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April 4, 2013

E. Roy Hawkens, Chair
Dr. Anthony J. Baratta
Dr. Gary S. Arnold
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Docket: Southern California Edison Company, San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361-CAL & 50-362-CAL

Re: Seventh Notification of Responses to RAIs

Dear Licensing Board Members:

The Nuclear Regulatory Commission (NRC) staff issued Requests for Additional Information (RAIs) to Southern California Edison Company (SCE) on December 26, 2012 (RAIs 1-32), March 18, 2013 (RAIs 33-67), and March 15, 2013 (RAIs 68-72) regarding SCE's October 3, 2012 response to the March 27, 2012 Confirmatory Action Letter for San Onofre Nuclear Generating Station Units 2 and 3.

The purpose of this letter is to provide notification to the Licensing Board of additional SCE responses to these RAIs. The responses are identified on the enclosure list. SCE submitted proprietary and non-proprietary versions of some of these RAI responses. Only the non-proprietary versions are enclosed. Please let us know if you would like us to send the Licensing Board copies of the proprietary versions pursuant to the Protective Order.

Respectfully submitted,

<u>*Signed (electronically) by Stephen J. Burdick*</u>
Stephen J. Burdick

Counsel for Southern California Edison Company

Enclosures

1. SCE Revised Response to RAI 11 (Apr. 2, 2013) (non-proprietary version)
2. SCE Revised Response to RAI 13 (Apr. 2, 2013) (non-proprietary version)
3. SCE Revised Response to RAI 18 (Mar. 22, 2013)
4. SCE Response to RAI 50 (Mar. 22, 2013) (non-proprietary version)
5. SCE Response to RAI 62 (Mar. 29, 2013) (non-proprietary version)
6. SCE Response to RAIs 68, 69, 70 and Revised Response to RAI 2 (Apr. 1, 2013)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

	)	
In the Matter of	)	
	)	Docket Nos. 50-361-CAL & 50-362-CAL
SOUTHERN CALIFORNIA EDISON COMPANY	)	
	)	
(San Onofre Nuclear Generating Station,	)	April 4, 2013
Units 2 and 3)	)	
	)	

CERTIFICATE OF SERVICE

I hereby certify that, on this date, a copy of the "Seventh Notification of Responses to RAIs" was filed through the E-Filing system.

Signed (electronically) by Stephen J. Burdick
Stephen J. Burdick
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Counsel for Southern California Edison Company

BOARD NOTIFICATION ENCLOSURE 1

SOUTHERN CALIFORNIA
EDISON®
An *EDISON INTERNATIONAL*® Company

Proprietary Information
Withhold from Public Disclosure

Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

April 2, 2013

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAI 11), Revision 1
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. 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

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

3. Letter from Mr. James R. Hall (USNRC) to Mr. Peter T. Dietrich (SCE), dated December 26, 2012, Request for Additional Information Regarding Response to Confirmatory Action Letter, San Onofre Nuclear Generating Station, Unit 2

4. Letter from Mr. Richard J. St. Onge (SCE) to NRC Document Control Desk, dated January 21, 2013, Response to Request for Additional Information (RAI 11) Regarding Confirmatory Action Letter Response, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By letter dated December 26, 2012 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. SCE provided the response to RAI 11 in a

P.O. Box 128
San Clemente, CA 92672

letter dated January 21, 2013 (Reference 4). The response to RAI 11 was revised to address questions raised by the NRC during the public meeting on February 27, 2013. Enclosure 2 of this letter provides Revision 1 to the RAI 11 response.

Enclosure 2 of this submittal contains proprietary information. SCE requests that this proprietary enclosure be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosure 1 provides a notarized affidavit from Westinghouse, which sets forth the basis on which the information in Enclosure 2 may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Enclosure 3 provides the non-proprietary version of Enclosure 2.

Enclosure 4 provides a list of commitments identified in this submittal. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Deborah Lindbeck for

Enclosures:

1. Notarized Affidavit
2. Response to RAI 11, Revision 1 (Proprietary)
3. Response to RAI 11, Revision 1 (Non-Proprietary)
4. List of Commitments

cc: A. T. Howell III, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

Notarized Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: NF-SCE-13-10

CAW-13-3680

April 1, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Proprietary Content for, "Follow-on Response to NRC Confirmatory Action Letter RAI #11 for SONGS Unit 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-13-3680 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Southern California Edison.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-13-3680, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

Thomas Rodack, Director
Licensing and Engineering Programs

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared Thomas Rodack, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Thomas Rodack

Thomas Rodack, Director
Licensing and Engineering Programs

Sworn to and subscribed before me
this 1st of April 2013

Anne M. Stegman
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Director, Licensing and Engineering Programs, in Nuclear Fuel, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is contained in, "Follow-on Response to NRC Confirmatory Action Letter RAI #11 for SONGS Unit 2" (Proprietary), dated April 1, 2013, being transmitted by Southern California Edison letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by SCE to the NRC is that associated with a response to NRC RAI #11with respect to fuel-clad modeling and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Support for SONGS Unit 2 enabling SCE to responds to NRC RAIs.

Further this information has substantial commercial value as follows:

(a) Westinghouse can sell support and defense of analyses involving Westinghouse fuel-clad modeling to other licensees, as necessary.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

ENCLOSURE 3

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to RAI 11, Revision 1

(NON-PROPRIETARY)

RAI 11

Please submit an operational impact assessment for operation at 70% power. The assessment should focus on the cycle safety analysis and establish whether operation at 70% power is within the scope of SCE's safety analysis methodology, and that analyses and evaluations have been performed to conclude operation at 70% power for an extended period of time is safe. The evaluation should also demonstrate that the existing Technical Specifications, including limiting conditions for operation and surveillance requirements, are applicable for extended operation at 70% power.

RESPONSE – Revision 1

Note: This response includes information requested in RAI 14 associated with the operational impact assessment for operation at 70% power. RAI 14 states: "Provide a summary disposition of the U2C17 calculations relative to the planned reduction in power operation."

SCE has evaluated the extended reduced power operation for its impacts on the Unit 2 Cycle 17 reload core design and safety analysis. The power levels evaluated range from 50% to 100% rated thermal power, which bounds the planned operation at the 70% power level. The assessments were performed in accordance with NRC approved SONGS reload methodology and topical reports referenced in the UFSAR and Technical Specification (TS) 5.7.1.5, and the SONGS Core Reload Analyses and Activities Checklist procedure.

The impacts of extended reduced power operation on Unit 2 Cycle 17 core design and reload analyses, including UFSAR Chapter 15 safety analyses are summarized in Table 1, the impact assessment table. The impact assessment table is organized consistent with the SONGS Core Reload Analyses and Activities Checklist procedure. For each analysis, the Reload Checklist item number is listed in the second column from the left; when applicable, the second column also lists the UFSAR Chapter 15 safety analysis section number. The determination of impact for each analysis is summarized in the right column of the table.

Tables 1 and 2 were revised to provide additional details and clarification to address issues raised during the February 27, 2013 public meeting. Revisions are annotated in the tables by change bars.

Safety Analysis Methodology

The NRC approved safety analysis methods, as described in TS 5.7.1.5, are used to establish the core operating limits specified in the Core Operating Limits Report (COLR) which encompass from Mode 6 up to Mode 1 operation at the rated thermal power. Therefore, operating at the 70% power level is within the scope of SCE safety analysis methodology. No change to the safety analysis methodology is required for extended reduced power operation.

Safety Analysis

The reload and safety analyses determined to be impacted by extended reduced power operation were re-analyzed. The conclusions of the reload analyses, including safety analyses, for extended reduced power operation are as follows: (1) All safety analyses results meet the established acceptance criteria, and (2) The radiological dose consequences for all safety analyses remain bounded by the dose consequences reported in the UFSAR.

Technical Specifications

The existing TS, including limiting conditions for operation (LCO) and surveillance requirements, are applicable for extended operation at 70% power. The impact assessment for TS surveillance requirements is described in the following section.

Impact Assessment for Technical Specification Surveillance Requirements

The TS surveillance requirements were evaluated for the impacts of reduced power operation. The evaluation concluded all TS surveillance requirements under the reactor core design and monitoring program that would have been performed at approximately 82% power or at full power will be performed with the plant operating at approximately 70% power. The evaluation is summarized in Table 2.

Two surveillance procedures related to monitoring Reactor Coolant System (RCS) flow ~~were revised to (1) reduce the minimum power required to perform the surveillances from 85% to 68% power, and to (2) account for the slightly increased RCS flow uncertainty at reduced power operation~~that had been required to be performed only above 85% power were revised to require performance of the surveillances at 70% power. No other surveillances were identified to be impacted by plant operation at 70% power.

Conclusions

Extended reduced power operation at 70% power has been evaluated and determined to be acceptable with respect to Unit 2 Cycle 17 reload core design and safety analysis. Reload analyses needed to support reactor startup and operation at 70% power have been completed. All TS LCO and surveillance requirements under the reactor core design and monitoring program normally performed at or above 70% power will be performed with the plant operating at approximately 70% power. The above evaluations demonstrate that the existing TSs, including limiting conditions for operation and surveillance requirements, are applicable for extended operation at 70% power.

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
1	0.1	Reload Ground Rules (RGR) Review	No change to analysis is required. No change to Rated Thermal Power (RTP). RGR still addresses 0% to 100% RTP operation. RGR addresses the full range of power independent and power dependent operating parameters, including those applicable at reduced power. The RGR Analysis Value defines the maximum or minimum value which must be bounded in the safety analysis. The number is not necessarily equivalent to the value used in an analysis (or Technical Specification) but will be conservative with respect to that value. The RGR Analysis Value includes applicable uncertainties and margins for which the safety analyses must be bounding.
2	1.1.3	Design Models and Depletions	Re-analysis was performed to determine impact, and all results were acceptable. Calculation revised to document depletion at 50% power from Beginning of Cycle (BOC) to End of Cycle (EOC) and comparison to 100% power. The results are expected, since radial power distributions are primarily a function of fuel burnup distribution, burnable poison loading, and control rod configuration. Depletion at reduced power level and corresponding reduced moderator temperature has a negligible impact on core average radial power distributions. Since the radial power distributions changed negligibly at 50% power compared to 100% power, an extended operation at any power level between these two points would also yield insignificant changes to these parameters. The lead fuel assembly (LFA) integrated radial power peaking factors remain below 95% of the core maximum integrated radial power peaking factor at all times in life. The maximum pin burnup remains below the peak pin burnup limit (60,000 MWD/T). ~~As the radial power distributions and distortion factors have been determined to be valid, no downstream analyses are impacted.~~ Impact of extended reduced power operation on generic axial shapes and scram curves is addressed in Item 10 (1-D HERMITE model.)

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
3	1.1.4	Design Parameters and F_R Versus Power	No change to analysis is required. Radial power distributions and generic axial shapes remain applicable. Individual Control Element Assembly (CEA) worth, CEA bank worth, scram worth, peaking factors, distortion factors that are strongly dependent on the radial power distribution remain applicable. Extended reduced power operation results in less Pu-239 inventory. As such, generic bounding parameters (i.e., Fuel Temperature Coefficient (FTC), Moderator Temperature Coefficient (MTC), kinetics parameters) remain applicable. Critical Boron Concentrations (CBC) at Beginning of Cycle (BOC) are not affected. CBC at End of Cycle (EOC) is similar. Therefore, bounding boron concentration requirements and Inverse Boron Worths (IBW) are not impacted. Representative design parameter and Fr values for Reload Analysis Report (RAR) are not impacted.
4	1.1.5	Physics Input to LOCA, TORC, and FATES Analysis (including Pin Census)	No change to analysis is required for the physics inputs to LOCA analysis and TORC code analysis. BOC, limiting boron concentration, reactivity are not affected. Radial power distribution and peaking data remain applicable. Generic LOCA and TORC input parameters remain applicable. Re-analysis was performed for the physics input to Fuel Performance Analysis (FATES) code analysis. Radial fall-off curves, Fr, and fast flux data were regenerated for reduced power operation. Generic axial shapes remain applicable. The impact of the revised input on FATES is addressed in Item 19.
5	1.1.6	Physics Input to Fuel Mechanical Design	Re-analysis was performed to determine impact, and all results were acceptable. Calculation revised to provide power history data for AREVA Lead Fuel Assembly (LFA) mechanical design analysis. Also updated maximum core residence time for Westinghouse analysis. Other generic parameters for Westinghouse mechanical design analysis remain applicable due to similar radial power distribution.
6	1.1.7	Physics Input to ASGT	No change to analysis is required. Physics Input to Asymmetric Steam Generator Transient (ASGT) is performed at EOC with most negative Technical Specification MTC. Calculations performed at multiple power levels (90%, 70%, 50%, and 20%). Due to similar power distributions, results remain applicable.
7	1.1.8	Physics Input to Post-Trip Steam Line Break Analysis	No change to analysis is required. Analysis performed at EOC. Radial power distributions (at the same power level and burnup) are essentially identical. The MTC is tuned to the most negative Tech Spec value (-3.7E-4 $\Delta k/k/^{o}F$). Cooling down adds reactivity. More reactivity is added cooling from 100% power (higher T-fuel and T-mod) than reduced power to lower temperatures (e.g., $545^{o}F$, $300^{o}F$, $200^{o}F$, $68^{o}F$)

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
8	1.1.9	Physics Input to CEA Ejection Analysis	No change to analysis is required. Physics data in this analysis were generated at multiple power levels and the reduced power operating range is covered. Since the reduced power operation results in power distributions essentially identical to those from 100% power operation, the data generated from the original analysis are applicable to reduced power operation.
9	1.1.10	Physics Input to CEA Withdrawal	No change to analysis is required. Calculations performed at multiple power levels. Radial power distributions (at the same power level and burnup) are essential identical. CEA worth remains applicable since it is strongly dependent on power distribution. Limiting axial power shapes from axial shape index (ASI) search remain applicable.
10	1.1.11	1-D HERMITE Model	Re-analysis was performed to determine impact, and all results were acceptable. Analysis is revised to establish applicability of the generic axial shapes used in the design analyses and applicability of the SCRAM curves used in the design analyses. Analysis also shows that depletion at reduced power leads to essentially the same limiting shapes from ASI search as those selected for the analyses of the design depletions.
11	1.1.12	Physics Input to Steam Line Break Return-to-Power ~~for Cycle N-1 Configuration~~	No change to analysis is required. This EOC event begins at 0% power. Radial power distributions (at the same power level and burnup) are essentially identical.
12	1.1.13	F_R Versus Temperature for Cooldown Events	No change to analysis is required. Bounding distortion factors were determined based on multiple CEA configurations, temperature ranges at BOC and EOC. Radial power distributions (at the same power level and burnup) are essentially identical.
13	1.1.14	Boron Requirement for SITs and BAMU Tanks	No change to analysis is required. The case run for this calculation is performed at hot zero power (HZP). The Xenon starting condition is Hot Full Power (HFP) which is conservative.
14	1.1.15	LOCA and Non-LOCA Source Term	No change to analysis is required. This analysis tests the Cycle 17 conditions of interest against the parameters required for applicability of the LOCA and Alternative Source Term (AST) source terms. The power level is used as a maximum not to be exceeded. Running Cycle 17 at reduced power results in less “short half-life” nuclides. Increase in “long half-life” nuclides due to extended calendar time is bounded by the lower production from extended reduced power.

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
15	1.1.16	Tritium Production	No change to analysis is required. Reduced power results in a decrease in tritium production. The analysis at 100% power is conservative.
16	1.1.17	STAR Physics Verification	No change to analysis is required. This analysis uses BOC (HZP) conditions (Mode 3) for an assessment for S2C17 inclusion in the Startup Test Activity Reduction (STAR) program.
17	1.1.18	Digital Setpoints Physics Data	No change to analysis is required. The case sets encompass LCO and Limiting Safety System Settings (LSSS) ASI ranges. Power level does not impact axial shapes significantly, so reduced powers are covered by the case set.
18	1.1.19	Physics RAR Inputs	Re-analysis was performed to determine impact, and all results were acceptable. RAR has been updated to reflect actual Cycle 16 EOC burnup and Cycle 17 reduced power operation.

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
19	1.2.1	Fuel Performance Analysis (FATES)	Re-analysis was performed to determine impact, and all results were acceptable. Reduced power results in fuel performance data that is not bounded when compared to the Generic Fuel Performance data generated for ZIRLO™ in Cycle 14 (data used in LOCA Analysis). A revision to the Fuel Performance and Setpoints Analyses was performed to determine the appropriate penalty factors such that the Generic Fuel Performance data remained bounding. Operation at reduced power impacts several of the fuel modeling parameters and mechanisms within the FATES fuel performance code. **[** **]**. Fuel Performance results are utilized in the ECCS LOCA Analysis (item #71). **[** **]**. The Fuel Performance results are also used in the CEA Ejection Analysis (item #44). New data were transmitted and addressed in that analysis.
20	1.2.2	T-H Input Summary	No change to analysis is required. Calculation is a collection of input data that are not impacted by reduced power.

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
21	1.2.4	T-H Limiting Assembly and CETOP Benchmarking Analysis	No change to analysis is required. Power is not an input. Calculation is a benchmark of CETOP to TORC computer codes at reference departure from nucleate boiling (DNBR) points rather than a benchmark at a given power. This benchmark is mainly driven by power distributions from physics. Physics Models & Depletions has validated the power distributions used in the original calculation.
22	1.2.5	Mechanical Design Analysis (Fuel Vendor)	Re-analysis was performed to determine impact, and all results were acceptable. Westinghouse performed calculations to determine the impact of reduced power on the fuel mechanical design. AREVA performed calculations to determine the impact of reduced power on the Lead Fuel Assembly fuel mechanical design.
23	1.2.6	Power Operating Limit Partial Derivative Verification	No change to analysis is required. The calculation is driven by a large family of axial shapes, which are not impacted by the power reduction.
24	1.2.7	Setpoints Input Summary	Re-analysis was performed to determine impact, and all results were acceptable. Calculation has been revised to address the increased reactor coolant system (RCS) flow uncertainty at reduced power.
25	1.2.8	RCS Flow Uncertainties	Re-analysis was performed to determine impact, and all results were acceptable. Has been reanalyzed. RCS flow uncertainty increases due to reduced delta-temperature and increased secondary calorimetric power uncertainty. More details of this analysis are provided in the RAI 12 Response.
26	1.2.9	Fuel Mechanical Design Verification	No change to analysis is required. The objective of the fuel mechanical design verification calculation is to document the design of the fuel based on the fuel vendor Bill of Materials, Design Drawings and the design and material specifications transmitted from the fuel vendor. Reduced power operation has no impact on this analysis.
27	1.2.11	Secondary Calorimetric Power Uncertainty	No change to analysis is required. Intermediate powers were explicitly analyzed in the original calculation.

Table 1 – Revision 1
SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
28	1.2.12	Delta-T/Turbine Power Uncertainties	No change to analysis is required. The analysis uses a reference power error of 1.3% at full power. The increase in reference power (i.e., secondary calorimetric power) associated with performing delta-t/turbine power calibrations at reduced power would increase the uncertainties. The bounding results include ~0.50% of conservatism; therefore, the analysis of record (AOR) remains bounding. Intermediate powers were explicitly analyzed in the original calculation.
29	1.2.13	Cycle Independent Data and Setpoints Assumptions List (CIDSAL)	No change to analysis is required. CIDSAL provides cycle independent values to use or to be verified in downstream analyses. Reduced power operation does not impact the requirements for downstream analysis verification. None of the calculations explicitly performed in the analysis section are dependent upon nominal plant operating conditions or the power shapes/distributions at reduced power operation.
30	1.2.16	Core Protection Calculator (CPC) Calibration Allowances	No change to analysis is required. Intermediate powers were explicitly analyzed in the original calculation. Due to less decalibration, full power bounds lower power levels.
31	1.2.17	Fuel Duty Index	No change to analysis is required. Full power bounds lower power levels.
32	1.2.18	T-H MSCU Verification	No change to analysis is required. Power is not an input. Calculation is a verification of response surface at reference DNBR points rather than a benchmark at a given power.
33	1.2.19	CEA STAR Verification	No change to analysis is required. Radial power distributions (at the same power level and burnup) are essentially identical. At reduced power the plan is to continue to operate with all rods out. The duration and depth of lead bank CEA insertion beyond the typical all-rods-out position is monitored per the core follow procedure with notification/action to review the conservative CEA life analysis when insertion exceeds an insertion assumption within the analysis.
34	1.3.1	Summary of Transients	Re-analysis was performed to determine impact, and all results were acceptable. Calculation was revised to perform an evaluation of all Updated Final Safety Analysis Report (UFSAR) Chapter 15 events for extended reduced power operation. Individual events are addressed in subsequent entries to this table.
35	1.3.2	CENTS Cycle Update and Action Modules	No change to analysis is required. Calculation and associated computer files already accommodate power levels from 0 to 100 percent.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
36	1.3.3 (15.10.1.3.1.1)	Main Steam Line Break (MSLB) Pre-Trip	No change to analysis is required. Pre-trip SLB is analyzed @100% power (with uncertainty). The generic physics inputs remain unchanged. Since the VOPT is generated on the rate of change in power setpoint (DELSPV), the actual trip occurs at the same power rise, independent of the starting power level. As this is a Required Over Power Margin (ROPM) event, the actual initial power level chosen is not significant to the event.
37	1.3.4 (15.10.1.3.1.2)	Main Steam Line Break (MSLB) Post-Trip	No change to analysis is required. This event is limiting at hot zero power (HZP). HZP cases show greatest return to power since there is minimum initial stored energy, decay heat and scram worth at HZP conditions. There is no impact to the HZP cases since HZP physics inputs and initial conditions do not change. A reactivity balance for reduced power showed that net reactivity change remained negative. HZP cases result in greatest return to criticality since initial stored NSSS energy, decay heat, and scram worth are minimized while steam generator pressure and mass are at maximum. This minimizes RCS mass and maximizes cooldown potential. Consequently, the HZP MSLB event bounds MSLB initiated from power conditions. Operation at intermediate power levels does not alter these key parameters that make HZP limiting.
38	1.3.5 (15.10.4.1.4)	Chemical Volume Control System (CVCS) Malfunction - Boron Dilution	No change to analysis is required. This is a BOC event that is not analyzed in Mode 1. The reactivity addition due to a boron dilution event is less adverse than the CEA Withdrawal event at Power and therefore Mode 1 and the higher power portion of Mode 2 are not explicitly addressed.
39	1.3.6 (15.10.4.1.1)	CEA Bank Withdrawal from Subcritical (CEAW @ SC)	No change to analysis is required. Event is evaluated at subcritical conditions. Note that this event is being re-evaluated to address the extended shut down.
40	1.3.6 (15.10.4.1.1)	CEA Bank Withdrawal at Low Power (CEAW @ HZP)	No change to analysis is required. Event is evaluated at hot zero power conditions.
41	1.3.6 (15.10.4.1.2)	CEA Bank Withdrawal at Power (CEAW @ Power, 50% & 100%)	No change to analysis is required. CEAW at reduced power is enveloped by CEAW @ 50% Power and CEAW @ 100% Power; and the results are acceptable.
42	1.3.8 (15.10.1.1.3)	Increased Main Steam Flow (IMSF)	No change to analysis is required. The system response is the same as IMSF+SF.

