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Morgan Lewis
COUNSELORS AT LAW

Stephen J. Burdick
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January 28, 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: Notification of Responses to RAIs

Dear Licensing Board Members:

On December 26, 2012, the Nuclear Regulatory Commission (NRC) issued Requests for Additional Information (RAIs) to Southern California Edison Company (SCE) 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 that SCE has begun to respond to the RAIs. The enclosed letters provide SCE's responses to RAIs 1, 5, 7, 9, 10, 11, 12, 13, 15, 16, 17, 18, 19, 27, 28, and 30. SCE submitted proprietary and non-proprietary versions of RAIs 1, 10, 15, 17, and 30 to the NRC. 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.

Morgan Lewis
COUNSELORS AT LAW

Respectfully submitted,

Signed (electronically) by Stephen J. Burdick
Stephen J. Burdick

Counsel for Southern California Edison Company

Enclosures

1. SCE Response to RAI 1 (Jan. 8, 2013) (non-proprietary version)
2. SCE Response to RAIs 5, 7, 9 (Jan. 25, 2013)
3. SCE Response to RAIs 10, 17 (Jan. 17, 2013) (non-proprietary version)
4. SCE Response to RAI 11 (Jan. 21, 2013)
5. SCE Response to RAI 12 (Jan. 18, 2013)
6. SCE Response to RAI 13 (Jan. 18, 2013)
7. SCE Response to RAI 15 (Jan. 9, 2013) (non-proprietary version)
8. SCE Response to RAI 16 (Jan. 10, 2013)
9. SCE Response to RAI 18 (Jan. 24, 2013)
10. SCE Response to RAI 19 (Jan. 16, 2013)
11. SCE Response to RAI 27 (Jan. 25, 2013)
12. SCE Response to RAI 28 (Jan. 21, 2013)
13. SCE Response to RAI 30 (Jan. 9, 2013) (non-proprietary version)

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, Units 2 and 3)	)	January 28, 2013

CERTIFICATE OF SERVICE

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

Signed (electronically) by Stephen J. Burdick

Stephen J. Burdick
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Avenue, N.W.
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Counsel for Southern California Edison Company

BOARD NOTIFICATION ENCLOSURE 1

SOUTHERN CALIFORNIA
EDISON®
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Proprietary Information
Withhold from Public Disclosure

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

January 8, 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 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

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. Enclosure 2 of this letter provides the response to RAI 1.

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 Mitsubishi Heavy Industries (MHI) and AREVA, which set forth the basis on which the information in Enclosure 2 may be withheld from public

P.O. Box 128
San Clemente, CA 92672

IE36
NRR

disclosure by the NRC and address with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Proprietary information identified in Enclosure 2 was extracted from MHI Report L5-04GA567, Evaluation of Stability Ratio for Return to Service and AREVA report 51-9187230, SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear, which are addressed in the affidavits. 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 Affidavits
2. Response to RAI 1 (Proprietary)
3. Response to RAI 1 (Non-proprietary)

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

ENCLOSURE 1

Notarized Affidavits

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 MHI technical documentation to determine whether it contains 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 determined that the following MHI document contains MHI proprietary information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4). Those pages of the document containing proprietary information have been bracketed with an open and closed bracket as shown here "[ ]" / and should be withheld from public disclosure.

   MHI document
   Document: L5-04GA567

3. The information identified as proprietary in the enclosed document has 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 it describes 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 14 day of September, 2012.

Jinichi Miyaguchi

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

Sworn to and subscribed

Before me this 14 day

of September, 2012

277
SEP. 14, 2012

Masahiko Kubota

Notary Public

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

My Commission Expires ________________

登簿平成２４年第 277 号

認　　　証

嘱託人　三菱重工業株式会社　原子力事業部　原子力誠三総括部　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。

よって認証する。

平成２４年９月１４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 277 Date SEP. 14. 2012

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)

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Gayle F. Elliott. 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 Southern California Edison document entitled "SONGS U2C17 Steam Generator Operational Assessment," dated October 2012, 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(b) and 6(c) 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.

SUBSCRIBED before me this 2nd
day of October 2012.

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2015
Reg. #110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2015

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 1

(NON-PROPRIETARY)

RAI 1

The Operational Assessment (OA) in Attachment 6, Appendix A (Reference 2), reports the 3 times normal operating pressure differential as being 4290 psi for 100% power conditions. This is the same value assumed in the Condition Monitoring Assessment provided in Attachment 2. This value is significantly higher than the values ranging from 3972-3975 psi for 100% power reported in Attachment 6, Appendices B, C, and D (References 3-5). Describe the reason for the differences.

RESPONSE

Table 1 summarizes the bases for the values of 3 times normal operating pressure differential (3xNOPD) used in each operational assessment (OA). Steam generator (SG) secondary side pressure, and consequently the 3xNOPD, is influenced by power, T_{COLD} (see discussion at end of this response), and SG tube plugging. Each of the OAs assumes a different set of relevant conditions in their respective analysis. The reference numbers in the discussion below correspond to the RAI reference numbers.

Reference 2 (AREVA document number 51-9182833-002, *SONGS U2C17 Outage – Steam Generator Operational Assessment*) assesses tube integrity due to all wear mechanisms, with the exception of tube-to-tube wear (TTW). The OA analysis uses a 3xNOPD equal to 4290 psid, which was selected to bound the potential operating conditions for the next operating interval including possible changes due to tube plugging and/or a decrease in power level. As shown in Table 1, the NOPD is based on the operating pressures in the SGs during the previous operating cycle, which included operation at 100% power, increased $T_{COLD,}$ and no tube plugging (same conditions used in *SONGS 2C17 Steam Generator Condition Monitoring Report*). The 0% plugging condition is bounding because as plugging increases, secondary side pressure increases reducing NOPD.

Reference 3 (AREVA document number 51-9187230-000, *SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear*) uses a conservative approach to demonstrate that TTW due to fluid elastic instability will be prevented while operating at 70% power. This OA utilizes probability of stability rather than 3xNOPD to determine SG inspection interval. However, the defense in depth section of this OA uses a 3xNOPD equal to [ ] psid, which is derived from the secondary side pressure in Table 6.2-1 "Basic parameters for calculation for 2A SG evaluation after plugging" of MHI document L5-04GA567 R6. It is based on operation at 70% power, increased T_{COLD}, and approximately 3% tube plugging. When power level is decreased, the secondary side pressure increases resulting in a reduction in differential pressure.

