\,D}} = \boldsymbol{K}\left[\frac{\boldsymbol{m(2\pi\zeta)}}{\boldsymbol{\rho\,D^2}}\right]^{\boldsymbol{b}} \tag{2}$$

where $\boldsymbol{U_{cr}}$, is the critical gap velocity through tube array, $\boldsymbol{f_n}$, is the natural frequency of the tube, $\boldsymbol{D}$, is the diameter of the tube, $\boldsymbol{m}$, is the mass of the tube per unit length including the mass of fluid in the tube and the hydrodynamic mass, $\boldsymbol{\zeta}$, is the damping factor, which depends on the fluid conditions, and the manner in which tubes are supported, and, $\boldsymbol{\rho}$, is the density of the fluid. The constant $\boldsymbol{K}$ and exponent $\boldsymbol{b}$ were found by Connors while fitting the data to be 9.9 and 0.5 respectively.

Although Connors obtained the correlation from single phase data, subsequent experimental work by a number of investigators involving single and two phase flows, different types of tube arrays, and tube supports has shown that general form of equation (2) still holds. However, different investigators have found that to correlate their data, the magnitude of constant $\boldsymbol{K}$ varies significantly from the value suggested by Connors.

There are a number of reasons for the large scatter that is observed in the data of various investigators when it is correlated in terms of Connors' equation (2).

i. The natural frequency of a tube depends on the size, configuration, and mass of the tube, the manner in which the tube is held, its length, and tube material properties. Any ambiguities in defining tube parameters will be reflected in variations in calculation of the critical velocity from a correlation such as equation (2).

ii. The critical velocity, $\boldsymbol{U_{cr}}$, is the gap velocity of the two phase mixture. Aside from gas and liquid flow rates, the mixture velocity will depend on void fraction and geometrical arrangement of the tube array. Any uncertainty in void fraction from different correlations will affect the magnitude of calculated $\boldsymbol{U_{cr}}$.

iii. The total mass of the tube includes tube mass, mass of fluid in the tube and the hydrodynamic mass (added mass or the fluid mass that moves with the tube). The tube mass depends on the size, thickness and density of tube material. The mass of fluid in the tube depends on the density of the primary coolant and inside diameter of the tube. Void fraction, densities of the two phases, the pitch to diameter ratio and arrangement of the tube array (e.g., triangular, square, etc.) will affect the added or hydrodynamic mass.

iv. The degrees of freedom of the tubes have influence on the observed critical velocity for onset of FEI. Generally, it is found that the threshold for out-of-plane instability (lift direction) is lower than that for the in-plane instability (drag direction). Janzen *et al*[iv] were the first to experimentally observe U-tube bundle vibrations both in-plane and out-of-plane under air-water two phase flow conditions. They noted that instability constant $\boldsymbol{K}$ in Connors' equation was generally higher than 3 and for in-plane instability the flow velocity was about twice that for out-of-plane instability. Value of $\boldsymbol{K}$ for in-plane instability was also found to be twice that for out-of-plane instability when values of damping factor observed in the experiments were used. However when a nominal value of 1.5% was used for the damping factor, the in-plane value of $\boldsymbol{K}$ was only 17% higher than the out-of-plane value. This is probably a result of the uncertainty in measurement of damping factor and variability in the damping factor data of various investigators. Violette *et al* [v] investigated tube arrays preferentially flexible in the in-flow direction and subjected to air-water two phase cross flow. The tubes were either allowed to be flexible in the flow direction or were allowed to be axi-symetrically flexible. In-plane (drag direction) instability occurred at higher velocities than that for a tube flexible in all directions. The critical velocity increased with increase of stiffness or frequency but not in direct proportion to frequency. Although they found a value of $\boldsymbol{K}$ higher than 3 in Connors' equation for FEI in the in-plane direction, (some of the data, however, could be correlated with $\boldsymbol{K} = \mathbf{3}$), the increase was attributed to a transition at values of parameter $\frac{\boldsymbol{2\pi m\zeta}}{\boldsymbol{\rho D^2}} \geq \mathbf{1}$.

The transition was identified by noting two different trends in the critical velocity data at onset of FEI when plotted as a function of $\frac{\boldsymbol{2\pi m\zeta}}{\boldsymbol{\rho D^2}}$.

FEI phenomenon in a tube bundle in which tube flexibility direction is in general different from either the flow approach direction or transverse to it has been studied by Khalvatti *et al.*[vi] Flow direction was found to affect the onset of instability. Critical velocity decreased when the flow was normal to the direction in which the tube was most flexible (lift direction).

v. The total damping factor, $\boldsymbol{\zeta}$, includes contributions from internal material damping, tube support damping including squeeze film damping, viscous damping, and two phase fluid damping. Internal material damping is due to internal energy dissipation within a material, support damping or structural damping includes friction and motion of trapped fluid between tube and support. Viscous damping is due to fluid drag and viscous dissipation. Because of the increased compliance of a gas-liquid mixture, two phase fluid leads to additional damping. Two phase fluid damping depends on the density of the surrounding fluid, void fraction of the two phase mixture and geometry of the tube array including orientation and spacing of tubes. Since the critical velocity depends approximately on the square root of the damping factor, any uncertainty in the damping factor would be reflected accordingly in the calculation of critical velocity at the onset of FEI. This is further discussed below in the context of determination of total damping factor.

vi. The two phase mixture density depends on the system pressure and temperature and on the void fraction. Thus depending on the type of correlation that is used in obtaining void fraction for given liquid and vapor velocity (unless measured), the calculated critical velocities could differ. This is especially important at very high void fractions where two phase damping decreases very rapidly to near zero as void fraction reaches unity.