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SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
43	1.3.8 (15.10.1.2.3)	IMSF with Single Failure (SF)	No change to analysis is required. IMSF+SF (fast & slow) analyzed @100% power. The generic physics inputs remain unchanged. The fast case credits the VOPT which is generated on the rate of change in power (DELSPV) setpoint, as such the actual trip occurs at the same power rise, independent of the starting power level. Since the fast case is a Required Over Power Margin (ROPM) event, the actual initial power level chosen is not significant to the event. The limiting event is the slow trip, which is initiated from a Power Operating Limit. As such, the actual initial power level chosen is not significant to the event.
44	1.3.9 (15.10.4.3.2)	CEA Ejection	Re-analysis was performed to determine impact, and all results were acceptable. The event is normally analyzed at multiple power levels. It was reanalyzed to address reduced power data from the fuel performance analysis. The CEA Ejection event is normally analyzed at multiple power levels according to the power dependent insertion limits (PDIL) using fuel performance data based on 100% power operation. As discussed in Item 19, [ ] used in the CEA Ejection analysis was determined with a conservative approach, the event was re-analyzed to address the impact on fuel performance from reduced power operation. The fuel performance analysis (Item 19) generated additional fuel performance data based on reduced power operation at 70% and at 50%. The CEA Ejection cases were re-analyzed using the fuel performance data based on 70% power operation, as well as those based on 50% power operation. The re-analysis showed [ ]. The re-analysis bounds the planned reduced power operation for Unit 2 and all results meet the acceptance criteria.
45	1.3.10 (15.10.3.3.1)	Reactor Coolant Pump Shaft Seizure	No change to analysis is required. Bounded by Reactor Coolant Pump Sheared Shaft (RCPSS).

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SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
46	1.3.10 (15.10.3.3.2)	Reactor Coolant Pump Sheared Shaft (RCPSS)	No change to analysis is required. This is a margin/ fuel failure calculation event. The thermal margin loss for this event is initiated by the loss of flow from one pump (either seized rotor or sheared shaft). The reduction of thermal margin due to the loss of flow from one pump is not a function of the initial power (i.e., is constant at any power level). In addition, at reduced power, the initial thermal margin is larger than at the 100% power condition. Therefore, the analysis at full power is bounding.
47	1.3.11 (15.10.2.1.3)	Loss of Condenser Vacuum (LOCV)	No change to analysis is required. Bounded by LOCV+SF
48	1.3.11 (15.10.2.2.3)	LOCV with Single Failure	No change to analysis is required. This event is driven by plant response and not by detailed core physics. There are two criteria (peak RCS pressure and peak secondary pressure). At lower powers, there is less internal energy in the reactor core, which translates into a slower RCS pressure transient that is more rapidly mitigated by main steam safety valves (MSSVs). The peak secondary pressure event is evaluated at multiple power levels to establish the allowed power level as a function of the number of gagged MSSVs (Tech Spec 3.7.1).
49	1.3.12 (15.10.6.3.2)	Steam Generator Tube Rupture (SGTR)	No change to analysis is required. The SGTR is a slow event and not sensitive to initial power. Furthermore, at lower powers there is a higher secondary pressure that translates to lower primary-to-secondary rupture flow (i.e., lower activity release).
50	1.3.13 (15.10.1.1.4)	Inadvertent Opening of a Steam Generator Safety or an Atmospheric Dump Valve (IOSGADV)	No change to analysis is required. See IOSGADV+SF
51	1.3.14 (15.10.1.2.4)	IOSGADV with Single Failure	No change to analysis is required. The IOSGADV+SF is analyzed at a power level of 1 MWt.
52	1.3.15 (15.10.9.1.1)	Asymmetric Steam Generator Transient (ASGT)	No change to analysis is required. The ASGT event was analyzed in the AOR at multiple power levels (90%, 70%, 50%, and 20%).
53	1.3.16 (15.10.1.1.1)	Decrease in Feedwater Temp (DFWT)	No change to analysis is required. Since feedwater heating is reduced at reduced power, the potential loss in feedwater heating is also reduced. Impact at reduced power is also mitigated by increased mass in RCS and Steam Generators (SGs) and increased recirculation in SGs at lower power.

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SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
54	1.3.17 (15.10.1.2.1)	DFWT with Single Failure	No change to analysis is required. Since feedwater heating is reduced at reduced power, the potential loss in feedwater heating is also reduced. Impact at reduced power is also mitigated by increased mass in RCS and Steam Generators and increased recirculation in SGs at lower power.
55	1.3.18 (15.10.1.1.2)	Increase in Feedwater Flow (IFF)	No change to analysis is required. Primary to secondary heat transfer is dominated by heat of vaporization (Hfg) which is considerably greater than steam generator enthalpy rise resulting from sensible heat. Consequently, cool downs resulting from Increases in Feedwater Flow events are limited by Increases in Main Steam Flow events. Further, Increases in Steam Flow events occur more rapidly as changes in Feed Water are mitigated by the liquid mass and recirculation flow in the steam generators. Further factors that mitigate Increasing Feedwater Flow events at reduced power include greater RCS / SG mass, increased recirculation flow in the steam generators, greater steam generator pressure and earlier reactor trip from increased feedwater flow - steam flow mismatch.
56	1.3.18 (15.10.1.2.2)	IFF with Single Failure	No change to analysis is required. The most adverse single failure postulated for IFF is the opening of all Steam Bypass Control System (SBCS) valves. Because the Increase in Main Steam Flow (IMSF) event postulates the opening of all SBCS valves and assumes that Main Feedwater flow increases to match steam flow, the IFF with Single Failure is the essentially the same event as the IMSF event. Therefore, conclusions regarding IMSF are applicable to IFF with Single Failure.
57	1.3.19 (15.10.2.1.1)	Loss of External Load (LOL)	No change to analysis is required. The system response to the Loss of External Load, Turbine Trip, and the Loss of Condenser Vacuum are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such these events remain bounded by LOCV.
58	1.3.19 (15.10.2.2.1)	LOL with Single Failure	No change to analysis is required. The system response to the Loss of External Load with single failure, Turbine Trip with single failure, and the Loss of Condenser Vacuum with single failure are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such these events remain bounded by LOCV+SF.
59	1.3.19 (15.10.2.1.2)	Turbine Trip (TT)	No change to analysis is required. The system response to the Loss of External Load, Turbine Trip, and the Loss of Condenser Vacuum are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such these events remain bounded by LOCV.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
60	1.3.19 (15.10.2.2.2)	TT with Single Failure	No change to analysis is required. The system response to the Loss of External Load with single failure, Turbine Trip with single failure, and the Loss of Condenser Vacuum with single failure are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such these events remain bounded by LOCV+SF.
61	1.3.20 (15.10.2.1.4)	Loss of Normal AC Power (LONAC)	No change to analysis is required. See LONAC+SF
62	1.3.20 (15.10.2.2.4)	LONAC with Single Failure	No change to analysis is required. Operation at lower power level is less challenging with respect to maintaining an adequate heat sink.
63	1.3.21 (15.10.2.2.5)	Loss of Normal Feedwater (LONF or LOFW)	No change to analysis is required. See LOFW+SF
64	1.3.21 (15.10.2.3.2)	LOFW with Single Failure	No change to analysis is required. Operation at lower power level is less challenging with respect to maintaining an adequate heat sink.
65	1.3.22 (15.10.2.3.1)	Feedwater System Pipe Breaks (FSPB or FWLB)	No change to analysis is required. Peak primary and secondary pressure events were analyzed at the least negative MTC value and main feedwater enthalpy corresponding to full power. The slightly higher MTC corresponding to reduced power is offset by the lower main feedwater enthalpy at reduced power. Operation at lower power level is less challenging with respect to maintaining an adequate heat sink. The energy in the plant is less at reduced power relative to full power, and therefore pressurizer overfill is bounded by the full power response.
66	1.3.23 (15.10.5.1.1)	CVCS Malfunction	No change to analysis is required. See CVCS Malfunction+SF.
67	1.3.23 (15.10.5.2.1)	CVCS Malfunction with Single Failure	No change to analysis is required. The energy in the plant is less at reduced power relative to full power, and therefore pressurizer overfill is bounded by the full power response. Operation at lower power level is less challenging with respect to maintaining an adequate heat sink.
68	1.3.24	Pressurizer Spray Malfunction	No change to analysis is required. See Core Protection Calculator (CPC) Dynamic Filter Analysis.
69	1.3.25 (15.10.4.1.5)	Reactor Coolant Pump (RCP) - Start Up of an Inactive Loop	No change to analysis is required. Modes 1 and 2 were not analyzed because operation in these Modes is only allowed with all 4 RCPs running.

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SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
70	1.3.27 (15.10.4.3.2)	CEA Ejection (peak pressure analysis)	No change to analysis is required. The event is limiting at hot zero power (HZP).
71	1.4 (15.10.6.3.3)	Emergency Core Cooling System (ECCS) Analyses including LBLOCA, SBLOCA and LTC	Re-analysis was performed to determine impact, and all results were acceptable. Impact assessment addressed in analyses performed by Fuel Vendors. Details are provided in the response to RAI 13.
72	(15.10.5.1.2)	Inadvertent Operation of ECCS at Power (IOECCS)	No change to analysis is required. The system response to the IOECCS and CVCS malfunction events are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such this event continues to be bounded by CVCS malfunction event.
73	(15.10.5.2.2)	IOECCS with Single Failure	No change to analysis is required. The system response to the IOECCS with single failure and CVCS malfunction with single failure events are essentially the same. Therefore, the relationship between the events will remain the same at reduced power. As such this event continues to be bounded by CVCS malfunction event with single failure.
74	(15.10.6.3.1)	Primary Sample or Instrument Line Break (PSILB)	No change to analysis is required. Mass releases are driven by energy in the primary system which is highest following operation at HFP. The event does not fail fuel, and there is no ROPM requirement.
75	(15.10.6.3.4)	Inadvertent Opening of a PSV (IOPSV)	No change to analysis is required. The IOPSV event is bounded by small break LOCA.
76	1.5.1	Applicability Evaluation of Source Terms in Dose Analyses	No change to analysis is required. There is no change to core activity inventory source term.
77	1.5.2	Cycle Specific Dose Analysis	No change to analysis is required. No Cycle 17 event-specific dose analysis was performed, therefore no impact for reduced power.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
78	1.5.4	Applicability Evaluation of Dose Analyses	Re-analysis was performed to determine impact, and all results were acceptable. Revised to document that the currently modeled radial peaking factors are conservatively greater than the increased radial peaking factors at reduced power. The transient analyses and mass release analyses are evaluated at the current 8% steam generator (SG) tube plugging limit. The dose calculation uses mass release data per the transient analyses and their assumed 8% SG tube plugging models. The calculation is revised with discretionary conservatism to model 20% SG tube plugging in the calculation of the RCS dilution volume and mass considered for non-LOCA events which have clad damage. Evaluated RCS dilution mass at RCS temperatures for both 50% and 100% power, which envelopes powers between 50% and 100%. The mass release calculations are evaluated for a core inlet temperature (Tcold) of 560F, which maximizes core average temperature (Tave). Currently modeled mass release values in the Summary of Transients (SOT) correspond to full power operation. The SOT did not identify an increase in the amount of steam released from the secondary side because it remains more limiting compared to operation at lower power level due to lower sensible heat in the RCS and lower post trip decay heat. Technical Specification Action Statement Figure 3.4.16-1 allows for larger short term elevated primary coolant dose equivalent iodine-131 (I-131) activity for plant operation at 70% Rated Thermal Power (RTP). For UFSAR chapter 15 radiological safety analyses that model a pre-existing iodine spike, the SONGS licensing basis is an initial primary coolant concentration of 60 μCi/gm dose equivalent I-131 at 100% RTP. SCE commits to administratively control the RCS dose equivalent I-131 specific activity described in TS LCO 3.4.16 Action A1 to no more than 60 μCi/gm (see Enclosure 4). No changes to these analyses are required for reduced power operation.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
79	n/a	Fuel Corrosion and Oxide Thickness (BOA Code) analysis	No change to analysis is required. The Westinghouse BOA code analysis for cycles 15, 16 and 17 was performed as part of the Zinc Injection project. This calculation compared predicted values for corrosion and oxide thickness, Fuel Duty Index and crud dryout to the Westinghouse Chemistry Guideline limits. Maximum values of Fuel Duty Index and Crud Dryout are driven by fresh fuel operating at high power. Operation at reduced power would be bounded by the 100% power cases run in the analysis of record (AOR). Maximum values of corrosion and oxide thickness are driven by both power level and effective full power days (EFPD). The AOR assumed a core operating strategy which would maximize corrosion and oxide; running fuel for three full cycles, a total of 1830 EFPD. Table 2-1 of the AOR showed that the maximum predicted oxide thickness for U2C17 is 28.4 microns, well below the 100 micron limit. Operation at reduced power for longer time would not significantly change the fuel rod corrosion rate, and there is substantial margin to the 100 micron limit.
80	n/a	AREVA Lead Fuel Assembly (LFA) compatibility	Re-evaluation was performed to determine impact, and all results were acceptable. Compatibility was verified by AREVA as documented in revised U2C17 Reload Analysis Report (RAR).
81	n/a	WEC Lead Fuel Assembly (LFA) compatibility	Re-evaluation was performed to determine impact, and all results were acceptable. Compatibility was verified by Westinghouse as documented in revised U2C17 RAR.
82	n/a	AREVA and WEC Chemistry concurrence	Re-evaluation was performed to determine impact, and all results were acceptable. Concurrence for reduced power operation was performed by Westinghouse and AREVA as documented in revised U2C17 RAR.
83	1.6.1	Reload Analysis Report (RAR)	Re-analysis was performed to determine impact, and all results were acceptable. Revised to address extended operation at reduced power.
84	1.6.2	Engineering Change Package (ECP) and 10CFR50.59 Review	Re-evaluation was performed to determine impact, and all results were acceptable. 10CFR50.59: New 10CFR50.59 review issued to address the extended operation at reduced power. ECP: Affected Section Change (ASC) issued to address the extended operation at reduced power.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
85	2.1.2	Physics Input to FLCEA Drop Analysis and PFDTME Verification	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analysis performed at multiple power levels.
86	2.1.3	Physics Input to PLCEA Drop Analysis	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analysis performed at multiple power levels.
87	2.1.5	Physics Input to CEA Deviation Within CPC Deadband	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analysis performed at multiple power levels.
88	2.1.9	Refueling Boron Concentration	No change to analysis is required. Analyzed at BOC, Mode 6.
89	2.1.10	CIDSAL Physics Verification	No change to analysis is required. Radial power distributions (at the same power level and burnup) are essentially identical. T-inlet program remain unchanged.
90	2.2.1 (15.10.4.1.3)	CEA Misoperation - Deviation within Dead Band (DWDB)	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analysis performed at multiple power levels.
91	2.2.2 (15.10.4.1.3)	CEA Misoperation - PLR Drop - Power ≤ 50%	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Event scenario is defined at ≤ 50% Power. Scenarios at >50% power are discussed in “CEA Misoperation - Single Part Length CEA Drop (PLR Drop) - Power > 50%.”
92	2.2.3 (15.10.4.1.3)	CEA Misoperation - Single Full Length CEA Drop (FLCEA Drop)	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analyzed at multiple power levels.
93	2.2.3 (15.10.4.1.3)	CEA Misoperation - Single Part Length CEA Drop (PLCEA Drop) - Power > 50%	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analyzed at multiple power levels.
94	2.2.3 (15.10.4.1.3)	CEA Misoperation - Sub Group CEA Drop	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analyzed at multiple power levels.
95	2.2.4	AOPM Analysis	No change to analysis is required. Power distributions at the same power level and burnup are essentially identical. Analyzed at multiple power levels.
96	2.2.5	Transient Thermal Margin Summary	No change to analysis is required. Analyzed at multiple power levels.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
97	2.2.6 (15.10.3.1.1)	Partial Loss of RCS Flow (PLOF)	No change to analysis is required. Bounded by TLOF.
98	2.2.6 (15.10.3.2.2)	PLOF with Single Failure	No change to analysis is required. Bounded by RCPSS.
99	2.2.6 (15.10.3.2.1)	Total Loss of Forced Reactor Coolant Flow (TLOF)	No change to analysis is required. The total loss of coolant flow event was analyzed for a bounding scenario at 100% power and a MTC of +0.5×10-4 $\Delta\rho$/ºF. This scenario bounds all powers from 0 to 100%.
100	2.2.6 (15.10.3.3.3)	TLOF with Single Failure	No change to analysis is required. Bounded by RCPSS.
101	2.2.7	CPC Dynamic Filter Analysis (including the Pressurizer Spray Malfunction)	No change to analysis is required. The bounding events considered include CEA Withdrawal, Excess Load events, etc. As the system response time for these events has not changed, the dynamic filter analysis remains conservative.
102	2.3.4	MSOUA Database and Files	No change to analysis is required. The impact of RCS flow uncertainty changes has been captured in MSOUA Post-Processor.
103	2.3.5	CPC Reload Data Block (RDB) Update	No change to analysis is required. Reduced power has been implemented through CPC Type 2 addressable constants, and not CPC RDB.
104	2.3.6	MSOUA Post Processor	Re-analysis was performed to determine impact, and all results were acceptable. Calculation has been revised for RCS flow uncertainty and the change in UNCERT from the FATES fuel performance analysis. The COLSS and CPC Departure from Nucleate Boiling-Ratio (DNBR) penalties increased by approximately 2% due to the RCS flow uncertainty increase (see Item 25) and another 3% increase for licensee's discretionary conservatism; bringing the total DNBR penalty increase to approximately 5%. The 70% RTP based COLSS Linear Heat Rate (LHR) penalty increased by approximately 8% due to an increase in the fuel performance uncertainty factor from the revised fuel performance analyses (see Item 19). Discretionary conservatism of 3% was added to the COLSS LHR penalty for a total COLSS LHR penalty increase of approximately 11%. The CPC Linear Power Density (LPD) penalty increased by 3% due to the addition of 3% discretionary conservatism. These COLSS and CPC overall uncertainty analysis based penalty increases are acceptable because adequate core DNBR and Local/Linear Power margins exist.

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SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Reload and UFSAR Chapter 15 Safety Analyses

ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
105	2.3.7	Core Operating Limits Supervisory System (COLSS) & CPC Operating Margin Assessment	No change to analysis is required. Calculation is a prediction of operating margin at full power. Reduced power increases operating margin.
106	2.3.8	COLSS Database	No change to analysis is required. No changes are being made to the manner in which COLSS functions or responds. Therefore the cycle independent constants do not require change. The installed Primary ΔT power Block I constants were verified to be bounding. The cycle specific constants that are impacted by reduced power operation have been addressed in the COLSS As-built Database and Test Cases calculation.
107	3.1.1	Full Core Load Map	No change to analysis is required. Fuel management not changed.
108	3.1.3	As-Built Models and Depletions	Re-analysis was performed to determine impact, and all results were acceptable. Calculation was revised to address extended reduced power operation and to verify Lead Fuel Assembly (LFA) compatibility operational requirements.
109	3.1.4	CECOR Coefficients	Impacted, and all results were acceptable. Calculation was revised to address extended reduced power operation.
110	3.1.5	As-Built Mini Depletion	Re-analysis was performed to determine impact, and all results were acceptable. Calculation revised to address extended reduced power operation and to verify LFA compatibility operational requirements.
111	3.1.6	Decay Heat	No change to analysis is required. Decay heat was evaluated at end of Cycle 16 condition. The calculation specifically addresses outage times past 99 days.
112	3.1.7	Simulator Data	Re-analysis was performed to determine impact, and all results were acceptable. Calculation revised to address extended reduced power operation.
113	3.1.8	Special Nuclear Material Database Update	No change to analysis is required. The change to Cycle 17 operating power will have no effect on prior cycle spent fuel and its characteristics.
114	3.1.9	Plant Physics Data Book	Re-analysis was performed to determine impact, and all results were acceptable. Data Book has been revised to address extended reduced power operation.