Reference 4 (APTECH OA document number AES 12068150-2Q-1, *Operational Assessment for SONGS Unit 2 Steam Generators for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16*) uses established industry methods to predict TTW degradation based on Unit 3 TTW data and a 70% power. The table in Section 4.2 of Reference 4 lists three values of NOPD. The NOPD of [ ] psid (3xNOPD = [ ] psid) was used for the OA analysis, which was the same value used in Reference 3 and described above. The other two NOPD values in the Section 4.2 table were not used in the OA analysis. The first value of NOPD equal to [ ] psid (3xNOPD = [ ] psid) is for Cycle 16 actual secondary side pressures at 100% power. The second value of NOPD equal to [ ] psid (3xNOPD = [ ] psid) was based on secondary side pressure of [ ] psia (rounded to [ ] psia, conservative) at 100% power, increased T_{COLD}, and approximately 3% tube plugging.

Reference 5 (Westinghouse OA document number SG-SGMP-12-10, Rev 3, *Operational Assessment of Wear Indications in the U-bend Region of San Onofre Nuclear Generating Station Unit 2 Replacement Steam Generators*) uses a 3xNOPD equal to [ ] psid. This OA uses the 100% power NOPD, which is derived from the secondary side pressure in Table 6.2-1 "Basic parameters for calculation for 2A SG evaluation after plugging" of MHI Document L5-04GA567 R4, and is based on operation at 100% power, increased T_{COLD}, and approximately 3% tube plugging.

At the end of Cycle 16, the Unit 2 original reactor vessel head (ORVH), comprised of Alloy 600, was replaced with a replacement reactor vessel head (RRVH), comprised of Alloy 690, due to Alloy 690's enhanced resistance to stress corrosion cracking (SCC) at elevated temperatures. Prior to the reactor vessel head replacement, T_{COLD} had been lowered to reduce Alloy 600's susceptibility to SCC. Increasing T_{COLD} increases secondary side pressure and decreases NOPD. With the installation of the RRVH, T_{COLD} will be increased to a nominal value of [ ] °F, reducing effective velocities and increasing margin to fluid elastic instability.

Table 1 Summary of 3xNOPD Differences in Operational Assessments

	RTSR Attachment 6, Appendix A (RAI Reference 2)	RTSR Attachment 6, Appendix B (RAI Reference 3)	RTSR Attachment 6, Appendix C (RAI Reference 4)	RTSR Attachment 6, Appendix D (RAI Reference 5)
Primary Side Pressure (psia)	2250	2250	2250	2250
Secondary Side Pressure (psia)	820	[ ]	[ ]	[ ]
NOPD (psid)	1430	[ ]	[ ]	[ ]
3xNOPD (psid)	4290	[ ]	[ ]	[ ]
Power Level (%)	100	70	70	100
Plugging (%)	0	≈ 3	≈ 3	≈ 3
T_{COLD} (°F)	541	[ ]	[ ]	[ ]
Summary of Operating Parameters Used and basis	NOPD is based on operating pressures in the SGs during the previous operating cycle at 100% power and reduced T_{COLD}. Parameters were selected to bound possible conditions for the next operating cycle at the time the document was issued. The 0% plugging condition is bounding because as plugging increases, secondary side pressure increases reducing NOPD.	NOPD based on 70% power at increased T_{COLD} with approximately 3% tube plugging.	NOPD based on 70% power at increased T_{COLD} with approximately 3% tube plugging.	NOPD based on 100% power at increased T_{COLD} with approximately 3% tube plugging.

BOARD NOTIFICATION ENCLOSURE 2

SOUTHERN CALIFORNIA
EDISON®
An EDISON INTERNATIONAL® Company

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

January 25, 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 5, 7, and 9)
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

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. Enclosure 1 of this letter provides the responses to RAIs 5, 7, and 9.

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,

Enclosure:

1. Response to RAIs 5, 7, and 9

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 RAIs 5, 7, and 9

RAI 5

Regarding Reference 4, third paragraph from the bottom of page 4-3, why is non-detected wear only assigned to no degradation detected (NDD) tubes and not to NDD tube/AVB intersections in tubes with detected wear at other intersections?

RESPONSE

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.

Both tube populations had non-detected (undetected) wear assigned but two different techniques were used in assigning undetected wear. This was done because of the distinct differences between the two groups. The tubes with no degradation detected (NDD) received 100% bobbin examination. The tubes with detected wear received a 100% bobbin examination followed by a special examination using the +Point^{TM} probe. Also, a very conservative method for assigning undetected wear locations in the population of NDD tubes was used in order to simplify the assessment of that large group of tubes.

The population of 1350 NDD tubes was assumed to have undetected wear to account for the possibility of having active wear at some tube/support locations in these tubes at the beginning of the next operating cycle. Depths for undetected wear at these active wear locations were determined from the probability of detection (POD) performance for the bobbin probe.

Tubes with detected wear at anti-vibration bar (AVB) intersections were assigned active wear locations to account for the initiation of new wear sometime during the next operating cycle. This topic is discussed in Section 4.4.2 of RAI Reference 4. It is conservatively assumed that the new active wear locations all exist at the start of the next operating.

Further discussion of the method by which new wear locations are added is given in the response to RAIs 7 and 8.

RAI 7

Regarding Reference 4, page 4-5, what is meant by the words, "each active wear location" in the 1350 NDD tubes? How are the "active wear" locations determined?

RESPONSE

The population of 1350 NDD tubes in SG 2E-089 may have undetected wear in some tubes. As a conservative treatment of this tube population, every tube is assumed to have some amount of undetected wear. The locations of possible undetected wear, either at AVB or at TSP intersections, are referred to as "active wear locations."

Active wear locations are assigned to each tube using the data from the Unit 2 tube population with detected AVB and/or TSP wear. The locations that are assumed to have undetected active wear are randomly assigned based on the cumulative distribution function (CDF) for a number of support locations with detected wear in SG 2E-089. The CDFs for assigning active wear locations were developed from the Unit 2 histograms discussed in RAI Reference 4 as shown in

Figure 3-3 for AVB supports and Figure 3-4 for TSP supports. The model algorithm assigns five active wear locations, on average, in each NDD tube (two minimum).