Two parameters that strongly influence the critical velocity for onset of FEI are natural frequency and damping factor. They are discussed next.

Natural Frequency: Natural frequency for lateral vibrations of a simply supported rod or tube is given as[vii,viii]

$$f_n = \frac{n^2\pi}{2L}\sqrt{\frac{EI}{mL^2}} \tag{3}$$

where n is the mode of excitation, L, is beam or tube span, E, is the Young's modulus of elasticity of the rod or tube material, I, is the moment of inertia of the cross sectional area of the rod or tube about the axis of bending, and m, is mass per unit length of the tube including fluid filling the tube. As the natural frequency varies inversely with the square of the length of the tube, an increase in unsupported span of the tube will lead to reduction in natural frequency and in turn to reduced velocity for onset of FEI. In the U-bend region of steam generators anti-vibration bars are placed to limit the amplitude of out-of-plane (lift direction) vibrations. In case there is no gap between tubes and anti-vibration bars or the amplitude of vibrations is such that tubes hit the anti-vibration bars, new nodes are created at anti-vibration bars and span length of the tubes decreases. This in turn leads to an increase in the natural frequency of the tubes.

Damping: Pettigrew and Taylor[ix,x], and Pettigrew, *et al* [xi] from their review of two phase flow induced vibration studies, have made design recommendations for avoidance of FEI of tubes subjected to two phase cross flow. Total damping, as described earlier, includes viscous damping, ζ_v, internal material and support damping, ζ_s, and, two phase mixture damping, ζ_{tp}. Viscous damping is due to fluid that adheres to the tube and the damping of two phase mixture is in addition to it. As such an expression for total damping is written as:

$$\zeta = \zeta_v + \zeta_s + \zeta_{tp} \tag{4}$$

To obtain viscous damping in two phase flow, Pettigrew and Taylor[1] used an expression similar to that used for single phase while replacing single phase kinematic viscosity with the two phase kinematic viscosity. Two phase kinematic viscosity, ν, was defined as

$$\nu = \frac{\nu_l}{1-\propto\ (\frac{\nu_l}{\nu_g} - 1)} \tag{5}$$

where ν_l, is the liquid kinematic viscosity, ν_g, is the gas or vapor kinematic viscosity, and $\propto$ is the void fraction. An expression for viscous damping in terms of two phase kinematic viscosity was written as

$$\zeta_v = \frac{\pi}{\sqrt{8}}\left(\frac{\rho D^2}{m}\right)\left(\frac{2\nu_{TP}}{\pi f D^2}\right)^{1/2}\left\{\frac{\left[1 + (D/D_e)^3\right]}{\left[1 - (D/D_e)^2\right]^2}\right\} \tag{6}$$

where, ρ, is the fluid density on secondary side, D, is the tube diameter, m, is the mass of the tube per unit length, f, is the tube frequency, and D_e is the equivalent diameter. The equivalent diameter ratio, D/D_e, for a triangular array was defined as

$$\frac{D_e}{D} = \frac{(0.96 + 0.5\, P/D)}{P/D} \qquad (7)$$

where $\mathbf{P}$ is the tube pitch. Table 1 gives the value of ζ_v calculated for a secondary pressure of about 58 bars or 853 psia as a function of $\propto$. For primary liquid filled tube including the hydrodynamic mass, the tube mass was calculated as $m = 0.4711 + .00478\, \rho\, lbm/ft$. Tube frequency was taken to be 34 sec^{-1}.

Viscous Damping Factor

Table 1

$\propto$	$\rho, lbm/ft^3$	ζ_v[4]
0	47.6	1.36×10^{-3}
0.2	38.5	1.28×10^{-3}
0.4	29.3	1.15×10^{-3}
0.6	19.8	0.97×10^{-3}
0.8	11.1	0.7×10^{-3}
1.0	1.9	0.19×10^{-3}

Structural (internal or material) damping in steam generator tubes is generally much smaller than the damping introduced by supports such as tube support plates (TSPs) and anti-vibration bars. As such structural or internal damping may be neglected. For tubes that pass through holes drilled in plates or broached holes or through egg crate type of supports, there is a gap between the tubes and supports. A liquid film resides in the gap at not very high qualities. For a vibrating tube liquid film may be squeezed as the tube moves towards the solid surface. During its sliding motion in TSP, a tube may experience viscous shear imposed by the liquid film between two solid surfaces. In case there is no liquid on the shell side, or the tube has a rocking motion, solid to solid contacts can occur and lead to friction force in addition to viscous shear. Thus the supports can provide additional mechanisms for dissipation of energy during vibration of tubes and to the damping experienced by tubes. Generally tubes are not held axi-symmetric in the gap. This uncertainty along with the complexity of interaction between the squeeze film, viscous shear and sold to solid friction has limited the validity of the application of theoretical models such as by Mulcahy.[xii]

A liquid film can also exist between tubes and anti-vibration bars placed in the U-bend region of steam generators if there is no contact between the tubes and the anti-vibration bars. For out-of-plane vibrations (lift direction) of the tubes, squeezing out of the liquid film will occur as the tubes come closer to or impact the anti-vibration bars. During the in-plane motion the tubes will slide over the liquid film between anti-vibration bars and tubes. Thus in the latter case only viscous shear damping will be important. Janzen *et al* [4] have experimentally studied the effect of flat bar U-bend restraints (FURs) on in-plane and out-of-plane FEI. They have noted that amplitude of vibrations was limited by the gap between tubes and FURs. No discernible effect of the presence of FURs and the size of the gap between tubes and FURs on the onset of in-plane or out-of-plane FEI was observed. This observation is true as long as no physical contact between tubes and FURs occurs. In an earlier study Weaver and Schneider[xiii] studied the effect of flat bar supports on FEI of U-tube heat exchangers under cross-flow of air. They found that in the presence of little or no gap between tubes and flat bar supports, higher nodes for out-of-plane vibrations were created. However, when

[4] **In practical application to nuclear steam generators the contribution of viscous damping is small and can be neglected.**

there was a large gap between the tubes and supports, the tubes behaved as if there was no support until the amplitude of vibrations was such that tubes started to impact the support. Instability occurred at low flow velocities. The repeated impact could lead to fretting failure of tubes with time. However, after the tubes started to impact the supports, an increase in flow velocity triggered higher nodes of instability.

