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October 3, 2012

Elmo E. Collins, Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 East Lamar Blvd.
Arlington, Texas 76011-4511

Subject: **Docket No. 50-361**
Confirmatory Action Letter - Actions to Address Steam Generator Tube Degradation
San Onofre Nuclear Generating Station, Unit 2

References:
1. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated March 23, 2012, Steam Generator Return-to-Service Action Plan, San Onofre Nuclear Generating Station
2. Letter from Mr. Elmo E. Collins (USNRC) to Mr. Peter T. Dietrich (SCE), dated March 27, 2012, Confirmatory Action Letter 4-12-001, San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation

Dear Mr. Collins:

On March 23, 2012, Southern California Edison (SCE) submitted a letter (Reference 1) to the NRC describing actions it planned to take with respect to issues identified in the steam generator (SG) tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. On March 27, 2012, the NRC responded by issuing a Confirmatory Action Letter (CAL) (Reference 2), describing the actions that the NRC and SCE agreed would be completed to address those issues and ensure safe operations. The purpose of this letter is to report the completion of the Unit 2 CAL actions, which are to be completed prior to entry of Unit 2 into Mode 2 (as defined in the SONGS technical specifications).

Completion of the Unit 2 CAL actions is summarized below. Detailed information demonstrating fulfillment of Actions 1 and 2 of the CAL is provided in SCE's Unit 2 Return to Service Report which is included as Enclosure 2 of this letter. Enclosure 1 provides a list of new commitments identified in this letter.

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CAL ACTION 1:

"Southern California Edison Company (SCE) will determine the causes of the tube-to-tube interactions that resulted in steam generator tube wear in Unit 3, and will implement actions to prevent loss of integrity due to these causes in the Unit 2 steam generator tubes. SCE will establish a protocol of inspections and/or operational limits for Unit 2, including plans for a mid-cycle shutdown for further inspections."

COMPLETION OF CAL ACTION 1:

SCE has determined the causes of tube-to-tube interactions that resulted in SG tube wear in Unit 3, as summarized below. In addition, SCE implemented actions to prevent loss of tube integrity due to these causes in the Unit 2 SGs and established a protocol of inspections and operational limits, including plans for a mid-cycle shutdown. These are summarized under CAL Action 2.

Causes of Tube-to-Tube Interactions in Unit 3

As noted in Reference 1, the SG tube wear that caused a Unit 3 SG tube to leak was the result of tube-to-tube interaction. This type of wear was confirmed to exist in a number of other tubes in the same region in both Unit 3 SGs. Subsequent inspections of the Unit 2 SGs found this type of wear also existed in a single pair of tubes (one contact location) in one of the two Unit 2 SGs (SG 2E-089).

To determine the cause of the tube-to-tube wear (TTW), SCE performed extensive inspections and analyses, and commissioned the assistance of experts in the fields of thermal-hydraulics and in SG design, manufacturing, operation, and maintenance. Based on the results of these inspections and analyses, SCE determined the cause of the TTW in the two Unit 3 SGs was fluid elastic instability (FEI), resulting from the combination of localized high steam velocity, high steam void fraction, and insufficient contact forces between the tubes and the anti-vibration bars (AVBs). The FEI caused vibration of SG tubes in the in-plane direction that resulted in TTW in a localized area of the SGs. Details of SCE's investigation and cause evaluation are provided in Section 6 of Enclosure 2.

Corrective and Compensatory Actions, Inspections, and Operational Limits

To prevent loss of integrity due to FEI and TTW in Unit 2, SCE implemented corrective and compensatory actions and established a protocol of inspections and operational limits, including plans for a mid-cycle shutdown. These are described in CAL Action 2 below.

CAL ACTION 2:

"Prior to entry of Unit 2 into Mode 2, SCE will submit to the NRC in writing the results of your assessment of Unit 2 steam generators, the protocol of inspections and/or operational limits, including schedule dates for a mid-cycle shutdown for further inspections, and the basis for SCE's conclusion that there is reasonable assurance, as required by NRC regulations, that the unit will operate safely."

COMPLETION OF CAL ACTION 2:

Assessment of Unit 2 Steam Generators

SCE evaluated the causes of TTW in the Unit 3 SGs and the applicability of those causes to Unit 2 and inspected the Unit 2 SGs for evidence of similar wear. SCE determined the TTW effects were much less pronounced in Unit 2 where two adjacent tubes were identified with TTW indications. The wear depth was less than 15% through-wall wear, which is below the threshold of 35% through-wall at which tube plugging is required. These two tubes are located in the same region of the SG as those with TTW in Unit 3. Given that the thermal hydraulic conditions are essentially the same in both units, the significantly lower level of TTW in Unit 2 has been attributed to manufacturing differences that resulted in greater contact between the tubes and AVBs in Unit 2, providing greater tube support. Details of SCE's investigation and cause evaluation are provided in Section 6 of Enclosure 2.

Actions to Prevent Loss of Integrity due to TTW in Unit 2 SG Tubes Including Protocol of Inspections and Operational Limits

SCE has taken actions to prevent loss of Unit 2 SG tube integrity due to TTW including establishing a protocol of inspections and operational limits to provide assurance that Unit 2 will operate safely. These actions are summarized below, with details provided in Section 8 of Enclosure 2. The operational assessments performed to confirm the adequacy of these operational limits are described in Section 10 of Enclosure 2.

1. SCE will administratively limit Unit 2 to 70% reactor power prior to a mid-cycle shutdown (Commitment 1). Limiting Unit 2 power to 70% eliminates the thermal hydraulic conditions that cause FEI from the SONGS Unit 2 SGs by reducing the steam velocity and void fraction. Further, at 70% power, the SONGS Unit 2 SGs will operate within an envelope of steam velocity and void fraction that has proven successful in the operation of other SGs of similar design. Thus, limiting power to 70% ensures that loss of tube integrity due to FEI will not occur.

2. SCE plugged the two tubes with TTW in Unit 2. As a preventive measure, additional tubes were plugged in the Unit 2 SGs. Tubes were selected for preventive plugging using correlations between wear characteristics in Unit 3 tubes and actual wear patterns found in Unit 2 tubes. Removing these tubes from service will prevent any further wear of these tubes from challenging tube integrity.

3. SCE will shut down Unit 2 for a mid-cycle SG inspection outage within 150 cumulative days of operation at or above 15% power (Commitment 2). This shortened inspection interval will ensure that any potential tube wear will not challenge the structural integrity of the in-service tubes. The protocol for mid-cycle inspections is provided in Section 8.3 of Enclosure 2.

To ensure that these actions are effective in preventing a loss of tube integrity due to FEI, SCE retained the experience and expertise of AREVA NP, Westinghouse Electric Company LLC, and Intertek/APTECH. These companies routinely perform operational assessments (OAs) of SGs for the U.S. nuclear industry. AREVA and Westinghouse also have extensive steam generator design experience. SCE retained these companies to develop independent OAs using different methodologies to evaluate whether, under the operational limits imposed by SCE, SG tube integrity will be maintained until the next SG inspection. Each of these independent OAs demonstrates that operating at 70% power will prevent loss of tube integrity beyond the 150 cumulative day inspection interval.

The actions to operate at reduced power and shut down for a mid-cycle inspection within 150 cumulative days of operation are interim compensatory actions. SCE will reevaluate these actions during the mid-cycle inspection based on the data obtained during the inspections. In addition, SCE has established a project team to develop and implement a long term plan for repairing the SGs.

Defense-in-depth measures were developed to provide increased safety margin in the unlikely event of tube-to-tube degradation in the Unit 2 SGs during operation at 70% power. These actions, identified in Section 9 of Enclosure 2, will facilitate early detection of a SG tube leak and ensure immediate and appropriate plant operator and management response.

Basis for Conclusion of Reasonable Assurance

SCE has evaluated the causes of TTW in the Unit 3 SGs and, as described in response to CAL Action 2 above, has completed corrective and compensatory actions in Unit 2 to prevent loss of tube integrity due to these causes. Tubes within regions of the Unit 2 SGs that might be susceptible to FEI have been plugged. In addition, as described in response to CAL Action 2 above, SCE has established operational limits that eliminate the thermal-hydraulic conditions associated with FEI from the SONGS Unit 2 SGs. Specifically, operation of Unit 2 will be administratively limited to 70% power. Within 150 cumulative days of operation at or above 15% power, Unit 2 will be shut down for inspection to confirm the condition of the SG tubes. The analyses and OAs performed by SCE and independent industry experts demonstrate that under these conditions, tube integrity will be maintained. On this basis, SCE concludes that Unit 2 will operate safely.

We understand that the NRC will conduct inspections at SONGS to confirm the bases for the above information.

Please call me or Mr. Richard St. Onge at (949) 368-6240 should you require any further information.

Sincerely,

Enclosures: 1. List of Commitments
2. Unit 2 Return to Service Report

cc: NRC Document Control Desk
R. Hall, NRC Project Manager, San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, Region IV

ENCLOSURE 1

List of Commitments

Enclosure 1
List of Commitments

This table identifies actions discussed in this letter that Southern California Edison commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

	Description of Commitment	Scheduled Completion Date
1	Prior to a mid-cycle shutdown of Unit 2, SCE will administratively limit operation of Unit 2 to 70% power (refer to cover letter, Completion of CAL Action 2).	mid-cycle shutdown of Unit 2
2	SCE will shut down Unit 2 for a mid-cycle steam generator (SG) inspection outage. During this outage, inspections of Unit 2 SG tubes will be performed to confirm the effectiveness of the corrective and compensatory actions taken to address tube-to-tube wear in the Unit 2 SGs. (refer to cover letter, Completion of CAL Action 2).	within 150 cumulative days of operation at or above 15% power
3	SCE will install a temporary N-16 radiation detection system (refer to Enclosure 2, Section 9.2). The temporary N-16 detectors will be located on the Unit 2 main steam lines and be capable of detecting an increase in steam line activity.	prior to Unit 2 entry into Mode 2
4	SCE Plant Operators will receive training on use of the new detection tools for early tube leak identification and on lessons learned from response to the January 31, 2012, Unit 3 shutdown due to a steam generator (SG) tube leak (refer to Enclosure 2, Section 9.4.2).	prior to Unit 2 entry into Mode 2
5	SCE will upgrade the Unit 2 Vibration and Loose Part Monitor System (refer to Enclosure 2, Section 11.1). The new system will provide additional monitoring capabilities for steam generator secondary side noise.	prior to Unit 2 entry into Mode 2
6	SCE will install analytic and diagnostic software (GE Smart Signal) utilizing existing instrumentation (refer to Enclosure 2, Section 11.2).	prior to Unit 2 entry into Mode 2

ENCLOSURE 2

San Onofre Nuclear Generating Station

Unit 2 Return to Service Report

[Proprietary Information Redacted]

SOUTHERN CALIFORNIA EDISON
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SAN ONOFRE NUCLEAR GENERATING STATION
UNIT 2 RETURN TO SERVICE REPORT
October 3, 2012

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
0	Entire Document	Initial Issue

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ATTACHMENTS

* [Proprietary Information Redacted]

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ABBREVIATIONS AND ACRONYMS

2E-088	Unit 2 Steam Generator E-088
2E-089	Unit 2 Steam Generator E-089
3E-088	Unit 3 Steam Generator E-088
3E-089	Unit 3 Steam Generator E-089
AILPC	Accident Induced Leakage Performance Criterion
Ar	Argon
ATHOS	Analysis of Thermal-Hydraulics of Steam Generators
AVB	Anti-Vibration Bar
CDP	Core Damage Probability
CM	Condition Monitoring
DA	Degradation Assessment
DID	Defense in Depth
ECT	Eddy Current Testing
EFPD	Effective Full Power Days
EPRI	Electric Power Research Institute
ETSS	Examination Technique Specification Sheet
FEI	Fluid Elastic Instability
FIV	Flow Induced Vibration
FO	Foreign Object
FOSAR	Foreign Object Search and Retrieval
gpd	Gallons Per Day
INPO	Institute of Nuclear Power Operations
LERP	Large Early Release Probability
MHI	Mitsubishi Heavy Industries, Ltd.
MSLB	Main Steam Line Break
MWt	Megawatt Thermal
N-16	Nitrogen – 16
NEI	Nuclear Energy Institute
NODP	Normal Operating Differential Pressure
OA	Operational Assessment
OSG	Original Steam Generator
post-trip SLB	Steam Line Break Post-Trip Return-To-Power Event
PRA	Probabilistic Risk Assessment
RB	Retainer Bar
RCPB	Reactor Coolant Pressure Boundary
RCE	Root Cause Evaluation
RCS	Reactor Coolant System
Ref.	Reference
RSG	Replacement Steam Generator
SCE	Southern California Edison
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIPC	Structural Integrity Performance Criterion
SLB	Steam Line Break
SGP	Steam Generator Program
SONGS	San Onofre Nuclear Generating Station
SR	Stability Ratio
T/H	Thermal-Hydraulics
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
TSP	Tube Support Plate

TTW	Tube-to-Tube Wear
TW	Through Wall
TWD	Through Wall Depth
U2C17	Unit 2 Cycle 17
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
WEC	Westinghouse Electric Company

1.0 EXECUTIVE SUMMARY

On January 31, 2012, a leak was detected in a steam generator (SG) in Unit 3 of the San Onofre Nuclear Generating Station (SONGS). Southern California Edison (SCE) operators promptly shut down the unit in accordance with plant operating procedures. The leak resulted in a small radioactive release to the environment that was well below the allowable federal limits. Subsequently, on March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Ref. 1) to SCE describing actions that the NRC and SCE agreed must be completed prior to returning Units 2 and 3 to service.

To address the tube leak and its causes, SCE assembled a technical team including experts in the fields of thermal hydraulics (T/H) and in SG design, manufacture, operation, and maintenance. The team performed extensive investigations into the causes of the tube leak and developed compensatory and corrective actions that SCE has implemented to prevent recurrence of the tube-to-tube wear (TTW) that caused the leak. SCE also implemented defense-in-depth (DID) measures to provide additional safety margin. SCE has planned SG inspections following a shortened operating interval to confirm the effectiveness of its compensatory and corrective actions.

As required by the SONGS technical specifications (TSs), the SONGS Steam Generator Program (SGP), and industry guidelines, an Operational Assessment (OA) must be performed to ensure that SG tubing will meet established performance criteria for structural and leakage integrity during the operating period prior to the next planned inspection. Because of the unusual and unexpected nature of the SG TTW, SCE commissioned three independent OAs by experienced vendors. These vendors applied different methodologies to ensure a comprehensive and diverse evaluation. An additional OA was performed to evaluate SG tube wear other than TTW. Each of these OAs independently concluded that the compensatory and corrective actions implemented by SCE are sufficient to address tube wear issues so that the Unit 2 SGs will operate safely.

The purpose of this report is to provide detailed information demonstrating completion of CAL actions required prior to entry of Unit 2 into Mode 2. The report also describes in detail the basis for the conclusion that Unit 2 will continue to operate safely after restart.

This report describes:

- Results of inspections of the SG tubes
- Causes of the tube wear in the Unit 2 and Unit 3 SGs
- Compensatory and corrective actions that SCE has taken to address tube wear in Unit 2
- OAs that have been performed to demonstrate that those compensatory and corrective actions ensure that TTW will be prevented until the next SG inspections
- Additional controls and DID actions that SCE is implementing to ensure health and safety of the public in the unlikely event of a loss of SG tube integrity

1.1 Occurrence and Detection of the Unit 3 Tube Leak

New SGs were placed into service at SONGS Units 2 and 3 in 2010 and 2011, respectively. The replacement steam generators (RSGs) were installed to resolve corrosion and other degradation issues present in the original steam generators (OSGs). The RSGs were designed and manufactured by Mitsubishi Heavy Industries (MHI). On January 9, 2012, after 22 months of operation, Unit 2 was shut down for a routine refueling and SG inspection outage. This was the first inspection of the Unit 2 SG tubes performed following SG replacement. The condition monitoring (CM) assessment performed to evaluate the results of this inspection confirmed that the SG performance criteria were satisfied during the operating interval .

On January 31, 2012, while the Unit 2 outage was in progress, SONGS Unit 3 was operating at 100 percent power when a condenser air ejector radiation monitor alarm indicated a primary-to-secondary leak. Unit 3 was promptly shut down in accordance with plant operating procedures and placed in a stable cold shutdown condition. The TS limit for operational leakage (150 gallons per day (gpd)) was not exceeded during the event. A small, monitored radioactive release to the environment occurred, resulting in an estimated 0.0000452 mrem dose to the public. This estimated dose was well below the allowable federal limit specified in 10 CFR 20 of 100 mrem per year to a member of the public.

1.2 Inspections of the Steam Generator Tubes and Cause Evaluations of Tube Wear

Subsequent to the reactor cooldown, extensive inspection, testing, and analysis of SG tubes was performed in both Unit 3 SGs. This was the first inspection of the Unit 3 SG tubes performed following SG replacement after approximately 11 months of operation. The leak was identified in SG 3E-088 and was caused by TTW in the U-bend portion of the tube in Row 106 Column 78. Additional inspections revealed significant TTW in many tubes in Unit 3.

In accordance with SGP requirements for unexpected degradation, SCE initiated a cause evaluation of the TTW phenomenon. The Root Cause Evaluation (RCE) Team used significant input from the SG Recovery Team which included the services of MHI and industry experts in the fields of T/H and in SG design, manufacturing, operation, and repair. The mechanistic cause of the TTW in Unit 3 was identified as fluid elastic instability (FEI), caused by a combination of localized high steam velocity (tube vibration excitation forces), high steam void fraction (loss of ability to dampen vibration), and insufficient tube to anti-vibration bar (AVB) contact to overcome the excitation forces. The FEI resulted in a vibration mode of the SG tubes in which the tubes moved in the in-plane direction parallel to the AVBs in the U-bend region. This resulted in TTW in a localized region of the Unit 3 SGs.

Although no TTW had been detected during the routine inspections of all tubes in Unit 2, the unit was not returned to service pending an evaluation of the susceptibility of the Unit 2 SGs to the TTW found in Unit 3. In March 2012, as part of this evaluation, additional inspections using a more sensitive inspection method were performed on the Unit 2 tubes. Shallow TTW was identified between two adjacent tubes in SG 2E-089.

1.3 Compensatory, Corrective, and Defense-in-Depth Actions

SCE has implemented compensatory and corrective actions that will prevent loss of integrity due to TTW in Unit 2, including:

1. Limiting Unit 2 to 70% power prior to a mid-cycle SG inspection outage
2. Preventively plugging tubes in both SGs
3. Shutting down Unit 2 for a mid-cycle SG inspection outage within 150 cumulative days of operation at or above 15% power

SCE has also implemented conservative DID measures to provide an increased safety margin in the unlikely event of tube-to-tube degradation in the Unit 2 SGs during operation at 70% reactor power. Additionally, SCE has provided enhanced plant monitoring capability to assist in evaluating the condition of the SGs.

1.4 Operational Assessments

As required by the CAL (Ref. 1), SCE has prepared an assessment of the Unit 2 SGs that addresses the causes of TTW wear found in the Unit 3 SGs, prior to entry of Unit 2 into MODE 2.

Due to the significant levels of TTW found in Unit 3 SGs, SCE assessed the likelihood of additional TTW in Unit 2 from several different perspectives, utilizing the experience and expertise of AREVA NP, Westinghouse Electric Company, LLC (WEC), and Intertek/APTECH. Each of these companies routinely prepare OAs to assess the safety of operation of SGs at U.S. nuclear power plants. These companies developed independent OAs to evaluate the TTW found at SONGS and the compensatory and corrective actions being implemented to address TTW in the Unit 2 SGs. These OAs apply different methodologies to ensure a comprehensive and diverse evaluation. Each of these OAs concluded that the compensatory and corrective actions implemented by SCE are sufficient to address tube wear issues so that the Unit 2 SGs will operate safely. The results of these analyses fulfill the TS requirement to demonstrate that SG tube integrity will be maintained over the reduced operating cycle until the next SG inspection.

1.5 Conclusion

On the basis of the compensatory and corrective actions, DID actions, and the results of the OAs, SCE concludes that Unit 2 will operate safely at 70% power for 150 cumulative days of operation with substantial safety margin and without loss of tube integrity. Reducing power to 70% eliminates the thermal hydraulic conditions that cause FEI and associated TTW from the SONGS Unit 2 SGs. After this period of operation, Unit 2 will be shut down for inspection of the steam generator tubes to confirm the effectiveness of the compensatory and corrective actions that have been taken. SCE will continue to closely monitor steam generator tube integrity and take corrective actions as appropriate to ensure the health and safety of the public is maintained.