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ITEM #	CHECKLIST ITEM (UFSAR SECTION)	DESCRIPTION	SUMMARY OF IMPACT ASSESSMENT
115	3.1.10	Startup Physics Test Predictions	Re-analysis was performed to determine impact, and all results were acceptable. Calculation has been revised to address changes to startup testing power plateaus.
116	3.2.1	COLSS As-built Database and Test Cases	Re-analysis was performed to determine impact, and all results were acceptable. Calculation has been revised to address extended reduced power operation impact on the cycle specific COLSS reload constants for DNBR & Linear Heat Rate (LHR) penalties.
117	3.2.2	CEFAST Database Analysis	Re-analysis was performed to determine impact, and all results were acceptable. Calculation has been revised to address extended reduced power operation impact on the cycle specific CPC reload constants for DNBR & Local Power Density (LPD) penalties.

Table 2 – Revision 1

SONGS Unit 2 Cycle 17 Reduced Power Operation – Summary of Impact Assessment of Core Design and Monitoring Technical Specification Surveillance Requirements

Surv #	Surveillance Topic	Power Applicability and Surveillance Frequency	Summary of Impact Assessment for Performing at 68-70% Power
3.1.3.1	Reactivity Balance	Every 31 EFPD	Steady state power (not full power) is required
3.1.4.1	MTC within positive limit	Prior to Mode 1	Performed at Hot Zero Power and projected to BOC 70% conditions
3.1.4.2	MTC within negative limit	Within 14 EFPD of peak Boron @ RTP	Peak boron occurs at BOC, – performed at Hot Zero Power and projected to HFP EOC conditions
3.1.4.2	MTC within negative limit	Within ± 30 EFPD of 2/3 of expected core burnup	Steady state power (not full power) is required; projected to HFP EOC conditions
3.2.2.1	CPC & COLSS Fxy > measured Fxy (CECOR)	Between 40% - 85% (i.e., prior to exceeding 85%)	68%-70% is within the power range required for surveillance
3.2.2.1	CPC & COLSS Fxy > Measured Fxy (CECOR)	Every 31 EFPD	Steady state power (not full power) is required
3.2.3.3	CPC Azimuthal Tilt > Measured Tilt (CECOR)	Every 31 EFPD	Steady state power (not full power) is required
3.3.1.2	RCS Flow in CPCs < Measured RCS Flow	Every 12 hours (not required until 12 hours after power > 85% RTP)	Procedure changed to perform surveillance at ≥ ~~68~~70% power
3.3.1.5	RCS Flow by calorimetric	Every 31 days (not required until 12 hours after power > 85% RTP)	Procedure changed to perform surveillance with additional margin at ≥ ~~68~~70% power~~., and to require additional margin when surveillance is performed during extended operation at < 95% power~~
3.3.1.11	CPC Shape Annealing Matrix (SAM) Verification	Prior to exceeding 85%	A minimum ASI change, rather than a specific power level, is required
N/A	Startup Test Activity Reduction Program Reactivity Balance HZP - HFP	Normally performed after reaching full power	Results are already adjusted from actual test conditions to RTP conditions as a part of the test method

ENCLOSURE 4

List of Commitments

Enclosure 4
List of Commitments

This table identifies an action 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.

Commitment	One-Time Only	Sustainable	Due Date
Administratively control the RCS dose equivalent I-131 specific activity described in Technical Specification LCO 3.4.16 Action A1 to no more than 60 µCi/gm.	X		Prior to Unit 2 Cycle 17 Mode 2 operation

BOARD NOTIFICATION ENCLOSURE 2

SOUTHERN CALIFORNIA
EDISON®
An EDISON INTERNATIONAL® Company

Richard J. St. Onge
Director, Nuclear Regulatory Affairs and Emergency Planning

April 2, 2013

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAI 13), Revision 1
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. 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

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

3. Letter from Mr. James R. Hall (USNRC) to Mr. Peter T. Dietrich (SCE), dated December 26, 2012, Request for Additional Information Regarding Response to Confirmatory Action Letter, San Onofre Nuclear Generating Station, Unit 2

4. Letter from Mr. Richard J. St. Onge (SCE) to Document Control Desk (USNRC), dated January 18, 2013, Response to Request for Additional Information (RAI 13) Regarding Confirmatory Action Letter Response, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By letter dated December 26, 2012 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. SCE provided a response to RAI 13 in a letter dated January 18, 2013 (Reference 4). The response to RAI 13 was revised to address questions raised by the NRC during the public meeting on February 27, 2013. Enclosure 2 of this submittal provides Revision 1 of the RAI 13 response.

P.O. Box 128
San Clemente, CA 92672

Enclosure 2 of this submittal contains proprietary information. SCE requests that this proprietary enclosure be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosure 1 provides notarized affidavits from Westinghouse and Mitsubishi Heavy Industries (MHI), which set forth the basis on which the information in Enclosure 2 may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Proprietary information in Enclosure 2 was extracted from Westinghouse document LTR-LAM-13-23-P-Attachment, "Follow-on Response to NRC Confirmatory Action Letter RAI #13 for SONGS Unit 2" (Proprietary), which is addressed in the Westinghouse affidavit. Enclosure 3 provides the non-proprietary version of Enclosure 2.

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Deborah Lindbeck for

Enclosure:

1. Notarized Affidavits
2. Response to RAI 13, Revision 1 (Proprietary)
3. Response to RAI 13, Revision 1 (Non-Proprietary)

cc: A. T. Howell III, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

Notarized Affidavits

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter:

CAW-13-3662

March 15, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-LAM-13-23-P-Attachment, "Follow-on Response to NRC Confirmatory Action Letter RAI #13 for SONGS Unit 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-13-3662 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Southern California Edison.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-13-3662, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J.A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 15th day of March 2013

Anne M. Stegman

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-LAM-13-23-P-Attachment, "Follow-on Response to NRC Confirmatory Action Letter RAI #13 for SONGS Unit 2" (Proprietary), dated March 15, 2013 for submittal to the Commission, being transmitted by Southern California Edison letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with a response to NRC RAI #13 and may be used only for that purpose.

This document also contains appropriately marked third party proprietary information from MHI, used with permission. The scope of this affidavit does not include withholding this information and must be addressed separately by MHI.

This information is part of that which will enable Westinghouse to:

(a) Adequately support the response to the NRC RAI.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of supporting responses to NRC RAIs.

(b) Westinghouse can sell support and defense of safety analysis services.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Jinichi Miyaguchi, state as follows:

1. I am Director, Nuclear Plant Component Designing Department, of Mitsubishi Heavy Industries, Ltd. ("MHI"), and have been delegated the function of reviewing the referenced documentation to determine whether it contains MHI's information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information that is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the following documentation and have determined that it contains MHI proprietary information that should be withheld from public disclosure. Those pages containing proprietary information have been bracketed with an open and closed bracket as shown here "[ ]" / and should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

   SCE's document
   - Responses to the Request for Additional Information issued by NRC, regarding response to March 27, 2012 Confirmatory Action Letter for San Onofre Nuclear Generating Station Unit 2 (TAC NO.ME9727) (RAI No. ; #13)

3. The information identified as proprietary in the document have in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that they describe unique design, manufacturing, experimental and investigative information developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it is the result of an intensive MHI effort.

5. The referenced information was furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design and manufacture of nuclear plant components without incurring the costs or risks associated with the design and the manufacture of the subject component. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. and world nuclear markets:

   A. Loss of competitive advantage due to the costs associated with development of technologies relating to the component design, manufacture and examination. Providing public access to such information permits competitors to duplicate or mimic the methodology without incurring the associated costs.

   B. Loss of competitive advantage of MHI's ability to supply replacement or new heavy components such as steam generators.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 25 day of March, 2013.

Jinichi Miyaguchi

Jinichi Miyaguchi,
Director- Nuclear Plant Component Designing Department
Mitsubishi Heavy Industries, LTD

Sworn to and subscribed

Before me this 25 day

of March, 2013

Masahiko Kubota

Notary Public

NOTARY

MAR. 25. 2013 62

My Commission does Not expire

登簿平成２５年第６２号

認　　証

嘱託人　三菱重工業株式会社　原子力事業本部
原子力製造総括部　原子力機器設計部　部長　宮
口仁一　は本職の面前で添付書面に　署名　した。

Masahiko Kubota
Kobe District Legal Affairs Bureau
NOTARY
44 Akashimachi, Chuo-ku,
Kobe, Japan

よって認証する。

平成２５年３月２５日

本職役場に於て

神戸市中央区明石町４４番地

神戸地方法務局所属

公証人　窪田正彦

公　証　人　役　場

Registered Number 6 2 Date MAR. 25. 2013

NOTARIAL CERTIFICATE

This is to certify that **JINICHI MIYAGUCHI , Director-Nuclear Plant Component Designing Department MITSUBISHI HEAVY INDUSTRIES, LTD** has affixed his signature in my very presence to the attached document.

Masahiko Kubota
Kobe District Legal Affairs Bureau
NOTARY
44 Akashimachi, Chuo-ku,
Kobe, Japan

Masahiko Kubota

MASAHIKO KUBOTA

Notary

44 Akashimachi, Chuo-Ku,

Kobe, Japan

Kobe District Legal Affairs Bureau

(面前法2)

ENCLOSURE 3

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to RAI 13, Revision 1

(Non-Proprietary)

RAI 13

The installation of new steam generators involved changes to the steam generator heat transfer characteristics, which could affect the performance of the plant under postulated loss of coolant accident conditions. Please explain how the existing ECCS analysis accounts for these changes, and how considerable steam generator tube plugging has been addressed in the ECCS evaluation. Provide the ECCS evaluation that will apply to the planned operating cycle.

RESPONSE - Revision 1

Summary

Note: Response (2) below includes information requested in RAI 14 associated with the Emergency Core Cooling System (ECCS) evaluation. RAI 14 states: "Provide a summary disposition of the U2C17 calculations relative to the planned reduced-power operation."

(1) Evaluation of Impact of Replacement Steam Generators on Emergency Core Cooling System (ECCS) Performance Analyses

Replacement steam generators (RSGs) were installed in SONGS Units 2 and 3 for Cycle 16. The Cycle 16 ECCS performance for SONGS Units 2 and 3 with the RSGs was evaluated to demonstrate conformance to the ECCS acceptance criteria for light water nuclear power reactors contained in 10 CFR 50.46. The evaluation considered the impact of the RSGs on the Analyses of Record (AORs) for Large Break Loss-of-Coolant Accident (LBLOCA), Small Break Loss-of-Coolant Accident (SBLOCA), and post-Loss-of-Coolant Accident (LOCA) Long-Term Cooling (LTC), which are based on the original steam generators (OSGs).

The impact of the RSGs on the SONGS Units 2 and 3 ECCS performance AORs was evaluated through a two-step process. First, the design data of the RSGs, including thermal hydraulic characteristics, were compared to those of the OSGs as modeled in the ECCS performance AORs. Second, differences in design data, which were identified from the comparison, were evaluated for their impact on ECCS performance. The scope of the comparison considered all design features of the steam generators (SGs) that are modeled in the ECCS performance analysis. The most significant parameters are discussed below.

(i) Rated Thermal Power

The OSGs and the RSGs were evaluated at the same core power level, as there was not a power uprate associated with installation of the RSGs.

(ii) RSG Tube Plugging and RCS Volume

The RSGs have more RCS volume than the OSGs. The amount of assumed tube plugging in the RSGs is less than the OSGs. These factors result in a net increase in the total reactor coolant system (RCS) volume. This is a beneficial feature since, for example, it results in more RCS inventory available to drain into the reactor vessel during a SBLOCA, thereby delaying the time that the core begins to uncover. A larger water volume increases the amount of water available to flow through the core during the blowdown period of a LBLOCA. This increases the amount of stored energy removed from the core during the blowdown period. The increase in water volume has an insignificant impact on the post-

LOCA LTC analysis. The maximum number of plugged tubes per SG for the RSG was assumed to be 779 tubes (8%) per SG for the RSG Cycle 16 ECCS evaluation.

(iii) RSG Heat Transfer Characteristics

The maximum assumed number of plugged tubes per SG is used in conjunction with the total number of tubes per SG to establish the minimum number of unplugged tubes per SG This is used to establish SG primary side volume, tube bundle flow area, and tube bundle heat transfer area. The RSGs have more tubes (9,727 versus 9,350) than the OSGs and a smaller value for the maximum number of plugged tubes (779 versus 2,000). RSG tubes have a larger average heated length (729.56 in. versus 680.64 in.) than the OSG tubes. These features result in larger values for the RSG for heat transfer area, tube bundle flow area, and tube bundle water volume. This is beneficial in the short and long term for SBLOCAs, which rely upon the steam generators for RCS heat removal.

The RSG tube bundle material is Inconel 690 whereas the OSG tube bundle material is Inconel 600. While the thermal conductivity of Inconel 690 is less than that of Inconel 600, the impact is not significant in the context of ECCS performance. First, the RSGs have larger heat transfer areas which compensate for the decrease of thermal conductivity. Second, after the subcooled forced convection mode of SG heat transfer early in a LOCA transient, the primary coolant-to-wall resistance, and not the wall resistance, is the limiting resistance for SG heat transfer during a LOCA. Therefore, the difference in thermal conductivity does not have a significant impact on ECCS performance given the overall design of the RSGs relative to the OSGs and the nature of SG heat transfer during a LOCA.

Sensitivity studies have shown that the impact due to SG heat transfer area changes is insignificant for LBLOCAs. Heat transfer characteristic differences have an insignificant impact on post-LOCA LTC.

(iv) RSG Pressure Drop / Flow Resistance

The RSGs have a smaller flow resistance and, consequently, a smaller pressure drop than the OSGs based on the same set of conditions and the maximum number of plugged tubes assumed by the ECCS performance analyses. A smaller total SG pressure drop is beneficial for ECCS performance.

The evaluation of the impact of the RSGs on the SONGS Units 2 and 3 ECCS performance analyses demonstrates that the RSGs have a beneficial impact on ECCS performance. Consequently, the results and conclusions of the SONGS Units 2 and 3 ECCS performance AORs for LBLOCA, SBLOCA, and post-LOCA LTC, performed for the OSGs, are applicable to SONGS Units 2 and 3 for operation with the RSGs.

(2) ECCS performance evaluations for SONGS Unit 2 Cycle 17

An ECCS performance analysis was performed for SONGS Unit 2 Cycle 17 to demonstrate conformance to the ECCS acceptance criteria for light water nuclear power reactors. The major changes evaluated in the Unit 2 Cycle 17 ECCS performance analysis are discussed as follows.

(i) Increase in T_{COLD}

The RCS temperature at the inlet to the core, i.e., T_{COLD}, has increased for Unit 2 Cycle 17 to 550°F from the previous Unit 2 Cycle 16 value of 541°F ("T_{COLD} Restoration"). The effect of the change in T_{COLD} is bounded by the Unit 2 Cycle 17 ECCS performance analysis.

(ii) SG Tube Plugging

The maximum number of plugged tubes per SG for Unit 2 Cycle 17 operation is 3%, which is bounded by the maximum number of plugged tubes per SG (8%) assumed in the RSG ECCS performance evaluation.

(iii) Extended Operation at Power Levels Between 50% and 100%

SONGS Unit 2 Cycle 17 safety analyses and LOCA analyses were evaluated for acceptability of plant operating at power levels between 50% and 100%, which bounds the planned operation at 70% power level. The impact of the extended reduced power operation was evaluated to determine the continued applicability of SONGS Units 2 and 3 ECCS performance AORs. It was concluded that the power operation range between 50% and 100% remains bounded by the current SONGS Units 2 and 3 ECCS performance AORs for LBLOCA, SBLOCA, and post-LOCA LTC.

DETAILED SUPPORTING INFORMATION

1. Introduction

The current AORs for SONGS Units 2 and 3 were performed for ZIRLO® implementation with OSGs and supports up to 21.4% steam generator tube plugging (SGTP). In order to support application of the AORs to SONGS Units 2 and 3 with RSGs, an evaluation of the impact of RSGs with 8% SGTP on the SONGS Unit 2 and 3 ECCS performance analysis was prepared. The evaluation considered the impact of the RSGs on the AORs for LBLOCA, SBLOCA, and post-LOCA LTC, and demonstrated conformance to the ECCS acceptance criteria for light water nuclear power reactors contained in 10 CFR 50.46. A reload analysis was prepared for SONGS Unit 2 Cycle 16 with RSGs. Then, for the following cycle, a reload analysis was prepared for SONGS Unit 2 Cycle 17.

The impact of the RSGs on the SONGS Units 2 and 3 ECCS performance AORs was evaluated through a two-step process. First, the design data of the RSGs, including thermal hydraulic characteristics, were compared to those of the OSGs as modeled in the ECCS performance AORs. Second, differences in design data, which were identified from the comparison, were evaluated for their impact on ECCS performance. The scope of the comparison considered all design features of the SGs that are modeled in the ECCS performance analysis.

This response to RAI 13 summarizes:

i. the results of current AORs (Section 2).

ii. a comparison of SG parameters for ECCS Performance analysis (Section 3).

iii. the results of the evaluation of the impact of RSGs on the SONGS ECCS performance analyses (Section 4).

iv. the applicability of the AORs to SONGS Unit 2 Cycle 16 (Section 5).

v. the applicability of the AORs to SONGS Unit 2 Cycle 17 (Section 6).

vi. the applicability of the AORs to SONGS Unit 2 Cycle 17 with reduced power operation (Section 7).

2. The Current AORs

The current LBLOCA, SBLOCA, and Post-LOCA LTC AORs for SONGS Unit 2 supported operation of the OSGs with up to 21.4% SGTP. The limiting LBLOCA break was the 0.6 double ended guillotine break in the pump discharge leg (DEG/PD) and assumes 21.4% SGTP. The limiting SBLOCA was the 0.04 ft^2 break and assumes 30% SGTP. The post-LOCA Long-Term Cooling analysis assumed 30% SGTP. Peak Cladding Temperature (PCT), Peak Local Oxidation (PLO), and Core Wide Oxidation (CWO) for those analyses satisfied 10 CFR 50.46 Acceptance Criteria as given in Table 1.

Table 1 - SONGS Units 2 and 3 AOR Results for OSGs

Analysis	PCT	PLO	CWO
LBLOCA	2170°F	15.3%	<1%
SBLOCA	2077°F	14.1%	<1%

3. Comparison of the OSGs and RSGs

RSGs were installed in SONGS Units 2 and 3 for Cycle 16. Although the RSGs are considered to be in-kind replacements of the OSGs in terms of form, fit and function, the RSGs have more tubes compared to the OSGs (approximately 4% more), longer tubes (approximately [ ]), thinner tube walls (approximately [ ]), larger inside diameter tubes [ ], [ ] outside diameter tubes, lower thermal conductivity material (Inconel 690, approximately 10% lower), larger primary side liquid volume (approximately [ ]).

Table 2 shows a comparison of parameters and percent differences between the RSGs and OSGs.

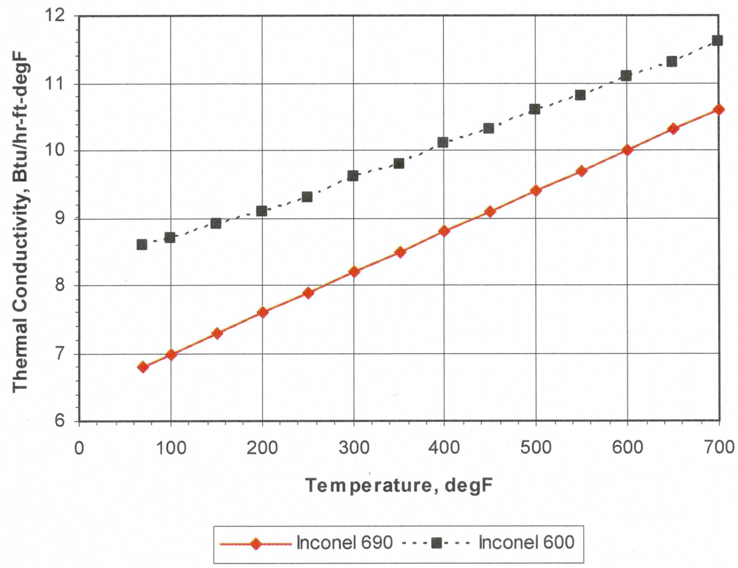
Table 2 - Steam Generator Parameters for ECCS Performance Analysis

Description	RSG	OSG	Percent Difference
1. Number of tubes/SG (0% SGTP)	9,727	9,350	4.0
2. Plugged tubes/SG (assumed in ECCS analysis)	8%	21.4% LBLOCA 30% SBLOCA 30% LTC	---
3. Heat transfer area (0% SGTP), ft^2	[ ]*	[ ]	[ ]
4. Tube bundle flow area (0% SGTP), ft^2	[ ]*	[ ]	[ ]
5. Secondary side fluid (water and steam) volume, ft^3	[ ]*	[ ]	[ ]
6. Total SG metal mass, lbm	[ ]*	[ ]	[ ]
SG Primary Side Water Volume			
7. SG Primary Side Water Volume (0% SGTP), ft^3	[ ]*	[ ]	[ ]
SG Tube Elevations			
8. Top of tube sheet elevation, inch	[ ]*	[ ]	---
9. Top of longest tube, at centerline, inch	[ ]*	[ ]	---
SG Tube Material			
10. SG tube material	Inconel 690	Inconel 600	---
Other Components			
11. SG Outlet Nozzle Flow Restrictor	Yes	No	---

[ ]* – Mitsubishi Heavy Industries, Ltd. Proprietary Class B information used with permission

Figure 1 shows a comparison as a function of temperature of the RSG and OSG Tubes Thermal Conductivity. The RSGs have Inconel 690. The OSGs have Inconel 600. The thermal conductivity of Inconel 690 is approximately 10% to 12% lower than Inconel 600.