Pettigrew *et al*[11] have correlated available data for tube support damping due to mechanisms of squeeze film and friction which presumably includes the contribution of viscous shear. The data were from both laboratory experiments and field tests and the correlation was developed for multi-span tubes. Their semi-empirical correlation for multi-span tubes accounted for the thickness of the support plate relative to the tube span and was written as

$$\zeta_s = \frac{N-1}{N} \cdot \left[\frac{1460}{f_n}\left(\frac{\rho D^2}{m}\right)\left(\frac{L}{lm}\right)^{0.5} + 0.5\left(\frac{L}{lm}\right)^{0.5}\right] \quad (8)$$

The first term in equation (8) accounts for squeeze film damping whereas the second term for friction between tubes and tube support. In equation (8), $\mathbf{N}$, is the number of tube supports, f_n, is the natural frequency of tubes, ρ, is the fluid density, D, is tube diameter, m is the tube mass per unit length, L, is the thickness or width of the support and, lm, is the characteristic (average) span length. It should be noted that the squeeze film damping is correlated with inverse of frequency and the data used in developing the correlation covered a frequency range from 30 to 500 Hz. It should be noted however, in an earlier work, Kim *et al*[xiv] correlated the squeeze film damping data as $f^{-0.6}$. Because of inverse dependence of squeeze film damping on frequency in eq. (8), very high values of support damping are given by the correlation at low frequencies ($f_n \ll 30\ Hz$). As such one must be very careful in extending the correlation beyond the range of the available data. In the absence of additional data it may be prudent to limit the upper value of the maximum squeeze film damping at a frequency of $30\ Hz$. Also, it should be noted that the correlation is applicable to tubes passing through eggcrates and circular support plates and currently no justification exists for applying the results to flat anti-vibration bars.

Damping factor for two phase flow has been reported by a number of investigators. These damping factors have been deduced from the data for variation of amplitude of vibrations with frequency. Often a significant scatter in the data is found because of the uncertainty in the evaluation of damping factor from the vibration amplitude frequency signal, using half-power bandwidth method. Also, often it is not clear if other components of damping are included or excluded in the reported data. Based on the data reported in the literature up to 1994, Pettigrew and Taylor[1] have proposed a correlation for two phase flow damping as

$$\zeta_{tp} = A\left(\frac{\rho_l D^2}{m}\right) f(\propto) s \left[\frac{1+\left(D/D_e\right)^3}{\left(1-\left(\frac{D}{D_e}\right)^2\right)^2}\right] \quad (9)$$

where A is an empirical constant that was chosen to have a value of 5. The function $f(\propto)$ was correlated as

$$\begin{aligned} f(\propto) &= \frac{\propto}{0.4} \ \text{ for } \propto \le 0.4 \\ &= 1.0 \ \text{ for } 0.4 \le \propto \le 0.7 \end{aligned} \quad (10)$$

$$= \mathbf{1.0} - \frac{(\propto -\mathbf{0.7})}{\mathbf{0.3}} \ \mathbf{for} \ \propto \geq \mathbf{0.7}$$

The parameter $\boldsymbol{s}$ was included to account for the effect of variation of surface tension with temperature and was defined as

$$\boldsymbol{s} = \left[\frac{\boldsymbol{\sigma(T)}}{\boldsymbol{\sigma(20°C)}}\right]^{\boldsymbol{C}} \tag{11}$$

Because of the limited steam-water data at different pressures, the definition of parameter $\boldsymbol{s}$ is very tenuous. Recently Mitra, Dhir and Catton[xv] obtained FEI data on flexible tube bundles in both air-water and steam-water mixtures. From their work, they found little systematic combined effect of change in surface tension with temperature and presence of vapor versus air in the two phase mixture. Thus at present no definitive data are available to substantiate the form and the value of exponent $\boldsymbol{C}$ in the expression for $\boldsymbol{s}$ proposed by Pettigrew and Taylor. It appears reasonable to assume $\boldsymbol{C}$ to be zero or $\boldsymbol{s}$ to have a value of unity.

Critical velocity for onset of FEI: It has been suggested by Pettigrew and Taylor[9,10] that for design purposes, Connors' equation (2) could be used to determine the critical velocity for onset of FEI by assuming $\boldsymbol{K} = \mathbf{3}$ and $\boldsymbol{b} = \mathbf{0.5}$. Pettigrew and Taylor[9] also noted that value of $\mathbf{3}$ was recommended for tube bundles with $\boldsymbol{P/D} > 1.47$. However, for $\boldsymbol{P/D}$ less than $\mathbf{1.47}$, they recommended the correlation

$$\boldsymbol{K} = \mathbf{4.76}\frac{\boldsymbol{P} - \boldsymbol{D}}{\boldsymbol{D}} + \mathbf{0.76} \tag{12}$$

for $\boldsymbol{P/D}$ of $\mathbf{1.33}$, eq. (12) gives a value of about $\mathbf{2.3}$ for $\boldsymbol{K}$. Mohany et al[xvi] have reported instability data for multi-span U-tubes under two phase flow of Freon. They conclude that for out-of-plane instability a value of $\boldsymbol{K} = \mathbf{3}$ represents the lower bound of data for tubes with $\boldsymbol{P/D} = \mathbf{1.5}$. We assume that value of $\boldsymbol{K} = \mathbf{3.0}$ is a realistic value for out-of-plane instability in RSGs. It should be noted that in obtaining critical velocity in the U-bend region, only components of velocity normal to the tube need to be considered.