2.0 INTRODUCTION

On March 27, 2012, the NRC issued a CAL (Ref. 1) to SCE describing actions that the NRC and SCE agreed would be completed prior to returning Units 2 and 3 to service. The purpose of this report is to provide detailed information to demonstrate fulfillment of Actions 1 and 2 of the CAL, which are required to be completed prior to entry of Unit 2 into Mode 2. The actions as stated in the CAL are as follows:

> CAL ACTION 1: *"Southern California Edison Company (SCE) will determine the causes of the tube-to-tube interactions that resulted in steam generator tube wear in Unit 3, and will implement actions to prevent loss of integrity due to these causes in the Unit 2 steam generator tubes. SCE will establish a protocol of inspections and/or operational limits for Unit 2, including plans for a mid-cycle shutdown for further inspections."*
>
> *CAL ACTION 2: "Prior to entry of Unit 2 into Mode 2, SCE will submit to the NRC in writing the results of your assessment of Unit 2 steam generators, the protocol of inspections and/or operational limits, including schedule dates for a mid-cycle shutdown for further inspections, and the basis for SCE's conclusion that there is reasonable assurance, as required by NRC regulations, that the unit will operate safely."*

This report describes the actions SCE has taken to return Unit 2 to service while ensuring that the unit will operate safely. Because the SGs in Units 2 and 3 have the same design, the causes of the tube leak in Unit 3 and the potential susceptibility of Unit 2 SGs to the same mechanism are also addressed. This report will demonstrate that actions have been completed to prevent loss of integrity in the Unit 2 SG tubes due to these causes.

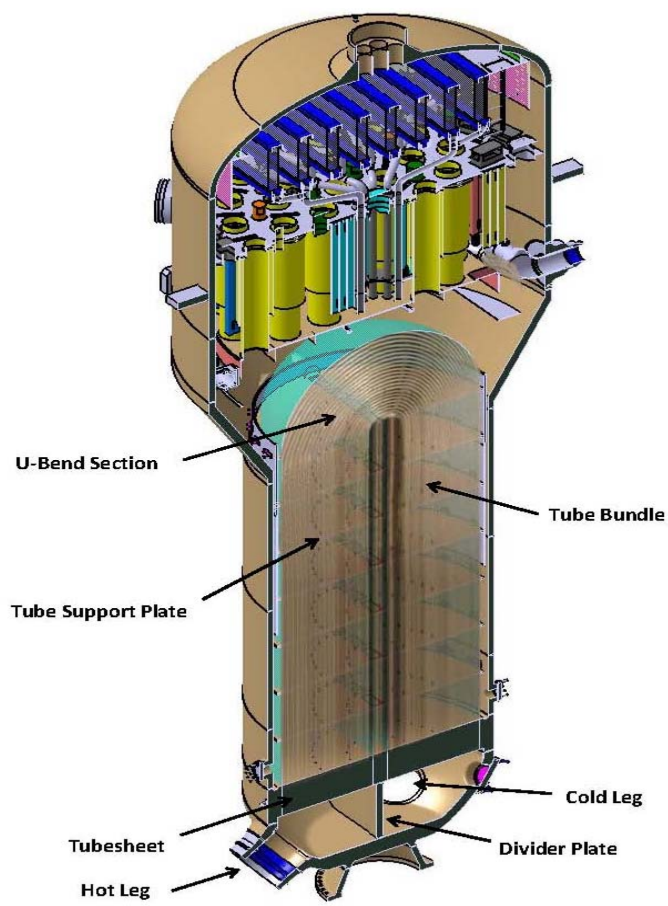
3.0 BACKGROUND

3.1 Steam Generator Tube Safety Functions

The Reactor Coolant System (RCS) circulates primary system water in a closed cycle, removing heat from the reactor core and internals and transferring it to the secondary side main steam system. The SGs provide the interface between the RCS and the main steam system. Reactor coolant is separated from the secondary system fluid by the SG tubes and tube sheet, making the RCS a closed system and forming a barrier to the release of radioactive materials from the core. The secondary side systems also circulate water in a closed cycle transferring the waste heat from the condenser to the circulating water system. However, the secondary side is not a totally closed system and presents several potential release paths to the environment in the event of a primary-to-secondary leak.

The SG tubes have a number of important safety functions. As noted above, the SG tubes are an integral part of the Reactor Coolant Pressure Boundary (RCPB) and, as such, are relied on to maintain primary system pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes act as the heat transfer surface that transfers heat from the primary system to the secondary system. Figure 3-1 provides a section view of a SONGS SG.

Figure 3-1: Replacement Steam Generator Section View



3.2 SG Regulatory/Program Requirements

The nuclear industry and the NRC have instituted rigorous requirements and guidelines to ensure that SG tube integrity is maintained such that the tubes are capable of performing their intended safety functions. Title 10 of the Code of Federal Regulations (10 CFR) establishes the fundamental regulatory requirements with respect to integrity of the SG tubes. The SONGS TSs include several requirements relating to the SGs including the requirement that SG tube integrity is maintained and all SG tubes reaching the tube repair criteria are plugged in accordance with the SGP (TS 3.4.17), that a SGP is established and implemented to ensure that SG tube integrity is maintained (TS 5.5.2.11), that a report of the inspection and CM results be provided to the NRC following each SG inspection outage (TS 5.7.2.c), and that the primary-to-secondary leakage through any one SG is limited to 150 gpd (TS 3.4.13). These TSs are provided in their entirety in Attachment 1.

TS 5.5.2.11, Steam Generator Program, requires the establishment and implementation of a SGP to ensure that SG tube integrity is maintained. The SGP ensures the tubes are repaired, or removed from service by plugging the tube ends, before the structural or leakage integrity of the tubes is impaired. TS 3.4.13 includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shutdown. Should a flaw exceeding the tube repair limit not be detected during the periodic tube inspections, the leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired.

TS 5.5.2.11 requires the SGP to include five provisions, which are summarized below

a. CM assessments shall be conducted during each SG inspection outage to evaluate the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The purpose of the CM assessment is to ensure that the SG performance criteria have been met for the previous operating period.

b. SG tube integrity shall be maintained by meeting the specified performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

c. Tubes found by in-service inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.

d. Periodic SG tube inspections shall be performed as specified in the TS. The inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection.

e. Provisions shall be made for monitoring operational primary-to-secondary leakage.

TS 3.4.13, RCS Operational Leakage, limits primary-to-secondary leakage through any one SG to 150 gpd. The limit of 150 gpd per SG is based on the operational leakage performance criterion in the Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines (Ref. 2). The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage.

3.3 The SONGS Steam Generator Program

The purpose of the SGP is to ensure tube integrity and compliance with SG regulatory requirements. The program contains a balance of prevention, inspection, evaluation and repair, and leakage monitoring measures. The SONGS SGP (Ref. 10), which implements the requirements specified in TS 5.5.2.11, is based on the NEI 97-06, Steam Generator Program Guidelines (Ref. 2) and its referenced Electric Power Research Institute (EPRI) guidelines. Use of the SGP ensures that SGs are inspected and repaired consistent with accepted industry practices.

The SGP requires assessments of SG integrity. This assessment applies to SG components which are part of the primary pressure boundary (e.g., tubing, tube plugs, sleeves and other repairs). It also applies to foreign objects (FOs) and secondary side structural supports (e.g., tube support plates (TSPs)) that may, if severely degraded, compromise pressure-retaining components of the SG. Three types of assessments are performed to provide assurance that the SG tubes will continue to satisfy the appropriate performance criteria: (1) Degradation Assessment (DA); (2) CM Assessment; and (3) OA.

The DA is the planning process that identifies and documents information about plant-specific SG degradation. The overall purpose of the DA is to prepare for an upcoming SG inspection through the identification of the appropriate examinations and techniques, and ensuring that the requisite information for integrity assessment is obtained. The DA performed for Unit 2 Cycle 17 (U2C17) SG Inspection Outage is discussed in Section 7.1 of this report.

The CM is backward looking, in that its purpose is to confirm that adequate SG tube integrity has been maintained during the previous inspection interval. The CM involves an evaluation of the as-found condition of the tubing relative to the integrity performance criteria specified in the TS. The tubes are inspected according to the EPRI Pressurized Water Reactor SG Examination Guidelines (Ref. 3). Structural and leakage integrity assessments are performed and results compared to their respective performance criteria. If satisfactory results are not achieved, a RCE is performed and appropriate corrective action taken. The results of this analysis are factored into future DAs, inspection plans, and OAs of the plant. The CM results for U2C17 are presented in Section 7 of this report.

The OA differs from the CM assessment in that it is forward looking rather than backward looking. Its purpose is to demonstrate that the tube integrity performance criteria will be met throughout the next inspection interval. During the CM assessments, inspection results are evaluated with respect to the appropriate performance criteria. If this evaluation is successful, an OA is performed to show that integrity will be maintained throughout the next interval between inspections. If any performance criterion is not met during performance of CM, a RCE is required to be performed and the results are to be factored into the OA strategy. The results of the OA determine the allowable operating time for the upcoming inspection interval. The OA addressing all degradation mechanisms found during U2C17 is discussed in Section 10 of this report.

4.0 UNIT 2 AND 3 REPLACEMENT STEAM GENERATORS

New SGs were placed into service at SONGS Units 2 and 3 in 2010 and 2011, respectively. The RSGs were intended to resolve corrosion and other degradation issues present in the OSGs. The RSGs were designed and manufactured by MHI.

The steam generator is a recirculating, vertical U-tube type heat exchanger converting feedwater into saturated steam. The steam generator vessel pressure boundary is comprised of the channel head, lower shell, middle shell, transition cone, upper shell and upper head. The steam generator internals include the divider plate, tubesheet, tube bundle, feedwater distribution system, moisture separators, steam dryers and integral steam flow limiter installed in the steam nozzle. The channel head is equipped with one reactor coolant inlet nozzle and two outlet nozzles. The upper vessel is equipped with the feedwater nozzle, steam nozzle and blowdown nozzle. In the channel head, there are two 18 inch access manways. In the upper shell, there are two 16 inch access manways. The steam generator is equipped with six handholes and 12 inspection ports providing access for inspection and maintenance. In addition, the steam generators are equipped with several instrumentation and minor nozzles for layup and chemical recirculation intended for chemical cleaning.

5.0 UNIT 3 EVENT – LOSS OF TUBE INTEGRITY

5.1 Summary of Event

On January 31, 2012, while the Unit 2 refueling and SG inspection outage was in progress, SONGS Unit 3 was in Mode 1 operating at 100 percent power, when a condenser air ejector radiation monitor alarm indicated a primary-to-secondary leak. A rapid power reduction was commenced when the primary-to-secondary leak rate was determined to be greater than 75 gpd with an increasing rate of leakage exceeding 30 gpd per hour. The reactor was manually tripped from 35 percent power, and placed in a stable cold shutdown condition in Mode 5. The TS 3.4.13 limit for RCS operational leakage (150 gpd) was not exceeded. A small, monitored radioactive release to the environment occurred, resulting in an estimated 0.0000452 mrem dose to the public, which was well below the allowable federal limit specified in 10 CFR 20 of 100 mrem per year to a member of the public.

Subsequent to the reactor cooldown, extensive inspection, testing, and analysis of SG tube integrity commenced in both Unit 3 SGs. This was the first inspection of the Unit 3 SG tubes performed following SG replacement after approximately eleven months of operation. The work scope included the following activities: bobbin probe and rotating probe examinations using eddy current testing (ECT), secondary and primary side visual examinations, and in-situ pressure testing. The location of the leak in SG 3E-088, which resulted in the Unit 3 shutdown, was determined to be in the U-bend portion of the tube in Row 106 Column 78. ECT was subsequently performed on 100% of the tubes in both Unit 3 SGs. During these inspections, unexpected wear was discovered in both SGs including wear at AVBs, TSPs, RBs, and significant TTW in the U-bend area of the tubes. The TTW in Unit 3 was found to be much more extensive than in Unit 2, where only two tubes in one SG were determined to be affected.

The EPRI guidelines (Ref. 4) allow assessment of the structural and accident induced leakage integrity to be performed either analytically or through in-situ pressure testing. In accordance with EPRI guidelines and the SGP, in-situ pressure testing was performed on a total of 129 tubes in Unit 3, (73 in SG 3E-088 and 56 in SG 3E-089) in March 2012. The pressure tests were performed to determine if the tubes met the performance criteria in the TS (Attachment 1). The testing resulted in detected leaks in eight tubes in SG 3E-088 at the pressures indicated in Table 5-1. The failure location for all eight tubes was in the U-bend portion of the tube bundle in the tube freespan area. The locations of the tubes that were pressure tested and the tubes that failed the pressure tests are shown in Figure 6-7 and Figure 6-8. The first tube listed in the table (location 106-78) was the tube with the through-wall leak which resulted in the Unit 3 shutdown on January 31, 2012. No leaks were detected in the remaining 121 tubes tested in Unit 3. For the eight tubes indicating leakage, three tubes failed both the accident induced leakage performance criterion (AILPC) and the structural integrity performance criterion (SIPC); and 5 tubes passed the AILPC but failed the SIPC. All tubes met the operational leakage performance criterion of TS Limiting Condition for Operation 3.4.13. Details of the Unit 3 inspections and in-situ testing results are documented in the Unit 3 CM Report included as Attachment 3.

Additional testing performed to identify the extent and cause of the abnormal wear is presented in Section 6. Required reports in response to the reactor shut down and in-situ test failures were made to the NRC in accordance with 10 CFR 50.72 and 50.73 (Refs. 5-8).

Table 5-1: SONGS Unit 3 SG 3E-088 In-Situ Pressure Tests with Tube Leakage

Test Date, Time	Tube Location (row-column)	Maximum Test Pressure Achieved (see Note 1)	Performance Criteria Not Met (see Note 2)
03/14/12, 1120PDT	106-78	2874 psig	Accident Induced Leakage
03/14/12, 1249PDT	102-78	3268 psig	Accident Induced Leakage
03/14/12, 1425PDT	104-78	3180 psig	Accident Induced Leakage
03/15/12, 1109PDT	100-80	4732 psig	Structural Integrity
03/15/12, 1437PDT	107-77	5160 psig	Structural Integrity
03/15/12, 1604PDT	101-81	4889 psig	Structural Integrity
03/15/12, 1734PDT	98-80	4886 psig	Structural Integrity
03/16/12, 1216PDT	99-81	5026 psig	Structural Integrity

Note 1
Test Pressures: (Calculated)
Normal Operating Differential Pressure (NODP) Test Pressure = 1850 psig
Accident Induced Leakage DP (Main Steam Line Break) Test Pressure = 3200 psig
Structural Integrity Limit (3 x NODP) Test Pressure = 5250 psig

Note 2
Performance Criteria:
Structural Integrity – No burst at 3 x NODP test pressure
Accident Induced Leakage - leak rate < 0.5 gpm at MSLB test pressure
Operational Leakage - TS Limiting Condition for Operation 3.4.13

5.2 Safety Consequences of Event

As discussed above, the Unit 3 shutdown on January 31, 2012, due to a SG tube leak, resulted in a small, monitored radioactive release to the environment, well below allowable limits. The potential safety significance of the degraded condition of the Unit 3 SG tubes is discussed below.

5.2.1 Deterministic Risk Analyses

The SONGS Updated Final Safety Analysis Report (UFSAR) Section 15.10.1.3.1.2 presents the current licensing basis steam line break (SLB) post-trip return-to-power event (post-trip SLB). Based on the actual plant RCS chemistry data, the accident-induced iodine spiking factor of 500, and the estimated SG tube rupture leakage rate, the calculated dose would have been at least 32 percent lower than the dose consequences reported in the UFSAR for the post-trip SLB event with a concurrent iodine spike. The postulated post-trip SLB with tube rupture and concurrent iodine spike Exclusion Area Boundary, Low Population Zone, and Control Room doses would be less than 0.068 Rem Total Effective Dose Equivalent (TEDE), which is well below the post-trip SLB Control Room limit of 5 Rem TEDE, and the Exclusion Area Boundary and Low Population Zone limit of 2.5 Rem TEDE.

The potential for a seismically-induced tube rupture was also evaluated. The analysis determined the equivalent flaw characteristics of the most limiting degraded tube in Unit 3 SG 3E-088 from its in-situ pressure test result. This tube, Row 106 Column 78 (the leaking tube), sustained an in-situ test pressure of 2,874 psi before exceeding leakage limits. This in-situ test pressure, which is slightly more than twice the operating differential pressure on the tube, corresponds to the limiting stress for crack penetration or plastic collapse with large deformation. The combined stresses due to operating differential pressure and seismic forces corresponding to SONGS Design Basis Earthquake (DBE) are lower than this limiting stress and are also less than the allowable stress for the faulted condition (i.e., including DBE) according to the American Society of Mechanical Engineers Code. Therefore, the degraded tube would not have burst under this worst case loading.

5.2.2 Probabilistic Risk Assessment

A Probabilistic Risk Assessment (PRA) was performed to analyze the risk impact of the degraded SG tubes on SONGS Unit 3 SG 3E-088 with respect to two cases: (1) any increased likelihood of an independent SG tube rupture (SGTR) at normal operating differential pressure (NODP), or (2) due to a SGTR induced by an excess steam demand event, also referred to as a main steam line break (MSLB). The SONGS PRA model was used to calculate the increases in Core Damage Probability (CDP) and Large Early Release Probability (LERP) associated with each case. In both cases, all postulated core damage sequences are assumed to result in a large early release since the containment will be bypassed due to the SGTR; therefore, the calculated CDP and LERP are equal. The total Incremental LERP (ILERP) due to the degraded SG tubes (i.e., the sum of the two analyzed cases) was determined to be less than 2×10^{-7}. This small increase in risk is attributed to two factors. First, the exposure time for the postulated increased independent SGTR initiating event frequency case was very short (0.1 Effective Full Power Month (EFPM)). Second, a MSLB alone does not generate sufficient differential pressure to cause tube rupture in Case 2. The differential pressure across the SG tubes necessary to cause a rupture will not occur if operators prevent RCS re-pressurization in accordance with Emergency Operating Instructions.

6.0 UNIT 3 EVENT INVESTIGATION AND CAUSE EVALUATION

6.1 Summary of Inspections Performed

Following the identification of SG tube leakage in the Unit 3 SG 3E-088, extensive inspections were performed to determine the location and cause of the leak. The location of the leak was identified by filling the SG secondary side with nitrogen and pressurizing to 80 psig. The test identified the tube located at Row 106, Column 78 (R106 C78) as the source of the leakage. Using eddy current bobbin and rotating probes, the tube at R106 C78 and those immediately adjacent to it were inspected and the leakage location was confirmed. The leak location was in the U-bend portion of the tube in the "freespan" area between AVB support locations (refer to Figure 6-1).

To determine the extent of the wear that had resulted in a leak, an eddy current bobbin probe examination of the full-length of all tubes in both Unit 3 SGs was performed. The locations of tubes with TTW are shown on Figures 6-7 and 6-8. Based on the results of the bobbin probe examinations, TTW indications were then examined using a more sensitive +Point™ rotating probe. Figure 6-6 illustrates a comparison of the sensitivity of the two types of examinations. The more sensitive rotating probe examinations were also performed on a region of tubes adjacent to the tubes with detected TTW. This region is also shown on Figures 6-7 and 6-8. TTW indications were identified in 161 tubes in 3E-088 and 165 tubes in 3E-089. All of the TTW flaws were located in the U-bend portion of the tubes between TSPs 7H and 7C (shown on Figure 6-1).

The more sensitive eddy current rotating probe provided an estimated depth and overall length of TTW flaws on each tube examined. The examination technique (EPRI Examination Technique Specification Sheet, ETSS 27902.2) was site validated by building a test specimen with flaws similar to the TTW flaws observed in Unit 3. Comparison of estimated wear depths with actual wear depths of the specimen supported the conclusion that ETSS 27902.2 conservatively estimated the depths across the entire range of depths tested (from 5% through-wall to 81% through-wall).

The tubes with flaws identified by ECT were analyzed to determine if they were capable of meeting the SONGS TS tube integrity performance criteria (Attachment 1). Tubes that did not meet the performance criteria based on analysis were tested via in-situ pressure testing. As described in detail in Section 5 and in the CM report (Attachment 3), a total of 129 tubes in the Unit 3 SGs were selected for in-situ pressure testing. Three tubes failed both the AILPC and the SIPC, and 5 tubes passed the AILPC but failed the SIPC as defined in TS 5.5.2.11. These eight tubes are listed in Table 5-1. Figure 6-7 and Figure 6-8 show the locations of the tubes that were in-situ tested and the eight tubes that did not meet the performance criteria.

Secondary side remote visual inspections were performed to supplement the eddy current results and provide additional information in support of the cause evaluation. The inspections included the 7th TSP and inner bundle passes at AVBs B04 and B09 (shown on Figure 6-1). The 7th TSP inspection revealed no unexpected or unusual conditions. The inner bundle passes included several inspections between columns 73 and 87 and showed instances of wear indications that extended outside the AVB intersection. This was confirmed by eddy current data. Additional passes were made between columns 50 and 60. These inspections did not show any AVB wear outside the AVB intersections.