RAI 9

It is stated in Reference 4, at the top of page 4-9 that the simulation results of the bench marking process are shown in Figure 4-6. Provide additional detail on what Figure 4-6 is showing and how it relates to the benchmarking process. As part of this additional detail, explain the meaning of the ordinate label "number of observations" in the figure.

RESPONSE

The initiation model for TTW for SONGS Unit 3 was developed from Unit 3 data using a Beta distribution to represent the probability of the presence of TTW at a given tube wear index value. A similar model was then developed for Unit 2 by modifying the Unit 3 model. Both of these models are shown in Figure 4-4 of RAI Reference 4. The Unit 2 model was developed by benchmarking the model predictions for Unit 3 against the two detected TTW indications for Unit 2. Another benchmark condition required that the Unit 2 initiation model approaches the Unit 3 model behavior as the input wear index distribution approaches that of Unit 3.

The benchmarking process involved a probabilistic simulation to predict the number of Unit 2 TTW indications based on Unit 2 wear index values. The simulation results were compared to what was actually observed for Unit 2 (two detected TTW indications in SG 2E-089). Final benchmarking was achieved when the model produced two detected indications at the estimated threshold detection level for the +Point™ probe.

The benchmarking was performed in a simulation process of 1000 trial calculations. Figure 4-6b of RAI Reference 4 shows the histogram for the results of the 1000 trials using the Unit 3 initiation model with the Unit 2 wear indices as model input. The "number of observations" in the ordinate label of Figure 4-6 is the number of trials out of 1000 that produces the corresponding TTW occurrences (initiations). The summation of all observations equals 1000. The average number of TTW initiations from this simulation is about 34, producing approximately 5-6 detections. This number of detections is larger than the value that would be required to conservatively benchmark Unit 2. After the Unit 2 model was benchmarked as discussed on page 4-8 of RAI Reference 4, the simulation produced the histogram shown in Figure 4-6a of RAI Reference 4.

BOARD NOTIFICATION ENCLOSURE 3

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

January 17, 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 10 and 17)
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

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. Enclosure 2 of this letter provides the response to RAIs 10 and 17.

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 (Electric Power Research Institute), which sets forth the basis on which the information in Enclosure 2 may be withheld from public disclosure

P.O. Box 128
San Clemente, CA 92672

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.

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 Affidavits
2. Response to RAIs 10 and 17 (Proprietary)
3. Response to RAIs 10 and 17 (Non-proprietary)

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

ENCLOSURE 1

Notarized Affidavits

ELECTRIC POWER RESEARCH INSTITUTE

January 14, 2013

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding Values in Table "AOI Steam Generator Tube Leak Actions" in SONGS Response to NRC Request for Additional Information No. 10

Reference: *Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4. EPRI, Palo Alto, CA: 2011. 1022832*

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the information identified in the enclosed Affidavit consisting of the commercial information owned by Electric Power Research Institute, Inc. ("EPRI") identified above (the "Table"). Copies of SONGS response to NRC RAI No. 10 and the Affidavit in support of this request are enclosed.

EPRI desires to disclose values in the Table in confidence as a means of exchanging technical information with the NRC. The values in the Table are not to be divulged to anyone outside of the NRC nor shall any copies be made of the values in the Table provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 595-2732. Questions on the content of the Table should be directed to Helen Cothron of EPRI at (865) 773-4033.

Sincerely,

Steven M. Swilley
Senior Business Operations Manager

CHARLOTTE OFFICE
1300 West W.T. Harris Boulevard, Charlotte, NC 28262-8550 USA • 704.595.2000 • Fax 704.595.2860
Customer Service 800.313.3774 • www.epri.com

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

Request for Withholding Values in Table "AOI Steam Generator Tube Leak Actions" from SONGS Response to NRC Request for Additional Information No. 10

Reference: *Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4. EPRI, Palo Alto, CA: 2011. 1022832*

I, Steven M. Swilley, being duly sworn, depose and state as follows:

I am the Senior Business Operations Manager at Electric Power Research Institute, Inc. whose principal office is located at 1300 W WT Harris Blvd, Charlotte North Carolina ("EPRI") and I have been specifically delegated responsibility for the values in the above-listed Table that is sought under this Affidavit to be withheld (the "Table"). I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the values in the Table on behalf of EPRI.

EPRI requests that the values in the Table be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The values in the Table are taken from the referenced report that is owned by EPRI and constitutes commercial information which has not been placed in the public domain by EPRI.

b. EPRI made a substantial economic investment to develop the values in the Table and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of fees charged for the sale of the information. The values in the Table are entitled to the protection of the United States copyright laws. If the values in the Table were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry at no cost, these entities would be able to use the values in the Table for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the values.

c. EPRI made a substantial investment of both money and employee hours over an extended period of time in the development of the values in the Table. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the values in the Table are highly valuable to EPRI.

d. A public disclosure of the values in the Table would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to sell the values in the Table both domestically and internationally. If a party does not purchase the referenced report from EPRI, it would require an investment of money, time and effort equivalent to that expended by EPRI for the party to duplicate the values in the Table.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 1 / 14 / 2013

Steven M. Swilley

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 14th day of January, 2013, by Steven M. Swilley, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016,

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 RAIs 10 and 17

(NON-PROPRIETARY)

RAI 10

Technical Specification (TS) 3.4.12.d allows 150 gallons per day primary to secondary leakage. The Return to Service Report (Enclosure 2 of Reference 1), Section 9.4.1 states, "The plant operating procedure for responding to a reactor coolant leak has been modified to require plant Operators to commence a reactor shutdown upon a valid indication of a primary to secondary SG tube leak at a level less than allowed by the plant's TSs. This procedure change requires earlier initiation of operator actions in response to a potential SG tube leak." Does this mean that a reactor shutdown would be commenced upon any valid indication of primary to secondary leakage? Provide a description of the action levels in the procedure. Discuss any additional actions, planned or taken, such as simulator testing, operator training, and/or any evaluations to assess potential impacts of the revised procedure.

RAI 17

Reference 1, Section 9.4.1, page 50 – Provide the procedural action levels/statements (for SO23-13-14, Reactor Coolant Leak).

RESPONSE

The following discussion provides the response to RAIs 10 and 17.