Table 2 lists the values of $\boldsymbol{\zeta}$ and the dimensionless critical velocity for onset of FEI corresponding to expected thermal hydraulic conditions on the secondary side of RSGs in Units 2 and 3.

Table 2

$\propto$	$\rho, \frac{lbm}{ft^3}$	$\frac{m}{\rho D^2}$	ζ_v	ζ_{tp}	ζ_s	ζ	$\frac{U_{cr}}{f_n D}$
0	47.6	3.74	1.36×10^{-3}	0	-	0.136×10^{-2}	0.54
0.2	38.5	4.35	1.28×10^{-3}	1.67×10^{-2}	-	1.8×10^{-2}	2.1
0.4	29.3	5.31	1.15×10^{-3}	3.33×10^{-2}	-	3.45×10^{-2}	3.22
0.6	19.8	7.28	0.97×10^{-3}	3.33×10^{-2}	-	3.43×10^{-2}	3.76
0.8	11.1	12.08	0.7×10^{-3}	2.33×10^{-2}	-	2.4×10^{-2}	4.04
1.0	1.9	61.5	0.19×10^{-3}	0	-	0.019×10^{-2}	0.81

There is a significant uncertainty in the magnitude of squeeze film and anti-vibration bar support damping. Table 3 shows the dimensionless critical velocity when support damping of 1% and 2% are assumed.

Table 3

$\propto$	ζ_s	ζ	$\frac{U_{cr}}{f_n D}$	ζ_s	ζ	$\frac{U_{cr}}{f_n D}$
0	2×10^{-2}	2.14×10^{-2}	2.13	1×10^{-2}	1.14×10^{-2}	1.55
0.2	2×10^{-2}	3.8×10^{-2}	3.06	1×10^{-2}	2.8×10^{-2}	2.26
0.4	2×10^{-2}	5.45×10^{-2}	4.04	1×10^{-2}	4.45×10^{-2}	3.65
0.6	2×10^{-2}	5.43×10^{-2}	4.73	1×10^{-2}	4.43×10^{-2}	4.27
0.8	2×10^{-2}	4.4×10^{-2}	5.48	1×10^{-2}	3.4×10^{-2}	4.81
1.0	2×10^{-2}	2.02×10^{-2}	8.38	1×10^{-2}	1.02×10^{-2}	5.95

The calculated dimensionless critical velocity for onset of FEI is plotted in Fig. 1 as a function of void fraction, with or without support damping. In this figure the expected two phase mixture velocity through the steam generator (RSG) tube bundle is also plotted as a function of $\propto$ for an assumed two phase mass flow rate of $\mathbf{65lbm / ft^2/sec}$ or $\mathbf{315kg/m^2/sec}$ and natural frequency of 34 $sec.^{-1}$ It is noted that for void fractions less than 0.8 in the U-bend region of the bundle (for cross flow), the critical velocity for onset of instability is higher than the mixture velocity. As such probability of onset of FEI will be low. However, for void fractions greater than 0.8, the mixture velocity can be substantially higher than the critical velocity, indicating a strong possibility of occurrence of FEI. In RSGs for Units 2 and 3, there is a significant area in the upper U-bend region where calculated void fractions exceed 0.8. When support damping of 1% and 2% is included, the predicted dimensionless critical velocity for onset of FEI increases for all values of $\propto$. However, the increase is substantial for high void fractions (greater than 0.8) because now the support damping dominates viscous and two phase damping. With support damping of 1%, the two phase mixture velocity can exceed the critical velocity for onset of FEI when the void fraction approaches unity. (Although this is not the case for the results plotted in Fig. 1 when support damping of 2% is included in the calculations.) Thus to reduce the propensity of FEI it is essential that very high void fraction regions ($\propto \simeq \mathbf{1}$) in the upper U-bend region of RSGs should be avoided. Figure 1 can be used to determine for individual tubes or clusters of tubes the potential for FEI in Units 2 and 3 RSGs. It should also be noted that lack of contact forces and damping due to anti-vibration bars may be the dominant reason for existence of FEI and severe tube damage in Unit 3 RSGs. As a result of FEI, initially out-of-plane vibrations are expected to occur in the lift direction normal to plane of the tubes. However, this instability in the absence of restraining force from anti-vibration bars can excite in-plane vibration that can be the cause of tube to tube damage.