6.2 Summary of Inspection Results

This section provides a summary of the different types of tube wear found in the SONGS Unit 2 and 3 SGs. Wear is characterized as a loss of metal on the surface of one or both metallic objects that are in contact during movement.

The following types of wear were identified in the SONGS Units 2 and 3 SG tubes:

- AVB wear - wear of the tubing at the tube-to-AVB intersections
- TSP wear - wear of the tubing at the tube-to-TSP intersections
- TTW - wear in the tube free-span sections between the AVBs located in the U-bend region.
- RB wear - wear of the tubing at a location adjacent to a RB (RBs are not designed as tube supports for normal operation)
- FO wear - wear of the tubing at a location adjacent to a FO.

Most of the tube wear identified in the SGs is adjacent to a tube support. Figure 6-1 is a side view of an SG, showing the relationship of the tubes to the two types of tube supports: TSPs in the straight portions and AVBs in the U-bend portions of the tubes. All tubes are adjacent to many of these two types of tube supports. The RB supports are not shown because a very small number of tubes are adjacent to them.

TTW indications occurred in the free span sections of the tubes. The “free span” is that secton of the tube between support structures (AVBs and TSPs shown in Figure 6-1). TTW occurred almost exclusively in Unit 3 and is located on both the hot and cold leg side of the U-tube. In many cases, the region of the tube with TTW has two separate indications on the extrados and intrados of the tube. The wear indications on neighboring tubes have similar depth and position (ranging from 1.0 to 41 inches long and 4% to 100% throughwall) along the U-bend, confirming the tube-to-tube contact.

Table 6-1 provides the Wear Depth Summary for each of the four SGs based on eddy current examination results. Detailed results of the examinations performed are provided in the Units 2 and 3 CM reports included as Attachments 2 and 3. Figures 6-2 through 6-5 provide distributions of wear at AVB and TSP supports for all four SGs.