1. **Does this mean that a reactor shutdown would be commenced upon any valid indication of primary to secondary leakage?**

Yes, a reactor shutdown will be commenced upon a valid primary to secondary leak.

2. **Provide a description of the action levels in the procedure.**

The Abnormal Operating Instruction (AOI), "Reactor Coolant Leak" specifies actions to be taken in response to indications of a primary to secondary leak. The AOI includes standard actions consistent with Electric Power Research Institute (EPRI) guidelines and additional SONGS specific administrative actions. The table below summarizes the required actions in the AOI following a steam generator tube leak (SGTL). As shown on the table, the actions depend on the leak rate, rate of change of the leak, and time and availability of radiation monitors.

Summary of AOI Steam Generator Tube Leak Actions

SGTL RATE: (1 gpm = 1440 gpd)	**ACTION(S)*** (consistent with EPRI guidelines)
Leakage is [ ] **AND** increasing by [ ] OR Leakage is [ ]	1. INITIATE Attachments 1, 2 & 3 2. Concurrently COMMENCE shutdown to be[ ]per procedure, "Rapid Power Reduction" (RPR), Attachment for RPR at a rate of [ ]
Leakage is [ ] and the rate of change in leakage is increasing by [ ]	1. INITIATE Attachments 1, 2 & 3 2. Concurrently COMMENCE shutdown to be in [ ] per "Rapid Power Reduction" procedure, Attachment for RPR at a rate of [ ]
Leakage is [ ] sustained for [ ] **AND** increasing [ ]	1. INITIATE Attachments 1, 2 & 3 2. Concurrently COMMENCE a controlled shutdown to be in [ ] per procedure, "Power Operations," Attachment for Power Descension 3. Monitor [ ]
Any confirmed SG leakage	1. Contact Management 2. INITIATE Attachments 1, 2 & 3 3. [ ] 4. INITIATE a controlled shutdown per procedure, "Power Operations," Attachment for Power Descension, and be in [ ]

* Attachments 1 and 2 are used to calculate SG tube leakage using RE-7870 and RE-7818A, respectively. Attachment 3 is used to minimize contamination during a SG tube leak.

3. Discuss any additional actions, planned or taken, such as simulator testing, operator training, and/or any evaluations to assess potential impacts of the revised procedure.

SONGS Operators have received additional classroom and simulator training to enhance Operator response to a primary to secondary SGTL. The lessons learned from Operator response to the Unit 3 SGTL, simulator tests and the new detection methods (Argon 40 injection, Nitrogen 16 monitors) were incorporated into new training.

To capture lessons learned, the plant operators involved in the Unit 3 SGTL on January 31, 2012 were interviewed to identify strengths and improvement areas. The enhancements identified by this process were incorporated into AOIs, "Reactor Coolant Leak" and "Rapid Power Reduction."

The training department Licensed Operator Requalification group performed classroom training sessions with licensed operators on the use of methods and techniques for detection of leaks using Argon 40 injection and Nitrogen 16 monitors, calculation of low level tube leak events, and operation at reduced power. In the classroom, AOIs "Reactor Coolant Leak" and "Rapid Power Reduction" were covered along with Operations Standard Manuals which were revised to capture new steam generator tube leak practices.

Simulator scenarios were conducted using the revised AOIs, "Reactor Coolant Leak" and "Rapid Power Reduction," which included the following:

- SGTL of 50 gpd in Mode 4 at normal operating pressure (NOP)
- Steam Generator Tube Rupture (SGTR) of 300 gpm in Mode 3 at NOP and normal operating temperature (NOT)
- SGTR of 12 gpm in Mode 2 at NOP/NOT
- SGTR of 355 gpm in Mode 1, 69% power

Training based on the Unit 3 SGTL event and plant upgrades was completed and is documented in the Nuclear Training Division Crew Performance Monitoring System.

An additional SGTL Simulator session for Licensed Operator Continuing Training will be completed prior to restart of Unit 2. This training session will include a small SGTL and use the AOI which specifies shutting down upon any confirmed SGTL. Additional Just-In-Time classroom training for operators will be completed prior to plant startup.

BOARD NOTIFICATION ENCLOSURE 4

SOUTHERN CALIFORNIA
EDISON®
An *EDISON INTERNATIONAL®* Company

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

January 21, 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)
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

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. Enclosure 1 of this letter provides the response to RAI 11.

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 11

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 11

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

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.

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. 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
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. SONGS Unit 2 Cycle 17 (S2C17) at 50% power results in radial power distributions (at the same power level and burnup) essentially identical to depleting the core at 100% power. 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.)
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.

Table 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
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.
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)
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.

Table 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
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.
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.

Table 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
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.
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.
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.
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.

Table 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
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.
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.

Table 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
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.
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.
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)	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.
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.

Table 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
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.
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).
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).

Table 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
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.
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.

Table 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
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.
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.

Table 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
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.
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.
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.

Table 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
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.
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.

Table 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
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.

Table 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
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.

Table 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
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 $+0.5\times10\text{-}4\ \Delta\rho/^{o}F$. 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.
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.

Table 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
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.
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.
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

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% 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 at ≥ 68% 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

BOARD NOTIFICATION ENCLOSURE 5

SOUTHERN CALIFORNIA
EDISON®
An *EDISON INTERNATIONAL®* Company

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

January 18, 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 12)
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

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. Enclosure 1 of this letter provides the response to RAI 12.

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 12

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 12

RAI 12

Operation at lower power level could introduce additional uncertainty in measuring reactor coolant flow. Please provide a detailed evaluation of RCS flow uncertainty, identify how RCS flow uncertainty is affected by operation at 70% power, and discuss the overall treatment of the RCS flow uncertainty, actual and indicated, in the context of the remaining safety analyses. Provide similar information for secondary flow uncertainty, as well.

RESPONSE

Note: This response also includes information requested in RAI 14 associated with the Reactor Coolant System (RCS) Flow Uncertainty Analysis. RAI 14 states: "Provide a summary disposition of the U2C17 calculations relative to the planned reduction in power operation."

A detailed RCS flow uncertainty analysis for extended operation at 70% Rated Thermal Power (RTP) has been performed. The results were used to bound extended operation at greater than or equal to indicated 68% RTP. The methodology used is consistent with the methodology described in the SONGS Units 2 and 3 Reload Topical Report.