Figure 1
Comparison of Thermal Conductivity of Inconel 690 (RSG) and Inconel 600 (OSG)



4. Impact of the RSGs on the AOR ECCS Performance Analysis

The impact of the RSGs on ECCS performance is evaluated from the effect of the following key parameters that summarize the impact of the differences noted in Table 2.

i. Effect of the changes in SG tube plugging.

ii. Effect of the differences in SG primary and secondary side liquid inventory.

iii. Effect of the differences in elevation of SG components.

iv. Effect of the differences in SG total metal mass.

4.1 Impact of the RSGs on the LBLOCA AOR

i. Effect of the changes in SG tube plugging.

[

]

[

]

ii. Effect of the differences in SG primary side liquid inventory.

The RSGs with 8% SGTP have [ ] in the SG primary side than the OSGs with 21.4% SGTP. As noted in Item 4.1.i, [

]

iii. Effect of the differences in SG secondary side liquid inventory.

The nominal [
] This [ ] LBLOCA ECCS performance analyses performed with the 1999 EM. This is [

]

iv. Effect of the differences in elevation of SG components.

The average SG tube length for the RSGs is [ ] The vertical rise for the longest tube for the RSGs is [ ]
The differences in elevation [ ] since the flow in a LBLOCA is mostly homogeneous and [ ]

v. Effect of the differences in SG total metal mass.

The SG metal mass for the RSGs is [ ] analyses performed with the 1999 EM. This is demonstrated [ ]

4.2 Impact of the RSGs on the SBLOCA AOR

i. Effect of the changes in SG tube plugging.

[ ]

The [ ] of SGs for the RSGs has the [ ] due to the decrease of the core decay heat with time. The [ ] RSG tubes have the [ ] of loss of natural circulation.

ii. Effect of the differences in SG tube geometry and material.

[ ]

$$R_{wall} = \frac{D_O}{2k_{wall}} \ln\left(\frac{D_O}{D_I}\right)$$

where:

R_{wall} = tube wall resistance to heat transfer, $(Btu/hr\text{-}ft^2\text{-}°F)^{-1}$
D_O = tube outside diameter, ft
D_I = tube inside diameter, ft
k_{wall} = tube material thermal conductivity, Btu/hr-ft-°F

shows [ ] compared to the OSGs at 550°F. This [ ] since after the cessation of the subcooled forced convection mode of SG heat transfer early in the LOCA transient, the [ ] is generally the limiting resistance for SG heat transfer during a LOCA.

iii. Effect of the differences in SG primary side liquid inventory.

The RSGs with 8% SGTP have approximately [ ] SG primary side than the OSGs with 21.4% SGTP. As noted in Item 4.2.i [ ] in the primary sides of SGs for the RSGs [ ] of core uncovery and the subsequent less severe core uncovery due to the decrease of the core decay heat at a later time.

iv. Effect of the differences in SG secondary side liquid inventory.

The nominal secondary side liquid inventory for the RSGs is approximately [ ] SBLOCA calculations with the Supplement 2 Model (S2M, CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April, 1998) are [ ]. The impact of the difference on the SG heat transfer [ ] and its impact on core uncovery or time of core uncovery is [ ]

v. Effect of the differences in elevation of SG components.

The average SG tube length for the RSGs is [ ] The vertical rise for the longest tube for the RSGs is [ ] the OSGs. These differences in elevation [ ] on a SBLOCA event since the impact on the elevation head difference between the hot and cold sides of the SGs is negligible.

vi. Effect of the differences in SG total metal mass.

The SG metal mass for the RSGs is [ ] OSGs. This [ ] ECCS performance analyses performed with the S2M (CENPD-137, Supplement 2-P-A). SG secondary side wall heat is [ ] the secondary side and has [ ] on the RCS.

vii. Impact of the SG outlet nozzle flow restrictor on the Main Steam Safety Valves (MSSVs) flow rate.

The RSGs SG outlet nozzle flow restrictors that can potentially reduce the flow rate past the MSSVs. The OSGs do not have SG outlet nozzle flow restrictors. However, the OSG SBLOCA analysis [ ] would occur if the SG outlet nozzle flow restrictor [ ] Thus, the impact of the flow restrictors is [ ]

viii. Impact of RSGs on the break spectrum analysis.

The SBLOCA AOR determined the PCT by running the break spectrum analysis for the following breaks with OSGs and 30% SGTP,

Break Size (ft^2)	0.03	0.04	0.05
Peak Cladding Temperature (°F)	1676	2077	1900
Maximum Cladding Oxidation (%)	5.14	14.11	10.96
Core-Wide Cladding Oxidation (%)	<0.34	<0.74	<0.61

[ ]

[ ]

The evaluation described above examined the impact of differences between OSGs and RSGs and concluded that the impact of RSGs with 8% SGTP [ ] The shape of the PCT vs. break size curve remains valid with the RSGs [ ] Thus, the OSGs maximum PCT [ ]

4.3 Impact of the RSGs on the post-LOCA Long Term Cooling AOR

The AOR post-LOCA LTC analysis with the OSGs was performed with the CENPD-254-P-A Evaluation Model (CENPD-254-P-A, "Post-LOCA LTC Evaluation Model," June 1980). The analysis is done with 30% SGTP.

Implementation of the RSGs [ ] The impact of the RSGs is as follows:

i. Impact of SG heat transfer parameters.

In the LTC analysis, the amount of SG heat transfer is determined by the controlled cooldown, which is accomplished with the Atmospheric Dump Valves (ADVs). Given the simple RCS hydraulic models and primary-to-secondary heat transfer models of the CENPD-254-P-A evaluation model in conjunction with the small heat loads during the cooldown (in comparison to the full power heat load for which the SGs are designed), the differences in data shown in Table 2 do not significantly impact the RCS response calculated in the post-LOCA LTC analysis. That is, the RSGs with 8% SGTP has [ ] than the OSGs with 30% SGTP and thus [ ] assumed in the AOR with OSGs.

ii. Effect of the differences in SG primary side liquid inventory.

The increase in water volume [ ] In the boric acid precipitation portion of the post-LOCA LTC analysis, the increase in water volume impacts the initial boric acid concentration in the sump. The impact is insignificant since the increase in volume [ ] in comparison to the total volume of all the sources that contribute to the sump inventory (i.e., [

] Also, the boric acid concentration of the RCS [ ] all the sources of boric acid so it acts [ ] sump and, therefore, an increase in volume would tend [ ] (although, in this case, [ ]).

For the post-LOCA LTC analysis with OSGs and 30% SGTP [ ] In the SBLOCA decay heat removal portion of the post-LOCA LTC analysis, [ ] RCS for those break sizes that are predicted to refill. However, the increase in water volume [ ] significant impact on the range of break sizes that are predicted to refill (since it is the break size itself and the HPSI pump delivery curve, not the RCS water volume, that primarily determines for which break sizes the RCS is predicted to refill).

iii. Effect of the differences in SG total metal mass.

The SG metal mass for the [ ] Increasing the SG metal mass has an adverse impact on the post-LOCA SG/RCS cooldown (i.e., it prolongs the time required to cooldown the RCS to the shutdown cooling system entry temperature). However, the AOR has an [ ] This [ ] metal mass is more than sufficient to cover the approximately [ ] the SG/RCS metal mass [ ] mass of the RSGs relative to the OSGs.

4.4 Conclusions for the Evaluation of RSGs on AOR ECCS Performance Analysis

The RSG LBLOCA evaluation with 8% SGTP is bounded by the LBLOCA AOR for OSGs with 21.4% SGTP and complies with the 10 CFR 50.46 Acceptance Criteria.

The RSG SBLOCA evaluation with 8% SGTP is bounded by the AOR SBLOCA analysis for OSGs with 30% SGTP and complies with the 10 CFR 50.46 Acceptance Criteria.

The RSG post-LOCA LTC evaluation with 8% SGTP is bounded by the AOR post-LOCA LTC evaluation with OSGs with 30% SGTP.

5. Evaluation for SONGS Unit 2 Cycle 16

RSGs and AREVA Lead Fuel Assemblies (LFAs) were installed in SONGS Unit 2 prior to the beginning of Cycle 16. The AOR analyses with OSGs bound and remain applicable for RSGs with up to 8% SGTP as assumed for the RSGs. An increase in the PCT is added to bound mixed core effects caused by the addition of AREVA LFAs. PCT, PLO, and CWO for SONGS Unit 2 Cycle 16 satisfy 10 CFR 50.46 Acceptance Criteria as given in Table 3.

Table 3 - SONGS Unit 2 Cycle 16 Results

Analysis	PCT	PLO	CWO
LBLOCA	2174°F	15.3%	<1%
SBLOCA	2077°F	14.1%	<1%

The impact of RSGs is [ ] LBLOCA and SBLOCA analyses [ ] for the post-LOCA LTC analysis as described Section 4.

The impact of the presence of the AREVA LFAs was evaluated by means of a [ ] analysis for LBLOCA that represented the difference in flow resistance of the Westinghouse and AREVA fuel assemblies. A [ ]

The presence of AREVA LFAs has [ ] on the SBLOCA and post-LOCA LTC analyses since there [ ] effects in these events.

The impact of [ ] fuel performance data was explicitly evaluated for the LBLOCA analysis by recalculating input data from the Cycle 16 fuel performance data and rerunning the AOR with the Cycle 16 fuel performance data that demonstrate [ ] the AOR.

The impact of [ ] fuel performance for the SBLOCA was demonstrated by recalculating input data with the Cycle 16 fuel performance data and showing [ ] by the AOR.

[ ] fuel performance data has [ ] the post-LOCA LTC analysis.

Other cycle specific changes, like changes in the minimum containment spray implemented for SONGS Cycle 16, [ ] on the post-LOCA LTC Analysis.

5.1 Conclusions for SONGS Unit 2 Cycle 16

The SONGS Unit 2 LBLOCA, SBLOCA, and post-LOCA LTC AORs, with a PCT delta of 4°F for the LBLOCA analysis remain bounding for SONGS Unit 2 Cycle 16.

6. Evaluation for SONGS Unit 2 Cycle 17

The SONGS Unit 2 Cycle 17 analysis includes eight first burn AREVA LFAs, eight second burn AREVA LFAs, and eight Westinghouse first burn Modified Standard Design (MSD) LFAs. In addition the analysis supports the cold leg temperature restoration from 541°F to 550°F. Similar to the analysis for Cycle 16, an increase in PCT was added to bound the mixed core effect caused by the addition of AREVA and Westinghouse LFAs. PCT, PLO, and CWO for SONGS Unit 2 Cycle 17 satisfy 10 CFR 50.46 Acceptance Criteria as given in Table 4.

Table 4 - SONGS Unit 2 Cycle 17 Results

Analysis	PCT	PLO	CWO
LBLOCA	2174°F	15.4%	<1%
SBLOCA	2077°F	14.1%	<1%

The impact of the presence of the eight additional AREVA LFAs and eight Westinghouse MSD LFAs was evaluated by means of a LBLOCA [ ] that represented the difference in flow resistance of the Westinghouse and AREVA fuel assemblies. [ ]

The presence of AREVA and Westinghouse LFAs has no effect on the SBLOCA and post-LOCA LTC analyses since there [ ] effects in these events.

The impact of [ ] fuel performance data was explicitly evaluated for the LBLOCA analysis by recalculating input data from the Cycle 17 fuel performance data and rerunning the AOR with the Cycle 17 fuel performance data for T_{COLD} restored to 550°F that demonstrate that it is bounded by the AOR.

The impact of [ ] fuel performance for the SBLOCA was demonstrated by recalculating input data with the Cycle 17 fuel performance data and showing that it is bounded by the AOR.

Differences in fuel performance data have [ ] the post-LOCA LTC analysis.

The impact of the change in the initial pressurizer level for the LBLOCA and SBLOCA analyses for Cycle 17 as a result of the T_{COLD} restoration to 550°F was demonstrated [ ] by the AOR.

Restoration to T_{COLD} to 550°F [ ] the LBLOCA and SBLOCA analyses since the AORs are performed using [ ] cold leg temperatures that are not changed.

Other plant cycle specific changes, like differences in the condensate storage tank liquid temperature implemented for Cycle 17, have [ ] the post-LOCA LTC analysis.

6.1 Conclusions for SONGS Unit 2 Cycle 17

The SONGS Unit 2 LBLOCA, SBLOCA, and post-LOCA LTC AORs, with a PCT [ ] the LBLOCA analysis remain bounding for SONGS Unit 2 Cycle 17.

7. Evaluation for Extended Operation at Power Levels Between 50% and 100%

Appendix K ECCS Performance Analyses are performed at 100% power plus uncertainty. SONGS Unit 2 Cycle 17 LOCA analyses were evaluated for acceptability of plant operation at power levels between 50% and 100%, which includes the planned operation at 70% power level.

7.1. LBLOCA Evaluation at Power Levels Between 50% and 100%

The LBLOCA analysis for SONGS Unit 2 with the 1999 EM calculates PCT during reflood. Key parameters that impact the LBLOCA transient response are initial power, peak linear heat generation rate (PLHGR) and initial stored energy in the core. Initial stored energy in the core has a [ ] that is limiting for SONGS Unit 2. Lowering the initial power results in lower stored energy in the core at the end of blowdown, which coupled with the lower decay heat (a multiple of the initial power) [ ] steaming rate during reflood. This results in an [ ] reflood rate, which yields [ ] The initial stored energy in the core is calculated by [ ] the fuel performance time in life dependent FATES3B (CENPD-139-P-A, "Fuel Evaluation Model," July 1974) data file to the 1999 EM codes input files.

In summary, for the LBLOCA event, lower initial power yields [ ]

The AOR fuel performance input data for 100% power [ ] for low power operation.

7.2 SBLOCA Evaluation at Power Levels Between 50% and 100%

The SBLOCA analysis for SONGS Unit 2 with the S2M calculates PCT during the time of maximum core uncovery. Key parameters that impact the SBLOCA PCT are [ ] Reducing the initial power [ ] in the core. This [ ] amount of core uncovery and [ ] the uncovered region of the core yielding [ ]

Hence, for the SBLOCA event, lower initial power yields [ ]

The AOR fuel performance input data for 100% power [ ] for low power operation.

7.3 Post-LOCA Long Term Cooling Evaluation at Power Levels Between 50% and 100%

The boric acid precipitation analysis portion of the post-LOCA LTC analysis uses the decay heat for 100% power which is conservative since it produces larger evaporation which increases the core boron concentration. Thus, the AOR for 100% power [ ] for reduced power operation.

The decay heat portion of the post-LOCA LTC analysis accounts for the ADVs during calculation of the cooldown of the system. The cooldown at 100% power [ ] cooldown for lower initial powers since the decay heat for lower powers is proportionally lower. Thus, the AOR [ ] results for reduced power operation.

7.4 Conclusions for Operation at Power Levels Between 50% and 100%

The impact of the extended reduced power operation was evaluated to determine the continued applicability of SONGS Units 2 and 3 ECCS performance AORs at 100% power. It was concluded that the power operation range between 50% and 100% [

]

BOARD NOTIFICATION ENCLOSURE 3

SOUTHERN CALIFORNIA
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Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

March 22, 2013

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAI 18), Revision 1
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. 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

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

3. Letter from Mr. James R. Hall (USNRC) to Mr. Peter T. Dietrich (SCE), dated December 26, 2012, Request for Additional Information Regarding Response to Confirmatory Action Letter, San Onofre Nuclear Generating Station, Unit 2

4. Letter from Mr. Richard J. St. Onge (SCE) to NRC Document Control Desk, dated January 24, 2013, Response to Request for Additional Information (RAI 18) Regarding Confirmatory Action Letter Response, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By letter dated December 26, 2012 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. SCE provided the response to RAI 18 in a letter dated January 24, 2013 (Reference 4). Enclosure 1 of this letter provides Revision 1 to the RAI 18 response.

P.O. Box 128
San Clemente, CA 92672

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Enclosures:

1. Response to RAI 18, Revision 1

cc: E. E. Collins, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to RAI 18, Revision 1

RAI 18

Reference 1, Section 11.1, page 52 – SCE proposes to upgrade the vibration and loose parts monitoring system (VLPMS) as a defense-in-depth measure to enhance plant monitoring capability to facilitate early detection of a steam generator tube leak and ensure immediate and appropriate plant operator and management response.

Fluid Elastic Instability (FEI) was identified as a main cause of the tube wear for both the Unit 2 and 3 steam generators. The FEI experienced is due to a combination of the conditions of steam quality, secondary side fluid velocity in the vicinity of the tube bundle, and steam void fraction, and the degree of such fluid elastic instability is related to the damping provided by internal support structures. According to your report, "steam quality 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." However, the proposed plant VLPMS enhancement does not appear to directly monitor steam quality, secondary side fluid velocity, or steam void fraction.

Please provide the following information to address the effectiveness of the enhanced VLPMS:

a. Describe the specific purpose of using the enhanced VLPMS equipment for monitoring steam generator performance. For example, is it to be used for monitoring acoustic noise indicative of flow velocity, steam quality, and void fraction, or for the measurement of metallic noise indicative of vibration of tubes against each other or against tube support structures? Exactly how will this be done? What is the theory of operation? If it will be used to monitor an increase in ρv^2 leading to the onset of FEI, provide a description of the correlation of the velocity of steam voids through the secondary side of the steam generator and the relative changes in characteristics of the signal output from the various VLPMS accelerometers. If it is to be used for detecting actual tube vibration, provide a description of the process that will be used for discerning actual tube vibration noise from background noise, and the required threshold identification criteria that will be applied to reach the conclusion that tube vibration is occurring.

b. Identify the ranges of amplitudes and frequencies of the acoustic noise signals from each accelerometer that are indicative of an approach to the conditions leading to FEI or actual tube vibration, and the reasons for selection of the more sensitive accelerometers. Also, discuss the required response time of the signal processing equipment needed to detect and continuously monitor either fluid velocities within the steam generator or tube impact noise, depending on the intended use of the enhanced VLPMS, and the actual response time capabilities of the equipment, from sensor through processed signal output, that is being proposed for use.

c. Discuss the acceptance criteria (e.g., magnitude of signal, plant power level, etc.) that will be used to establish the setpoints for the alarms described in Section 11 of your report: "The signals from these sensors are compared with preset alarm setpoints." Provide a description of how the alarm setpoints were established, and at what point during the start-up of Unit 2 will these alarm setpoints be calibrated into the VLPMS. If the setpoints have not yet been determined, provide a description of your plan for determining and implementing these settings.

d. Describe the planned operator actions and any changes to the procedures for responding to alarms or signals potentially indicative of tube-to-tube contact, including time limits for analyzing the signals and taking any necessary action including plant shutdown. Describe the lessons learned that have been drawn from the signals of potential metal-to-metal contact experienced in Unit 3 and how these lessons have been factored into current procedures.

e. A description of how you determined that acoustic noise monitoring and predictive signal processing was the best method for monitoring either the onset of FEI or actual tube vibration, including a list of other methods (e.g., time domain reflectivity probes calibrated for steam void propagation monitoring) that had been considered for enhancing steam generator tube monitoring during start-up of Unit 2, and the reasons for their rejection.

RESPONSE - Revision 1

The purpose of the upgraded vibration and loose parts monitoring system (VLPMS) is to provide additional monitoring capabilities for steam generator (SG) secondary side acoustics. The upgraded VLPMS is capable of recording secondary side acoustic signals via accelerometers mounted on the external shell of the SGs at locations near the upper tube bundle and tubesheet. The upgraded VLPMS will be used as a backward looking analysis tool in subsequent inspection outages should unexpected wear be discovered. The upgraded VLPMS will enable SCE to evaluate historical SG secondary side acoustic signal data for events which may help with the understanding of the causes of unexpected tube wear. To gain experience with the analysis tools during the next cycle, the recorded data will be reviewed on a monthly basis in accordance with plant procedures.

The Unit 2 Return to Service (RTS) report describes the upgrades to the VLPMS in Section 11.1 as an additional action to provide monitoring capabilities for secondary side acoustic signals. SCE did not propose the upgrade of the VLPMS as a defense-in-depth measure nor as a means of monitoring steam quality, secondary side fluid velocity, or steam void fraction. Corrective measures to control these secondary side parameters are addressed in Section 8 of the RTS report. The defense-in-depth measures being taken in support of Unit 2 return to service are described in Section 9 of the RTS report.