Figure 6-1: Steam Generator Section View Sketch

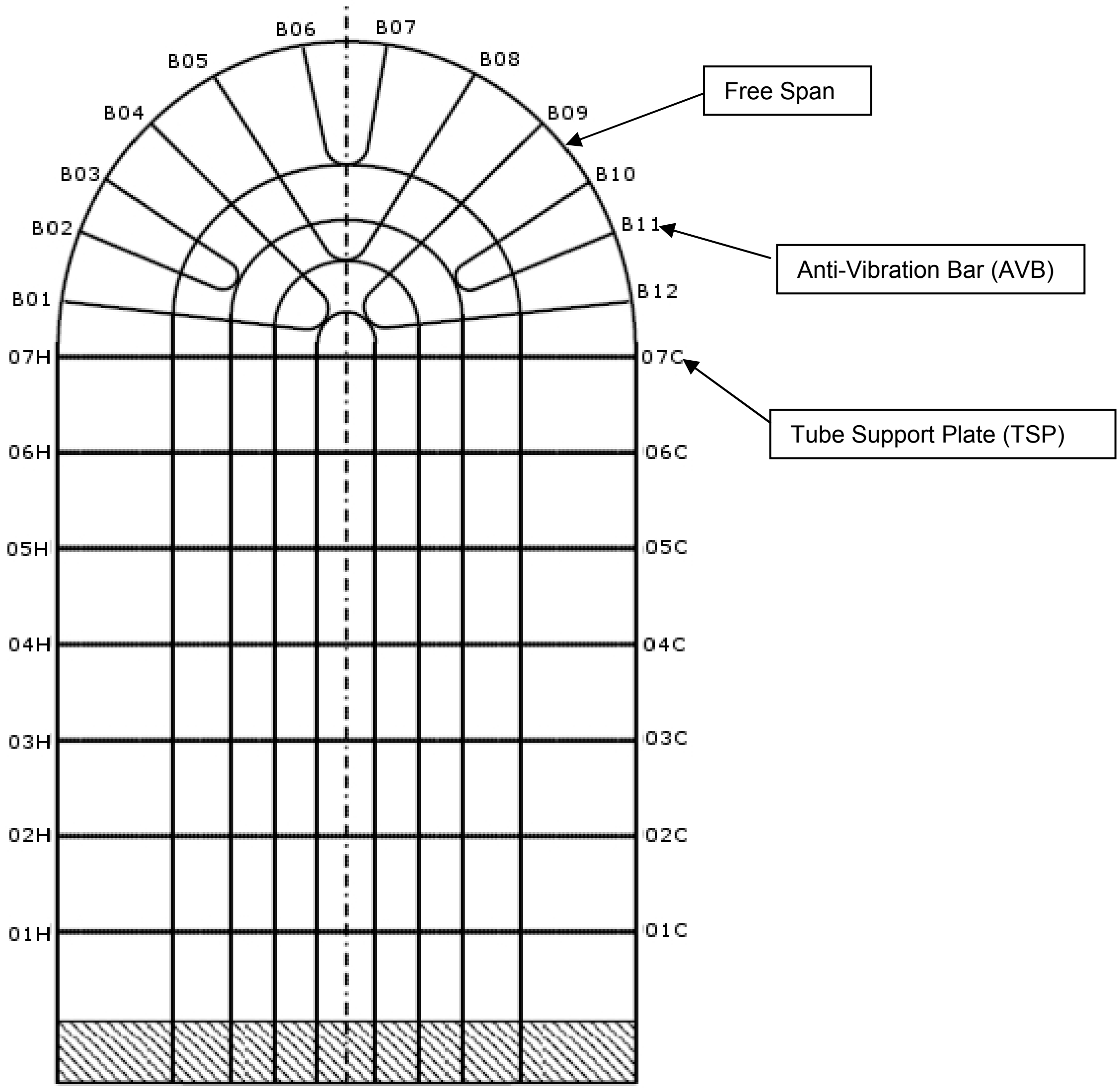


Figure 6-2: Unit 2 Distribution of Wear at AVB Supports

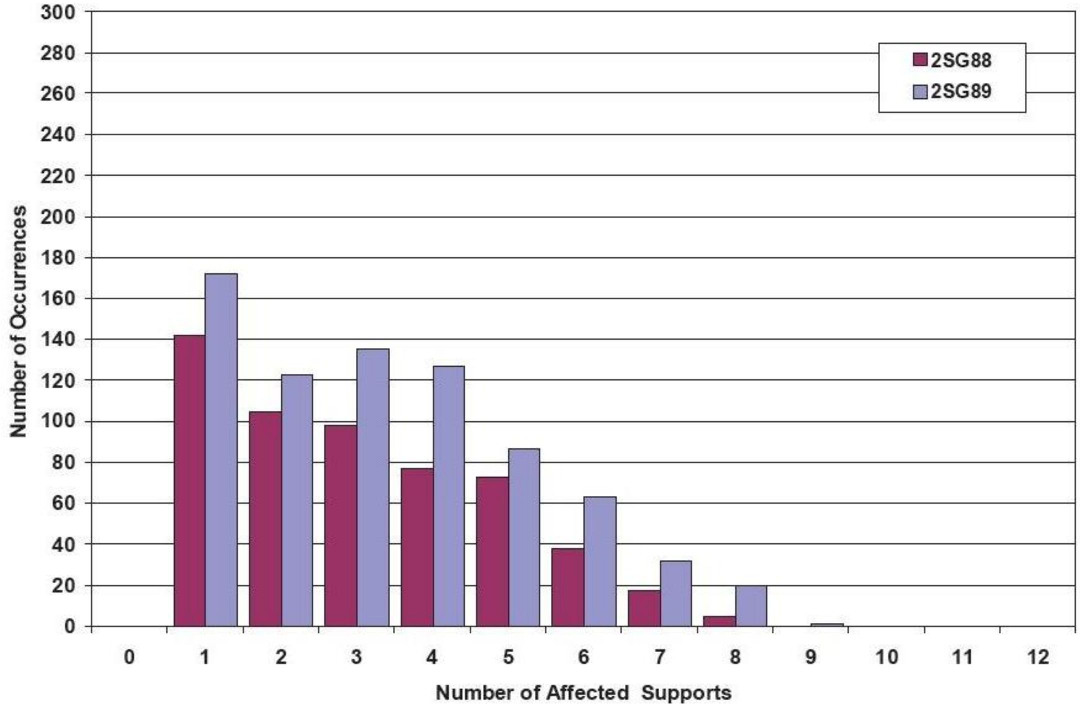


Figure 6-3: Unit 3 Distribution of Wear at AVB Supports

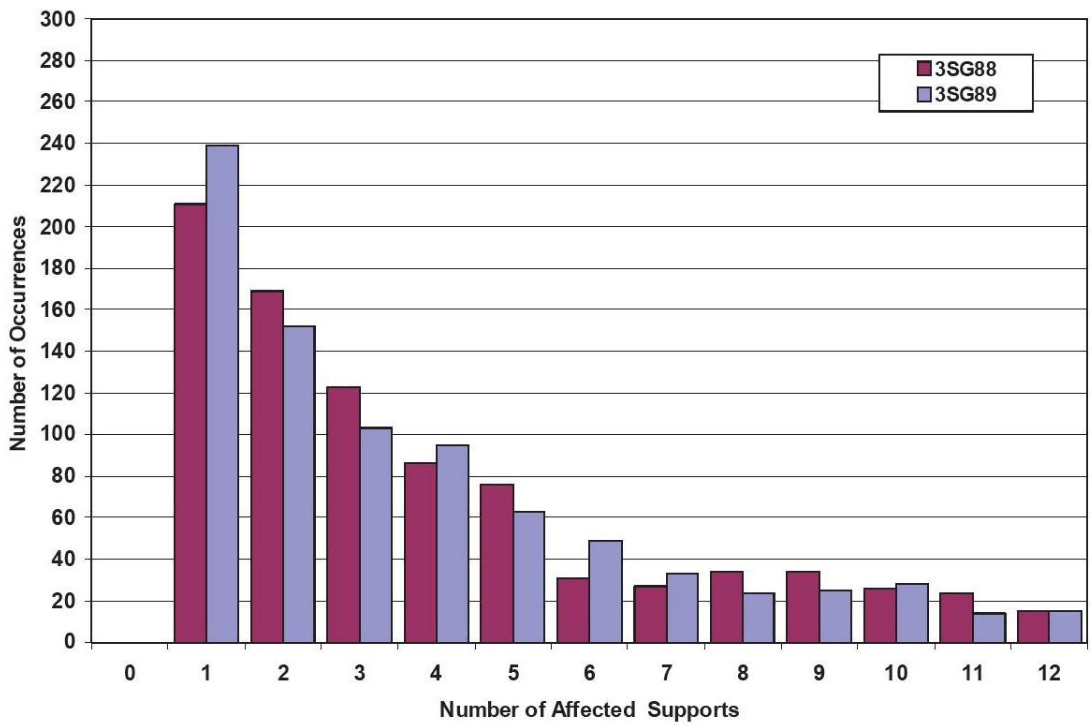


Figure 6-4: Unit 2 Distribution of Wear at TSP Supports

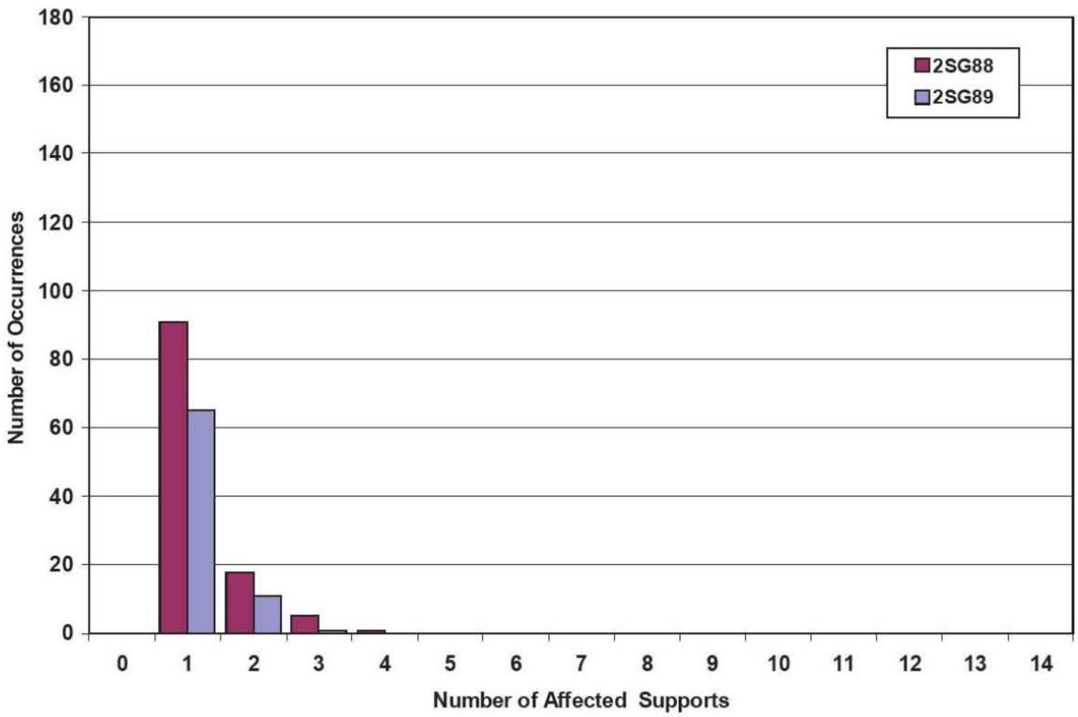


Figure 6-5: Unit 3 Distribution of Wear at TSP Supports

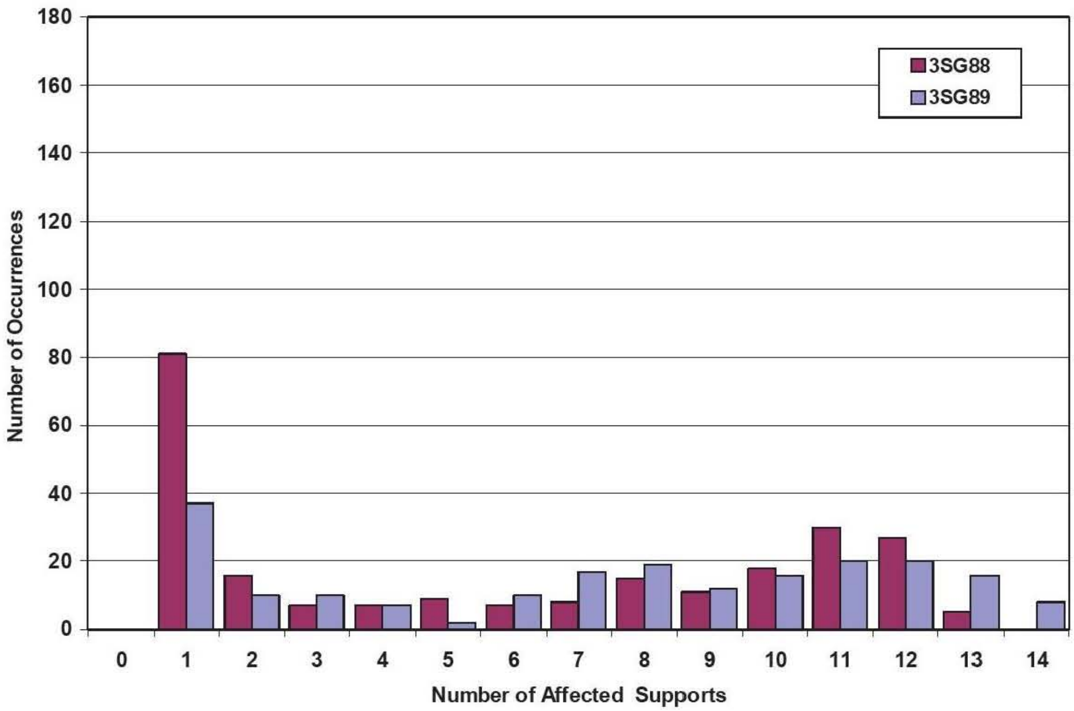


Figure 6-6: Probability of Detection for Tube Wear

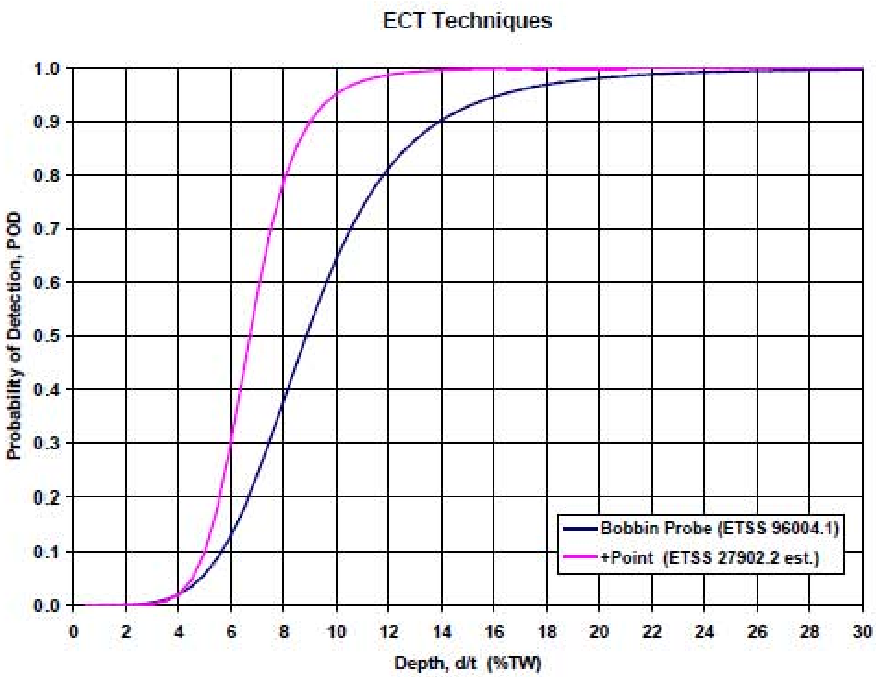


Figure 6-7: 3E-088 Rotating Coil Inspection Region

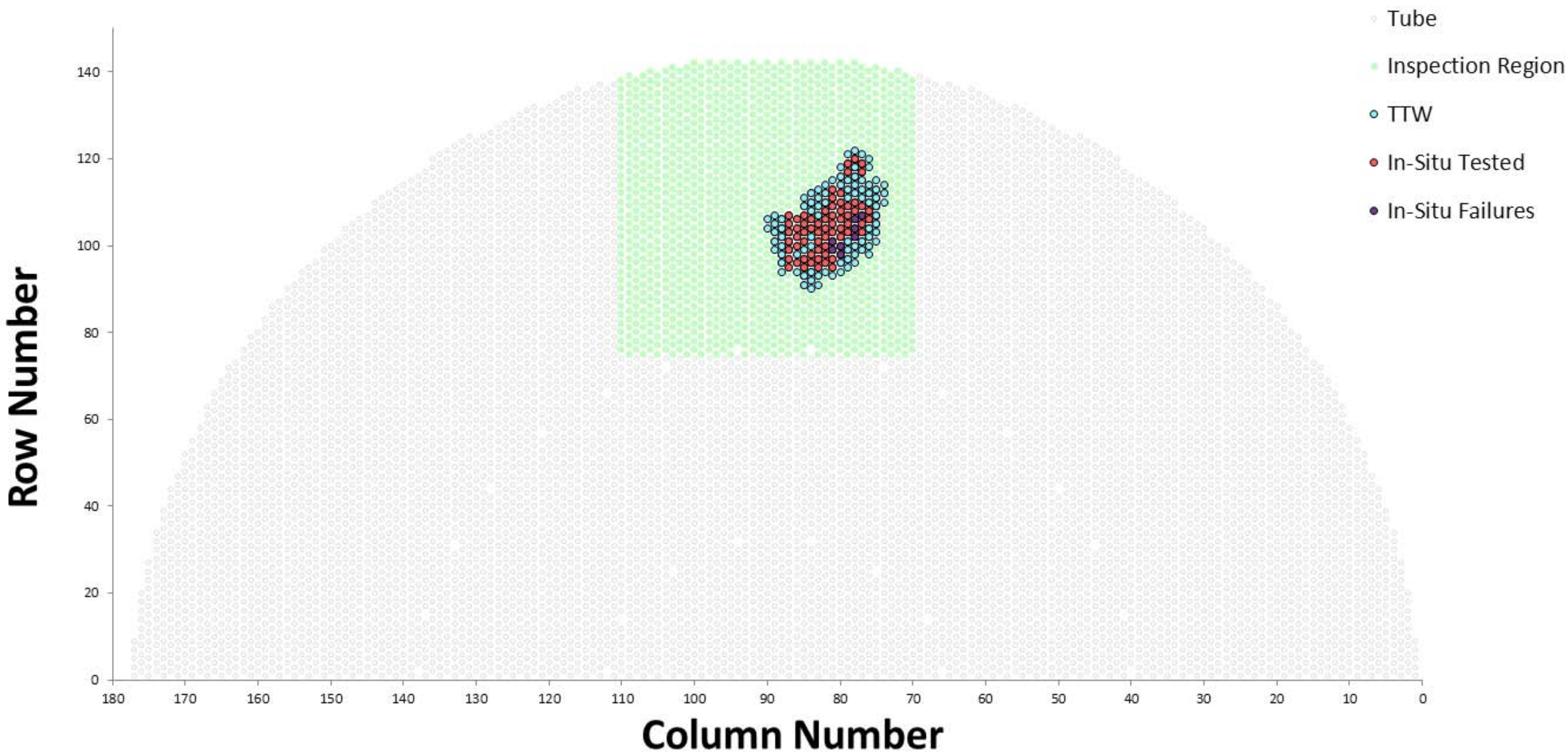


Figure 6-8: 3E-089 Rotating Coil Inspection Region

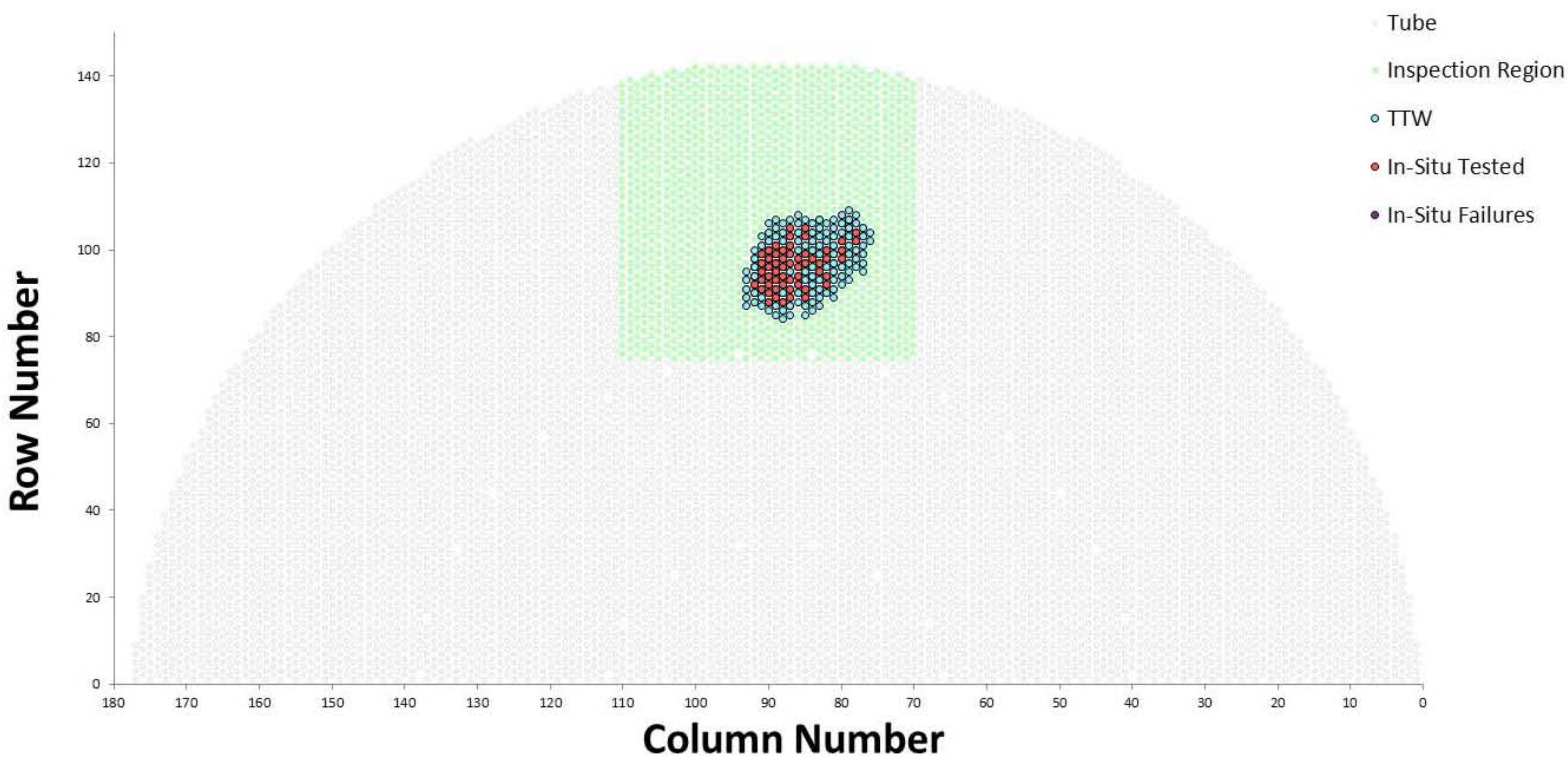


Table 6-1: Steam Generator Wear Depth Summary

TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
SG 2E-088							
TW ≥ 50%	0	0	0	1	0	1	1
35 - 49%	2	0	0	1	0	3	3
20 - 34%	86	0	0	0	2	88	74
10 - 19%	705	108	0	0	0	813	406
TW < 10%	964	117	0	0	0	1081	600
Total	**1757**	**225**	**0**	**2**	**2**	**1986**	**734***
SG 2E-089							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	0	0	1	0	1	1
35 - 49%	0	0	0	1	0	1	1
20 - 34%	78	1	0	3	0	82	67
10 - 19%	1014	85	2	0	0	1101	496
TW < 10%	1499	53	0	0	0	1552	768
Total	**2591**	**139**	**2**	**5**	**0**	**2737**	**861***
SG 3E-088							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	117**	48	0	0	165	74
35 - 49%	3	217	116	2	0	338	119
20 - 34%	156	506	134	1	0	797	197
10 - 19%	1380	542	98	0	0	2020	554
TW < 10%	1818	55	11	0	0	1884	817
Total	**3357**	**1437**	**407**	**3**	**0**	**5204**	**919***
SG 3E-089							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	91**	26	0	0	117	60
35 - 49%	0	252	102	1	0	355	128
20 - 34%	45	487	215	0	0	747	175
10 - 19%	940	590	72	0	0	1602	450
TW < 10%	2164	94	1	0	0	2259	838
Total	**3149**	**1514**	**416**	**1**	**0**	**5080**	**887***

* This value is the number of tubes with a wear indication of any depth at any location. Since many tubes have indications in more than one depth category, the total number of tubes with wear indications is not the additive sum of the counts for the individual depth categories.

** All TSP indications ≥50% TW were in tubes with TTW indications.

6.3 Cause Analyses of Tube-to-Tube Wear in Unit 3

6.3.1 Mechanistic Cause

SCE established a RCE team to investigate the condition, extent of condition, and cause of the event in Unit 3 and to determine corrective actions. The RCE was conducted, documented, and reviewed in accordance with the SONGS Corrective Action Program (CAP). The RCE Team used systematic approaches to identify the mechanistic cause of the TTW, including failure modes analysis (Kepner-Tregoe). The RCE team had access to and used significant input from the SG Recovery Team, which included the services of MHI and industry experts in the fields of T/H and in SG design, manufacturing, operation, and repair.

The failure modes analysis identified a list of 21 possible causes. The list was narrowed down, using facts, analysis, and expert input, to a list of eight potential causes that warranted further technical evaluation. The potential causes included manufacturing/fabrication, shipping, primary side flow induced vibration, divider plate weld failure and repair, additional rotations following divider plate repair, TSP distortion, tube bundle distortion during operation (flowering), and T/H conditions/modeling.

The eight potential causes underwent rigorous analysis using both empirical and theoretical data, and support-refute methodology. This approach identified likely causes and eliminated non-causes. Each of the potential causes was evaluated by engineering analysis of the supporting and refuting data. The mechanistic cause of the TTW in Unit 3 was identified as FEI, involving the combination of localized high steam velocity (tube vibration excitation forces), high steam void fraction (loss of ability to dampen vibration), and insufficient tube to AVB contact to overcome the excitation forces. A more detailed discussion of the cause of FEI in the Unit 3 SGs is provided in MHI's Technical Evaluation Report, which is included as Attachment 4.

6.3.2 Potential Applicability of Unit 3 TTW Causes to Unit 2

At the time of the Unit 3 SG tube leak, Unit 2 was in the first refueling outage after SG replacement and undergoing ECT inspections per the SGP. Following the discovery of TTW in Unit 3, additional Unit 2 inspections identified two tubes with TTW indications in SG 2E-089. The location of TTW in the Unit 2 SG was in the same region of the bundle as in the Unit 3 SGs indicating causal factors might be similar to those resulting in TTW in the Unit 3 SGs. Because of the similarities in design between the Unit 2 and 3 RSGs, it was concluded that FEI in the in-plane direction was also the cause of the TTW in Unit 2.

After the RCE for TTW was prepared, WEC performed analysis of Unit 2 ECT data and concluded TTW was caused by the close proximity of these two tubes during initial operation of the RSGs. With close proximity, normal vibration of the tubes produced the wear at the point of contact. With proximity as the cause, during operation the tubes wear until they are no longer in contact, a condition known as 'wear arrest'. This wear mechanism is addressed in Section 10 and Attachment 6.

As described in Section 8, the compensatory and corrective actions implemented to prevent loss of tube integrity caused by TTW in Unit 2 are sufficient to conservatively address both identified causes.

6.4 Industry Expert Involvement

Upon discovery of TTW in Unit 3, SCE commissioned the services of industry experts to assist in assessing the cause of this phenomenon. SCE selected experts based upon their previous experience in design, evaluation, tube vibration, testing and causal determinations related to SGs. Members included experts in T/H and SGPs from MPR Associates, AREVA, Babcock & Wilcox Canada, Palo Verde Nuclear Generating Station, EPRI, Institute of Nuclear Power Operations (INPO), and MHI, as well as experienced consultants including former NRC executives and a research scientist. A series of panel meetings were conducted during which testing and analysis results were presented. The panel members assessed whether the current work by SCE and its partners was sufficient in understanding the TTW phenomenon and whether the corrective actions developed were sufficient to ensure tube integrity in the future.

6.5 Cause Analysis Summary

SCE has determined the mechanistic cause of the TTW in Unit 3 was FEI, resulting from the combination of localized high steam velocity, high steam void fraction, and insufficient contact forces between the tubes and the AVBs. The FEI resulted in a vibration mode of the SG tubes in which the tubes moved in the in-plane direction, parallel to the AVBs, in the U-bend region. This resulted in TTW in a localized area of the SGs. As discussed in the following sections, SCE has identified actions to prevent loss of integrity due to FEI in the Unit 2 SG tubes. The extent of condition inspections performed in Unit 2 and differences identified between Units 2 and 3 are discussed in Section 7. The compensatory and corrective actions to prevent loss of integrity due to these causes in the Unit 2 SG tubes are discussed in Section 8.

7.0 UNIT 2 CYCLE 17 INSPECTIONS AND REPAIRS

On January 9, 2012, Unit 2 was shut down for a routine refueling and steam generator inspection outage after approximately 22 months of operation. As discussed in Section 3.3, the SGP requires a CM assessment to confirm that SG tube integrity has been maintained during the previous inspection interval. SCE conducted a number of inspections on each of the two Unit 2 SGs (2E-088 and 2E-089) in accordance with the SGP. Based on the inspection results, the Unit 2 CM assessment (included as Attachment 2) concluded that the TS SG performance criteria were satisfied by the Unit 2 SGs during the operating period prior to the current U2C17 outage. The TS performance criteria for tube integrity for all indications were satisfied through a combination of ECT examination, analytical evaluation, and in-situ pressure testing. The operational leakage criterion was satisfied because the Unit 2 SGs experienced no measurable primary-to-secondary leakage during the operating period preceding the Cycle 17 outage.

The Unit 2 outage was in progress on January 31, 2012, when Unit 3 was shut down in response to a tube leak. Although the SG performance criteria had been met by the Unit 2 SGs, the unit was not returned to service pending an evaluation of the tube leak in Unit 3. Subsequent to the discovery of TTW conditions in the U-bend region of the Unit 3 SGs, additional inspections were performed on the Unit 2 tubes and shallow TTW was identified in two adjacent tubes in SG 2E-089.

Section 7.1 provides a summary of results from the routine inspections performed in Unit 2 and Section 7.2 provides a summary of results from the additional Unit 2 inspections performed in response to the discovery of TTW in Unit 3. Details of all the inspections are provided in the Unit 2 CM report (Attachment 2). Section 7.3 summarizes the differences observed between Units 2 and 3.

7.1 Unit 2 Cycle 17 Routine Inspections and Repairs

The SGP requires that a DA be performed prior to a SG inspection outage to develop an inspection plan based on the type and location of flaws to which the tubes may be susceptible. This assessment was performed prior to the inspection and was updated when unexpected degradation mechanisms were found during the inspection. These unexpected degradation mechanisms included (1) RB wear and (2) the TTW observed in Unit 3.

Initially, eddy current bobbin probe examinations of the full length of each tube was performed on 100% of the tubes in both Unit 2 SGs. Selected areas were then inspected using a more sensitive rotating +Point™ examination. During the ECT examinations, wear was detected at AVBs, TSPs and RB locations. Six tubes with high wear indications (equal to or exceeding 35% of the tube wall thickness) were found. Four of those indications occurred in the vicinity of the RBs and two were associated with AVB locations as shown in Table 6-1. One in-situ pressure test was performed on a tube with RB wear, with satisfactory results. No other indications required in-situ pressure testing. Numerous smaller depth wear indications were also reported at other AVB and TSP locations. The ECT results are summarized in Table 6-1.

In accordance with TS 5.5.2.11.c, tubes that are found to have indications of degradation equal to or exceeding 35% through wall (TW) are removed from service by the installation of a plug in both ends of the tube. Once plugs are installed in both ends of a tube, they prevent primary system water from entering the tube. Plugs may also be used to preventively remove tubes from service. Use of preventive plugging is discussed in Section 8.2.

An RCE was completed for the unexpected RB wear. The RCE concluded that the RB size (diameter and length) was inadequate to prevent the RB from vibrating and contacting adjacent tubes during normal plant operation. The vibration source was a turbulent two phase flow (water and steam) across the RBs. As a corrective action, the 94 tubes adjacent to the RBs in each Unit 2 SG were plugged, including two tubes with RB wear in SG 2E-088 and four tubes with RB wear in SG 2E-089.

Four additional tubes were plugged due to wear at AVB locations. Two of these were plugged as required for wear depths equal to or exceeding 35% TW; the other two with through wall depths (TWDs) of approximately 32% were plugged as a preventive measure. A significant number of tubes were preventively plugged and removed from service using screening criteria based on TTW indications in Unit 3. Table 6-1 provides the total numbers of tubes and indications due to all types of wear in the Unit 2 SGs. The tubes and criteria used to select tubes to be removed from service by preventive plugging due to their susceptibility to TTW are discussed in Section 8.2 and Attachment 5.

During the eddy current inspection of SG 2E-088, FO indications and FO wear indications were reported in two adjacent tubes at the 4th TSP. A secondary side foreign object search and retrieval (FOSAR) effort was performed and the object was located and removed. A follow-up analysis identified the object as weld metal debris. The two adjacent tubes were left in service because the indications were below the TS plugging limit and the cause of the degradation had been removed.

Remote visual inspections were performed to confirm the integrity of the RBs. The results of these visual inspections are summarized below:

- No cracking or degradation of RBs or RB-to-retaining bar welds was observed
- No cracking or degradation of AVB end caps or end cap-to-RB welds was observed
- No FOs or loose parts were found in the RB locations

Post sludge lancing FOSAR examination at the top-of-tubesheet (periphery and the no-tube lane) found no evidence of degradation and no FOs.

7.2 Unit 2 Cycle 17 Inspection in Response to TTW in Unit 3

Subsequent to the discovery of TTW conditions in the U-bend region of Unit 3 SGs, an additional review of the U-bend region bobbin probe data was performed for the Unit 2 SGs. The tubes selected for review encompassed the suspected TTW zone as observed in Unit 3 and tubes surrounding that zone. Over 1,000 tubes in each Unit 2 SG were reviewed. The review included a two-party manual analysis (primary/secondary) of the complete U-bend with emphasis on the detection of low level freespan indications, which may not have been reported during the original analysis of the U2C17 bobbin coil data. No new indications were identified during this review.

Additional examinations of the U-bends were performed using rotating probe (+Point™) technology. The scope of this examination is identified on the tubesheet maps provided in Figure 7-1 and Figure 7-2. During this examination, two adjacent tubes with TTW indications were detected. The indications were approximately 6 inches long, located between AVBs B09 and B10 in tubes R111 C81 and R113 C81 in SG 2E-089. Figure 7-2 shows the location of the two tubes with TTW in 2E-089. The maps in Figure 6-7 and Figure 6-8 show the inspection region overlaying the locations of the TTW found in the Unit 3 SGs.

SCE notified the NRC of the discovery of the two tubes with TTW in a letter dated April 20, 2012. (Ref. 9)

Remote visual inspections of the secondary side upper tube bundle were conducted in the Unit 2 SGs. These inspections were similar to those performed in Unit 3 SGs to assist in the development of the mechanistic root cause of TTW and tube wear at RB locations. No indications of TTW or other conditions associated with the FEI in Unit 3 (i.e., AVB wear extending outside the supports) were observed.

Rotating Pancake Coil ECT and Ultrasonic Testing (UT) were performed to measure the tube-to-AVB gap sizes in the Unit 2 SGs. Tube-to-AVB gap data was used to validate the contact force distribution model used in the TTW OA, (Attachment 6, Appendix B).