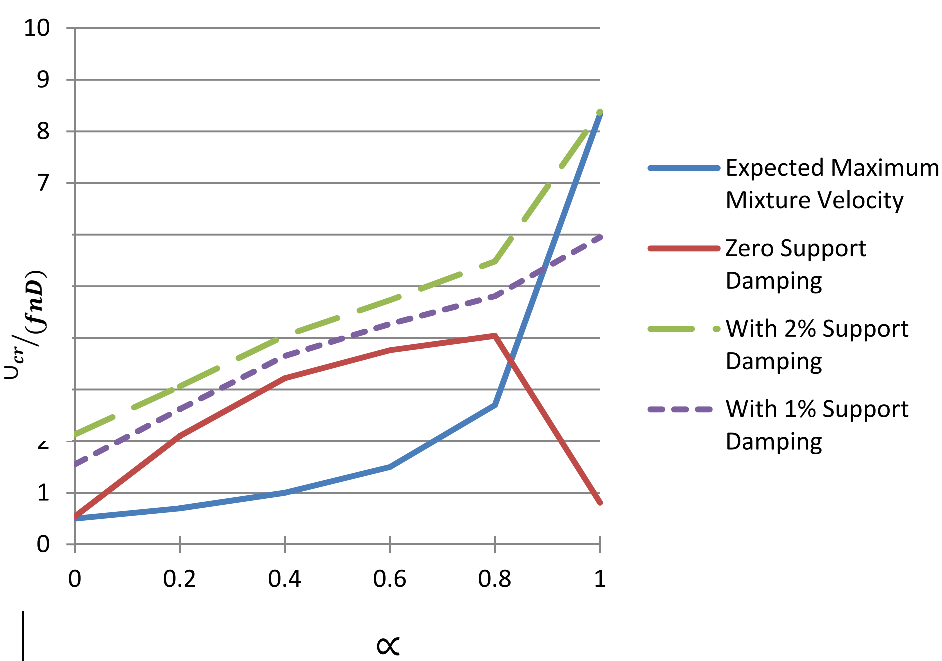


Fig. 1 Critical velocity for onset of FEI as a function of void fraction.

B. MHI methods have included assumption of certain number of inactive (non-contact) anti-vibration bar supports. This lack of engagement in combination with the “flowering” phenomena is believed to be the primary cause of wear in Unit 3. Comment on basis of these causal factors.

Critical velocity for onset of FEI is directly proportional to the natural frequency of the tube. Natural frequency of the tube depends on the manner it is supported (e.g., fixed) at the ends, span length between intermediate supports and the extent (size of gaps between tubes and supports) of the intermediate supports. An expression for natural frequency of vibration normal to the axis of a tube simply supported at the two ends as noted earlier is given as

$$f_n = \frac{n^2\pi}{2L}\sqrt{\frac{EI}{mL^2}} \qquad (13)$$

Thus we see that natural frequency of the tube will decrease as the unsupported length increases. Consequently onset of FEI of tubes of longer unsupported span will occur at lower fluid velocity. Gap between anti-vibration bars and tubes will limit the amplitude of tube vibrations in the lift direction (out-of-plane). This is expected to be the direction in which FEI will first occur. Existence of non-contact anti-vibration bars will lead to increased span length for tube vibration and in turn in reduced natural frequency. Since critical velocity is proportional to frequency, a corresponding reduction in the fluid gap velocity required for FEI will occur. Consequently in the presence of non-contact anti-vibration bars, certain number of tubes in the U-bend region where void fractions and superficial velocities are high will be prone to early FEI. It should be noted that liquid film between tubes and non-contact anti-vibration bars will provide some damping due to squeezing of the film for out-of-plane vibrations. When the amplitude of out-of-plane vibrations is such that tubes hit the anti-vibration bars, tube contact with anti-vibration bars can lead to addition of new nodes and reduced span length accompanied by increase in natural frequency. For in-plane (drag direction) vibrations, sliding of tubes directly over anti-vibration bars or over a liquid film will provide additional damping but it is expected to be much less than that for squeeze film damping in tube supports.

In summary the presence of non-active anti-vibration bars in Units 2 and 3 and repeated impact of tubes with anti-vibration bars can be a significant cause of tube wear in Units 2 and 3. The ‘flowering’ phenomena can occur due to differential thermal stresses but is not expected to be the main cause of severe damage to the tubes.

C. Provide some assessment of why Unit 2 and Unit 3 RSGs have behaved so differently? Do you concur with the premise of anti-vibration bars contact forces not being sufficient on Unit 3?

I see two possibilities. One is that anti-vibration bar gap is large for Unit 3 and as a result out of plane vibration of tubes occurs unrestrained until the magnitudes of vibrations become large. As discussed in item B, lack of contact leads to reduced natural frequency and in turn, lower fluid velocity for onset of FEI. Lack of contact force between tubes and anti-vibration bars and smaller damping due to the presence of anti-vibration bars can be the major cause of out-of-plane and consequently in-plane large amplitude vibrations in Unit 3. Other factor that can contribute is larger gap between tubes and TSPs.

The second possible cause, though much less probable, is the maldistribution of secondary flow resulting from non-uniformities in primary flow. Early onset of FEI in one part of the tube bundle can lead to tube instability in other parts. If the fluid velocity exceeds the critical velocity, the amplitude of vibration of tubes in the lift direction (out-of-plane) can substantially increase causing the tubes to hit noncontact anti-vibration bars. At onset of instability tubes would start to vibrate in the out-of-plane mode. Large magnitude out-of-plane vibrations in turn can trigger in-plane instability. Persistence of this condition over a long period of time can lead to thinning of tube wall with eventual tube failure. Thus FEI is considered to be the major cause of wear in Unit 3. In-plane flowering of the tubes due to thermal stresses is not considered to be the primary cause of tube failure. It may have exacerbated the potential for damage due to large amplitude vibrations.