The theory of operation of the VLPMS data acquisition equipment is provided as follows:

> The Vibration and Loose Parts Monitoring System (V&LPMS) is a stand-alone system designed to perform loose parts detection and vibration monitoring functions. The loose part detection function is designed to fulfill the requirements of the loose parts detection system as set forth in Regulatory Guide 1.133. The design objective of the loose parts detection portion of the V&LPMS is to detect the presence of loose parts in the reactor coolant system (RCS) and annunciate an alarm in the control room when a loose part is detected.
>
> The system consists of loose parts and vibration sensors, preamplifiers and a computerized data acquisition and processing system. The sensors and preamplifiers are located inside the containment and the data acquisition and processing equipment is housed in a cabinet located in the control room cabinet area. The equipment located inside containment consists of piezo-electric sensors, preamplifiers and associated

cabling at each of the following natural collection regions of each unit to detect loose parts:

- Upper reactor vessel; vessel head on head lift rig stopper
- Lower reactor vessel; outside wall
- Steam generator E088; outside wall
- Steam generator E089; outside wall

The RCS component vibration monitoring, reactor internal vibration monitoring, and vibration data analysis features are on-demand functions. The on-demand features provided with the VLPMS allow the selection of any two loose parts, vibration or reactor internal vibration channels for vibration monitoring or analysis. The on-demand data acquisition and analysis features also allow a live channel signal or historical data from the historical data file to be selected for time domain and/or frequency domain analysis, displayed, stored and/or printed.

The upgrades to the VLPMS consist of:

- Relocation of existing VLPMS accelerometers (2 per SG) from the support skirt to locations above and below the SG tubesheet. These will remain as VLPMS sensors to meet Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors"
- Installation of increased sensitivity accelerometers (2 per SG) at locations above and below the tubesheet
- Installation of increased sensitivity accelerometers (2 per SG) on an 8 inch hand hole on the side of the SGs to monitor for secondary side noises at the upper tube bundle

Relocation of the accelerometers enhances the acoustic monitoring of the SG secondary side by placing accelerometers closer to the upper tube bundle.

The new accelerometers have an increased sensitivity of 25 pC/g compared with 10 pC/g for the existing accelerometers. The acceptance criteria used to establish the setpoints for the alarms associated with the upgraded VLPMS accelerometers is the same as used with the Regulatory Guide 1.133 accelerometers. After Unit 2 reaches 70% power, background data is collected and alarm thresholds are established to compensate for high background noise as discussed in Regulatory Guide 1.133, section C.1.b, "System Sensitivity."

Operator actions for VLPMS alarms are controlled by the VLPMS trouble annunciator operations alarm response procedure. Upon an alarm, the VLPMS automatically records data for all of the VLPMS accelerometer channels. The Control Room Shift Technical Advisor is instructed by this procedure to immediately notify the system engineer supervisor for any loose parts channel alarms associated with the SGs. The Control Room Operator documents the VLPMS alarm in the site Corrective Action Program (CAP). The CAP requires an operability determination to be made within 24 hours of event discovery. Each loose parts alarm associated with SGs will be independently reviewed by an offsite vendor.

Following identification of tube-to-tube wear (TTW) caused by Fluid Elastic Instability (FEI) in Unit 3 and two indications of TTW in Unit 2, a review of the VLPMS alarms for the previous operating cycle of both units was performed. No potential metal-to-metal contact alarms were recorded for Unit 2. Potential metal-to-metal contact alarms were recorded for Unit 3. Analysis of data from Unit 3 VLPMS events concluded most of the events were the result of RCS temperature changes. A number of events were not directly associated with RCS temperature changes and were reviewed by on-site as well as independent off site personnel. The independent review concluded these alarms were caused by: "...true metallic impacts and not false indications from electrical noise or fluctuations in background noise." The review found the acoustic signals were similar to those that occur when the SGs shift during RCS temperature transients. None of the VLPMS alarms were attributed to SG tube vibration.

Since the VLPMS is not designed to detect tube to tube contact, the absence of tube vibration related VLPMS alarms is consistent with the capabilities of its design. The approach in the Unit 2 RTS plan is to eliminate the causes of TTW caused by FEI. Reducing power reduces SG secondary side thermal-hydraulic parameters to values within the industry's experience. While the RTS plan does not require a direct method to measure tube vibration, SCE determined it was appropriate to upgrade the existing VLPMS as discussed above.

BOARD NOTIFICATION ENCLOSURE 4

SOUTHERN CALIFORNIA
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Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

March 22, 2013

10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAI 50)
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. 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

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

3. Letter from Mr. James R. Hall (USNRC) to Mr. Peter T. Dietrich (SCE), dated March 18, 2013, San Onofre Nuclear Generating Station Unit 2 - Second Request for Additional Information Regarding Response to Confirmatory Action Letter

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By letter dated March 18, 2013 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. Enclosure 2 of this letter provides the response to RAI 50.

Enclosure 2 of this submittal contains proprietary information. SCE requests that this proprietary enclosure be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosure 1 provides a notarized affidavit from Westinghouse Electric Company (WEC), which sets forth the basis on which the information in Enclosure 2 may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Proprietary information identified in Enclosure 2 was extracted from WEC document LTR-SGDA-13-316, Revision 0, "San Onofre Nuclear Generating Station Unit 2 MHI Replacement Steam Generator Response to RAI 50," which is addressed in the affidavit. Enclosure 3 provides the non-proprietary version of Enclosure 2.

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Enclosures:

1. Notarized Affidavit
2. Response to RAI 50 (Proprietary)
3. Response to RAI 50 (Non-Proprietary)

cc: E. E. Collins, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

Notarized Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: CONO-13-24

CAW-13-3672

March 21, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-SGDA-13-31 P-Attachment, "San Onofre Nuclear Generating Station Unit 2 MHI Replacement Steam Generator Response to RAI 50" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-13-3672 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Southern California Edison.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-13-3672 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-13-3672

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

James A. Gresham, Manager

Regulatory Compliance

Sworn to and subscribed before me

this 21st day of March 2013

Renee' Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA

NOTARIAL SEAL

Renee Giampole, Notary Public

Penn Township, Westmoreland County

My Commission Expires September 25, 2013

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SGDA-13-31 P-Attachment, "San Onofre Nuclear Generating Unit 2 MHI Replacement Steam Generator Response to RAI 50," for submittal to the Commission, being transmitted by Southern California Edison Letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the calculation of fluidelastic excitation of steam generator tubes and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Respond to NRC Request for Additional Information on intermediate details of the Westinghouse flow induced vibration (FIV) calculations for several limiting tubes. The information provided will enable the NRC to perform a comparison between Westinghouse and Mitsubishi Heavy Metal Industries (MHI) FIV methods and results.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purpose of evaluating the impact of fluidelastic excitation on steam generator tube integrity.

(b) Westinghouse can sell support and defense of the thermal hydraulic analysis of secondary side flow field in the steam generator shell.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar information and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for additional information regarding stability ratios calculated for certain anti-vibration bar (AVB) support conditions for the San Onofre Nuclear Generating Station Unit 2 steam generators.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice

ENCLOSURE 3

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to RAI 50

(NON-PROPRIETARY)

RAI 50

In Reference 8, p. 102 through 137, Figures 4-5 through 4-40 show local SR results. Please provide a tabulated summary of the results for [

]

RESPONSE

Note: RAI Reference 8 is Westinghouse Electric Company (WEC) document LTR-SGDA-12-36, "Flow-Induced Vibration and Tube Wear Analysis of the San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators Supporting Restart," Revision 3.

Figures 4-5 through 4-40 from RAI Reference 8 provided in-plane and out-of-plane stability ratios at 70% and 100% power for various support conditions. Each case represents a unique support condition. Every case has four associated SR maps: 100% power out-of-plane SR, 70% power out-of-plane SR, 100% power in-plane SR, 70% power in-plane SR. Out of the nine cases contained in Figures 4-5 through 4-40, five cases have support conditions of 5 or more missing AVBs.

The limiting tubes would not provide a good representation of the bundle since these tubes are all adjacent. To provide a meaningful comparison, the following tables provide a summary of the results following the format in MHI L5-04GA567 Rev 6, Tables 2.2-1 to 2.2-4. The nine representative tube locations used in this response are the same nine representative tube locations used in response to RAI 29.

Note: The SRs in Figures 4-5 through 4-40 and in Tables 1 – 5 (below) assume all tubes are active (in-service). Two tubes, R95 C85 and R100 C80, are plugged and stabilized (out-of-service).

Table 1 – Stability Ratio with 5 Consecutive AVBs Missing (Case 45)
B03, B04, B05, B06, and B07

Row	Column	70%		100% (No Plug)	
		Out-of-Plane	In-Plane	Out-of-Plane	In-Plane
80	70				
80	80				
95(*)	85(*)				
100	70				
100(*)	80(*)				
120	70				
120	80				
125	85				
138	84				

Note (*): Tube plugged and stabilized.

Table 2 – Stability Ratio with 5 Consecutive AVBs Missing (Case 46) B04, B05, B06, B07, and B08

Row	Column	70%		100% (No Plug)	
		Out-of-Plane	In-Plane	Out-of-Plane	In-Plane
80	70				
80	80				
95(*)	85(*)				
100	70				
100(*)	80(*)				
120	70				
120	80				
125	85				
138	84				

Note (*): Tube plugged and stabilized.

Table 3 – Stability Ratio with 6 Consecutive AVBs Missing (Case 53) B03, B04, B05, B06, B07, and B08

Row	Column	70%		100% (No Plug)	
		Out-of-Plane	In-Plane	Out-of-Plane	In-Plane
80	70				
80	80				
95(*)	85(*)				
100	70				
100(*)	80(*)				
120	70				
120	80				
125	85				
138	84				

Note (*): Tube plugged and stabilized.

Table 4 – Stability Ratio with 6 Consecutive AVBs Missing (Case 54)
B04, B05, B06, B07, B08, and B09

Row	Column	70%		100% (No Plug)	
		Out-of-Plane	In-Plane	Out-of-Plane	In-Plane
80	70				
80	80				
95(*)	85(*)				
100	70				
100(*)	80(*)				
120	70				
120	80				
125	85				
138	84				

Note (*): Tube plugged and stabilized.

Table 5 – Stability Ratio with 7 Consecutive AVBs Missing (Case 60)
B03, B04, B05, B06, B07, B08, and B09

Row	Column	70%		100% (No Plug)	
		Out-of-Plane	In-Plane	Out-of-Plane	In-Plane
80	70				
80	80				
95(*)	85(*)				
100	70				
100(*)	80(*)				
120	70				
120	80				
125	85				
138	84				

Note (*): Tube plugged and stabilized.

To directly compare Westinghouse in-plane SRs to those from L5-04GA567, Tables 2.2-1 through 2.2-4, the same support conditions must be used. In L5-04GA567, the number of consecutive missing AVBs start from AVB B01 (hot leg side). The following Tables 6 - 9 provide the Westinghouse in-plane SRs and are based on identical support conditions from L5-04GA567. The 70% power SRs for tubes R95 C85 and R100 C80 in Tables 6 - 9 are out-of-service SR values.

Table 6 – In-Plane Stability Ratio with 6 Consecutive AVBs Missing (Case 51) B01 through B06

Row	Column	70%	100% (No Plug)
80	70		
80	80		
95(*)	85(*)		
100	70		
100(*)	80(*)		
120	70		
120	80		
125	85		
138	84		

Note (*): Tube plugged and stabilized.

Table 7 – In-Plane Stability Ratio with 8 Consecutive AVBs Missing (Case 64) B01 through B08

Row	Column	70%	100% (No Plug)
80	70		
80	80		
95(*)	85(*)		
100	70		
100(*)	80(*)		
120	70		
120	80		
125	85		
138	84		

Note (*): Tube plugged and stabilized.

Table 8 – In-Plane Stability Ratio with 10 Consecutive AVBs Missing (Case 73)
B01 through B10

Row	Column	70%	100% (No Plug)
80	70		
80	80		
95(*)	85(*)		
100	70		
100(*)	80(*)		
120	70		
120	80		
125	85		
138	84		

Note (*): Tube plugged and stabilized.

Table 9 – In-Plane Stability Ratio with 12 Consecutive AVBs Missing (Case 78)

Row	Column	70%	100% (No Plug)
80	70		
80	80		
95(*)	85(*)		
100	70		
100(*)	80(*)		
120	70		
120	80		
125	85		
138	84		

Note (*): Tube plugged and stabilized.

BOARD NOTIFICATION ENCLOSURE 5

SOUTHERN CALIFORNIA
EDISON®
An EDISON INTERNATIONAL® Company

Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

March 29, 2013 10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAI 62)
Regarding Confirmatory Action Letter Response
(TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:

1. 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

2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2

3. Letter from Mr. James R. Hall (USNRC) to Mr. Peter T. Dietrich (SCE), dated March 18, 2013, Second Request for Additional Information Regarding Response to Confirmatory Action Letter, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By letter dated March 18, 2013 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. Enclosure 2 of this letter provides the response to RAI 62.

P.O. Box 128
San Clemente, CA 92672

Enclosure 2 of this submittal contains proprietary information. SCE requests that this proprietary enclosure be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosure 1 provides a notarized affidavit from AREVA NP Inc., which sets forth the basis on which the information in Enclosure 2 may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Proprietary information in Enclosure 2 was extracted from AREVA document 51-9197672-003, SONGS Probability of FEI Operational Assessment RAI Responses, which is addressed in the affidavit. Enclosure 3 provides the non-proprietary version of Enclosure 2.

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Enclosures:

1. Notarized Affidavit
2. Response to RAI 62 (Proprietary)
3. Response to RAI 62 (Non-Proprietary)

cc: E. E. Collins, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

Notarized Affidavit

AFFIDAVIT

STATE OF NORTH CAROLINA)
) ss.
COUNTY OF MECKLENBURG)

1. My name is Dennis C. Williford. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the document titled "51-9197672-003, 'SONGS Unit 2 Probability of FEI Operational Assessment RAI Responses'," and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information":

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Dennis C. Williford

SUBSCRIBED before me this 20th

day of March 2013.

Thomas A. Casias

Thomas A. Casias
NOTARY PUBLIC, STATE OF NORTH CAROLINA, COUNTY OF MECKLENBURG
MY COMMISSION EXPIRES: 14 December 2014

ENCLOSURE 3

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to 62

(NON-PROPRIETARY)

RAI 62

In Reference 10, Figure 8-3, the staff understands that the stability ratio (SR) in the context of Figure 8-3 is a 95% upper bound estimate, both for the last operating period for both Units 2 and 3 and for the next operating period for Unit 2. Why wasn't a best estimate SR used for benchmarking the probability of SR>1 at the conclusion of the last operating period for both Units 2 and 3? (Benchmarking refers to selecting a contact force criterion for effective AVB support such as to produce probabilities of SR>1 at the end of the last operating period consistent with what was actually observed.) How would a best estimate SR have affected the curves presented for the last operating period? Discuss whether the use of a 95% upper bound estimate for benchmarking purposes essentially negates the conservatism of using 95% upper bound SR estimates for future operation of Unit 2?

RESPONSE

Note: RAI Reference 10 is "SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear," prepared by Areva NP Inc. Document No. 51- 9187230-000, Revision 0, October 2012.

The goal of mechanistically based probabilistic models is to accurately simulate service performance in a probabilistic manner. To achieve this goal, mechanistically based probabilistic models are benchmarked to the observed behavior. Benchmarking is successful when model results show agreement, not conservatism, compared to observed behavior.

Conservatism of estimates for future operation is typically established by demonstrating that an unwanted future event has an acceptably low probability of occurrence. Applying a factor of safety to the unwanted event provides additional conservatism. For example, in tube integrity probabilistic models, the industry established requirement is that the probability of a tube burst at 3 times the normal operating differential pressure, $3\Delta P$, considering all degraded tubes in the bundle, must be maintained below 0.05.

For the probabilistic evaluation of tube-to-tube wear, the unwanted event is the first onset of in-plane fluid elastic instability (FEI) in Unit 2 at 70% power. Benchmarking of the probabilistic FEI model is achieved by showing agreement with the observed FEI behaviors of both Unit 3 and Unit 2 in the previous cycle with a stability ratio (SR) threshold of 1.0.

Why wasn't a best estimate SR used for benchmarking the probability of SR>1 at the conclusion of the last operating period for both Units 2 and 3? How would a best estimate SR have affected the curves presented for the last operating period?

[

] Benchmarking to upper 95th percentile SRs produces support criteria that are physically realistic based on laboratory testing and field performance.

Discuss whether the use of a 95% upper bound estimate for benchmarking purposes essentially negates the conservatism of using 95% upper bound SR estimates for future operation of Unit 2?

Using upper 95th percentile SRs for benchmarking does not negate the conservatism of SR estimates for future operation of Unit 2 because benchmarking matches the observed FEI behavior in the previous cycle. [

]

BOARD NOTIFICATION ENCLOSURE 6

SOUTHERN CALIFORNIA
EDISON®
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Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

April 1, 2013 10 CFR 50.4

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket No. 50-361**
Response to Request for Additional Information (RAIs 68, 69, 70 and RAI 2 Revision 1) Regarding Confirmatory Action Letter Response (TAC No. ME 9727)
San Onofre Nuclear Generating Station, Unit 2

References:
1. 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
2. Letter from Mr. Peter T. Dietrich (SCE) to Mr. Elmo E. Collins (USNRC), dated October 3, 2012, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation, San Onofre Nuclear Generating Station, Unit 2
3. Email from Mr. James R. Hall (USNRC) to Mr. Ryan Treadway (SCE), dated March 15, 2013, Request for Additional Information (RAIs 68-72) Regarding Response to Confirmatory Action Letter, San Onofre Nuclear Generating Station, Unit 2
4. Letter from Mr. Richard J. St. Onge (SCE) to Document Control Desk (USNRC), dated February 25, 2013, Response to Request for Additional Information (RAIs 2, 3 and 4) Regarding Confirmatory Action Letter Response, San Onofre Nuclear Generating Station, Unit 2

Dear Sir or Madam,

On March 27, 2012, the Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) (Reference 1) to Southern California Edison (SCE) describing actions that the NRC and SCE agreed would be completed to address issues identified in the steam generator tubes of San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. In a letter to the NRC dated October 3, 2012 (Reference 2), SCE reported completion of the Unit 2 CAL actions and included a Return to Service Report (RTSR) that provided details of their completion.

By e-mail dated March 15, 2013 (Reference 3), the NRC issued Requests for Additional Information (RAIs) regarding the CAL response. Enclosure 1 of this letter provides the response to RAIs 68, 69 and 70. SCE provided the response to RAI 2 in a letter dated

P.O. Box 128
San Clemente, CA 92672

February 25, 2013 (Reference 4). Enclosure 1 of this letter provides Revision 1 to the RAI 2 response.

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please call me at (949) 368-6240.

Sincerely,

Enclosure:

1. Response to RAIs 68, 69, 70 and RAI 2 Response, Revision 1

cc: A. T. Howell III, Regional Administrator, NRC Region IV
J. R. Hall, NRC Project Manager, SONGS Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, NRC Region IV

ENCLOSURE 1

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER

DOCKET NO. 50-361

TAC NO. ME 9727

Response to RAIs 68, 69, 70 and

RAI 2 Response, Revision 1

RAI 68

Reference 1, Response to RAI 2 – Provide wear depth distributions for the following, for both Unit 3 SGs:

- Anti-vibration bar (AVB) wear depth distributions for the group of AVBs, B03 through B10, for tubes without tube-to-tube wear (TTW)
- AVB wear depth distributions for the group of AVBs, B03 through B10, for tubes with TTW
- AVB wear depth distribution for the group of AVBs, B01, B02, B11, B12, for tubes without TTW
- AVB wear depth distribution for the group of AVBs, B01, B02, B11, B12, for tubes with TTW
- Tube support plate (TSP) wear depth distribution (top TSP only) for tubes without TTW
- TSP wear depth distribution (top TSP only) for tubes with TTW

RESPONSE

Note: RAI Reference 1 is SCE's "Response to Request for Additional Information (RAIs 2, 3, and 4) Regarding Confirmatory Action Letter Response," dated February 25, 2013.

The distributions of tube wear depths for Unit 3 are provided in the attached figures in the form of cumulative distribution functions, and wear pattern distribution plots with actual wear location counts.

Figure 1 is a schematic of the tube support configuration for reference purposes.

Cumulative distribution functions (CDF) of the requested wear depths are provided in Figures 2 through 4. Figure 2 shows the CDF for the anti-vibration bar (AVB) wear depths at the upper supports (B03 through B10) for tubes with and without TTW. Figure 3 shows the CDF of AVB wear depths at the lower supports (B01, B02, B11, and B12) for tubes with and without TTW. Figure 4 shows the CDF of tube support plate (TSP) wear depths at the uppermost support plate elevation (07C and 07H) for tubes with and without TTW.

Plots that show the wear patterns and counts of the AVB and TSP support locations are also provided in Figures 5 and 6. Figure 5 shows the wear pattern (depths and numbers of affected supports) for AVB wear for the two requested tube groups; i.e., tubes with and without TTW. This figure collectively includes the AVB locations from the cold-leg side (B12 to B7) to the hot-leg side (B06 to B01).

Figure 6 shows the wear pattern (depth and number of affected supports) for TSP wear for the two requested tube groups; i.e., tubes with and without TTW. This figure collectively shows the TSP locations from the cold-leg side (01C to 07C) to the hot-leg side (07H to 01H).