Figure 7-1: 2E-088 Rotating Coil Inspection Region

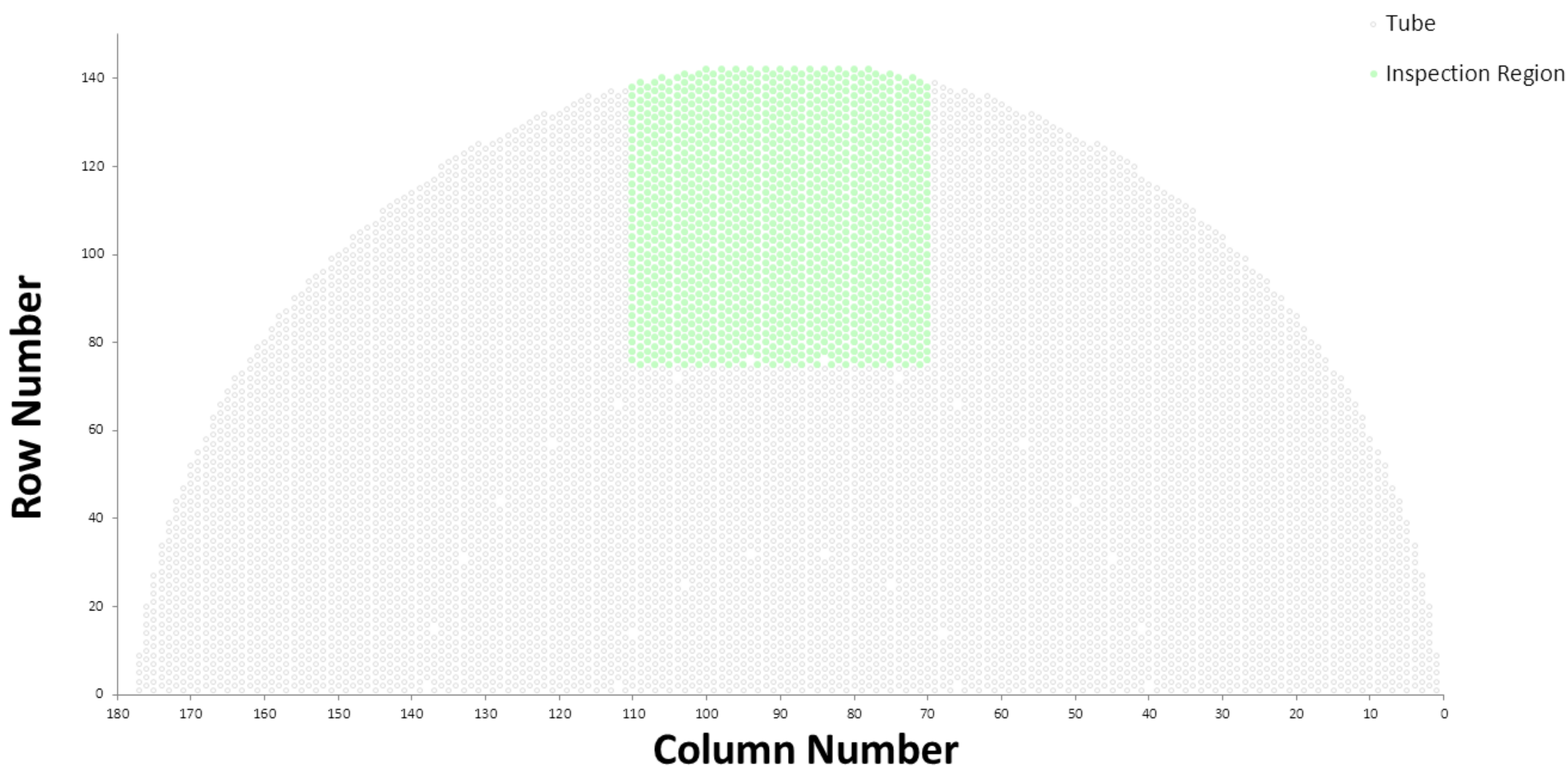
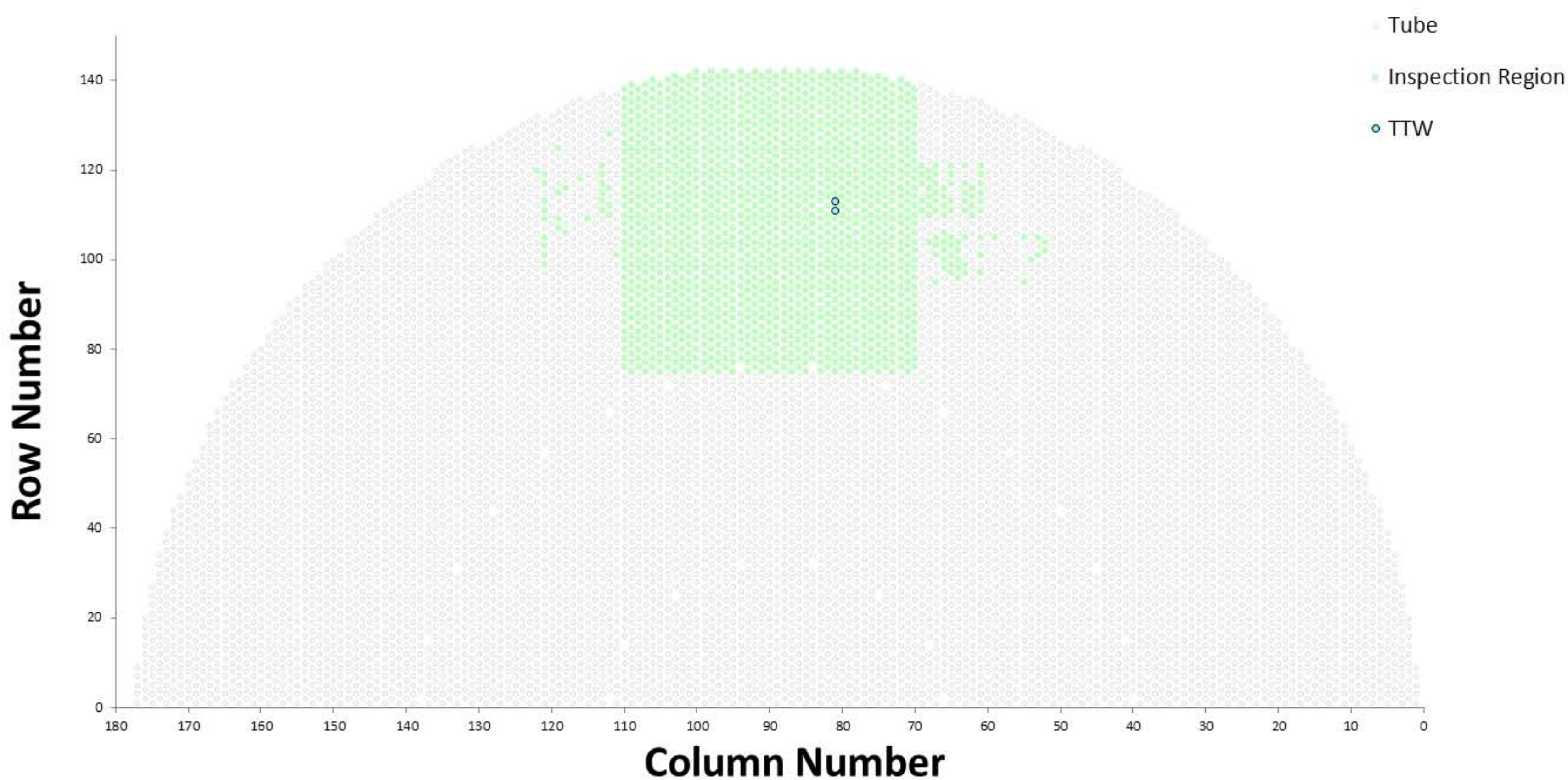


Figure 7-2: 2E-089 Rotating Coil Inspection Region



7.3 Differences between Units 2 and 3

As discussed in Section 6, inspections of the Unit 3 SG's found significant levels of TTW while Unit 2 SGs were limited to two shallow indications at one area of contact between two tubes.

A comparison of TTW and of factors associated with TTW between Unit 2 and Unit 3 SGs is provided below:

Table 7-1: TTW Comparison between Unit 2 and Unit 3 SGs

Description	Unit 2	Unit 3
TTW Indications	2	823
TTW Tubes	2	326
Max Depth (ECT %TW)	14%	99%
Max Length (inches)	~6	~41
TTW In-Situ Pressure Tests	0	129
TTW In-Situ Pressure Tests (Unsatisfactory)	0	8
Operating Period (EFPD)	627	338

In addition to the above parameters, differences in manufacturing dimensional tolerance dispersion (distribution of dimensional values for manufacturing parameters that remain within acceptable tolerances) exist between the Units 2 and 3 SGs. Manufacturing process improvements implemented during the fabrication of the Unit 3 SGs resulted in lower manufacturing dispersion than in the Unit 2 SGs. MHI concluded that the reduced manufacturing dispersion in the Unit 3 SGs resulted in smaller average tube-to-AVB contact force than in the Unit 2 SGs. Due to the smaller average tube-to-AVB contact force, Unit-3 was more susceptible to in-plane vibration.

8.0 UNIT 2 CORRECTIVE AND COMPENSATORY ACTIONS TO ENSURE TUBE INTEGRITY

SCE has implemented the following corrective and compensatory actions to prevent the loss of SG tube integrity due to TTW in Unit 2:

1. Limiting Unit 2 to 70% power prior to a mid-cycle SG inspection outage (CAL Response Commitment 1)

2. Preventively plugging tubes in both SGs (complete)

3. Shutting down Unit 2 for a mid-cycle SG inspection outage within 150 cumulative days of operation at or above 15% power (CAL Response Commitment 2)

The actions to operate at reduced power and perform a mid-cycle inspection within 150 cumulative days of operation are interim compensatory actions. SCE will reevaluate these actions during the mid-cycle inspection using data obtained during the inspections. In addition, SCE has established a project team to develop and implement a long term plan for repairing the SGs. SCE will keep the NRC informed of any findings or developments in the future.

SCE has performed an OA to assess the adequacy of the compensatory actions taken in Unit 2. The OA results demonstrate that operating at 70% power level will prevent loss of tube integrity due to TTW. In particular, reducing power to 70% eliminates the T/H conditions that cause FEI and associated TTW from the SONGS Unit 2 SGs. The OA and supporting analyses are summarized in Section 10 and provided in Attachment 6.

8.1 Limit Operation of Unit 2 to 70% Power

SCE will administratively limit Unit 2 to 70% reactor power prior to a mid-cycle SG inspection outage. The cause of the TTW in the Unit 3 SGs was in-plane tube vibration due to FEI, resulting in tube-to-tube contact and wear. An indication of whether a tube is susceptible to FEI is a calculated term defined as the stability ratio (SR). The SR calculation takes into account T/H conditions (including fluid flow and damping) and tube support conditions and provides a measure of the margin to a critical velocity value at which the tubes may experience the onset of instability due to FEI. The OA and its supporting analyses provided in Section 10 and Attachment 6 demonstrate that operating at 70% power will result in acceptable SRs in Unit 2.

Three independent comparisons were performed of the T/H parameters of SONGS RSGs operating at 100% and 70% power. SONGS RSG's were compared with five operating plants with recirculating SGs of similar design that have not observed TTW. The SONGS RSG's were also compared with the SONGS OSGs. The comparisons were conducted as follows:

(1) SCE Engineering conducted a study of average T/H parameters
(2) WEC performed an Analysis of Thermal-Hydraulics of Steam Generators (ATHOS) study of SONGS RSGs to OSGs
(3) An industry expert in SG design performed an independent ATHOS comparison of T/H parameters that can influence FEI

Based on these comparisons, Plant A was selected for detailed analysis due to similarity of design characteristics and thermal power rating. Both SONGS and Plant A SGs use a U-bend design with the same tube diameter and pitch. Plant A operates at 1355 megawatts thermal per SG (MWt/SG) bounding the SONGS RSGs at 70% power (1210 MWt/SG). Plant A RSGs and SONGS RSGs utilitize out-of-plane AVBs in the U-bend. Plant A RSGs have operated for two fuel cycles without indications of TTW.

Results of the comparisons of three T/H parameters (steam quality, void fraction, and fluid velocity) are presented in the following subsections. These results demonstrate that operating SONGS SGs at 70% power improves the T/H parameters to values lower than those in Plant A at 100% power.

Steam Quality

Steam quality, defined as mass fraction of vapor in a two-phase mixture, is an important factor used in determining SRs. Steam quality is directly related to void fraction for a specified saturation state. This description is important when considering effects on damping. Damping is the result of energy dissipation and delays the onset of FEI. Damping is greater for a tube surrounded by liquid compared to a tube surrounded by gas. Since quality describes the mass fraction of vapor in a two-phase mixture, it provides insight into the fluid condition surrounding the tube. A higher steam quality correlates with dryer conditions and provides less damping. Conversely, lower steam quality correlates with wetter conditions resulting in more damping, which decreases the potential for FEI.

Steam quality also directly affects the fluid density outside the tube, affecting the level of hydrodynamic pressure that provides the motive force for tube vibration. When the energy imparted to the tube from hydrodynamic pressure (density times velocity squared or ρv^2) is greater than the energy dissipated through damping, FEI will occur. When steam quality decreases, the density of the two-phase mixture increases, decreasing velocity. Since the hydrodynamic pressure is a function of velocity squared, the velocity term decreases faster than the density increases. Small decreases in steam quality significantly decrease hydrodynamic pressure and the potential for FEI.

Steam quality in the SONGS RSGs was calculated for 100% and 70% power using the industry expert's independent ATHOS model and compared to Plant A at 100% power. The results of the calculations are summarized in Table 8-1 and graphically presented in Figure 8-1.

Limiting SONGS power to 70% reduces steam quality and hydrodynamic pressure to values less than Plant A. Plant A has not experienced TTW.

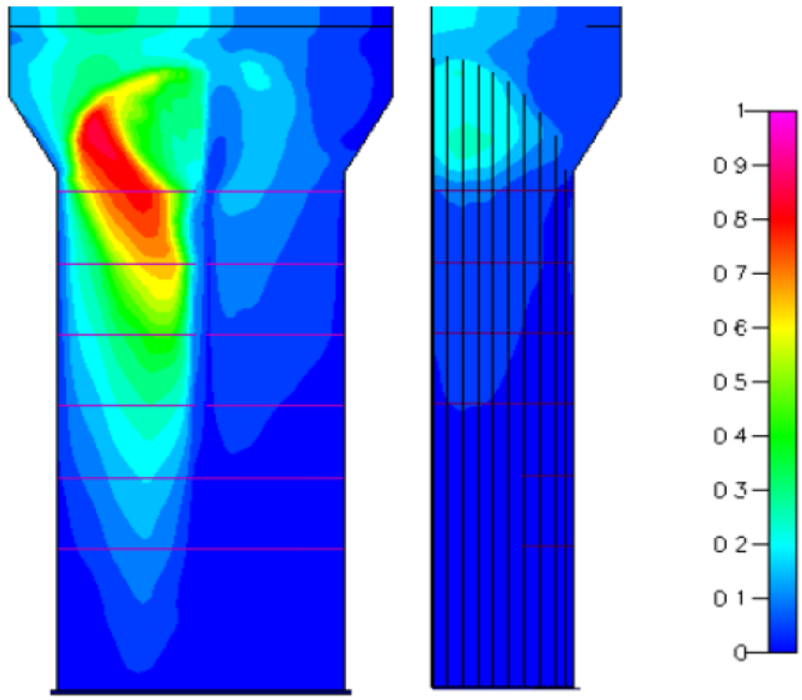
Table 8-1: Independent ATHOS Comparison Results – Steam Quality

	SONGS 100%	SONGS 70%	Plant A 100%
Thermal Power (MWt)	1715	1199	1368
Primary Inlet Temp (°F)	597.8	589.1	596.0
Maximum Mixture Density (kg/m3)	782	772	782
Minimum Mixture Density (kg/m3)	34	97	43
Maximum Dynamic Pressure (N/m2)	4140	2430	4220
Maximum Steam Quality	0.876	0.312	0.734

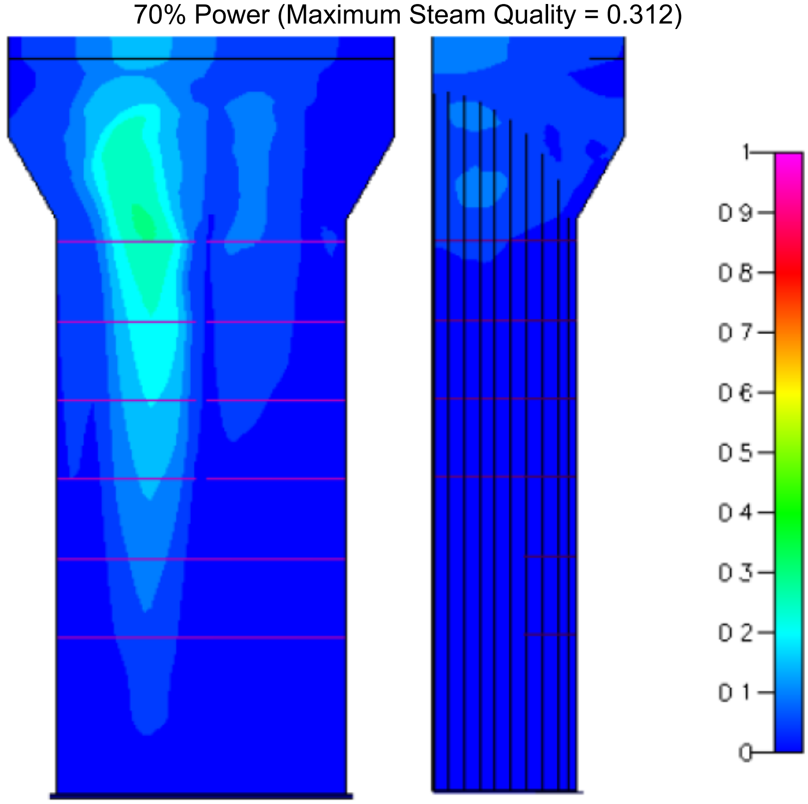
Note: The thermal power levels were calculated in the independent ATHOS comparison.

Figure 8-1: Steam Quality Contour Plots for 100% Power and 70% Power

100% Power (Maximum Steam Quality = 0.876 from Independent ATHOS T/H Comparision)



70% Power (Maximum Steam Quality = 0.312)



Void Fraction

Void fraction, defined as volume fraction of vapor in a two-phase mixture, is a factor used in determining SRs. A higher void fraction represents a lower percentage of liquid in the steam. Liquid in the steam dampens the movement of tubes. Higher void fractions result in less damping. Decreasing the void fraction in the upper bundle region during power operation increases damping and reduces the potential for FEI.

The void fraction in the SONGS RSGs was calculated at 100% and 70% power using ATHOS models from MHI, an independent industry expert, and WEC. The results are summarized in Table 8-2.

A significant effect of limiting power to 70% is the elimination of void fractions greater than Plant A. Plant A has not experienced TTW.

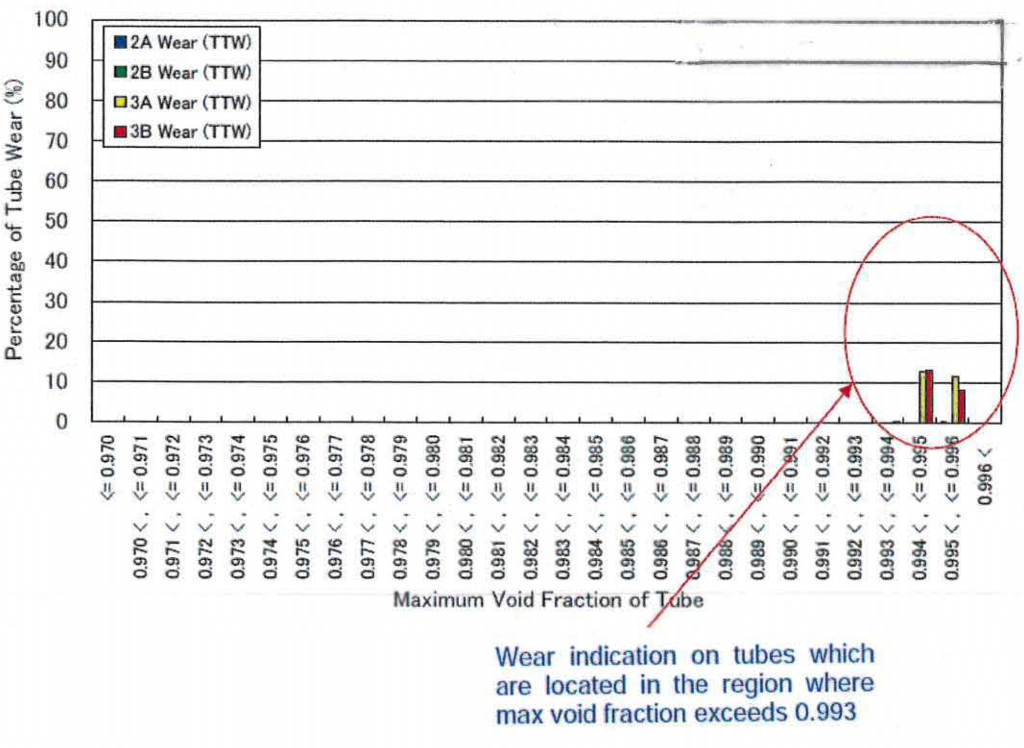
Table 8-2: Comparison of Maximum Void Fraction

	SONGS 100%	SONGS 70%	Plant A 100%	SONGS OSGs 100%
Thermal Power (MWt)	1729	1210	1355	1709
Bend Type	U-Bend	U-Bend	U-Bend	Square Bend
MHI ATHOS T/H Results	0.996	0.927	-	-
Independent ATHOS T/H Comparison	0.994	0.911	0.985	-
WEC ATHOS T/H Comparison	0.9955	0.9258	-	0.9612

Note: Not all sources had access rights to the ATHOS models of some of the comparison plants, resulting in blank cells in this table.

Void fractions at the locations of tubes with TTW in the RSGs are shown in Figure 8-2. The figure demonstrates that the occurrence of TTW was limited to tubes operating with maximum void fractions of greater than 0.993.

Figure 8-2: Maximum Void Fraction versus Power Level and Ratio of Tube Wear versus Maximum Void Fraction



By limiting power to 70% as presented in Table 8-2, void fractions are reduced to levels well below those associated with the TTW experienced at 100% power in the SONGS RSGs.

Fluid Velocity

The fluid velocity in a steam generator's secondary side is a factor in SR calculations. Hydrodynamic pressure is the fluid velocity squared multiplied by the fluid density (ρv^2) and is described in the "Steam Quality" section above.

The results of the velocity calculations are summarized in Table 8-3 and a graphical presentation of the results throughout a SG is shown in Figure 8-3. Interstitial velocity is a representative average velocity of flow through a porous media, which accounts for the structures and flow obstructions in the flow path.

