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Comment On: NRC-2013-0070-0001
Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generating Station, Unit 2

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Submitter Information

Name: Daniel Hirsch
Organization: Committee to Bridge the Gap

General Comment

Please find attached our comments and attachments thereto.

Attachments

BridgeTheGapComments

Pages from RAI32 OA

ASLB Order LBP 13 07

Far Outside the Norm

Comments
of the Committee to Bridge the Gap
Regarding Southern California Edison's
License Amendment Request for San Onofre
and Proposal that Any Adjudicatory Hearing on the Proposed Amendment
Occur Only After the Amendment is Granted
16 May 2013

Docket ID NRC–2013–0070

Introduction

Because of a short-sighted decision by Southern California Edison (SCE) to replace the steam generators (SGs) at its San Onofre Nuclear Generating Station (SONGS) with new SGs of substantially different design while trying to bypass the normal legally-required opportunity for the public to request an adjudicatory hearing, defective SGs were installed. Those SGs failed within a year in the case of Unit 3 and in less than two years for Unit 2. NRC staff allowed this bypass of required scrutiny.

Now, SCE and NRC staff propose to repeat their prior error. SCE wishes to restart its crippled Unit 2, without repairing or replacing the defective steam generators, and to do so without a prior adjudicatory hearing on the safety of the endeavor. The NRC staff has informed SCE, however, that is barred by its current license from doing so. The SONGS Technical Specifications, a part of the license, require that the steam generators be capable of operating at 100% power without a risk of any tube bursting, with a margin of safety. SONGS cannot do that, because of the design defects in the new SGs. For that reason, it wishes to operate for a time at 70% and see if its theory that this will slow the rate of continued damage in the SGs sufficiently. As the Atomic Safety and Licensing Board (ASLB) ruled on May 13, 2013, this is clearly a test or experiment—despite the statements by SCE and NRC staff, repeated over and over again, that they do not experiment with safety.

So SCE faces an obstacle to restarting the crippled reactor without fixing it: its license bars this. Therefore, in order to nonetheless restart, it proposes that that safety restriction in its license be eliminated. It has now submitted a license amendment application to that effect. But, not having learned from history, it now proposes to once again try to bypass the prospect of an evidentiary hearing where evidence that this is not safe could be presented by independent parties and ruled upon by independent decision-makers of an Atomic Safety and Licensing Board. Given the complete rejection of SCE and NRC staff's positions by the ASLB in the just-completed proceeding, one can perhaps understand SCE and NRC staff's eagerness to avoid such a review regarding the license amendment. But such avoidance would both violate the law and would be imprudent, because critical technical matters could be resolved by the ASLB that might result in determining grant of the license amendment and the restart it is designed to facilitate would be unsafe.

The license amendment application should be denied. The "no significant hazards

consideration" (NSHC) proposed determination—the mechanism by which it is proposed to avoid a prior safety hearing—should likewise be rejected.

The Proposed Amendments

At their core, the requested license amendments would eliminate a safety requirement in the current license that has the effect of barring operation of the reactor with damaged steam generators. The license prohibits operation if the SGs are incapable of full power operation with a factor of 3 safety margin for tube burst at that power. The SONGS replacement SGs are so defective--so riddled with design flaws—that they cannot do that, and for that reason SCE proposes running instead at 70% power and hoping for the best.[1]

The proposed license amendment would, for a period of two years, redefine 100% power as 70%, eliminating the prohibition against running with damaged SGs incapable of 100% power operations. It is important to note that this amendment would give SCE considerably more authority than it had previously stated it was requesting. In the past, SCE has said it only wished authority to run at 70% power for 5 months, after which it would shut down and inspect the SGs to see if its theory was working out. Now it asks for authority to run at 70% power with the damaged steam generators for 5 months, shut down and inspect-and then restart, and keep running in intervals of its choosing for 24 months. If the license amendment were granted, NRC would have no say in the matter. The license amendment would authorize operation for the two year period without having to ask NRC OK, with the decision left in SCE's hands based on its own internal assessments.