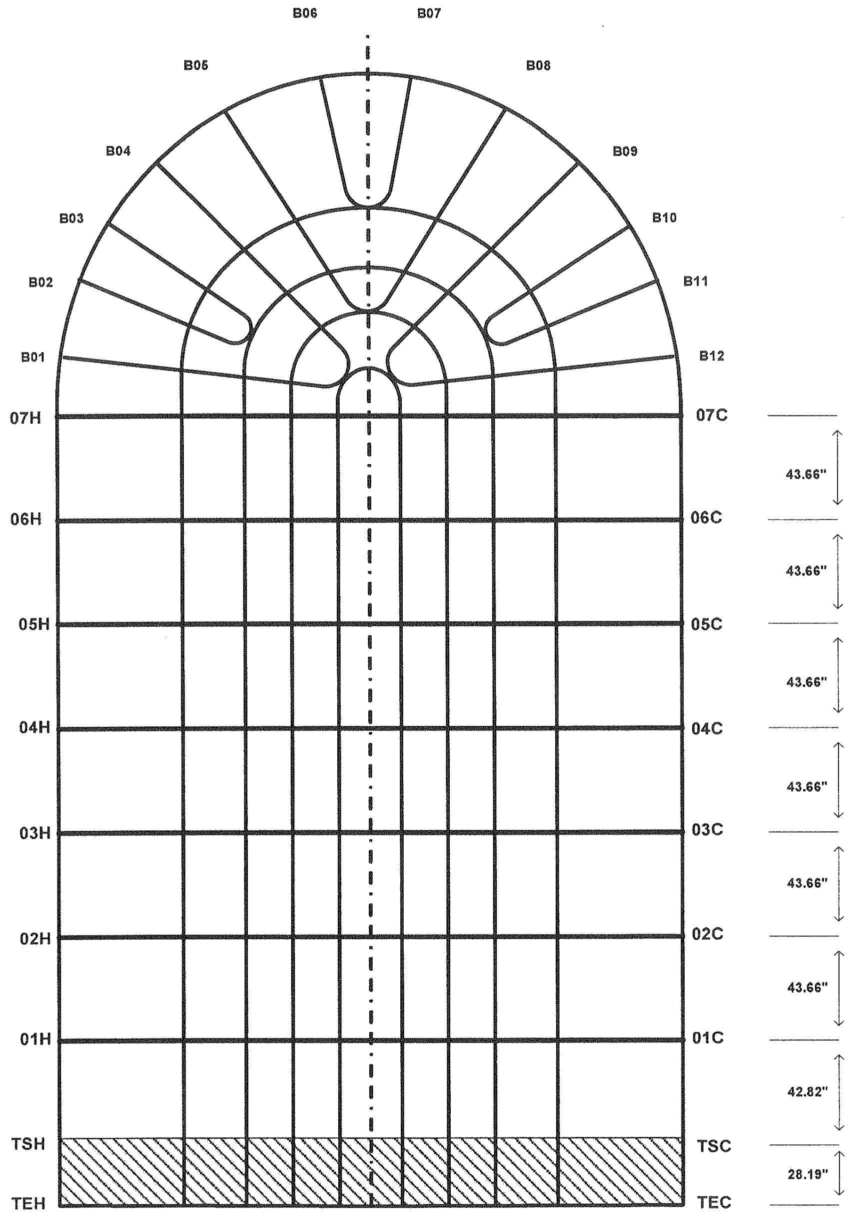


Figure 1 - SONGS Steam Generator Tube Support Structure Schematic

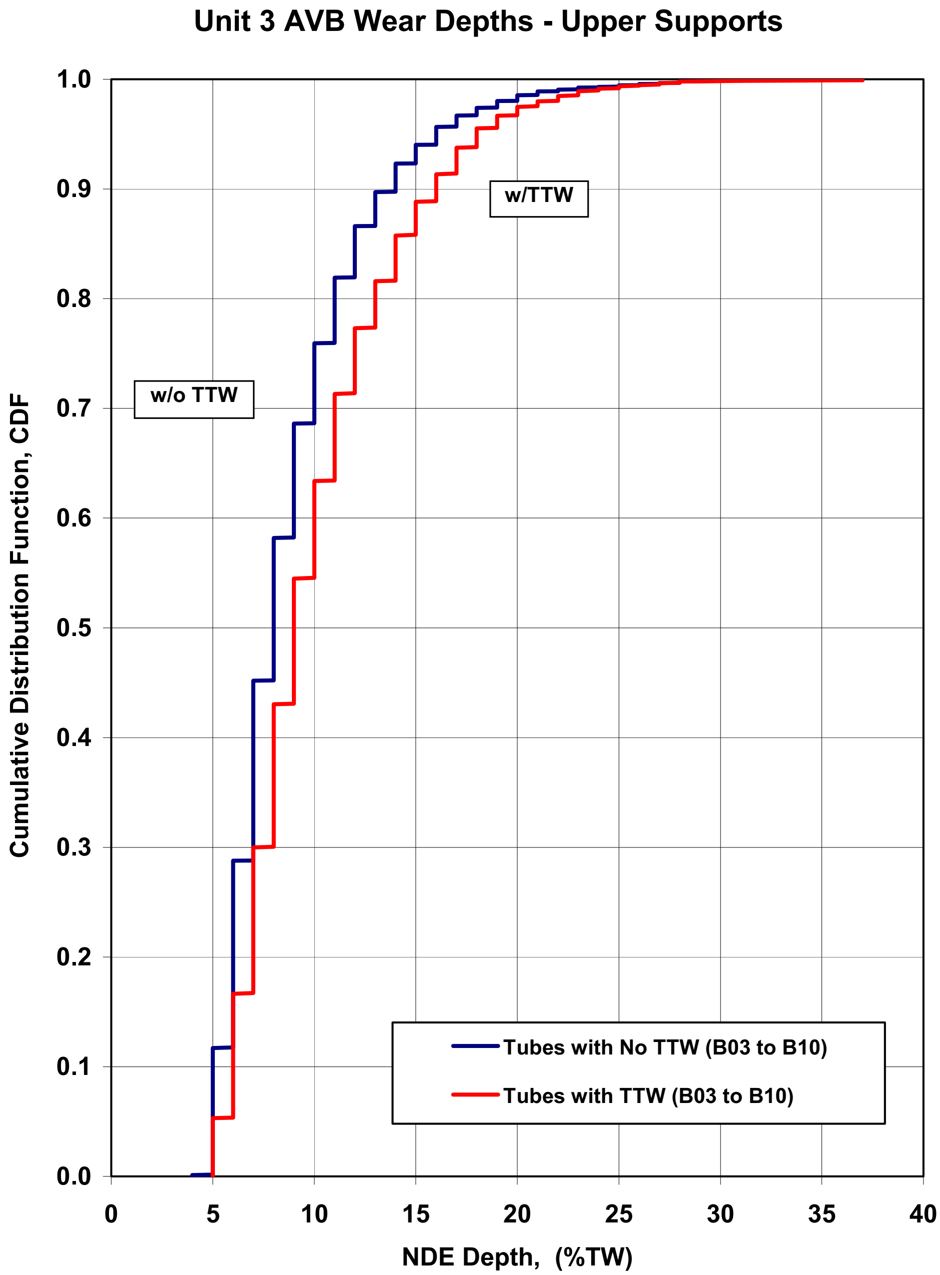


Figure 2 - Wear Depths for Tube/AVB Contacts at the Upper Supports after 0.926 Years at Power

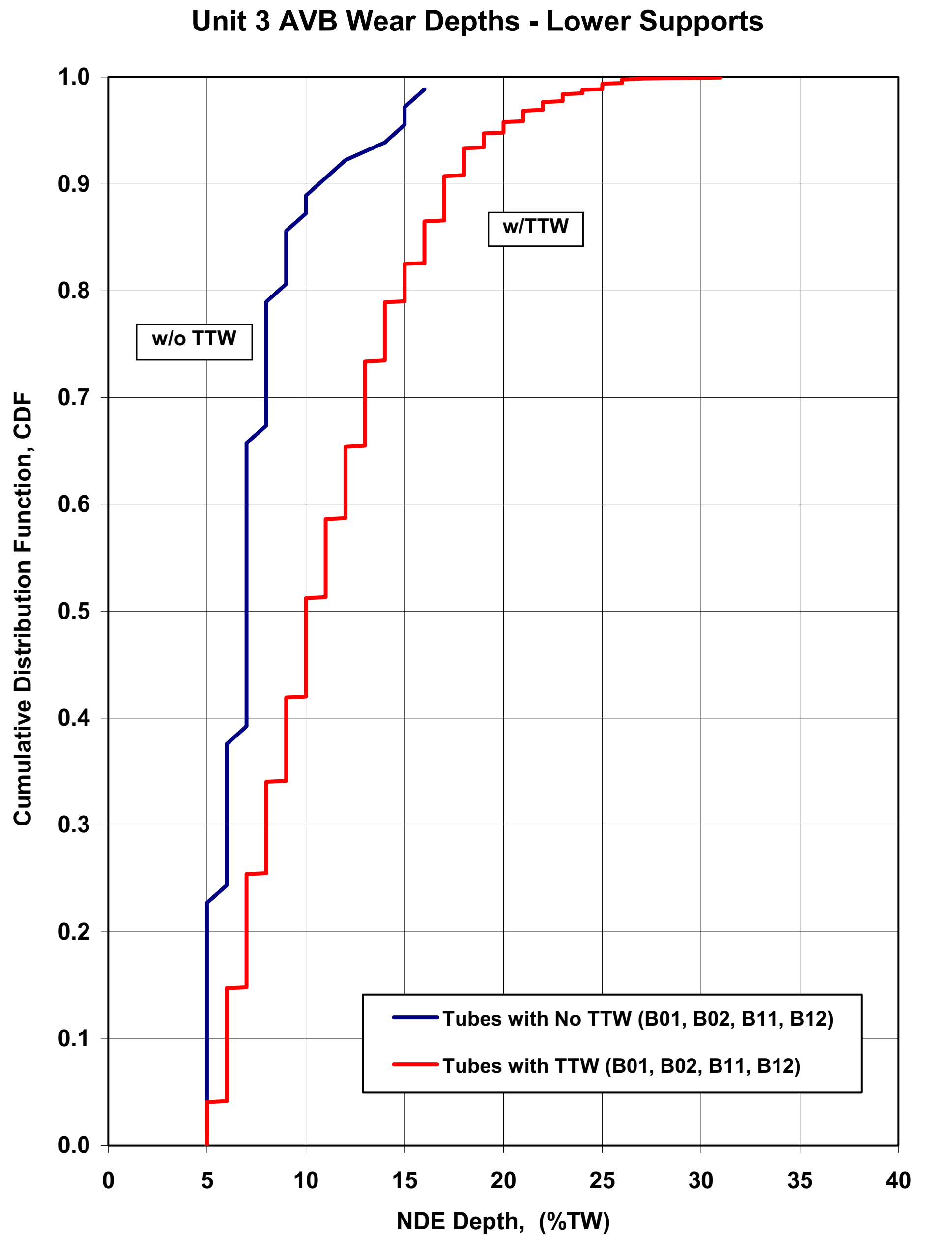


Figure 3 - Wear Depths for Tube/AVB Contacts at the Lower Supports after 0.926 Years at Power

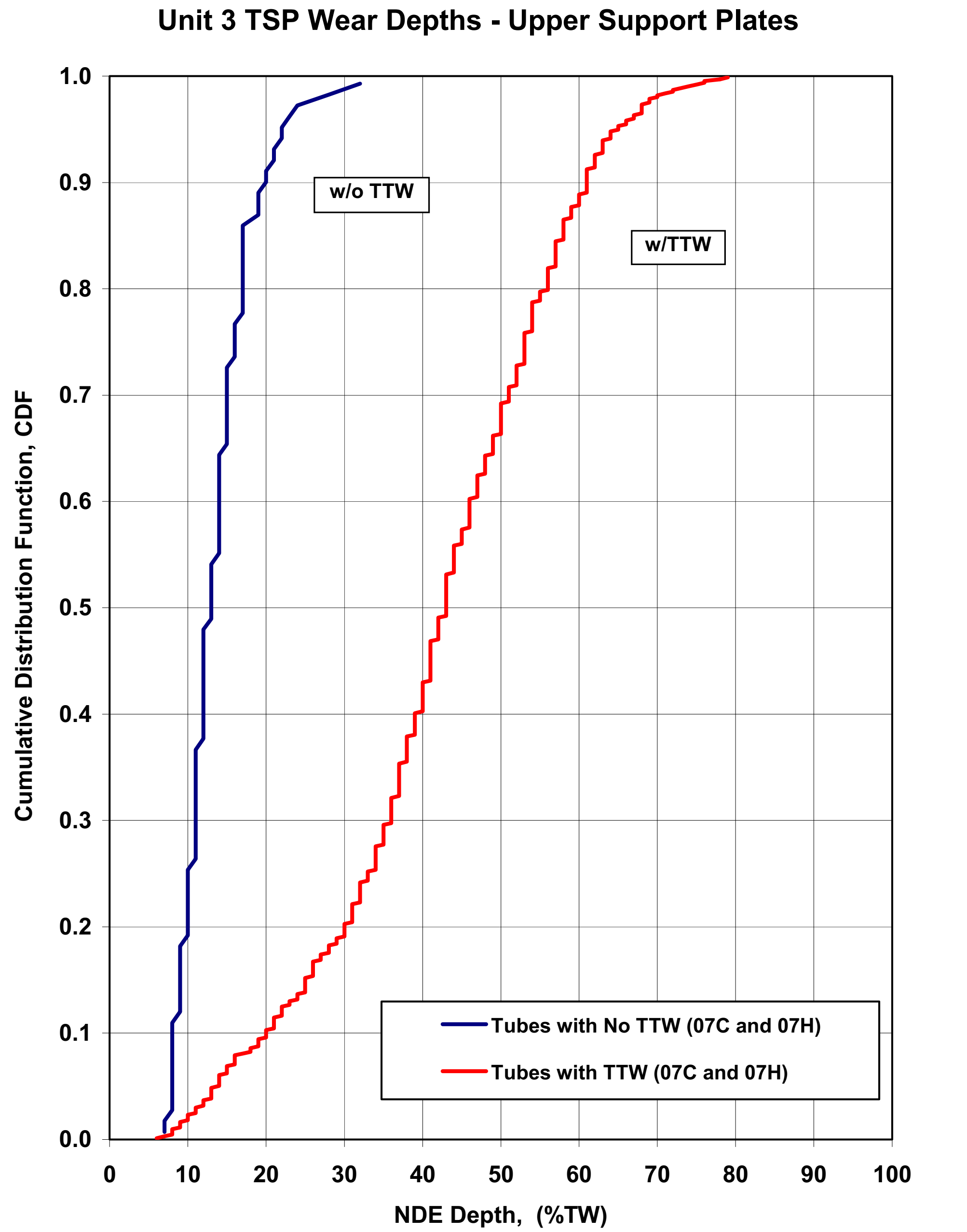
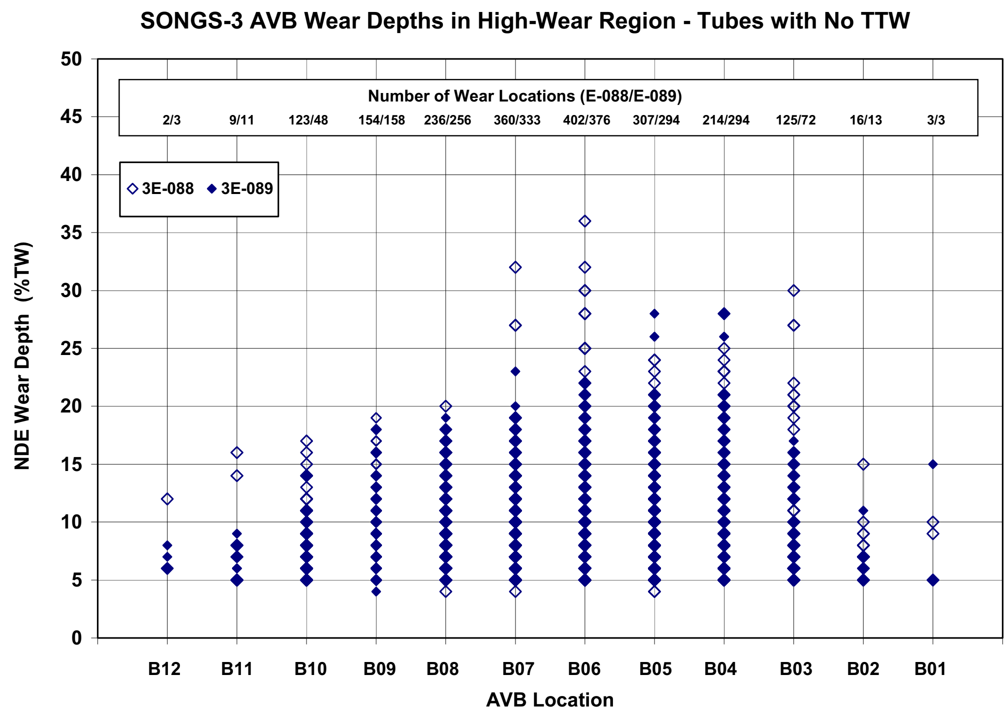


Figure 4 - Wear Depths for Tube/TSP Contacts at 07C and 07H Elevation after 0.926 Years at Power



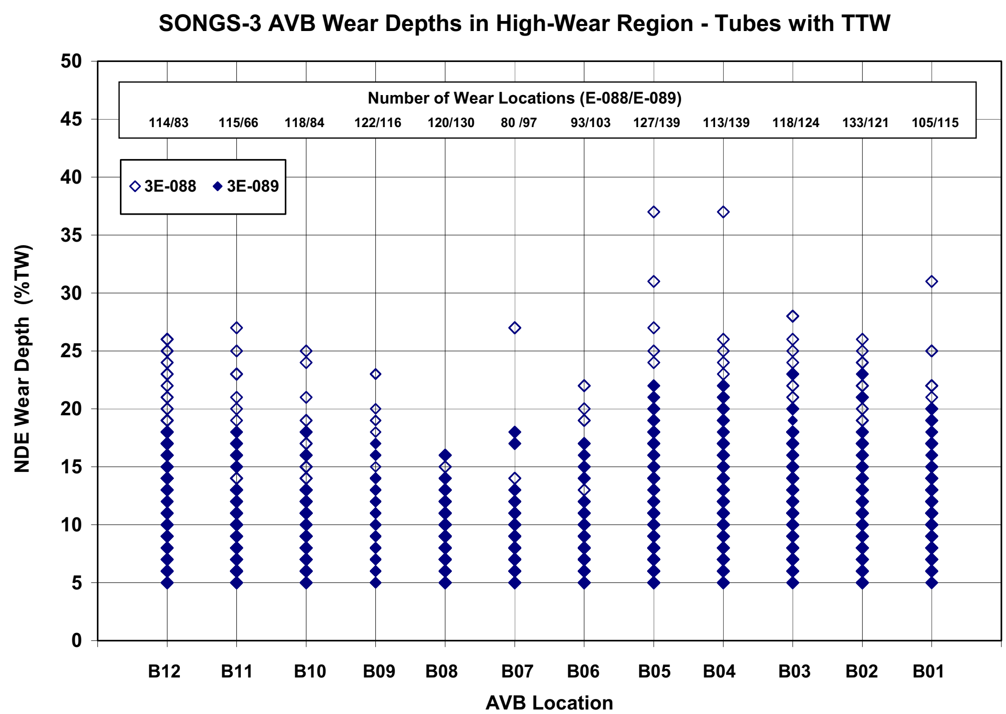
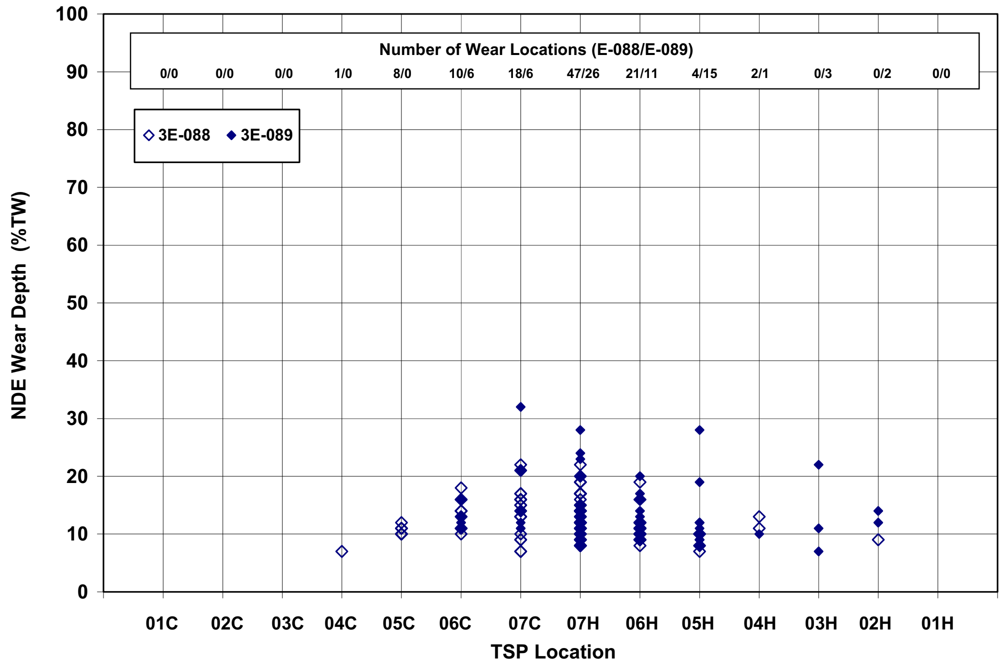