Table 8-3: Comparison of Maximum Interstitial Velocity (ft/s)

	SONGS 100%	SONGS 70%	Plant A 100%	SONGS OSGs 100%
Thermal Power (MWt)	1729	1210	1355	1709
Bend Type	U-Bend	U-Bend	U-Bend	Square Bend
MHI ATHOS T/H Results	23.60	13.38	-	-
Independent ATHOS T/H Comparision	22.08	11.91	17.91	-
WEC ATHOS T/H Comparison	28.30	13.28	-	22.90

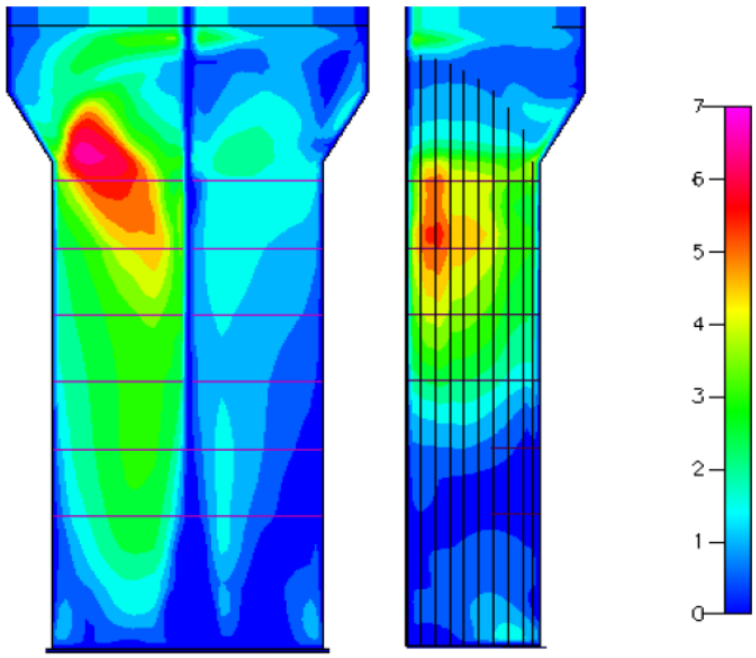
Note: Not all sources had access rights to the ATHOS models of some of the comparison plants, resulting in blank cells in this table.

An additional analysis of velocity at different locations along a tube at 100% and 70% power was performed by WEC. This analysis used gap velocity, which relates to interstitial velocity through the geometric arrangement of the tube bundle and the angle of incidence between the fluid flow and tube (interstitial velocity multiplied by a surface porosity based on the tube bundle geometry). Tube R141 C89 has the longest bend radius in the bundle and relatively high gap velocities. A significant reduction in gap velocity for this tube occurs in the U-bend (mainly the hot leg side) when power is limited to 70%. The results for 2E-088 are shown in Figure 8-4, and results for 2E-089 are shown in Figure 8-5. The slight differences in the plots for the two SGs are caused by differences in numbers and locations of plugged tubes.

Limiting power to 70% significantly reduces fluid velocity. The reduction in fluid velocity significantly reduces the potential for FEI.

Figure 8-3: Interstitial Velocity Contour Plots for 100% Power and 70% Power

100% Power (Maximum Interstitial Velocity = 6.73 m/s = 22.08 ft/s
from Independent ATHOS T/H Comparison)



70% Power (Maximum Interstitial Velocity = 3.63 m/s = 11.91 ft/s)

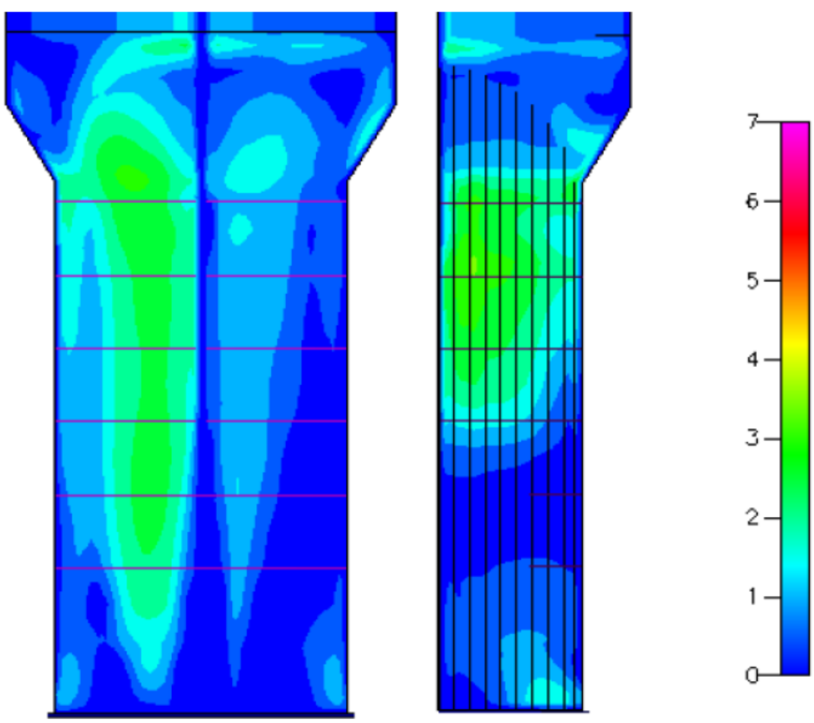
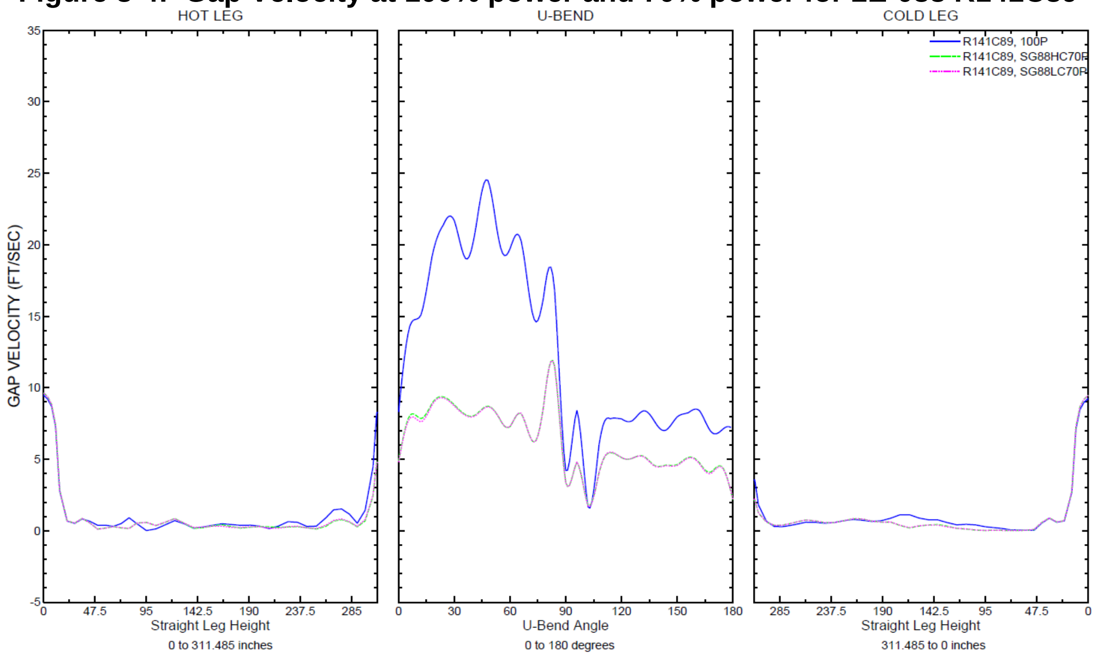
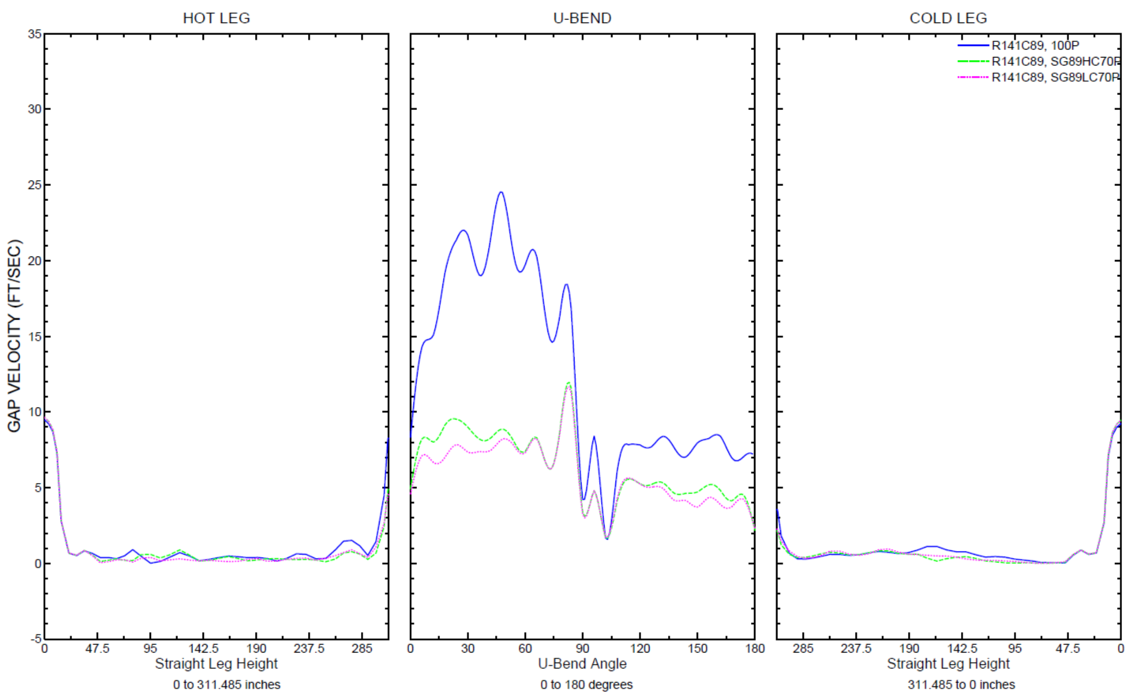


Figure 8-4: Gap Velocity at 100% power and 70% power for 2E-088 R141C89



* Note: Two lines are shown for 70% power because separate ATHOS simulations were run for each half of the tube bundle due to the asymmetrical plugging in the SG

Figure 8-5: Gap Velocity at 100% power and 70% power for 2E-089 R141C89



* Note: Two lines are shown for 70% power because separate ATHOS simulations were run for each half of the tube bundle due to the asymmetrical plugging in the SG.

MHI's ATHOS model was used to calculate the T/H input parameters for the SR calculations. ATHOS is an EPRI computer program used by SG design companies in North America. SCE commissioned two independent T/H analyses to verify the MHI ATHOS analysis. These independent verifications were performed by WEC using ATHOS and AREVA using their T/H computer code CAFCA4. MPR Associates compared the three T/H analyses (MHI ATHOS, WEC ATHOS, and AREVA CAFCA4) and concluded the models predicted similar void fraction, quality, and velocity results.

8.2 Preventive Tube Plugging for TTW

Tubes were identified for preventive plugging using correlations between wear characteristics in Unit 3 tubes and wear patterns at AVBs and TSPs in Unit 2. The screening criteria used to select these tubes is discussed in Section 8.2.1. Removing these tubes from service prevents future wear from challenging SG performance criteria for structural and leakage integrity. These tubes were plugged in addition to the 4 tubes plugged for AVB wear and the 182 tubes plugged as a preventive measure against potential RB wear (described in Section 7.1). A summary of all tubes selected for plugging in Unit 2 is provided in Table 8-4. The impact on operations of the plugged tubes is discussed in Section 8.2.2.

8.2.1 Screening Criteria for Selecting Tubes for Plugging

After identification of the TTW in Unit 3, additional examinations of the susceptible region in Unit 2 identified shallow TTW on two adjacent tubes. Although the 14% TW depth of these indications was below the TS plugging threshold of 35%, the tubes were stabilized and plugged to reduce the risk of tube failure due to continued wear. Using screening criteria developed by MHI from TTW indications in Unit 3, SCE selected 101 tubes in 2E-088 and 203 tubes in 2E-089 for preventive plugging. Nine screening criteria were identified using the quantity and location of AVB and TSP wear indications, length of AVB wear indications, average void fraction over the length of the tube, location of the tube within the tube bundle, and coupling between adjacent susceptible tubes. These criteria are provided in Attachment 5.

Table 8-4 provides a summary of all the tubes selected for plugging in Unit 2. The locations of the Unit 2 tubes selected for plugging and stabilization using the preventive plugging criteria are shown in Figure 8-6 and Figure 8-7. Additional screening criteria was provided by industry expert review (wear at 6 Consecutive AVBs) and WEC (TSP wear).

Table 8-4: Unit 2 Steam Generator Tube Plugging Summary

						TTW Preventive			
Steam Generator	TWD ≥ 35% at AVB	TWD 30-35% at AVB	Wear at RB	TTW	Preventive Retainer Bar	MHI Screening Criteria	Wear at 6 Consecutive AVBs	WEC Screening Additions	Total Tubes Selected
2E-088	2	2	2	0	92	101	6	2	207
2E-089	0	0	4	2	90	203	6	3	308

Figure 8-6: 2E-088 Plugging and Stabilizing Map

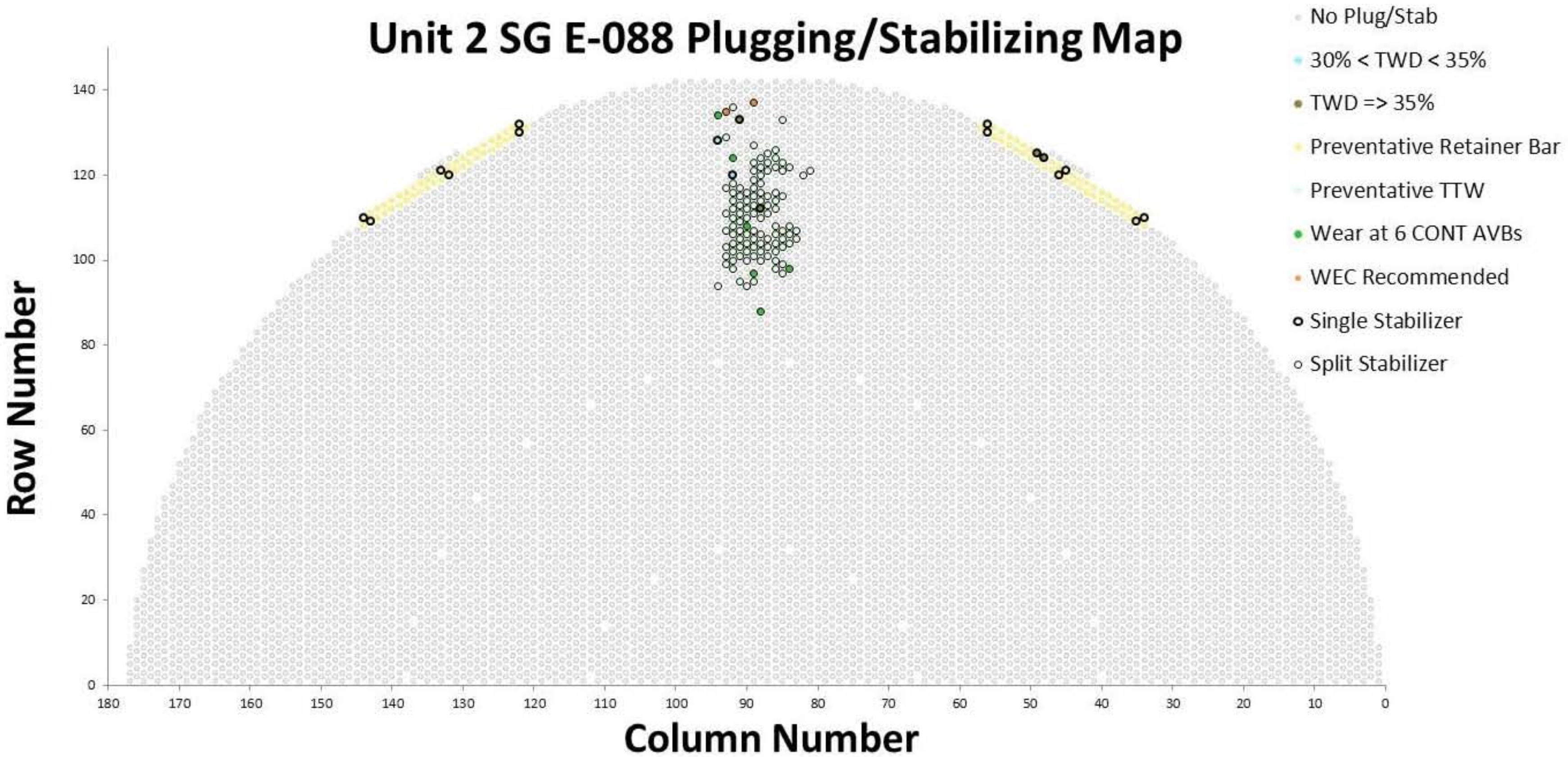
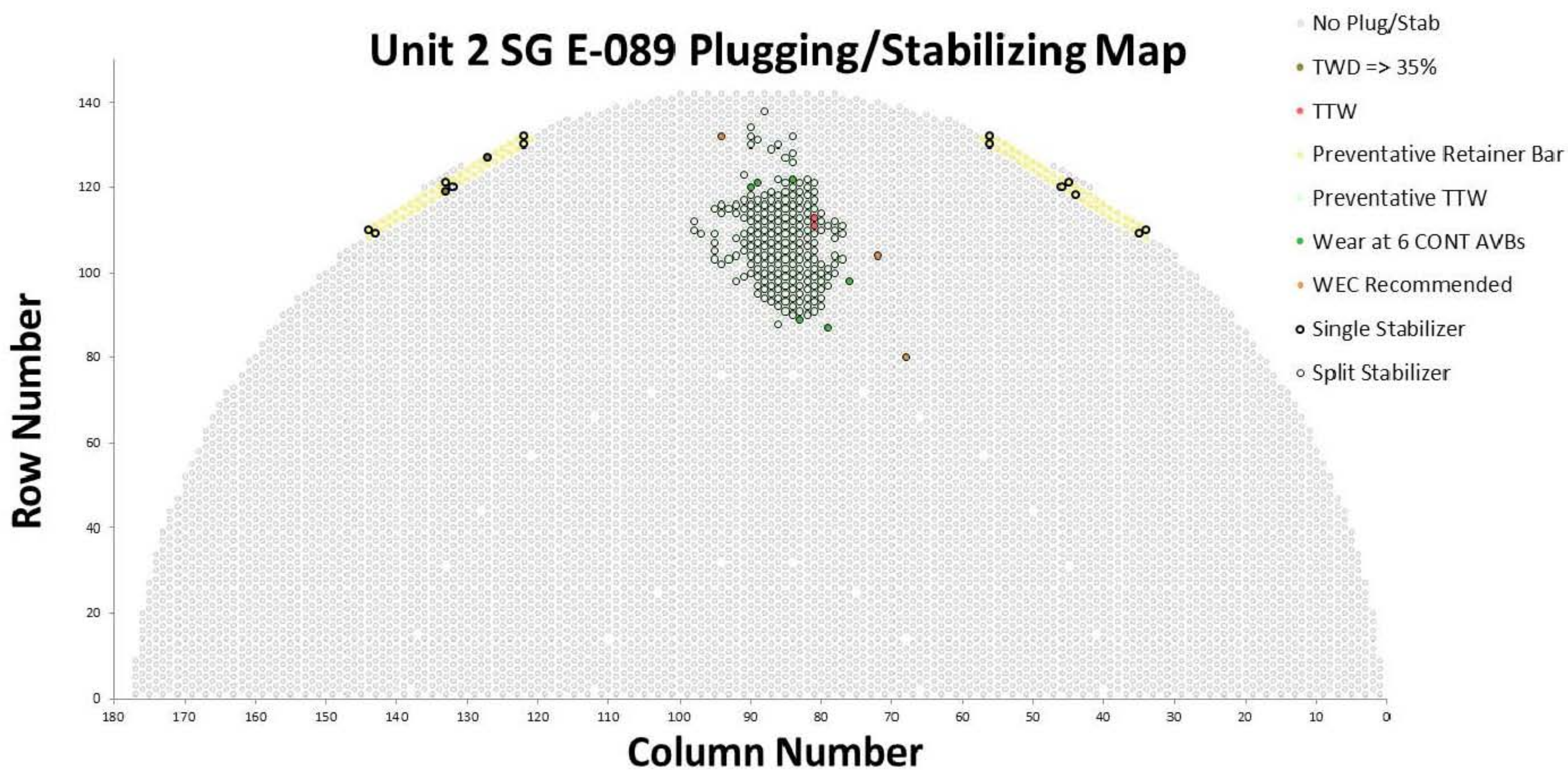


Figure 8-7: 2E-089 Plugging and Stabilizing Map



8.2.2 Plant Operations with Tubes Plugged in Unit 2

Results from MHI's ATHOS calculations were used to analyze the effect of the plugged region on tubes remaining in service. The T/H parameters evaluated were:

- Maximum void fraction, velocity, and hydrodynamic pressure along the U-bend
- Average void fraction, velocity, and hydrodynamic pressure along the hot leg portion of the U-bend
- Average void fraction, velocity, and hydrodynamic pressure along the U-bend

The effect of 4% tube plugging on the remaining in-service tubes was evaluated and determined to be insignificant.