The RCS flow uncertainty analysis determined RCS flow uncertainties used in both the verification of minimum RCS flow (i.e. safety analysis input) and as the input to the overall uncertainty analysis required by the digital setpoint process for the Core Protection Calculators (CPCs) and the Core Operating Limit Supervisory System (COLSS). The RCS flow uncertainty analysis simulates the RCS flow calculation based upon the calorimetric method. This method is the Technical Specification (TS) 3.3.1.5 required method to calibrate the RCS flow used in the CPC and COLSS computer systems. The flow rate uncertainties are determined by the "stochastic simulation" technique. Important input parameters to the RCS flow uncertainty based upon calorimetric methods are secondary calorimetric power, hot leg temperature, cold leg temperature, and input uncertainties including secondary calorimetric power uncertainty, hot leg temperature uncertainty, and cold leg temperature uncertainty. The analysis was performed at three different power levels (68%, 95%, and 100%) to bound extended operation at greater than or equal to indicated 68% RTP.

The calorimetric based RCS flow uncertainty increases as a result of extended operation at reduced power resulting in an increase in the CPC/COLSS RCS flow uncertainties. For example, the COLSS RCS flow uncertainty increases from 4.4% when calibrated to calorimetric based RCS flow at 95% RTP to 5.7% when calibrated to calorimetric based RCS flow at 68% RTP. Key contributors to the RCS flow uncertainty increase are an increase in primary coolant enthalpy change uncertainty and the secondary calorimetric power uncertainty. The secondary calorimetric power uncertainty input to the RCS flow uncertainty analysis increased from ± 2.0% RTP at 95% RTP to ± 2.3% RTP at 68% RTP.

The primary coolant enthalpy change and secondary calorimetric power uncertainties increase as power lowers because they are dominated by parameters that decrease as power decreases, but the uncertainty on those parameters is constant. The ± 3 °F RCS hot leg (T_{HOT}) and cold leg (T_{COLD}) temperature uncertainty has more effect as the difference between T_{HOT} and T_{COLD} approaches zero. Similarly, as feedwater flowrate decreases, the ± 5.6 inches H_2O uncertainty of the feedwater flowrate transmitter has more impact on secondary calorimetric power indication. The increase in RCS flow uncertainty has been accounted for in the overall uncertainty analysis required by the digital setpoint process for CPC/COLSS and did result in a small increase on a power related penalty term used within the CPC/COLSS Departure from Nucleate Boiling Ratio (DNBR) algorithms.

With respect to the safety analyses, the minimum RCS flow rate used is 376,200 gpm which is 95% of 396,000 gpm specified by TS 3.4.1 and the Core Operating Limit Report (COLR). During performance of SR 3.4.1.3, indicated RCS flow minus uncertainties will be required to be greater than or equal to the minimum required by the safety analyses (e.g., actual flow must be greater than or equal to 376,200 gpm); therefore, no COLR change is required. Satisfying the above requirement ensures the safety analyses remain bounded with respect to RCS flow requirements during extended operation at greater than or equal to indicated 68% RTP.

The primary indicator for monitoring power at 70% RTP is the secondary calorimetric power calculation in COLSS since it has the minimum uncertainty of available power indications. The analysis for the secondary calorimetric power indication uncertainty was performed over a range of power levels from 20% RTP to 100% RTP. As indicated above, an uncertainty of ± 2.3% RTP at 68% power was used as input to the RCS flow uncertainty analysis. Another product of the secondary calorimetric power uncertainty analysis is a power level dependent bounding uncertainty curve (e.g., applicable uncertainties from 20% RTP to 100% RTP), which is used by COLSS to conservatively monitor Linear Heat Rate (LHR) and DNBR power operating limits and is used within transient safety analyses. At 70% RTP, the secondary calorimetric power indication is based on the feedwater flow venturi. As is done at full power, the uncertainty in the indication is accounted for in the overall uncertainty analysis and the LHR and DNBR power operating limits. Therefore, the increase in secondary calorimetric power and RCS flow uncertainties as steady state power is decreased is accounted for in the overall uncertainty analysis required by the digital setpoint process for CPC/COLSS and remaining safety analyses.

BOARD NOTIFICATION ENCLOSURE 6

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

January 18, 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)
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

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. Enclosure 1 of this letter provides the response to RAI 13.

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,

Enclosure:

1. Response to RAI 13

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 13

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

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) <u>Evaluation of Impact of Replacement Steam Generators on Emergency Core Cooling System (ECCS) Performance Analyses</u>

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) <u>Rated Thermal Power</u>

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) <u>RSG Tube Plugging and RCS Volume</u>

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.

BOARD NOTIFICATION ENCLOSURE 7

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

January 9, 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 15)
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

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. Enclosure 2 of this letter provides the response to RAI 15.

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 Mitsubishi Heavy Industries (MHI), 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 the source document MHI Report L5-04GA561, Retainer Bar Tube Wear Report, 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 15 (Proprietary)
3. Response to RAI 15 (Non-proprietary)

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

ENCLOSURE 1

Notarized Affidavit

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 MHI technical documentation to determine whether it contains 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 determined that the following MHI documents and drawings contain MHI proprietary information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4). The drawings in their entirety are proprietary and those pages of the documents containing proprietary information have been bracketed with an open and closed bracket as shown here "[ ]" / and should be withheld from public disclosure.

   MHI documents and drawings
   Document: L5-04GA561, L5-04GA564, L5-04GA571, L5-04GA585, L5-04GA591
   Drawings: L5-04FU101 thru 108

3. The information identified as proprietary in the enclosed document has 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 it describes 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 2 day of August, 2012.

Jinichi Miyaguchi

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

220

AUG. -2. 2012

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

Sworn to and subscribed

Before me this 2 day

of August, 2012

Masahiko Kubota

Notary Public

My Commission Expires __________________

登簿平成２４年第２２０号

認　　証

嘱託人　三菱重工業株式会社　原子力事業部　原子力誠三総括部　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。

よって認証する。

平成２４年８月２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 220 Date AUG. -2. 2012

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 15

(NON-PROPRIETARY)

RAI 15

In Reference 1, Section 8.3.2, page 48 – How will the continued integrity of the non-stabilized, preventively-plugged tubes adjacent to the retainer bars be ensured? "Integrity" in this context refers to the tubes remaining intact and unable to cause damage to adjacent tubes.