Figure 5 - Unit 3 AVB Tube Supports for Tubes with and without TTW

SONGS-3 TSP Wear Depths in High-Wear Region - Tubes with No TTW



SONGS-3 TSP Wear Depths in High-Wear Region - Tubes with TTW

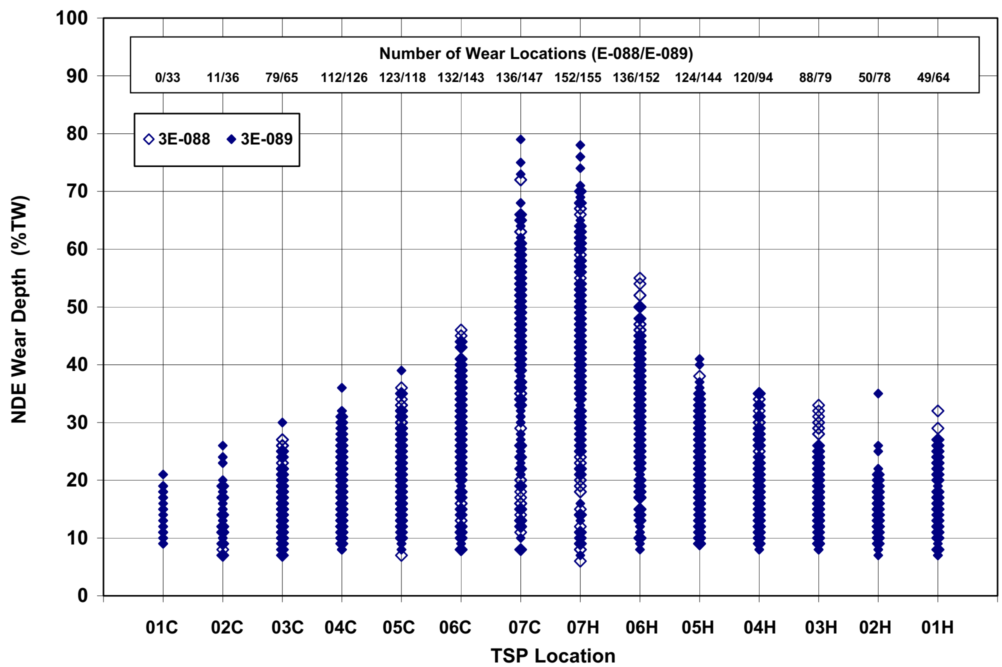


Figure 6 - Unit 3 TSP Tube Supports for Tubes with and without TTW

RAI 69

Reference 1, Response to RAI 2 – There is a statement in the middle of page 3 of 18, "The initiation-time model for TTW uses a wear index based only on tube to AVB wear in the upper supports (B03 through B10)." It is further stated on the same page (in item 3), "Thus the increase in AVB wear index after TTW is mainly due to the increase in locations with AVB wear, including wear locations at the lower supports (B01, B02, B11, and B12)." The second sentence appears inconsistent with the first (i.e., new wear index definition doesn't include wear from lower supports, yet wear at lower supports is causing wear index to increase). Please clarify the apparent inconsistency.

RESPONSE

Note: RAI Reference 1 is SCE's "Response to Request for Additional Information (RAIs 2, 3, and 4) Regarding Confirmatory Action Letter Response," dated February 25, 2013.

For the purpose of estimating the initiation time, it is shown below that the total wear index reduces to a wear index calculated from tube wear depths at the AVBs in the upper supports (B03 through B10). It was the intent of the first statement from RAI Reference 1 to make this point. The initiation-time analysis is a special case where only the wear in the upper supports is actually required to establish the distribution of initiation times for each tube in Unit 3 that had TTW. The technical basis for this special analysis case was discussed in the RAI 2 response. The tube support wear patterns that support the analysis conditions are provided in the response to RAI 68.

The second statement cited (in Item 3) at the bottom of Page 3 of RAI Reference 1 was provided in the context of the total wear index. The increase in the total wear index from the contribution of AVB wear after in-plane instability is the sum of three quantities:

a) continued tube wear at AVBs that were wearing prior to TTW initiation
b) additional locations with support wear at the upper AVB supports
c) additional support wear in the lower AVB supports (B01, B02, B11, and B12)

To eliminate the apparent inconsistency, the response to RAI 2 has been revised to remove the phrase "including wear locations at the lower supports (B01, B02, B11, and B12)" from Item 3. The revised RAI 2 response is enclosed.

The initiation-time model assumes that the AVB wear at the lower supports and the wear at the TSPs are tube degradation modes that primarily occur after in-plane tube instability. This assumption is based on comparisons of the tube support wear patterns in the high-wear region for Unit 3 tubes with and without TTW. The response to RAI 68 provides plots of the wear patterns that show this support wear behavior and forms the technical basis of the variable initiation-time model assumptions.

To provide additional explanation on the wear index and how it was developed for this analysis, the following derivation from the total wear index definition is provided. The total wear index at the end of the operating period (EOP) is defined as

$$WI_{EOP} = WI_{PRE} + WI_{POST} \tag{1}$$

where WI_{EOP} is the total wear index at EOP, and WI_{PRE} and WI_{POST} are the contributions to the total wear index during the operating period from the time up to TTW initiation and the time after TTW initiation. From the assumptions made based on the observation of support wear patterns, the WI_{PRE} is due to wear at AVB upper supports while the WI_{POST} is from the wear at AVB upper supports (US), lower supports (LS), and wear at TSPs. Therefore, for WI_{PRE},

$$WI_{PRE} = WI^{Pre}_{AVB-US}$$

The above equation is valid because of the relatively minor contributions to the wear index from the AVBs at the lower supports and at TSPs in the non-TTW population, so that WI^{Pre}_{AVB-LS}, and WI^{Pre}_{TSP} can be neglected in the analysis.

The post-initiation total wear index is the sum of the wear indices from AVB upper and lower supports and the TSP supports,

$$WI_{POST} = WI^{Post}_{AVB-US} + WI^{Post}_{AVB-LS} + WI^{Post}_{TSP}$$

Summing all contributions to the total wear index gives,

$$WI_{EOP} = WI^{Pre}_{AVB-US} + WI^{Post}_{AVB-US} + WI^{Post}_{AVB-LS} + WI^{Post}_{TSP} \tag{2}$$

Since the increase in the wear index due to tube wear at the AVB lower supports and TSPs is assumed to be post-initiation occurrences based on the observations of tubes with and without TTW, Eq. 2 can be re-written in terms of the wear index at the AVB upper supports only. For the purpose of establishing the initiation time, given the pre-and post-initiation observations, the total wear index is simplified to become,

$$WI_{EOP} - WI^{Post}_{AVB-LS} - WI^{Post}_{TSP} = WI^{Pre}_{AVB-US} + WI^{Post}_{AVB-US} = WI^{EOP}_{AVB-US}$$

The quantities WI_{EOP}, WI^{Post}_{AVB-LS}, and WI^{Post}_{TSP} are known for each tube with TTW since as discussed previously, WI^{Pre}_{AVB-LS}, and WI^{Pre}_{TSP} are negligible based on the observed wear patterns.

Therefore, for the prediction of TTW initiation time, the AVB wear index in the upper supports provides the governing empirical equation:

$$WI^{EOP}_{AVB-US} = WI^{Pre}_{AVB-US} + WI^{Post}_{AVB-US} \tag{3}$$

Figure 1, provided in RAI 70's response, is a graphical illustration of Eq. 3.

RAI 70

Reference 1, Response to RAI 2 – It is unclear to the staff how TTW initiation times were calculated (see description on pages 4 and 5 (of 18)). Describe each individual step, in sequential order, to calculating TTW initiation time for a given tube, for a given trial. Provide (or reference) in figure form all distributions that were sampled.

RESPONSE

Note: RAI Reference 1 is SONGS "Response to Request for Additional Information (RAIs 2, 3, and 4) Regarding Confirmatory Action Letter Response," February 25, 2013.

For the purpose of describing the simulation process, Figure 1 is provided to illustrate the increase in tube-AVB wear in the upper supports before and after tube instability in terms of the wear index. The analysis variables are also illustrated to aid in the description of the probabilistic solution for TTW initiation time. The important points on the wear index lines are labeled A through E.

Governing Equations

The equation that defines the initiation time is

$$t_{INIT} = \frac{WI_{AVB-US}^{EOP} - WIR_2\ t_{CYC}}{WIR_1 - WIR_2} \qquad (1)$$

where EOP is the end of the operating period for Unit 3, and

- t_{INIT} is the initiation time (Figure 1 Point B)
- WI_{AVB-US}^{EOP} is the wear index at EOP from AVB wear in the upper supports (Figure 1 Point D)
- WIR_1 is the growth rate in the WI prior to initiation (slope of Line AB in Figure 1)
- WIR_2 is the growth rate in the WI after initiation (slope of Line BD in Figure 1)
- t_{CYC} is the length of the operating period (0.926 years at power)

Equation 1 is the solution for the intersection of the two lines (AC and ED) and defines the time in the operating period when the change in the upper support wear index slope due to in-plane instability occurs. Point B in Figure 1 represents the calculated initiation time as determined by the model.

The growth in the wear indices in the AVB upper supports before in-plane instability (WIR_1) and after in-plane instability (WIR_2) are calculated from the following:

$$WIR_1 = \sum_{i=1}^{N_1} [WR_{AVB-US}^{Pre}]_i \quad (2)$$

$$WIR_2 = \sum_{i=1}^{N_2} [WR_{AVB-US}^{Post}]_i \quad (3)$$

where WR_{AVB-US}^{Pre} and WR_{AVB-US}^{Post} are the cumulative distribution functions (CDF) for the AVB wear rates derived from wear depth data for tubes without TTW and with TTW. The variable N_1 is the number of AVB upper-support locations in the tube before in-plane instability. The parameter N_2 is the number of AVB upper-support locations in the tube after in-plane instability.

Overall Simulation Process

A Monte Carlo simulation was performed to solve for t_{INIT} using Eq. 1. From this simulation process, a distribution of initiation times for each tube with TTW was developed. Each steam generator (SG) was evaluated separately. After completion of 1000 trials for each tube with TTW, a distribution of initiation times was saved in an array. The data array was used to determine the median initiation time for each tube using the calculated initiation times, ranked from shortest to longest (order-statistics).

Analysis Input

For tubes with TTW in Unit 3, the following input data are known on an individual tube basis and used in the simulation deterministically:

- Total number of AVB upper support locations having detected wear after 0.926 years at power (N_2).
- Wear index due to AVB wear in upper supports after 0.926 years at power, WI_{AVB-US}^{EOP} (Figure 1 Point D).

The variables that are treated statistically:

- Number of AVB upper support locations before in-plane instability (Figure 2)
- AVB wear rates before and after in-plane instability (Figure 3)

Procedure to Calculate TTW Initiation Time

For each trial in the simulation, the following steps were followed on a tube-by-tube basis:

1.0 Determination of N_1

1.1 At the start of a trial, the number of AVBs is selected by a random pick from a Poisson distribution developed from the number of affected AVBs observed in tubes without TTW. The data histogram and corresponding CDF are as shown in Figure 2. Both Unit 3 steam generators have similar distributions and these data were combined to obtain the mean parameter for the Poisson distribution.

1.2 If the number of AVB locations resulting from the random pick for N_1 is equal to zero, or if N_1 exceeds the number detected at EOP (N_2), the cumulative distribution in Figure 2 is re-sampled until $0 < N_1 \leq N_2$.

2.0 Determination of WIR_1

2.1 The wear index growth rate before instability is calculated by summing the individual AVB wear rates using Eq. 2. The AVB wear rates are determined by N_1 random selections from the AVB wear rate distribution in Figure 3 (pre-initiation curve).

2.2 If the slope of the Line AC exceeds the slope of Line AD, the AVB wear rates are re-sampled and a new value of WIR_1 is calculated from Eq. 2. The intercept point (Figure 1 Point A) is always zero. This defines the equation of Line AC.

3.0 Determination of WIR_2

3.1 The wear index growth rate after instability is calculated by summing the individual AVB wear rates using Eq. 3. The AVB wear rates are determined by N_2 random selections from the AVB wear rate distribution in Figure 3 (post-initiation curve).

3.2 It is assumed that after in-plane instability is initiated, there is a re-distribution in wear patterns and corresponding wear rates at the AVBs. Wear rates were assigned for N_2 locations (i.e., resetting N_1 AVB wear rate values as well as selecting wear rates for the new locations). This represents a physical re-distribution of support contact/interaction with the tube. The AVB wear rates are determined by N_2 random selections from the AVB wear rate distribution in Figure 3 (post-initiation curve).

3.3 The intercept point (Figure 1 Point E) is calculated from the expression $WI^{EOP}_{AVB-US} - WIR_2\ t_{CYC}$.

3.4 Point E is evaluated in each trial to confirm it is ≤ 0 on the wear index axis. This is to maintain a slope for Line ED that is always greater than or equal to the slope for Line AD. If Point E > 0, the AVB growth rates are re-sampled and WIR_2 is recalculated from Eq. 3 to satisfy this constraint. This defines the equation for Line ED, having a slope $WIR_2 \geq WIR_1$ and intercept ≤ 0.

4.0 Calculation of t_{INIT}

4.1 The point of intersection (Figure 1 Point B) is calculated from Eq. 1 and the time of initiation determined.

4.2 The above steps are repeated for 1000 trials. The 1000 initiation times (t_{INIT}) are ranked in ascending order and the median value (500th in ranking) is recorded.

The above procedure is repeated for each tube until all tubes in the high-wear region have been evaluated and median initiation times for each tube established.