As discussed in section A, contribution of support damping to total damping can be comparable to that from two phase mixture. Support damping includes squeeze film damping, viscous shear damping and solid to solid friction damping. Presence of any or all of these contributors and their interactions can affect the magnitude of support damping. As such a large scatter is seen in the data that have been correlated by Pettigrew *et al*[11]. However, one should be careful in using the correlation to support geometries and configurations that prevail in a given situation. For example, it is expected that magnitude of damping from anti-vibration bars will be less than that due to TSPs. Nevertheless, lack of anti-vibration bar support (no contact) force and little support damping and larger gaps between TSP and tubes in RSG 3 could have caused it to behave differently than RSG 2.

D. Review MHI root cause report and provide an assessment of the report findings and conclusions focusing on anti-vibration bar contact forces (damping impacts associated with Connors' equation) due to manufacturing differences and modeling errors.

As has been discussed earlier in connection with Item A, correlations for squeeze film damping have been developed from data from tubes passing through drilled and broached holes and egg crates. The geometrical configuration of tubes passing through holes is very different than that of a tube placed adjacent to a flat surface of anti-vibration bars. Because of the geometrical and orientation differences, the magnitudes of squeeze film and viscous shear damping in anti-vibration bars are expected to be different. For out-of-plane vibration of tubes in RSGs, squeeze film damping will occur but its magnitude has not yet been quantified. As the tubes' vibration amplitude equals the gap between tubes and anti-vibration bars, the tubes will start to impact the anti-vibration bars. This in turn will cause damage to the tubes and lead to inactive anti-vibration bars becoming active and creating additional vibrational nodes. For in-plane movement the damping due only to viscous shear caused by sliding of tubes against a liquid film between tubes and anti-vibration bars will take place. In this case, little contribution will come from squeeze film damping. One could theoretically model viscous shear contribution which amongst other variables will depend on the width of anti-vibration bars. However, any such model predictions must be validated under prototypical conditions.

E. Impact of recent testing by (CANDU Services 147-02120-505452-371-9001) AECL on the damping due to anti-vibration bars and return to service of SONGS RSGs.

In response to the questions that we (NRC) asked in connection with SONGS application for return to service of Unit 2, SONGS requested AECL to carryout tests to document the vibrational damping introduced by anti-vibration bars. Two key questions we had raised were:

1. What is the rationale for applying squeeze film damping correlations developed based on the data for circular supports to flat anti-vibration bars?

2. In the absence of any supporting data, what is the rationale for extrapolating squeeze film damping correlations for circular supports to tube frequencies less than 30Hz?

Tests for damping factors for circular supports and anti-vibration bars have been performed by AECL. In these tests prototypical single tubes of lengths varying from 4.2 meters to 1.5 meters have been used. The tubes have been placed in air or a pool of water at about room temperature and are externally excited. In the tests both in-plane and out-of-plane vibrational modes have been investigated. The width of gaps between tubes and supports has been varied parametrically. Damping factors with point values have been reported and the key observations from review of the data are:

1. The damping factors observed for squeeze film damping in circular supports are found to be a function of amplitude of vibrations and generally increase with increase in amplitude. For amplitude to gap width ratio of about 60%, the observed damping factors for tube frequencies of 31.9 and 8.7 and gap radial widths of 0.75 and 0.38mm are less than 1% and are also less than

those obtained from the correlations reported in the literature and used by SONGS. Although limited in scope, damping factors for a given gap width do not scale inversely with the frequency. One should note that these damping factors have been obtained with water at an atmospheric pressure and at about room temperature. During RSG operation pressure and temperature will be much higher. With increase in temperature liquid viscosity will decrease and as a result squeeze film damping factors may decrease further from those measured at room temperature.

2. For in-plane vibrations in the presence of anti-vibration bars, the damping is found to increase marginally. There is significant scatter in the data. The data show an increase in damping with frequency and decrease with gap size. For the highest frequency (31.9 $\mathbf{Hz}$) and smallest gap (0.0025 mm), the presence of anti-vibration bars in water leads to a maximum increase in damping factor due to sliding of the tube parallel to anti-vibration bars of less than 0.1 – 0.15%.

3. In experiments with out-of-plane vibrations in which vibrating tube pushes against the anti-vibration bars and in turn acts to squeeze out the intervening liquid film, damping factors have been observed to be higher than those in which tubes were only excited in-plane. Measurements were made for gap widths of 0.125mm, 0.025mm and 0.0025mm. For 0.125mm, the highest value of 0.45% for increase in damping factor was noted for RMS displacement to gap ratio of 1.4 and tube frequency of 4.5 $\boldsymbol{Hz}$. Tube with a natural frequency of 8.7 $\mathbf{Hz}$ yielded an increase of about 0.18% in damping ratio for RMS displacement/gap ratio of 2. For a gap of 0.0025mm, an increase in damping ratio of about 3% was observed for a tube with 4.54 $\mathbf{Hz}$ natural frequency and RMS displacement of 0.16mm. However, the data for the three gaps studied do not show a consistent trend with tube frequency. These data show that when out-of-plane motion exists, damping due to anti-vibration bars is higher than that for purely in-plane vibrations. However, in most cases, additional damping due to anti-vibration bars is less than 1%.

4. Damping factors, friction force and friction coefficient data have been reported when tubes are pre-loaded. Damping factors have been found to increase with pre-load as friction force between tubes and anti-vibration bars increases. The damping factors with friction have been found to increase as tube vibration frequency is decreased but decrease as the amplitude of vibrations is increased.