With power limited to 70%, there is no adverse impact on surrounding tubes of the preventive plugging in the Unit 2 SGs.

8.3 Inspection Interval and Protocol of Mid-cycle Inspections

As demonstrated in Section 8.1, limiting operations to 70% power significantly reduces the potential for FEI and improves tube stability margins. To provide additional safety margin, the Unit 2 inspection interval has been limited to 150 days of operation at or above 15% power. The protocol for the inspections to be performed during the mid-cycle outage is described below. (CAL Response Commitment 2)

8.3.1 Inspection of Inservice Tubes (Unplugged)

The following inspections will be performed during the mid-cycle SG inspection outage:

- Eddy Current Bobbin Coil Examinations of the full length of all in-service tubes
- Rotating Coil Examinations of the following areas:
    a. U-bend region – inspection scope will repeat the pattern used during the refueling outage. (~1300 tubes/SG)
    b. TSP and AVB wear bobbin coil indications ≥ 20%
- Visual inspection of small diameter RBs and welds

8.3.2 Inspection of Plugged Tubes

Plugged tubes will be inspected to determine if the compensatory and corrective actions (plugging and operating at reduced power) have been effective. The following inspections and evaluations are planned:

- Visual examination will be performed on all installed tube plugs
- 12 tubes in each SG will be unplugged and the stabilizer(s) removed to assess the effectiveness of the TTW compensatory and corrective actions. Following these inspections, all tubes will be re-plugged and stabilizers installed. The tubes will be selected as follows:
    - The 2 tubes with previous TTW indications
    - 5 tubes adjacent to tubes with TTW wear
    - 5 tubes selected from representative locations that were preventively plugged as part of the compensatory and corrective actions for TTW

- Any new TTW and TSP ECT indications will be assessed to determine if they are the result of FEI during the prior operating period or are cases of previously undetected wear (less than the probability of detection for the ECT probes used during the prior inspection).
- Confirmed new TTW or increases in TTW indication size beyond ECT uncertainty will require a review of the corrective actions implemented during the current inspection.

9.0 UNIT 2 DEFENSE-IN-DEPTH ACTIONS

As described in Section 8, Section 10, and Attachment 6, the compensatory and corrective actions taken by SCE eliminate the T/H conditions that cause FEI and associated TTW from the SONGS SGs. Nonetheless, SCE has developed DID measures to provide an increased safety margin even if tube-to-tube degradation in the Unit 2 SGs were to occur. The following actions have been taken to improve the capability for early detection of a SG tube leak and ensure immediate plant operator response.

9.1 Injection of Argon into the Reactor Coolant System (RCS)

Plant design has been modified to allow periodic injection of Argon (Ar-40) into the RCS. Ar-40 is activated over a short period of time to become Ar-41. The increased RCS activity makes it easier to detect primary-to-secondary tube leaks.

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

Plant design will be modified prior to Unit 2 startup (entry into Mode 2) by installing a temporary N-16 radiation detection system (CAL Response Commitment 3). This system is in addition to existing radiation monitoring systems and includes temporary N-16 detectors located on the main steam lines. This system provides earlier detection of a tube leak and initiation of operator actions.

9.3 Reduction of Administrative Limit for RCS Activity Level

The plant procedure for chemical control of primary plant and related systems has been modified to require action if the specific activity of the reactor coolant Dose Equivalent (DE) Iodine (I-131) exceeds the normal range of 0.5 µCi/gm, which is one-half of the TS Limit of 1.0 µCi/gm. In the event that the normal range is exceeded, Operations is required to initiate the Operational Decision Making process to evaluate continued plant operation.

9.4 Enhanced Operator Response to Early Indication of SG Tube Leakage

9.4.1 Operations Procedure Changes

The plant operating procedure for responding to a reactor coolant leak has been modified to require plant Operators to commence a reactor shutdown upon a valid indication of a primary-to-secondary SG tube leak at a level less than allowed by the plant's TSs. This procedure change requires earlier initiation of operator actions in response to a potential SG tube leak.

9.4.2 Operator Training

Plant Operators will receive training on use of the new detection tools for early tube leak identification (e.g., plant design changes described above), and lessons learned in responding to the January 31, 2012, Unit 3 shutdown due to a SG tube leak (CAL Response Commitment 4). This training will enhance operator decision making and performance in responding to an indication of a SG tube leak and will be completed prior to plant startup.

10.0 UNIT 2 OPERATIONAL ASSESSMENT

As defined in NEI 97-06 (Ref. 2), the OA is a “Forward looking evaluation of the SG tube conditions that is used to ensure that the structural integrity and accident leakage performance will not be exceeded during the next inspection interval.” The OA projects the condition of SG tubes to the time of the next scheduled inspection outage and determines their acceptability relative to the TS tube integrity performance criteria (Attachment 1).

As required by the CAL (Ref. 1), SCE has prepared an assessment of the Unit 2 SGs that addresses the causes of TTW wear found in the Unit 3 SGs, prior to entry of Unit 2 into MODE 2. The OA provided in Attachment 6 provides that assessment.

Due to the significant levels of TTW found in Unit 3 SGs, SCE has assessed the likelihood of additional TTW in Unit 2 from several different perspectives involving the experience and expertise of AREVA, WEC, and Intertek/APTECH. These companies developed independent OAs to address the TTW found at SONGS. These OAs apply different methodologies to ensure a comprehensive and diverse evaluation. The results of these analyses fulfill the TS requirement to demonstrate that SG tube integrity will be maintained until the next SG inspection. The OAs demonstrate that limiting operation to 70% power will prevent loss of tube integrity due to TTW. In particular, reducing power to 70% eliminates the T/H conditions that cause FEI and associated TTW from the SONGS Unit 2 SGs. The reduced 150 cumulative day inspection interval provides additional safety margin beyond the longer allowable inspection intervals identified in the OAs.

11.0 ADDITIONAL ACTIONS

As previously discussed, the OAs performed by AREVA, WEC, and Intertek/APTECH confirm that the compensatory and corrective actions implemented by SCE will result in continued safe operation of Unit 2 and that SG tube integrity will be maintained. SCE also implemented conservative DID measures to minimize the impact on public and environmental health and safety even if tube integrity were compromised. Additionally, SCE is establishing enhanced plant monitoring capability as described below.

11.1 Vibration Monitoring Instrumentation

The Vibration and Loose Parts Monitoring System (VLPMS) is designed in accordance with NRC Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors" to detect loose metallic parts in the primary system. VLPMS includes accelerometers mounted externally to the SGs. The VLPM sensors detect acoustic signals generated by loose parts and flow. The signals from these sensors are compared with preset alarm setpoints. Validated alarms are annunciated on a panel in the control room.

To improve sensitivity of the VLPMS, the system is being upgraded to WEC's Digital Metal Impact Monitoring System (DMIMS-DX) during U2C17 refueling outage (CAL Response Commitment 5). The following improvements will be implemented by the upgrade:

- Relocation of existing VLPMS accelerometers (2 per SG) from the support skirt to locations above and below the tubesheet. These will remain as VLPMS sensors to meet Regulatory Guide 1.133.
- Increased sensitivity accelerometers (2 per SG) will be installed at locations above and below the tubesheet.
- Increased sensitivity accelerometers (2 per SG) will be installed on an 8 inch hand hole high on the side of the SGs to monitor for secondary side noises at the upper tube bundle.

The upgraded system will provide SCE with additional monitoring capabilities for secondary side acoustic signals.

11.2 GE Smart Signal™

SCE will utilize GE Smart Signal™, which is an analytic tool that aids in diagnosis of equipment conditions (CAL Response Commitment 6). The tool will be used to analyze historical plant process data from the Unit 2 SGs following the inspection interval.

12.0 CONCLUSIONS

As noted in Reference 1, the SG tube wear that caused a Unit 3 SG tube to leak on January 31, 2012, was the result of tube-to-tube interaction. This type of wear was confirmed to exist in a number of other tubes in the same region in both Unit 3 SGs. Subsequent inspections of the Unit 2 SGs identified this type of wear also existed in two adjacent tubes in Unit 2 SG E-089.

To determine the cause of the TTW, SCE performed extensive inspections and analyses. SCE commissioned experts in the fields of T/H and in SG design, manufacturing, operation, and repair to assist with these efforts. Using the results of these inspections and analyses, SCE determined the cause of the TTW in the two Unit 3 SGs was FEI, caused by a combination of localized high steam velocity, high steam void fraction, and insufficient contact forces between the tubes and the AVBs. FEI caused in-plane tube vibration that resulted in TTW in a localized region of the SGs. The TTW in Unit 2 SG E-089 may have been caused by FEI, or alternatively, close proximity of the two tubes may have led to TTW from normal vibration.

SCE determined the TTW effects were much less severe in Unit 2 where two tubes were identified with TTW indications of less than 15% TW wear. These two tubes are located in the same region of the SGs as those with TTW in Unit 3. Given that the T/H conditions are essentially the same in both units, the less severe TTW in Unit 2 is attributed to manufacturing differences. Those differences increased tube-to-AVB contact forces in Unit 2, providing greater tube support.

To prevent loss of SG tube integrity due to TTW in Unit 2, SCE has implemented interim compensatory and corrective actions and established a protocol of inspections and operating limits. These include:

1. Limiting Unit 2 to 70% power prior to a mid-cycle SG inspection outage (CAL Response Commitment 1)
2. Preventively plugging tubes in both SGs (complete)
3. Shutting down Unit 2 for a mid-cycle SG inspection outage within 150 cumulative days of operation at or above 15% power (CAL Response Commitment 2)

On the basis of the compensatory and corrective actions discussed in Section 8, the DID actions presented in Section 9, and the results of the OAs presented in Section 10 and Attachment 6, SCE concludes that Unit 2 will operate safely at 70% power for 150 cumulative days of operation. Reducing power to 70% eliminates the T/H conditions that cause FEI and associated TTW from the SONGS Unit 2 SGs. SCE will continue to closely monitor SG tube integrity, perform SG inspections during the mid-cycle outage, and take compensatory and corrective actions to ensure the health and safety of the public.

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13.0 REFERENCES

1 Confirmatory Action Letter (CAL) – Letter from Elmo E. Collins (NRC) to Peter T. Dietrich (SCE), dated March 27, 2012, Confirmatory Action Letter 4-12-001, San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation

2 Nuclear Energy Institute NEI 97-06, Steam Generator Program Guidelines, Revision 3, January 2011

3 Electric Power Research Institute (EPRI), Pressurized Water Reactor Steam Generator Examination Guidelines

4 EPRI 1019038, 1019038 Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 3, November 2009

5 Event Notification Number 47628 - Telephone notification, Manual Trip Due to a Primary to Secondary Leak, made to the NRC Emergency Notification System (ENS) as required by 10 CFR 50.72(b)(2)(iv)(B)

6 Event Notification Number 47744 (including 2 followups) - Telephone notifications, Unit 3 Steam Generator Tubes Failed In-Situ Pressure Testing, made to the NRC ENS as required by 10 CFR 50.72(b)(3)(ii)(A)

7 Unit 3 LER 2012-001, dated March 29, 2012, Manual Reactor Trip Due to the SG Tube Leak as required by 10 CFR 50.73(a)(2)(iv)(A), actuation of the Reactor Protection System

8 Unit 3 LER 2012-002, dated May 10, 2012, SG Tube Degradation Indicated by Failed In-situ Pressure Testing as required by 10 CFR 50.73(a)(2)(ii)(A), a condition which resulted in a principal safety barrier being seriously degraded (i.e., serious SG tube degradation)

9 Letter from Peter T. Dietrich (SCE) to Elmo Collins (USNRC), dated April 20, 2012, Update of Unit 2 SG Tube Inspection Results

10 SONGS Steam Generator Program (SO23-SG-1)

ATTACHMENT 6

SONGS U2C17
Steam Generator Operational Assessment

[Proprietary Information Redacted]

SOUTHERN CALIFORNIA
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SONGS U2C17
STEAM GENERATOR OPERATIONAL ASSESSMENT

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ABBREVIATIONS AND ACRONYMS

2E-089	Unit 2 Steam Generator E-089
AILPC	Accident-Induced Leakage Performance Criteria
ASME	American Society of Mechanical Engineers
ATHOS	Analysis of Thermal-Hydraulics of Steam Generators
AVB	Anti-Vibration Bar
CE	Combustion Engineering
ECT	Eddy Current Testing
EFPY	Effective Full Power Year
EOC	End of Cycle (fuel)
EPRI	Electric Power Research Institute
ETSS	Examination Technique Specification Sheet
FEI	Fluid Elastic Instability
FOSAR	Foreign Object Search and Retrieval
gpd	Gallons Per Day
gpm	Gallons Per Minute
MHI	Mitsubishi Heavy Industries, Ltd.
NEI	Nuclear Energy Institute
NODP	Normal Operating Differential Pressure
NRC	Nuclear Regulatory Commission
OA	Operational Assessment
POB	Probability of Burst
RCS	Reactor Coolant System
SCE	Southern California Edison
SIPC	Structural Integrity Performance Criteria
SG	Steam Generator
SONGS	San Onofre Nuclear Generating Station
SR	Stability Ratio
T/H	Thermal-Hydraulic
TS	Technical Specifications
TSP	Tube Support Plate
TTW	Tube-to-Tube Wear
U2C17	Unit 2 Cycle 17
UNS	Unified Numbering System
WEC	Westinghouse Electric Company

EXECUTIVE SUMMARY

On January 31, 2012, a leak was detected in a Unit 3 Steam generator (SG) at San Onofre Nuclear Generating Station (SONGS). Southern California Edison (SCE) operators promptly shut down the unit in accordance with approved operating procedures. The resulting small radioactive release to the environment was well below the allowable federal limits. Subsequently, on March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter [1] to SCE describing actions that the NRC and SCE agreed would be completed prior to returning Units 2 and 3 to service. Since that time, SCE's technical team supplemented by a team of experts in the field of thermal-hydraulics and in SG design, manufacture, operation, and maintenance have performed extensive investigations into the causes of the tube leak and have assisted in the development of compensatory measures and corrective actions that will prevent a loss of SG tube integrity.

As required by the SONGS Technical Specifications (TS) [3], SONGS SG Program [2], and industry guidelines [5], an Operational Assessment (OA) must be performed to ensure that SG tubing will meet established performance criteria for structural and leakage integrity during the operating period prior to the next planned inspection. Because of the unusual and unexpected nature of the SG tube-to-tube wear (TTW) at SONGS, SCE commissioned three independent OAs [Appendices B, C, and D] by experienced vendors applying diverse methodologies. The non-TTW degradation mechanisms have been addressed by a separate OA included in this report [Appendix-A]. Each of these methodologies demonstrates that SCE has implemented compensatory measures and corrective actions to ensure that Unit 2 will operate safely with substantial conservative margin. This report contains the OAs that have been performed to demonstrate that those compensatory measures and corrective actions will prevent a loss of SG tube integrity.

1.0 PURPOSE

In accordance with the SONGS SG Program [2] an OA is performed to ensure that SG tubing meets established performance criteria for structural and leakage integrity during the interval prior to the next planned inspection. The OA projects and evaluates tube degradation mechanisms which have affected the SGs. The performance criteria are defined in plant TS [3] [4] and are based on NEI-97-06 [5].

This summary of the OAs establishes operational limits for Unit 2 and provides reasonable assurance, as required by NRC regulations, that Unit 2 will operate safely.

2.0 SONGS STEAM GENERATOR DESIGN FEATURES

The steam generator is a recirculating, vertical U-tube type heat exchanger converting feedwater into saturated steam. The steam generator vessel pressure boundary is comprised of the channel head, lower shell, middle shell, transition cone, upper shell and upper head. The steam generator internals include the divider plate, tubesheet, tube bundle, feedwater distribution system, moisture separators, steam dryers and integral steam flow limiter installed in the steam nozzle. The channel head is equipped with one reactor coolant inlet nozzle and two outlet nozzles. The upper vessel is equipped with the feedwater nozzle, steam nozzle and blowdown nozzle. In the channel head, there are two 18 inch access manways. In the upper shell, there are two 16 inch access manways. The steam generator is equipped with six (6) handholes and 12 inspection ports providing access for inspection and maintenance. In addition, the steam generators are equipped with several instrumentation and minor nozzles for layup and chemical recirculation intended for chemical cleaning (See Figure 2-1 and Figure 2-2).

Note: The SG design information is provided in References [6] [7] [8] [9] [10] [11] [12].

Figure 2-1: AVB Arrangement for SONGS Steam Generators

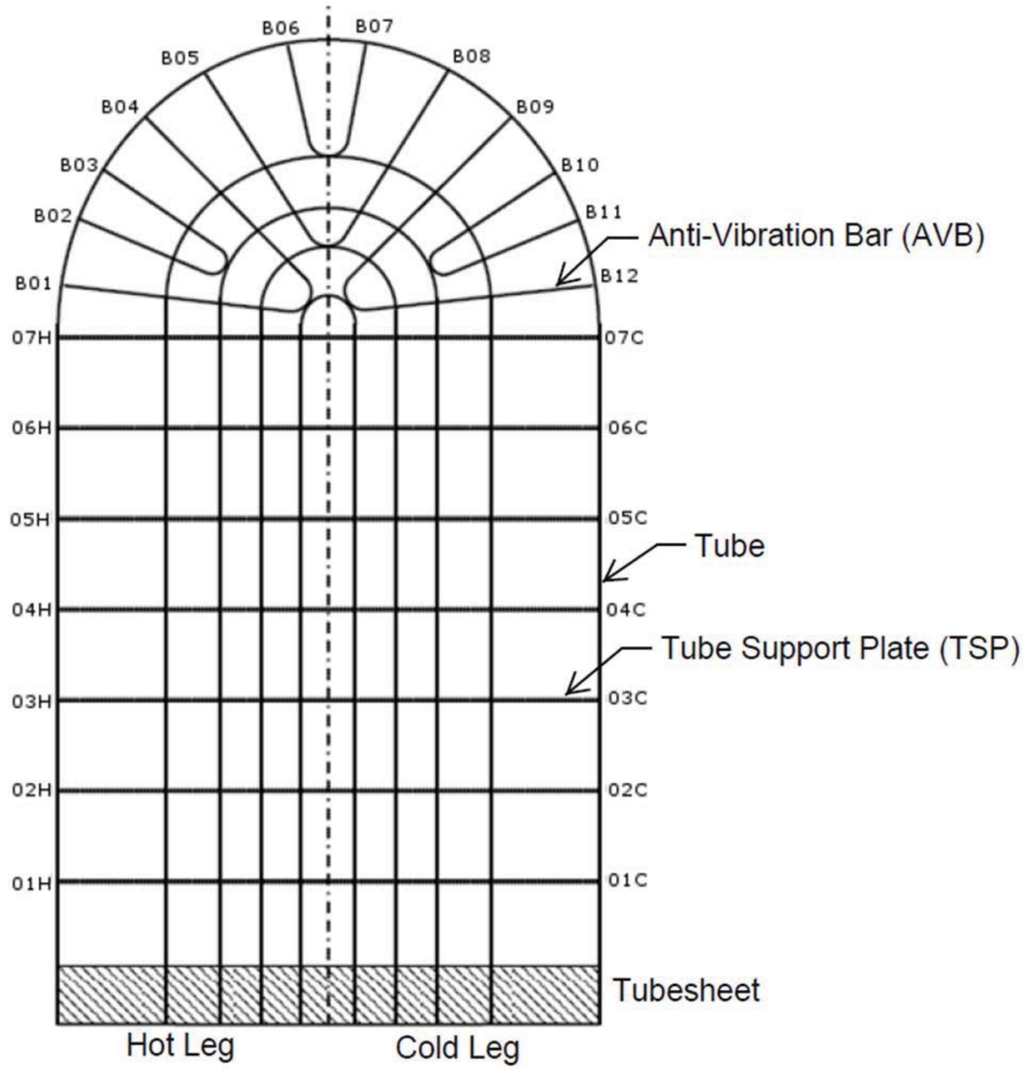


Figure 2-2: Details of AVBs, Retaining Bars, Bridges, and Retainer Bars

3.0 OPERATIONAL ASSESSMENT

As defined in NEI 97-06, the OA is a forward looking evaluation of the SG tube conditions that is used to ensure that the structural integrity and accident leakage performance will not be exceeded during the next inspection interval [5]. The OA projects the condition of SG tubes to the time of the next scheduled inspection outage and determines their acceptability relative to the TS tube integrity performance criteria.

As documented in the "SONGS U2C17 Steam Generator Condition Monitoring Report" [13], the Unit 2 SGs satisfied the three performance criteria specified in the TS for the previous operating period. The SG Program requires an OA to be completed for the next inspection interval within 90 days after initial entry into MODE 4 (MODE is defined in the station TS). This summary of the OAs establishes operational limits for Unit 2 and provides reasonable assurance, as required by NRC regulations, that Unit 2 will operate safely.