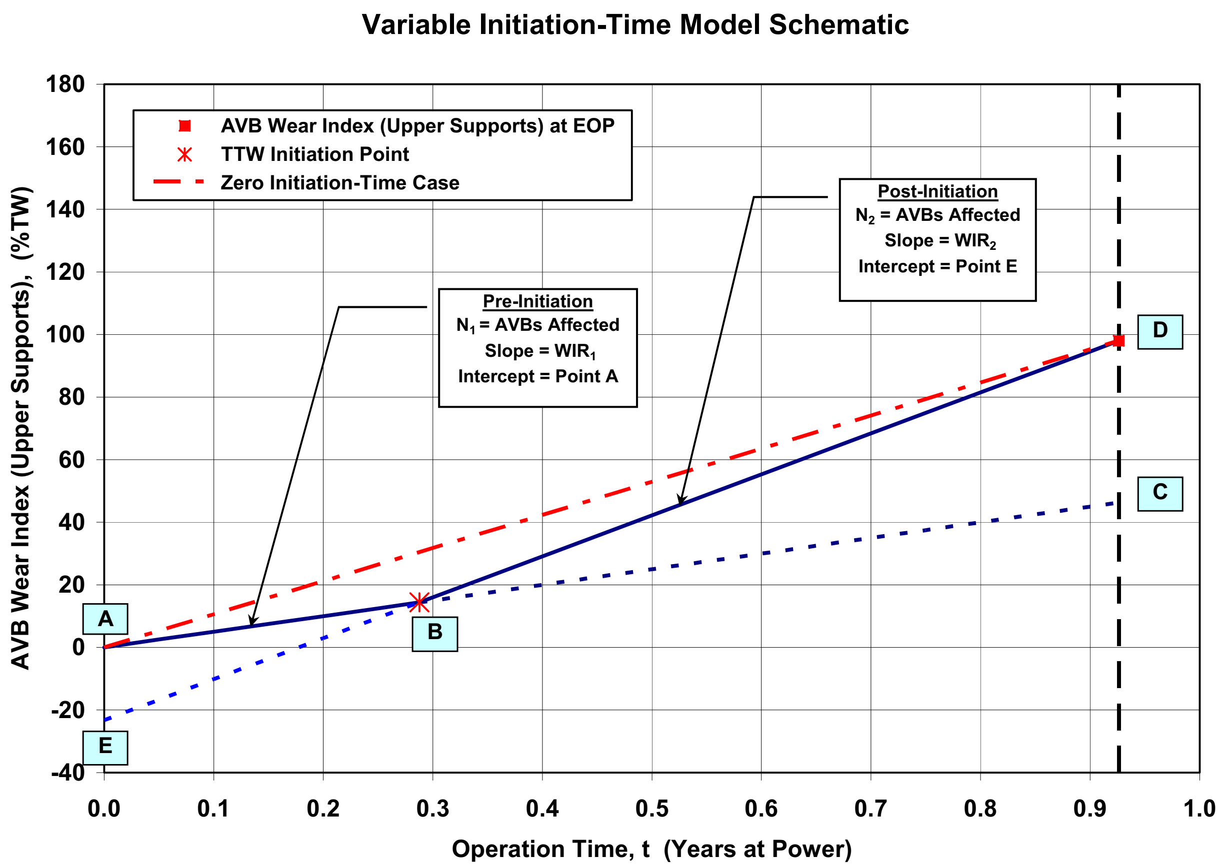


Figure 1 - Linear TTW Initiation-Time Model Variables

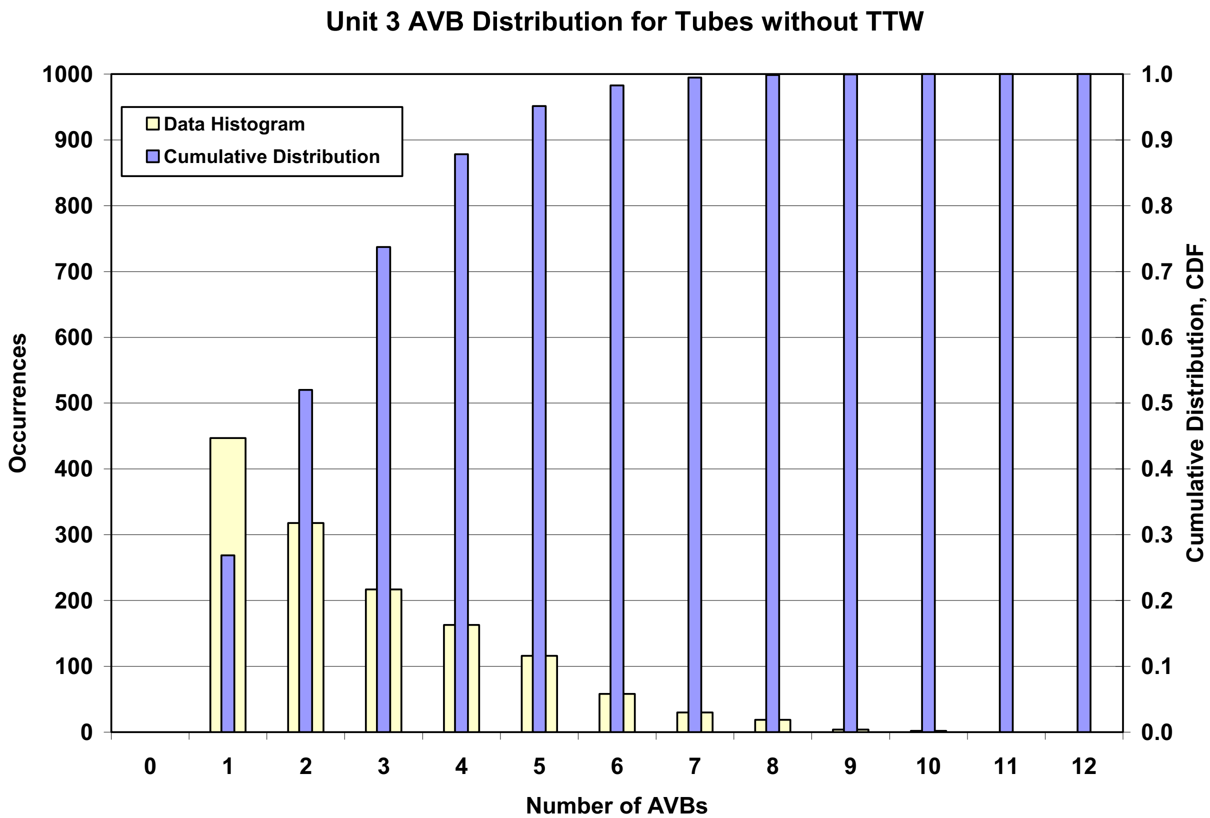


Figure 2 - Distribution of Affected AVBs in Unit 3 for Determining N_1

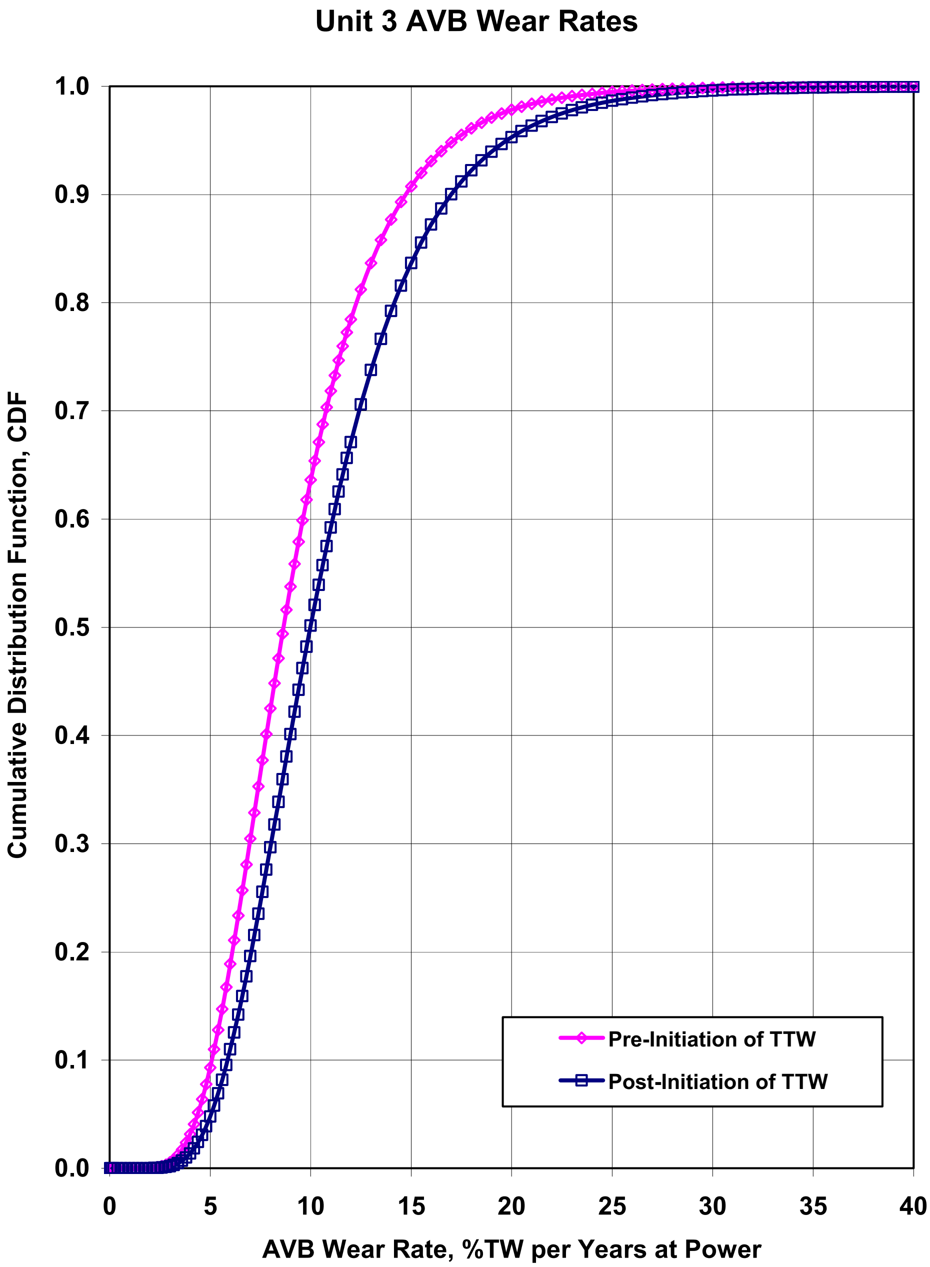


Figure 3 - Log-Normal AVB Upper Support Wear Rate Distributions for Unit 3

RAI 2

The Operational Assessment in Attachment 6, Appendix C (Reference 4), pages 3-2 and 4-12, appears to state that tube-to-tube wear (TTW) growth rates are based on the maximum TTW depths observed in Unit 3 at EOC 16 divided by the first Unit 3 operating period (0.926 years at power). Provide justification for the conservatism of this assumption. This justification should address the following:

a. Reference 4, page 3-2 defines “wear index” for a degraded tube and states that the existence of TTW and distribution of TTW depths are strongly correlated to the wear index. This is pictured in Figures 4-4 in terms of TTW initiation. This figure shows that TTW is not expected to have initiated until a threshold value of wear index is reached. This threshold value varies from tube to tube according to a cumulative probability distribution shown in the figure. This figure illustrates that TTW is not expected to have initiated until sometime after BOC 16. This suggests that the observed TTW depth at EOC 16 developed over a smaller time interval than the 0.926 years assumed in the analysis.

b. An independent analysis in Reference 3 also indicates an extremely low probability of instability onset at BOC 16 as illustrated in Figure 8-3. Reference 3, page 106 interprets this figure as indicating that the probability of instability only reaches 0.22 after 3 months and only becoming “high” after 4 months.

c. Reference 3 also considered a variety of different wear rate models to estimate how long it took to develop the observed TTW depths at Unit 3 after instability occurred. These analyses are documented in Appendix A of Reference 3 and produced estimates in the range of 2.5 to 11 months.

RESPONSE - Revision 1

Note: RAI Reference 4 is the “Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16,” prepared by Intertek APTECH for Areva, Report No. AES 12068150-2Q-1, Revision 0, September 2012.

RAI Reference 3 is the “SONGS U2C17 Steam Generator Operational Assessment for Tube to Tube Wear,” prepared by Areva NP Inc, Document number 51-9187230-000, Revision 0, October 2012.

Basis for Zero Time Initiation Model

The wear rate model in the SONGS Unit 2 operational assessment (OA) is based on the apparent tube-to-tube wear (TTW) growth rates observed in SONGS Unit 3. Industry guidelines define the apparent degradation rate as the change in observed degradation over the operating cycle length. Following standard industry practice, the wear rate model assumes that TTW began at the start of Cycle 16 for Unit 3. The wear rate distribution for SONGS Unit 2 was developed from the distribution of Unit 3 depths divided by the Cycle 16 operation period of 0.926 years at power.

It was recognized that a variation in the time when TTW initiated would affect the wear rate distribution. The OA model for TTW includes conservative assumptions to compensate for the variation in the time when TTW may have begun. These conservative modeling conditions are:

1) Only the maximum TTW depths were used in the determination of wear rate. This produces a faster growth model than one developed from all of the depth data.
2) The TTW rates are based on growth in through-wall depth (i.e., constant-depth growth). Wear processes are typically modeled as constant volume growth which leads to a decreasing depth growth in time.
3) The TTW growth rates are based on 100% power operation for Unit 3. No credit is taken for the potential reduction in TTW rate for 70% power operation.
4) The tube-AVB and tube-TSP growth rates are based on 100% power operation for Unit 3. No credit is taken for the potential reduction in wear rates at AVB and TSP supports for 70% power operation.
5) The TTW rate model is based on the wear index (WI) as an indicator of initiation and growth of TTW. The wear index is based on depth growth at the supports (AVB and TSP) which conservatively assumes constant depth growth.

By including these assumptions, the Unit 2 TTW growth rate model for 70% power operation was determined to be reasonable and conservative.

Treatment of Variable Initiation Time

To address the RAI question of TTW initiation time and the impact on the allowable inspection interval for Unit 2, additional analysis was performed. A time dependent predictive model (initiation-time model) for estimating the operating period prior to TTW initiation in Unit 3 was developed. This model is illustrated in Figure 1. This figure is a schematic example for a single tube where the rate of increase in the wear index is allowed to vary.

As shown in Figure 1, the growth in the wear index occurs in two stages: pre-initiation and post initiation. The initiation-time model for TTW uses a wear index based only on tube to AVB wear in the upper supports (B03 through B10). This model's wear index (WI) starts to increase at the beginning of the cycle. The basis for this approach was developed from the Unit 3 wear data as follows:

1) Tubes without TTW indications have very few or no AVB wear indications at the lower support bars (B01, B02, B11, and B12).
2) Most tubes with TTW have wear indications in the lower support bars. These tubes typically have significant wear at all four lower support bars. The wear at the lower supports is assumed to occur after TTW has initiated.
3) AVB wear depths (B03 through B10) for tubes with TTW are similar to the wear depths for tubes without TTW. This indicates that the wear rates at individual tube/AVB contact points before TTW initiates is not significantly elevated after TTW has initiated. Thus the increase in AVB wear index after TTW is mainly due to the increase in locations with AVB wear~~, including wear locations at the lower supports (B01, B02, B11 and B12)~~.

4) For tubes without TTW, few or no TSP wear indications were detected. Most TSP wear occurred in tubes with TTW.

Using the above Unit 3 NDE results, the initiation-time model assumes the development of wear at the lower AVB supports (B01, B02, B11, and B12) and TSPs occurs after TTW initiates. The TTW initiation time was determined from the change in the wear index in the upper tube supports (B03 through B10). The initiation-time model uses AVB wear in these upper supports as a predictor of TTW since the wear at the other supports is assumed to develop after TTW initiation.

The WI growth rate prior to initiation of TTW was developed from upper AVB support wear data (B03 through B10) for Unit 3 tubes without TTW. The increase in WI after TTW initiation uses AVB wear rate distributions sampled from the number of detected wear locations for tubes with TTW.

The predicted TTW initiation time is calculated by simulating two-stage growth in the WI. The initiation model is constrained to the end state for the calculated wear index from the NDE results. The slopes for the growth in the WI are also constrained to keep the two-stage process bounded by the constant WI growth case illustrated in Figure 1.

A Monte Carlo simulation was performed to compute the distribution of initiation times for each tube. After completion of 1000 trials for each tube, a distribution of initiation times was created and the median value recorded. The median value is used to provide the best-estimate time when TTW began in Cycle 16. The histogram of these median values is shown in Figure 2. By this analysis, a significant number of tubes initiated TTW early in the cycle, especially for SG 3E-088. The TTW growth rates were computed from these data using the maximum TTW depth divided by the cycle length minus the initiation time.

The zero-time initiation model in RAI Reference 4, the independently developed analysis in RAI Reference 3, and the initiation-time model developed in response to this RAI reach conclusions on either the onset of instability or the initiation of TTW in the next operating interval for Unit 2. The approaches used in these models differ in their use of the empirical information provided by the NDE inspection data and the use of steam generator thermal hydraulic properties and support conditions from Units 2 and 3. These differences lead to some variation in predicting the onset of instability or the initiation of TTW. SCE's conclusion is all three approaches provide diverse and comprehensive evaluations supporting the proposed operating interval.

Comparison of TTW Rate Behavior

The wear rate plots as a function of total wear index are shown in Figure 3 for both depth sizing techniques used in RAI Reference 4. The dotted line in each plot is the TTW rate model from Figure 4-13 in RAI Reference 4. This model assumed zero initiation time in establishing the wear rate. The solid line is the regression analysis of the results from the initiation-time model developed for this RAI. This comparison shows the TTW growth function based on the initiation-time model is more conservative than the model used in RAI Reference 4, without adjusting for the other conservative assumptions made in the development of the RAI Reference 4 model.

Significance of Initiation Time

The effect of the initiation-time model developed for this RAI on operating interval is shown in the table below. The allowable operating interval for maintaining structural integrity performance criteria margins for Unit 2 is 1.02 to 1.15 years at 70% power without changing any of the other conservative assumptions incorporated in the RAI Reference 4 model.

Unit 2 Years of Operation at 70% Power		
Case	**Reference 4 - Zero Initiation Time TTW Model**	**RAI TTW Initiation-Time Model**
ETSS Sized	1.33	1.02
AREVA Resized	1.48	1.15

These results represent a margin of at least 2.4 on the planned Unit 2 inspection interval of 150 days (0.42 years at 70% power).

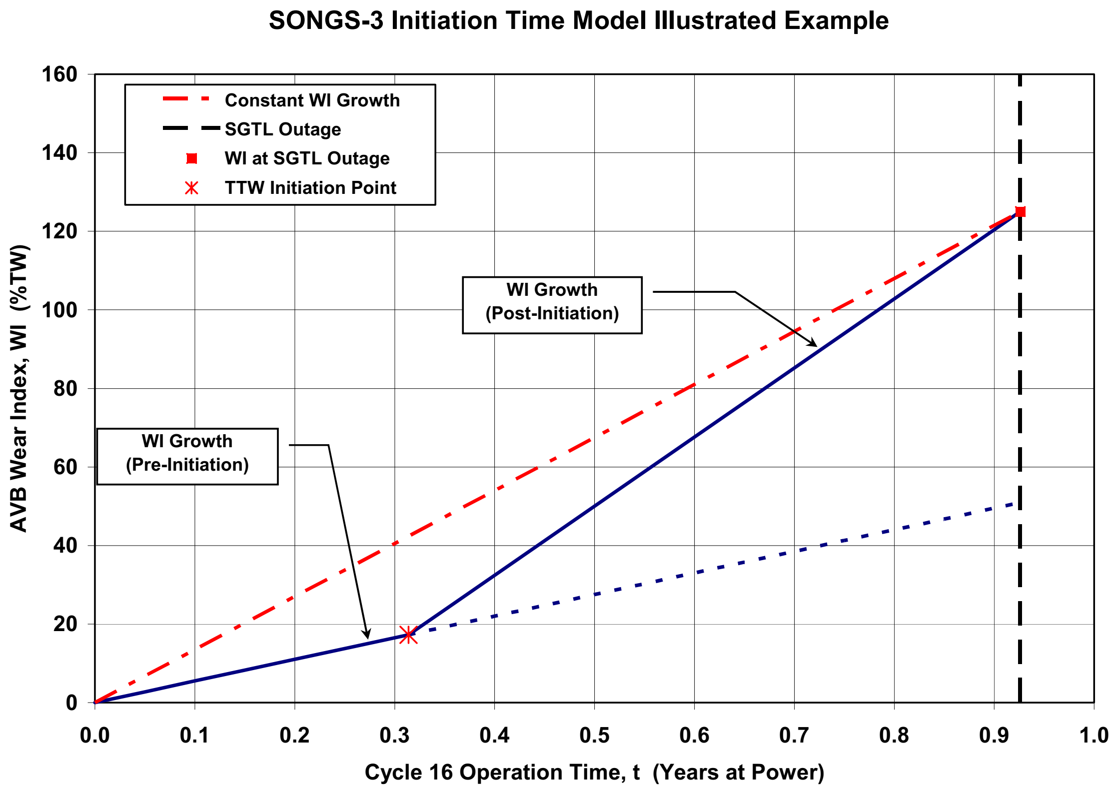


Figure 1 – Model for Estimating TTW Initiation Time

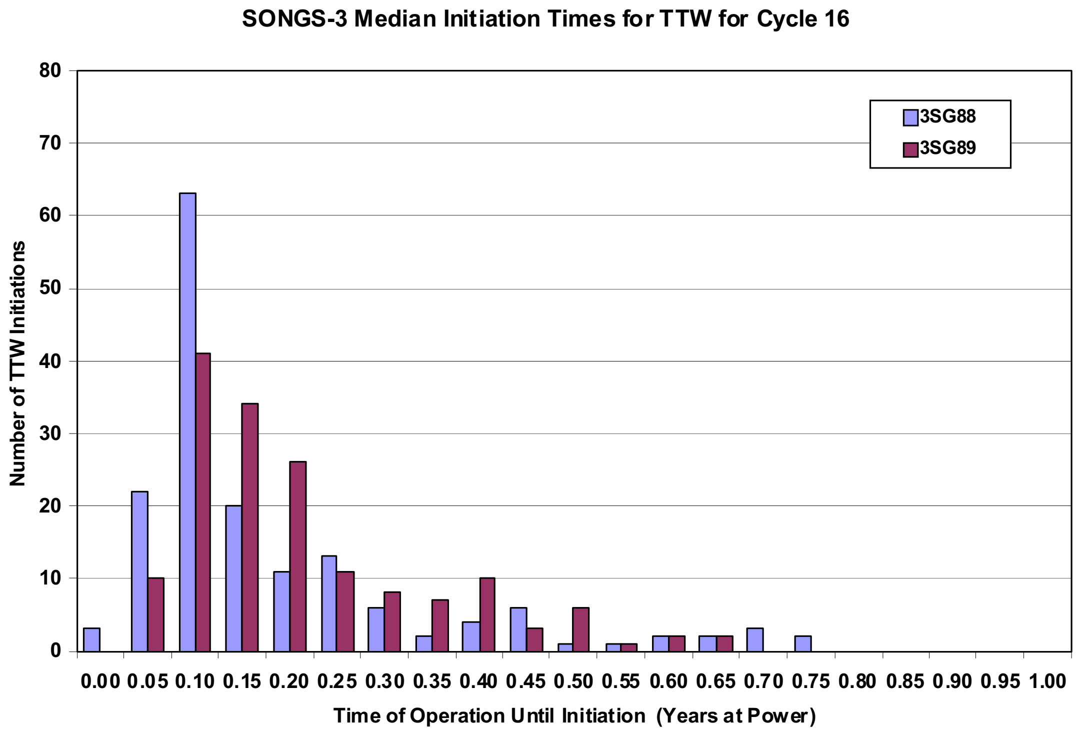
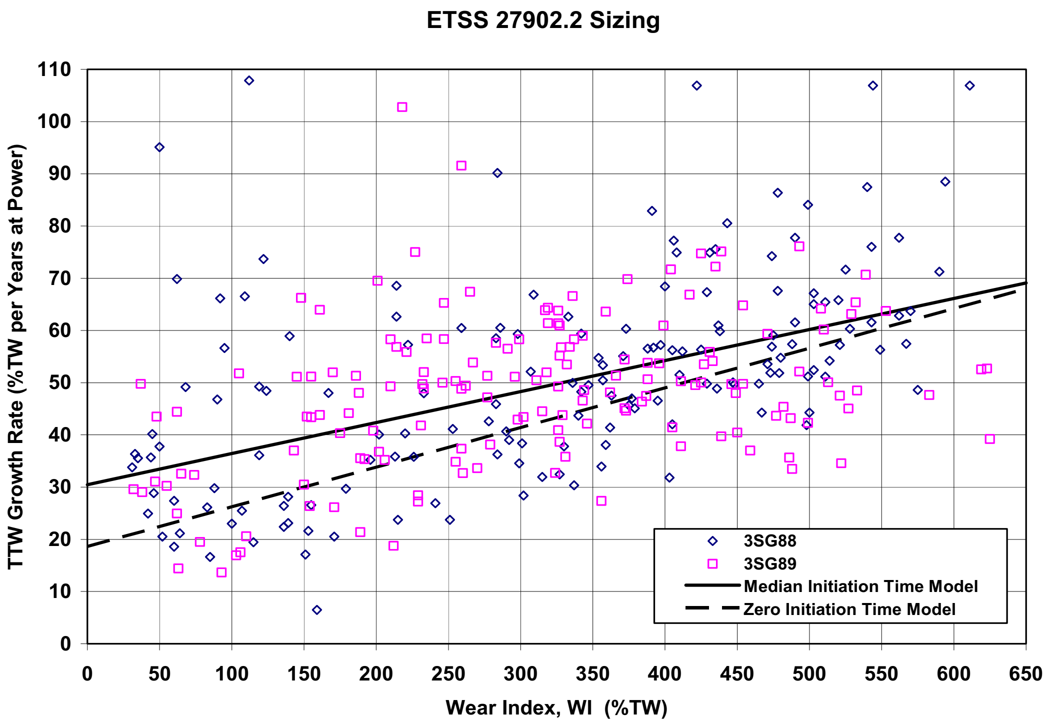


Figure 2 – Simulation Results from Time Dependent Initiation Model



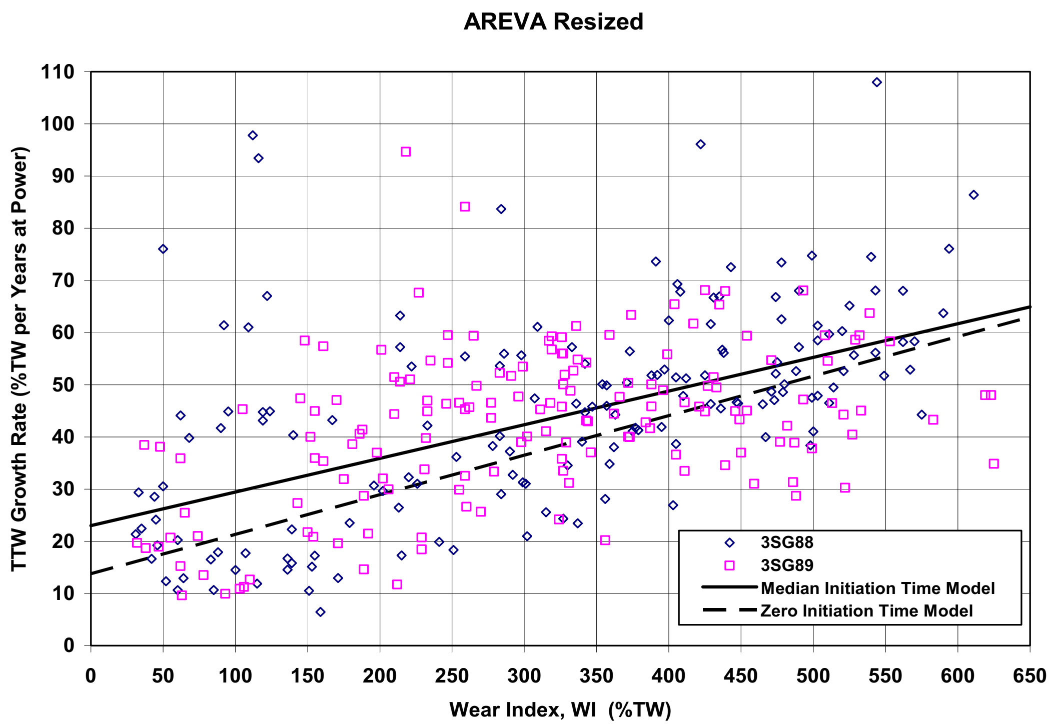


Figure 3 – Tube-to-Tube Wear Rate Comparisons