5. In terms of consequences of the above described experimental effort on the discussion in the earlier parts of the report, it is concluded that for in-plane tube vibrations, anti-vibration bars contribute little to additional damping. However, for out-of-plane vibrations, the additional damping (squeeze film) due to anti-vibration bars is higher. Although data show significant scatter, in most cases in the absence of any physical contact between tube and anti-vibration bars and any preload, the additional damping (squeeze film) due to anti-vibration bars is less than 1%. Furthermore this does not account for the reduction of viscosity of water with increase in temperature. Thus in Figure 1, the predicted critical velocity curve for 1% support damping seems to be more appropriate. Accordingly, with the information we have at this point, a certain number of tubes in the upper portion of the U-tube bundle could experience FEI. However, the exact number of tubes experiencing FEI would have to be determined by knowing the predicted flow velocities, frequencies and void fraction in each region of interest at 70% power.

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PRELIMINARY SIGNIFICANCE DETERMINATION
LOSS OF STEAM GENERATOR TUBE INTEGRITY

<u>Assumptions</u>

1. Unit 3 operated for 11.3 effective full power months with newly-installed steam generators prior to the tube leak. Using a t/2 assumption, the exposure period is assumed to be 5.65 months, or 172 days.

2. The risk of the condition consists of two elements:

   a) The increased frequency for a steam generator tube rupture, given the degraded state of the tubes.

   b) A main steam line break could have occurred during the exposure period, resulting in one or more tubes rupturing, resulting in a compound accident.

3. All steam line breaks that hypothetically could have occurred during the exposure period are assumed to result in a steam generator tube rupture, given no operator actions. Operators can take immediate actions according to their procedures that would mitigate the differential pressure across the tubes. These actions are primarily to prevent a re-pressurization of the RCS by cooling down using the intact steam generator and eventually shutting off any operating charging pumps. The probability that these actions will fail to preclude a tube rupture was assessed using the SPAR-H methodology, as follows:

	Diagnosis (0.01)	Action (0.001)
Available Time	Nominal (1.0)	Time available = time required (10)
Complexity	Moderate (2)	Nominal (1.0)
Stress	High (2.0)	High (2.0)
Procedures	Nominal (1.0)	Nominal (1.0)
	0.04	0.02
Total HEP[1]	0.06	

Note 1: human error probability

All other performance shaping factors are assumed to be nominal.

4. It is assumed that a steam generator tube rupture that results in core damage will always result in a large early release because all of the radiation barriers, including containment, are bypassed. This is a bounding assumption because there are some actions the plant can take that would limit the release to levels that would not meet the definition of a large, early release.

   Large early release frequency is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is the potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.

5. The degraded condition of the steam generator tubes was first manifested by a small leak that had little effect on core damage or large early release. However, there was a possibility that a tube rupture could have occurred as the first manifestation of a problem. For this analysis, it is assumed that the frequency of a steam generator tube rupture was doubled from the baseline value for the 172-day exposure period. The baseline value is 2.07E-3/year (1 in 483 years) and it was doubled to 4.1E-3/year (1 in 241 years).

6. The San Onofre SPAR model, Revision 8.22, was used for this analysis, assuming average test and maintenance, and a truncation limit of 1.0E-11.

<u>Analysis</u>

Steam Line Break/Steam Generator Tube Rupture

During the 172-day exposure period, if a steam line break had occurred, it is assumed that a steam generator tube rupture would have occurred concurrently, given no operator action. According to Assumption # 3 above, the probability that operator actions would fail to preclude a tube rupture is 0.06. If the steam line break does not cause a tube rupture, it is a baseline event having no significance to this finding.

It is most likely that only one tube would have ruptured, given that two of the three degraded tubes were marginally close to passing the test. This places the thermo-hydraulic considerations consistent with a steam generator tube rupture event.

The SPAR model event tree for main steam line breaks contains 8 sequences. Sequences 3, 4, 6, and 7 are core damage sequences. Sequences 1, 2, and 5 are “OK” sequences. Sequence 8 is a transfer to an anticipated transient without scram (ATWS) event tree.

The core damage sequences and the ATWS transfer were quantified with a result of 1.61E-6/yr. The delta-CDF for these sequences is zero (baseline) for this finding because core damage would have occurred even without the performance deficiency, but because of the possibility of a concurrent steam generator tube rupture, the delta-LERF must be considered. The baseline LERF for a steam line break is zero for this plant’s containment structure, so any core damage resulting from a concurrent steam generator tube rupture constitutes a delta-LERF. According to the HEP developed in Assumption #3, the delta-LERF for the core damage sequences is 1.61E-6 (0.06) = 9.65E-8/yr. For a 172-day exposure, this results in an ICLERP of 4.55E-8.

The “OK” sequences of the main steam line break sequences add to both the delta-CDF and the delta-LERF because they can result in a tube rupture with a probability of core damage from that situation independently. Therefore, main steam line break sequences 1, 2, and 5 were transferred to the steam generator tube rupture event tree. These sequences were quantified as 0.3825, 0.1275, and 0.3675 respectively, multiplied by the steam line break initiating event frequency of 7.7E-3.

The “OK” sequences of the main steam line break event tree all include a success of either main feedwater or auxiliary feedwater. Therefore, the core damage sequences of the steam generator tube rupture event tree that involve a failure of feedwater were not quantified for this portion of the analysis. This left steam generator tube rupture core damage sequences #s 3, 6, 7, 14, 15, 17, 18, and 20 to be quantified, with a result of a CCDP of 2.0E-4.

The same operator recovery of an HEP of 0.06 applies in this case.

The delta-CDF and delta-LERF resulting from a main steam line break that would normally have not resulted in core damage is therefore:

(0.3855 + 0.125 + 0.3675) (7.7E-3) (0.06) (2.0E-4) = 8.10E-8/yr. For a 172-day exposure, this results in an ICCDP and ICLERP of 3.82E-8.