The structural integrity performance criteria (SIPC) and accident-induced leakage performance criteria (AILPC) applicable to wear mechanisms are [14]:

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

> Accident-Induced Leakage — "The primary to secondary accident leakage rate for the limiting design basis accident shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rates for an individual steam generator."

The acceptance standard for structural integrity is [14]:

> The worst-case degraded tube shall meet the SIPC margin requirements with at least a probability of 95% at 50% confidence.

The acceptance standard for accident leakage integrity is [14]:

> The probability for satisfying the limit requirements of the AILPC shall be at least 95% at 50% confidence.

The OA may utilize either a deterministic (also known as simplified arithmetic) or a probabilistic methodology.

SCE has assessed all tube wear mechanisms in Unit 2, including TTW. Given the significance of TTW observed in Unit 3, SCE used the experience and expertise of multiple independent companies that routinely perform OAs for the US nuclear industry. AREVA, Westinghouse Electric Company (WEC), and Intertek developed independent OAs to address the TTW found at SONGS. These diverse analyses fulfilled the TS requirement to ensure that SG tube integrity is maintained until the next SG inspection.

- Section 3.1 provides a summary of the OA prepared by AREVA evaluating all degradation mechanisms found in Unit 2 SGs with the exception of TTW. This OA demonstrates there is reasonable assurance that the SIPC and AILPC for non-TTW will be satisfied for 18 months at 100% power.

- Section 3.2 provides a summary of the OA prepared by AREVA. This OA deterministically evaluates the potential for TTW for the limiting condition of no in-plane support. The OA also evaluates probabilistically the potential for in-plane Fluid Elastic Instability (FEI) occurring in Unit 2 based on an analysis of the contact forces between tubes and AVBs. The deterministic results demonstrate all tubes are stable (will not experience Thermal-Hydraulic (T/H) conditions that cause FEI) at 70% power for 18 months of operation without relying on the AVBs for in-plane support. Therefore, this OA demonstrates that the SIPC and AILPC for TTW will be satisfied for 18 months at 70% power. The probabilistic results demonstrate a low probability of FEI at 70% power for approximately 8 months of operation even when additional conservatisms are introduced.
- Section 3.3 provides a summary of the OA prepared by Intertek following "traditional" industry guidelines for assessing SG tube degradation. This OA evaluates the probability that TTW caused by FEI will not exceed the SG SIPC. This OA demonstrates there is a reasonable assurance that the SIPC and AILPC for TTW will be satisfied for 16 months at 70% power level.
- Section 3.4 provides a summary of the OA prepared by WEC based on an alternate interpretation of the inspection results. This OA determines the TTW in Unit 2 was caused by out-of-plane vibration between two tubes in close proximity. The OA evaluates the potential for in-plane instability and concludes the Unit 2 SG tubes were stable in-plane at 100% power. This OA demonstrates there is reasonable assurance that the SIPC and AILPC for TTW will be satisfied for 18 months at 70% power.

3.1 OA for Degradation Mechanisms Other than TTW

The "SONGS U2C17 Outage – Steam Generator Operational Assessment" report [Appendix-A] addresses all degradation mechanisms found in Unit 2 SGs with the exception of TTW. Due to the relatively large number of AVB and TSP wear indications, identified during the U2C17 outage, a probabilistic approach was used to complete the OA for these mechanisms, which included:

- Tube Wear at AVB Locations
- Tube Wear at TSP Locations
- Tube Wear at Retainer Bar Locations
- Tube Wear as a Result of Foreign Object Wear

The objective of this OA is to ensure that structural and leakage performance criteria will be met over the length of the upcoming inspection interval. The OA tube structural integrity requirement is that the projected worst case degraded tube for each existing degradation mechanism shall meet the limiting structural performance parameter with a 95% probability at 50% confidence [3].

AVB and TSP Wear

Because the tube wear indications are flat and long in the axial direction, the limiting requirement for the inspection interval length is structural integrity (i.e. tube burst at 3x NODP). The projected accident-induced leak rates for tube wear will not be limiting since leakage due to ligament pop-through will not precede burst condition at 3x NODP.

The OA uses a probabilistic method to calculate the growth at End of Cycle (EOC) of each indication by randomly sampling from the growth rate distribution yielding one estimate of the EOC depth for each indication. The burst pressure of the worst case degraded tube is calculated and compared with the value of 3 times NODP. This process is repeated thousands of times in order to develop a probability of burst for the worst case degraded tube. If the probability of burst of the worst case degraded tube is less than 5%, then the plugging criteria and inspection interval are satisfactory.

The projected EOC probabilities of burst for the population of indications in each damage mechanism category were calculated for Unit 2 at 100% power for a full cycle of operation (1.577 Effective Full Power Years, EFPY). The projected EOC probabilities are compared with the 95% probability 50% confidence EPRI guidelines [14] criteria to demonstrate the OA structural integrity criteria for AVB and TSP wear are satisfied for a full fuel cycle of operation at 100% reactor power.

Retainer Bar Wear

Because of the potential for continued retainer bar wear of Unit 2, tubes adjacent to retainer bars have been removed from service. Tubes with retainer bar wear indications were stabilized with U-bend cable stabilizers. The tubes on either side of all retainer bars, at each end of the retainer bars, and at the center of the retainer bars, were also stabilized with U-bend cable stabilizers. These corrective actions provide reasonable assurance that retainer bar wear will not challenge the structural and leakage integrity performance criteria during the remaining life of the SGs. In addition, the stabilization of these tubes provides reasonable assurance that a tube severance event will not occur as a result of retainer bar wear. The SG Program [2] will monitor the tubes adjacent to these plugged tubes during future SG inspections.

Foreign Object Wear

All Unit 2 SG tubes were examined full length with Eddy Current Testing (ECT) bobbin coil probes. Two adjacent tubes in SG 2E-089 were identified with foreign object wear indications. The foreign object was identified as weld slag and retrieved from the SG. No other foreign objects were found. The foreign object is not indicative of degradation of secondary side internals.

Because the foreign object has been removed, no potential exists for degradation to progress at these locations. After removal of the object, the affected indications were inspected with ECT. Since the indications are below the SONGS plugging limit and the object was removed, these tubes are left in service.

Based on ECT inspections, secondary side visual examinations, and FOSAR, no foreign objects capable of causing tube degradation remain in the Unit 2 SGs. There is reasonable assurance that foreign objects will not cause the structural or leakage integrity performance criteria to be exceeded prior to the next tube inspection in each SG.

OA for Degradation Mechanisms Other than TTW Conclusion

The OA demonstrates there is reasonable assurance that the SIPC and AILPC for non-TTW will be satisfied for 18 months at 100% power.

3.2 TTW OA Using Tube-to-AVB Support Conditions and Contact Force

The “SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear” [Appendix-B] assesses the TTW degradation mechanism deterministically, without taking credit for in-plane support. The OA also implements a probabilistic approach using tube to AVB contact forces for defining an effective tube support. The OA predicts the probability of in-plane FEI and compares this value to the probabilistic SIPC (95% probability at 50% confidence).

The deterministic approach uses Stability Ratios (SRs) as the criterion for susceptibility to FEI. The SR is calculated conservatively using Thermal-Hydraulic (T/H) and tube support conditions on the secondary side of the SG. The T/H conditions are determined using an ATHOS computer model.

The deterministic approach demonstrates in-plane stability (SR less than 1.0) at 70% power with no effective in-plane AVB supports. This demonstrates TTW will not occur and SIPC limits will be met.

As discussed above, a SR of less than 1.0 indicates the SG tubes will be stable. To demonstrate margin, a probabilistic evaluation was performed assuming instability may occur at a calculated SR as low as 0.75. In the probabilistic approach, the number of effective AVB supports for each tube uses a probabilistic contact force distribution and criteria for determining whether a support is effective for a given contact force. A finite element model of tubes, AVBs, tube-to-AVB gaps, and support structures is used to calculate contact forces at AVB locations. Tube wear inputs to the finite element model are determined from actual wear observed in Units 2 and 3. Results from published technical literature, confirmed by benchmarking the FEI probability model to Unit 3 TTW, indicate that effective supports have a contact force that exceeds a specified value.

SRs are determined for each U-bend tube as a function of the number of consecutive ineffective supports and power level. The distributions of contact forces are calculated for each AVB location in the bundle. Tube wear at AVB locations decreases the contact force at those locations. The required contact force for an AVB support to be considered effective is calculated for each AVB location.

Using the above as inputs, Monte Carlo trials of a SG are simulated. The probability of instability is the number of trials where the SG contained one or more unstable tubes divided by the total number of trials.

TTW OA Using Tube-to-AVB Support Conditions and Contact Force Assessment Conclusion

The deterministic approach demonstrates FEI will not occur. Using a SR of <1.0 at 70% power, the SIPC and AILPC are satisfied for an 18 month inspection interval. The probabilistic approach also demonstrates that there is safety margin in the planned inspection interval of 150 cumulative days at power. The approach demonstrates that if instability is assumed to initiate at a calculated SR of 0.75, rather than a value of 1.0, the SIPC acceptance standard is satisfied for approximately 8 months at 70% power.

3.3 "Traditional" Probabilistic OA for TTW

The "Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16" [Appendix-C] uses established industry methods for assessing degradation mechanisms. This OA uses empirical models for degradation growth and engineering models for determining burst pressure and through-wall leak rates. The non-traditional aspect of this OA is to characterize the presence and severity of TTW degradation indications using wear indices defined by the state of AVB and TSP wear for a specific tube.

Unit 3 wear data establish the initiation and growth of TTW indications in Unit 2 SG. An empirical correlation using a wear index (a measure of the state of wear degradation in each tube) provides the method for comparing the Unit 3 wear to Unit 2. A probabilistic model representing the high-wear region of the tube bundle evaluates TTW for inspection interval. Tube burst and leakage probabilities are calculated by Monte Carlo simulation for initiation and growth of TTW.

Two OA cases are evaluated using the sizing techniques that define the Unit 3 TTW depths. Case 1 evaluates eddy current indication sizing using EPRI ECT Examination Technique Specification Sheet (ETSS) 27902.2 to establish the TTW depth distributions. In Case 2, the TTW depths were determined using a more representative calibration standard.

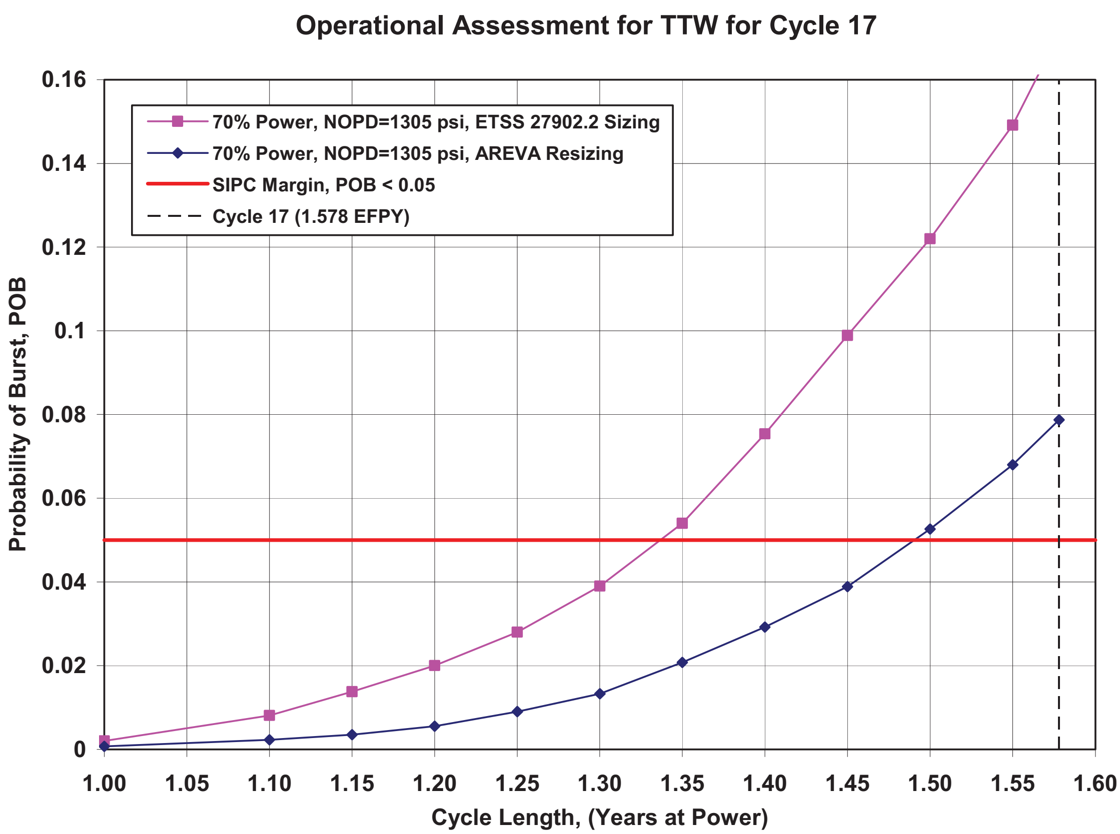
"Traditional" Probabilistic OA for TTW Conclusion

The results for Case 1 indicate that the SIPC margin requirements are satisfied for an inspection interval of 16 months at 70% power. In Case 2, the SIPC margins are met for a cycle length of 17 months at 70% power. The results of this analysis are displayed in Figure 3-1. The figure identifies the probability of burst as a function of operating cycle length (inspection interval) and power.

The SIPC (Tube burst at 3xNOPD) is the limiting requirement for the inspection interval. The AILPC is satisfied since burst margins at 3xNOPD are maintained during the inspection interval.

This OA demonstrates there is a reasonable assurance that the SIPC and AILPC for TTW will be satisfied for 16 months at 70% power level.

Figure 3-1: Traditional Operational Assessment Results



3.4 Deterministic TTW OA

A deterministic TTW OA [Appendix-D] was completed for tube wear at AVBs and TTW. Tube wear projections for in-service tubes confirm the SG performance criteria will be satisfied during the inspection interval. Tube wear projections for plugged tubes confirm that severance will not occur during the inspection interval.

Evaluation of TTW of the two tubes in SG 2E-089 concludes the wear did not result from in-plane vibration of the tubes. ECT data demonstrate the tube wear indications at AVBs did not extend beyond the width of the AVBs in Unit 2. Wear extending beyond the width of AVBs was strongly correlated with Unit 3 tubes with TTW. In-plane SRs indicate that the two Unit 2 tubes with TTW are stable at 100% power. Pre-service inspection data indicates these two tubes were in close proximity prior to SG operation. The OA postulates that during operation out-of-plane vibration and/or turbulence caused the two tubes to wear.

The potential for in-plane vibration leading to TTW in Unit 2 is evaluated by calculating in-plane SRs. The OA methodology predicts in-plane vibration in Unit 3 and confirms the absence of in-plane vibration in Unit 2.

This OA projects the depth of indications to the next inspection using current inspection data. ATHOS results provide the T/H inputs for flow velocity, density, and void fraction along the length of the tube. These conditions are used in the Flow Induced Vibration analysis to generate the SR for out-of-plane and in-plane vibration of the tube for various tube support conditions. The support conditions define whether or not a support location such as an AVB intersection is effective, meaning that the structure provides adequate support with respect to motion of the tube due to vibration. Presence of tube-to-AVB wear indicates an ineffective support.

The vibration analysis results and support conditions are used to make wear projections in the next operating cycle. This calculation is based on empirical test results and involves several input assumptions related to tube-to-AVB gap, the AVB twist, and the wear coefficient between the tube and AVB. The expected ranges of these parameters are known from test results, published data and experience. Wear depth projection is made taking into consideration the inspection results at the current outage. After setting the inputs to match the inspection results for a given indication, the wear calculations are extended to determine the projected wear depth at the next inspection.

Deterministic TTW OA Conclusion

The OA demonstrates there is reasonable assurance that the SIPC and AILPC for TTW will be satisfied for 18 months at 70% power.

3.5 Evaluation of Leakage Integrity

The AREVA non-TTW OA [Appendix-A], Section 6.3, discussed the evaluation of leakage integrity for both in-service and plugged tubes. Since the preparation of the AREVA non-TTW OA, SCE plugged five additional tubes. The five additional tubes resulted in a negligible change to the postulated operational and accident-induced leakage attributed to all of the tube plugs using the methodology from the AREVA non-TTW OA.

The operational leakage performance criterion is met through the plant monitoring program. The accident-induced leakage performance criterion is met by projecting leakage attributed to all degradation mechanisms along with postulated plug leakage and comparing the projected leakage to the allowable accident-induced leak rate limit. For tubes returned to service, the onset of pop-through and leakage for axially oriented indications with limited circumferential extent – the nature of the degradation identified in the Unit 2 SGs – is coincident with burst. None of the identified degradation mechanisms in Unit 2 are projected to exceed the structural performance criteria prior to the next scheduled inspection. The accident-induced leakage is only attributed to postulated plug leakage through out-of-service tubes. There is reasonable assurance the accident-induced leakage performance criteria will not be exceeded prior to the next inspection of the Unit 2 SGs.

3.6 Summary of All OA Conclusions

The OA provide reasonable assurance, as required by NRC regulations that Unit 2 will operate safely at 70% power for 150 cumulative days. The OAs (See Table 3-1) summarized in Sections 3.1 and 3.2 conclude the SIPC and AILPC are satisfied. The alternative OA methodologies summarized in Sections 3.3 and 3.4 also confirm the SG tube integrity will be maintained during the inspection interval.

Table 3-1: OA Approach and Results Comparison

OA Description	OA for Degradation Mechanisms Other Than TTW	TTW OA With No Effective AVB Supports	"Traditional" Probabilistic OA Prepared for TTW	Deterministic TTW OA
Reference Appendix	A	B	C	D
Degradation Mechanisms Addressed	All but TTW	TTW	TTW	TTW & AVB Wear
Type	Probabilistic	Deterministic	Probabilistic	Deterministic
Thermal Power Assumption	100%	70%	70%	70%
Resulting Inspection Interval	18 months	18 months	16 months	18 months

As identified in Table 3-1 above, the OAs result in an acceptable inspection interval of at least 16 months at 70% power. These OAs determined that at 70% power, the T/H conditions that cause FEI will be eliminated from the SONGS Unit 2 SGs. As discussed in Section 3.2, an additional probabilistic evaluation, assuming a calculated SR of 0.75, was performed to demonstrate margin. The approach assumes instability initiates at a calculated SR of 0.75 (rather than a SR of 1.0). Using this approach, the SIPC acceptance standard is satisfied for approximately 8 months at 70% power.

Accordingly, the 150 cumulative day inspection interval being implemented by SCE demonstrates substantial conservative margin using any of the OA methodologies.

4.0 REFERENCES

1. Confirmatory Action Letter 4-12-001 – "San Onofre Nuclear Generating Station, Units 2 and 3, Comments to Address Steam Generator Tube Degradation," March 27, 2012
2. SONGS Steam Generator Program, SO23-SG-1
3. SONGS Technical Specifications Sections 5.5.2.11, "Steam Generator (SG) Program," Amendment 204
4. SONGS Technical Specifications Section 3.4.12, "RCS Operational Leakage," Amendment 204
5. NEI 97-06, "SG Program Guidelines," Rev. 3, January 2011
6. AREVA NP Document 51-9176667-001, "SONGS 2C17 & 3C17 Steam Generator Degradation Assessment."
7. SCE Drawing SO23-617-1-D116 Rev. 2, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Tube Bundle 1/3" (MHI Drawing L5-04FU051 Rev. 1)
8. SCE Drawing SO23-617-1-D507 Rev. 5, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Anti-Vibration Bar Assembly 1/9" (MHI Drawing L5-04FU111 Rev. 2)
9. SCE Drawing SO23-617-1-D542 Rev. 9, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Anti-Vibration Bar Assembly 7/9" (MHI Drawing L5-04FU117 Rev. 9)
10. SCE Drawing SO23-617-1-D296 Rev. 3, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Tube Support Plate Assembly 3/3" (MHI Drawing L5-04FU108 Rev. 3)
11. SCE Drawing SO23-617-1-D117 Rev. 2, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Tube Bundle 2/3" (MHI Drawing L5-04FU052 Rev. 1)
12. SCE Drawing SO23-617-1-D118 Rev. 4, "San Onofre Nuclear Generating Station Unit 2 & 3 Replacement Steam Generators – Design Drawing – Tube Bundle 3/3" (MHI Drawing L5-04FU053 Rev. 3)
13. AREVA NP Document 51-9182368-003, "SONGS 2C17 Steam Generator Condition Monitoring Report"
14. EPRI Report 1019038, "Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines: Revision 3", November 2009.