RESPONSE

The integrity of the non-stabilized, preventively-plugged tubes is ensured by limiting the wear resulting from retainer bar vibration. The limited vibration amplitude of the tubes and retainer bars, combined with stabilizer deployment, prevents developing a displacement/wear geometry that could sever any of the tubes adjacent to retainer bars, either in the short term or long term.

Wear mechanism of tubes adjacent to retainer bars

There are 94 tubes in each steam generator adjacent to retainer bars. Each of these tubes has 7 hot leg tube support plate (TSP) support locations, 12 anti-vibration bar (AVB) support locations, and 7 cold leg TSP support locations. All 188 of these tubes in the Unit 2 steam generators (94 tubes per steam generator) were examined. No evidence of wear was found on any of these tubes at AVB and TSP intersections. Retainer bar wear was found on a total of 6 tubes with 7 wear locations (one tube in SG 2E-089, Row 120 Column 132, had retainer bar wear at two retainer bar locations, remaining 5 tubes had retainer bar wear at one location). The maximum wear depth of 90% tube wall thickness was found on SG 2E-089, Row 119 Column 133, in a location adjacent to a retainer bar.

The cause of tube wear at retainer bar locations has been evaluated by MHI. Wear marks at the AVB intersections would be evidence of out-of-plane displacement of the U-bend. Wear marks on the TSP intersections, especially the top TSP, would be evidence of in-plane displacement of the U-bend. The absence of wear at the AVBs and TSPs of all 188 tubes adjacent to the retainer bar is evidence that the tubes adjacent to the retainer bar are not vibrating. MHI concluded that the tube wear adjacent to retainer bars is caused by retainer bar vibration rather than tube vibration.

During steam generator operation, retainer bars are subject to flow induced vibration. MHI's analysis of the dynamic response of retainer bars to operating conditions found that the vibration amplitude is limited and much smaller than the tube diameter of 0.75". Consequently, these retainer bar motions may damage the wall of an adjacent tube but cannot sever these tubes. The retainer bar natural frequencies and vibration amplitudes for the first five modes are shown in Table 1 and the lowest three mode shapes are shown in Figure 1. The first mode moves in a direction parallel to the tubes. The second and third retainer bar modes are perpendicular to adjacent tubes. The maximum amplitude of the first mode due to steam generator operating conditions is between [ ]. Maximum amplitude of the second mode during steam generator operating conditions is between [ ]. All higher modes have negligible vibration amplitudes.

Table 1 - Retainer Bar Natural Frequencies and Vibration Amplitudes

	Mode 1	Mode 2	Mode 3	Mode 4	Mode 5
Frequency, Hz					
Amplitude					

Figure 1 – Retainer Bar Vibration Mode Shapes

Integrity of tubes adjacent to retainer bars

The six tubes with retainer bar wear indications in Unit 2 steam generators have been plugged, regardless of wear depth. To ensure that these tubes remain intact, ½" diameter braided stainless steel cable stabilizers have been installed in these six tubes.

As a preventive measure to ensure that no in-service tubes are subject to retainer bar wear, all tubes adjacent to retainer bars have been plugged.

Additionally, stabilizers have been deployed in six tubes at each retainer bar. Figure 2 shows a typical deployment. Three tubes on each side of the retainer bar have been stabilized: one tube near the center of the retainer bar and two tubes near both ends of the retainer bar. The stabilizers will arrest tube wear at the wear surface of the stabilizers. Since the tubes adjacent to retainer bars have no evidence of significant vibration and the retainer bar vibration amplitude is limited, the stabilizer deployment pattern prevents any possible retainer bar or tube displacement/wear geometry that could sever any of the tubes adjacent to the retainer bars.

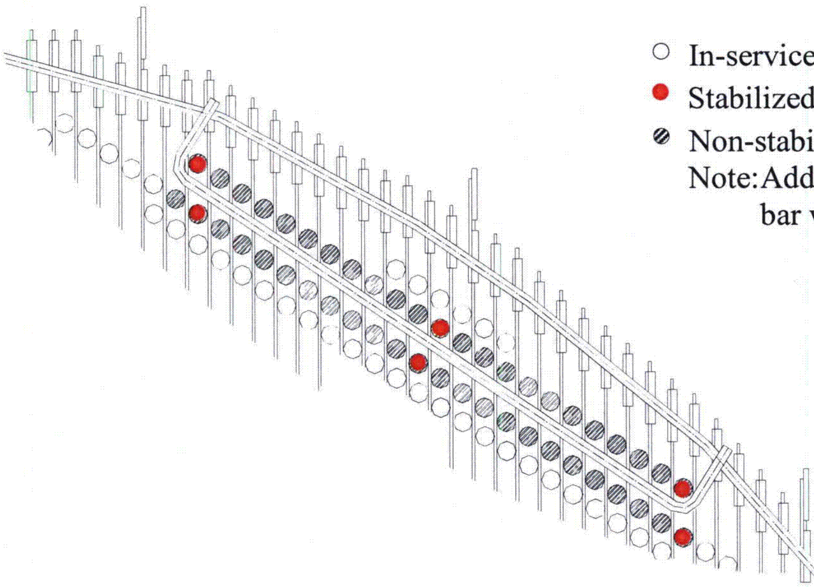

Figure 2 – Typical Stabilizer Deployment to Arrest Retainer Bar Wear

The integrity of the non-stabilized, preventively-plugged tubes is ensured by the limited vibration amplitude of the tubes and retainer bars, along with the number and arrangement of stabilized, preventively-plugged tubes at each retainer bar.

Future inspections of retainer bars

The steam generator retainer bar wear issue has been entered into the SONGS Corrective Action Program (CAP). An effectiveness review requires visual inspection of the smaller diameter retainer bars and welds during the upcoming Unit 2 mid-cycle outage. In addition, all in-service tubes will be inspected by eddy current testing in the upcoming Unit 2 mid-cycle outage. This inspection will confirm that the non-stabilized, preventively-plugged tubes adjacent to the retainer bars are not damaging adjacent in-service tubes.

BOARD NOTIFICATION ENCLOSURE 8

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

January 10, 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 16)
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

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. Enclosure 1 of this letter provides the response to RAI 16.