The following table presents the total risk resulting from main steam line breaks:

	ICCDP	ICLERP
Steam line breaks that directly result in core damage	0	4.55E-8
Steam line breaks that would normally not result in core damage, but can cause a steam generator tube rupture	3.82E-8	3.82E-8
TOTAL	3.82E-8	8.37E-8

Independent Steam Generator Tube Rupture

According to Assumption #5, it is assumed that the frequency of a steam generator tube rupture was twice the normal rate during the 172-day exposure period. This is not based on an analytical analysis, but represents a reasonable estimate.

The SPAR model nominal case delta-CDF for a steam generator tube rupture is 4.2E-7/yr. Therefore, the delta-CDF and delta-LERF resulting from independent steam generator tube ruptures would also be 4.2E-7. For a 172-day exposure period, the Delta-CDF and Delta-LERF was 2.0E-7.

Total Internal Risk

	Delta-CDF	Delta-LERF
Main Steam line Breaks	3.8E-8	8.4E-8
Independent Steam Generator Tube Ruptures	2.0E-7	2.0E-7
Total Internal Risk	2.4E-7	2.8E-7
Risk Significance	Green	White

The finding was green for the increase in core damage, but white for the increase in large early release.

External Events: Core damage frequency for high winds, floods, was not quantified in licensee's IPEEE because these events met the NRC screening criteria.

Seismic and fire events were possible contributors to CDF and LERF. However, the licensee's IPEEE stated that the median capacity of the steam generators was 8g. This was substantial when compared to other plant components. The design bases earthquake for SONGS was 0.67g.

The licensee's risk assessment stated, in part:

The potential for a seismically induced steam generator tube rupture was evaluated as part of the overall safety significance of the Unit 3 event. The deterministic analysis concluded that the most limiting degraded tube on Unit 3 would have been able to withstand a design basis earthquake based on in-situ test results. Therefore, the condition of the tubes was not considered an additional risk contributor in the PRA. More information on the seismic tube analysis is provided in Section 5.2.1 of San Onofre's Return to Service Report (which is enclosure 2 to San Onofre's Confirmatory Action Letter Response Letter).

The best estimate for the change to CDF and LERF from external events was 0.

External Events Sensitivity Studies: To evaluate fire and seismic events to the extent practicable, the analysts performed two risk evaluations considering different scenarios. The first considered the potential for a seismic induced tube rupture caused by the weakened tubes. Since fire scenarios would not directly cause a weakened tube to rupture, fire was not a contributor in this evaluation. The second evaluation increased the post core damage risk associated with the effect of a hot gas layer on the weakened tubes. Seismic and fire scenarios were considered here. Both evaluations required the use of significant judgment when specific statistics were lacking. Both evaluations were very conservative and bounding.

Seismic Induced Tube Rupture: The analyst considered an increase in core damage and large early release from a seismic induced tube rupture. The rupture is caused by vibrating the steam generators and rupturing a weakened portion of a tube.

Using the SPAR model, the CCDP of a steam generator tube rupture combined with an unrecoverable loss of offsite power (assumed in a large seismic event) was 1.64E-3.

The analyst qualitatively considered that the fragility of the degraded tubes would be at the approximate g-level earthquake that causes a 25% chance of a small break loss of coolant accident. This figure was 0.7 g, according to NUREG/CR-4840, Figure 3-6. The frequency of an earthquake of this magnitude or greater at San Onofre is 1.0E-4/yr.

Therefore, the delta-CDF and delta-LERF from a seismic event is approximately 1.6E-3 (1.0E-4/yr) = 1.6E-7/yr. For a 172-day exposure, the delta-CDF and delta-LERF was 7.7E-8.

Post-Core-Damage Hot Gas Layer: The SONGS "Individual Plant Examination of External Events (IPEEE)," dated December 15, 1995, reported the core damage frequency (CDF) for fire and seismic external events at SONGS was 3.3E-5 per year. The seismic contribution was 1.7E-5/yr. The seismic risk included accelerations beyond the SONGS design basis earthquake of 0.67 g peak ground acceleration.

The SONGS containment and pressure boundary response was reported in the IPEEE (Table 3.7-1). The SGTR can occur post-core-damage when the hot gas layer migrates to the steam generator tubes. The high reactor coolant system pressure coincident with the heat weakened tubes results in the tube failure.

The analyst qualitatively assumed that the degraded steam generator tubes would make the tube failure twice as likely as assumed in the licensee's IPEEE (from 2% to 4%).

The increase in LERF would be the total external event core damage frequency multiplied by the change in the probability of SGTR duration fraction, and the fraction of exposure time.

Delta-LERF = 3.3E-5 X (.04 -.02) X 0.47 = 3.1E-7/yr

Total Risk, Best Estimate

	Delta-CDF	Delta-LERF
Internal	2.4E-7	2.8E-7
External	0	0
Total	2.4E-7	2.8E-7
Classification	Green	White

Sensitivity for External Events – Vibration Induced Tube Rupture

	Delta-CDF	Delta-LERF
Internal	2.4E-7	2.8E-7
External	7.7E-8	7.7E-8
Total	3.1E-7	3.6E-7
Classification	Green	White

Sensitivity for External Events – Hot Gas Layer Failures

	Delta-CDF	Delta-LERF
Internal	2.4E-7	2.8E-7
External	3.1E-7	3.1E-7
Total	5.5E-7	5.9E-7
Classification	Green	White

Licensee Analysis

The licensee analysis was presented in PRA Report PRA-12-007, and used a different analytical technique using studies performed by a contractor used to predict a tube failure probability as a function of time, assuming linear degradation. The result was a Delta-LERF of 1.4E-7. This is also a White significance.