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 16

cc: E. E. Collins, Regional Administrator, NRC Region IV
R. Hall, NRC Project Manager, San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 and 3
R. E. Lantz, Branch Chief, Division of Reactor Projects, 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 16

RAI 16

Reference 1, Section 9.3, page 50 – Provide additional information concerning the "Operational Decision Making" process and describe how it would be applied if the proposed criterion is exceeded. Provide the procedural action statement.

RESPONSE

Reference 1, Section 9.3 states: "The plant procedure for chemical control of primary plant and related systems has been modified to require action if the specific activity of the reactor coolant Dose Equivalent Iodine (DE I-131) exceeds the normal range of 0.5 µCi/gm, which is one-half of the TS Limit of 1.0 µCi/gm. In the event that the normal range is exceeded, Operations is required to initiate the Operational Decision Making process to evaluate continued plant operation."

Operational Decision Making Process

The San Onofre Nuclear Generating Station (SONGS) Operational Decision Making (ODM) process is based on the Institute of Nuclear Power Operations (INPO) 2011 document, "Principles for Effective ODM," which was developed by an industry task force with assistance from INPO and U.S nuclear utility managers and executives.

SONGS ODM procedure provides the Operations Shift Manager (SM) and station leadership a framework for making decisions in response to degraded conditions to address long term protection of the public, plant personnel and station assets. Two scenarios are addressed by the ODM process: Type 1 off-normal conditions and Type 2 degraded conditions that fall below the action thresholds defined in license documents or may not be defined in existing procedures.

The ODM process provides checklists with specified considerations the SM uses to ensure a thorough review of the degraded condition is performed and documented. A list of stakeholders is included to facilitate involvement of required subject matter experts and personnel. The purpose of these lists is to provide an established format to address trends, assessments, risks, plant conditions, regulatory commitments, and impacts of the degraded condition to ensure the health and safety of the public. The ODM process results in operational decisions, implementation plans and thresholds to trigger actions.

Establishment of Administrative Limit for RCS DE I-131 Activity Level

Chemistry procedure, "Units 2/3 Chemical Control of Primary Plant and Related Systems," has been revised to lower the Reactor Coolant System (RCS) DE I-131 limit from 1.0 µCi/gm to 0.5 µCi/gm. The Technical Specification (TS) limit for DE I-131 is 1.0 µCi/gm. As a defense in depth action included in the Unit 2 return to service plan, if DE I-131 exceeds the Normal Range Limit of 0.5 µCi/gm, Chemistry personnel will perform sampling for DE I-131 once every four hours until the level is less than 0.5 µCi/gm. In addition, Chemistry personnel are directed to notify the on-shift Operators to initiate a Type 1 ODM evaluation to determine other appropriate actions to be taken. For this off-normal condition, the ODM will be performed to determine the impact of the increasing levels of DE I-131 and involve senior station management in evaluating continued plant operation. If DE I-131 exceeds 0.5 µCi/gm for greater than 48 hours, a plant shutdown to Mode 3 will commence with Tavg less than 500°F within 6 hours (as required by TSs when DE I-131 exceeds 1.0 µCi/gm). The procedural actions, frequency, and range are shown in the table below.

Using the ODM process, the SM obtains data from (1) Chemistry Department representatives as to the nature of the increasing DE I-131, (2) Nuclear Fuels group concerning the condition of the nuclear fuel pins that would be the source of the increased activity, and (3) engineering personnel regarding the condition of the steam generators. Based on the input from the stakeholders, the SM confers with Station Management to decide future actions, including a potential plant shutdown prior to being procedurally required.

Chemistry Procedure RCS Activity Level Action Statement

<table>
<tr><th>Parameter</th><th>Frequency</th><th>Normal Range</th><th>Comments/Corrective Action</th></tr>
<tr><td rowspan="2">DE I-131, µCi/gm

TS 3.4.16
SR 3.4.16.2</td><td>Weekly</td><td rowspan="2">≤ 0.5 µCi/gm</td><td rowspan="2">TS SR 3.4.16.2
Mode 1 requirement for non-transient sample is once per 14 days.

TS 3.4.16
VERIFY within 2 to 6 hours following a thermal power change ≥ 15% of the rated power within a one-hour period.

IF DE I-131 EXCEEDS the Normal Range (≤ 0.5 µCi/gm), <u>THEN</u> send a Memo to Operations requesting a Type 1 ODM be performed to continue plant operation considering Steam Generator health.

IF DE I-131 EXCEEDS the Normal Range Limit (≤ 0.5 µCi/gm), <u>THEN</u> SAMPLE and analyze the RCS every four hours until < 0.5 µCi/gm

NOTE: Actual Tech Spec Limit is > 1.0 µCi/gm, however, TS required Actions shall be performed at > 0.5 µCi/gm

IF DE I-131 is > 0.5 µCi/gm for greater than 48 hours <u>OR</u> greater than the limits of Figure 3.4.16-1 at any time, <u>THEN</u> be in Mode 3 with Tavg <500°F within 6 hours.</td></tr>
<tr><td>With >15% power change/hour</td></tr>
</table>

BOARD NOTIFICATION ENCLOSURE 9

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

January 24, 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)
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

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. Enclosure 1 of this letter provides the response to RAI 18.

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

cc: E. E. Collins, Regional Administrator, NRC Region IV
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

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

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.

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 10

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

January 16, 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 19)
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

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. Enclosure 1 of this letter provides the response to RAI 19.

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 19

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 19

RAI 19

Reference 1, Section 11.2, page 52 – Provide additional details on how the GE Smart Signal System will be used in the context of tube-to-tube wear and/or the circumstances associated with tube-to-tube wear. What information/data will the system be evaluating? For what purpose?

RESPONSE

As stated in the RAI Reference 1, Section 11.2, GE Smart Signal™ will be used as "an analytic tool that aids in diagnosis of equipment conditions. The tool will be used to analyze historical plant process data from the Unit 2 steam generators (SGs) following the inspection interval." GE Smart Signal™ will be used as a backward looking tool to assist in the investigation of any additional tube-to-tube wear identified during the mid-cycle inspection. GE Smart Signal™ is not included as a Defense in Depth measure in section 9 of RAI Reference 1.

GE Smart Signal™ is a software program used to trend and monitor variances in plant equipment (such as pressures, temperatures, flowrate, speed, feedwater pump and turbine vibration). GE Smart Signal™ will be monitoring the following equipment:

1. Unit 2 Steam Generators
2. Unit 2 Main Feedwater Pumps
3. Unit 2 Main Feedwater Turbines

BOARD NOTIFICATION ENCLOSURE 11

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

January 25, 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 27)
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

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. Enclosure 1 of this letter provides the response to RAI 27.

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 27

cc: E. E. Collins, Regional Administrator, NRC Region IV
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 27

RAI 27

Reference 6, Appendix 8, "SG Tube Flowering Analysis", page 8-2 (307 of 474) – MHI concludes, in part, that the tube-to-AVB gaps in the center columns increase due to hydrodynamic pressure by [...] when the manufacturing tolerance dispersion is not taken into account. MHI also concludes that the gap increase due to hydrodynamic pressure is small when the manufacturing tolerance dispersion is taken into account. Discuss whether this latter finding may simply reflect the hydrodynamic pressures acting to relieve the tube-to-AVB contact forces caused by the manufacturing tolerance dispersion, such that the gaps are relatively unchanged relative to the case where the hydrodynamic pressure is not considered. Reference 6, Appendix 9, "Simulation of Manufacturing Dispersion for Unit-2/3," does not seem to make specific mention of whether the calculated tube-to-AVB contact forces directly considered the effect of the hydrodynamic effect on tube-to-tube contact forces, but the staff understands that they did not. If the staff's understanding is correct, explain how the resulting contact forces are conservative.

RESPONSE

Note: RAI Reference 6 is MHI Document L5-04GA564, Tube Wear of Unit-3 RSG – Technical Evaluation Report, Revision 9, October 2012, prepared by Mitsubishi Heavy Industries, LTD. (ADAMS Accession Nos. ML12285A265, ML12285A266, and ML12285A267).

The staff's understanding is correct: the contact force analysis contained in RAI Reference 6, Appendix 9 does not consider the effect of hydrodynamic forces. Since hydrodynamic forces are very small in comparison to contact forces they were not included in the contact force analysis. To demonstrate this, a sensitivity study was performed in response to this RAI.

This study compared the probability of occurrence of in-plane fluid-elastic instability (FEI) for two cases: (1) contact force distribution including hydrodynamic forces and manufacturing dispersion and (2) contact force distribution based on manufacturing dispersion alone. For the 70% power condition, there was no statistically significant increase in the probability of in-plane FEI when hydrodynamic forces were included.

The consideration of hydrodynamic forces results in a slight reduction of average contact force at 70% power, but the tube-to-support gaps are relatively unchanged. Hydrodynamic forces are postulated to have little effect on tube-to-support gaps due to their low estimated magnitude.

The sensitivity study performed for this RAI response determined there is no statistically significant increase in the probability of in-plane FEI when the effects of hydrodynamic forces are included in the analysis.

BOARD NOTIFICATION ENCLOSURE 12

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

10 CFR 50.4

January 21, 2013

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 28)
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

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. Enclosure 1 of this letter provides the response to RAI 28.

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,

P.O. Box 128
San Clemente, CA 92672

Enclosures:

1. Response to RAI 28

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 28

RAI 28

Reference 5, Section 2.6.1 - What is the estimated growth rate of the tube to tube wear in steam generator 3E0-88, tube R106C78? Describe how it was determined.

RESPONSE

Note: Reference 5 in RAI 28 is the Westinghouse operational assessment.

Tube-to-tube wear (TTW) rates were not calculated for tubes in Unit 3 since that was not required as part of the operational assessment for Unit 2. Benchmarking of Unit 3 determined that all tubes with TTW were found to either be unstable in the in-plane direction, or were in contact with tubes that were unstable.

The Westinghouse evaluation determined that the wear associated with tube-to-tube contact in Unit 2 was not a result of in-plane instability. The analysis determined that the tube-to-tube contact was a result of a proximity condition, meaning that the tubes were closer together than what would be indicated in the design. Therefore, no significant TTW growth is predicted in Unit 2. Appendix A of Reference 5 contains a summary of the TTW that occurred in the free span of Unit 2. This appendix addressed the Unit 2 tubes with TTW and did not evaluate TTW in Unit 3.

In summary, no TTW calculations were performed for SG 3E-088, tube R106C78, as the tube wear in Unit 3 was produced from a different mechanism than the TTW that occurred in Unit 2. As a result, this was not a necessary calculation to support the Unit 2 operational assessment.

BOARD NOTIFICATION ENCLOSURE 13

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

January 9, 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 30)
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

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. Enclosure 2 of this letter provides the response to RAI 30.

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 Mitsubishi Heavy Industries (MHI), which sets

P.O. Box 128
San Clemente, CA 92672

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NRR

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.

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 Affidavits
2. Response to RAI 30 (Proprietary)
3. Response to RAI 30 (Non-proprietary)

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

ENCLOSURE 1

Notarized Affidavits

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 AREVA's technical documentations to determine whether they contain 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 MHI's letter "MHI/SCE-12-0161: Figure of Percentage of Tube Wear vs. Maximum Interstitial Velocity (Reply to RAI #30)" 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).

3. The information identified as proprietary in the documents has 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 it describes 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 28 day of December, 2012.

Jinichi Miyaguchi

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

Sworn to and subscribed

Before me this 28 day

of December, 2012

Masahiko Kubota

Notary Public

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

373 DEC. 2 8. 2012

My Commission Expires ______________

登簿平成２４年第３７６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 Akashi
Kobe, Ja

よって認証する。

平成２４年１２月２８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 3 7 6 Date DEC. 2 2012

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 30

(NON-PROPRIETARY)

RAI 30

Reference 1, Figure 8-2 – Provide similar figure for maximum interstitial velocities.

Response

The following figure shows the percentage of tubes with tube-to-tube wear (TTW) indications at each range of maximum interstitial velocity at 100% power. In the figure, 2A, 2B, 3A and 3B represent steam generators (SGs) 2E089, 2E088, 3E089 and 3E088, respectively. As indicated in Table 8-3 of Reference 1, operating Unit 2 at 70% power reduces the maximum interstitial velocity in the SG secondary side to well below the velocities which resulted in TTW at 100% power.