Additionally, the proposed license amendment would have SCE not provide to NRC for 60 days the results of its inspections and OAs after the inspection from the first five month test operation is completed. This means Unit 2 would be back up and running with the damaged steam generators and potentially very troubling results from the inspections and analyses before NRC staff would even receive those results, let alone have time to review them. There are significant potential hazards considerations in all of these proposals.

SCE is candid about the purpose of the license amendments: "The proposed changes are in support of re-start of SONGS Unit 2 following identification of steam generator tube wear during the SONGS Unit 2, Cycle 17 refueling and steam generator inspection outage." In other words, the proposed amendments are essential to permitting Unit 2 to restart with damaged steam generators that haven't been repaired. Clearly then, the safety considerations must include the potential impacts of what the amendments would help authorize. If the amendments are

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[1] SCE initially tried to get around this license requirement by asserting that 70% = 100%. When that absurdity didn't work, they claimed they could in fact operate at 100% power, which of course begs the question why don't they and why are they insisting on 70%. SCE submitted an Operational Assessment (OA) purporting to show Unit 2 could in fact run at 100%, but even its own analysis showed it would reach the danger point after less than a year's operation. NRC staff has not fully reviewed let alone approved the OA—which would be a remarkable step—and such a review would take considerable time, so SCE has adopted a third tack, this license amendment request.

granted, operation with the damaged steam generators cannot happen. Thus, granting the amendment request clearly can increase the likelihood and consequences of accidents previously evaluated; create the potential for accidents not previously analyzed; and significantly reduce safety margins.

We note one other matter related to the proposed license amendment request. SCE submitted a supplement to it on 9 April 2013—a single day before NRC staff signed off on the proposed amendment and NSHC determination, raising questions about the seriousness of the staff review. Additionally, the day before public comments were due on the license amendment request and NSHC proposed determination, NRC staff placed on the NRC's ADAMS database a 9 May 2013 memorandum from Brian Benney, Senior Project Manager, NRC's SONGS Special Projects Branch (and the NRC point of contact for this Federal Register notice about the license amendment). That memorandum summarizes a 29 April 2013 "clarification call with Southern California Edison" about its license amendment request. To the best of our knowledge, this call was not publicly noticed, made available for other parties to participate or even listen. The memorandum indicates NRC staff sought to get clarification from SCE about a number of aspects of its license amendment request—a troubling development given that NRC staff had already announced its intention to approve the license amendment request [LAR] and the NSHC request.

But more of concern is that "The licensee also indicated that it planned to supplement the LAR with additional license conditions.... In addition, the licensee plans to submit information with the supplement that will clarify the discussions provided in the LAR submittals." How can this possibly be? NRC staff has published in the Federal Register a month ago a notice soliciting public comments on the LAR and the NSHC determination that is predicated on it. Now we learn that SCE has indicated to NRC staff its intention to "supplement the LAR," include new "additional license conditions," and add other supplementation to the "discussions provided in the LAR submittals."

The entire process needs to stop and start over again. One can't engage in bait-and-switch tactics. One can't ask the public to comment on a License Amendment Request that is changed, supplemented, and has additional proposed license conditions added to it after the public notice is issued, or, indeed, after the public comment period expires. The whole process appears completely out of control.

Protesting the above violations of the required process, we nonetheless attempt to provide comments on the LAR as noticed in the Federal Register on 16 April 2013. We discuss below in more detail the three NSHC criteria that must be met in order for SCE and NRC staff to avoid the prospect of having to appear before another ASLB that might, as did the ASLB just concluded, reject SCE and NRC claims, and thus bar grant of the requested amendments.

<u>1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?</u> **YES**

SCE and NRC staff concede that SCE needs the requested license amendments in order to restart Unit 2 with its damaged steam generators without repairing or replacing them. Clearly,

then, the proposed change involves a significant increase in the probability or consequences of an accident previously evaluated. To the extent SCE has previously evaluated an accident involving failure of the SGs, the probability and/or consequences are significantly increased if it runs with defective SGs. It cannot run with these defective SGs without the license amendments; therefore granting them will increase those probabilities and/or consequences.

Unit 2 has 400 times more worn tubes and more than 1000 times more indications of wear on those tubes than the typical replacement U-tube steam generator nationally. It has had to plug more tubes than the rest of the country combined for a comparable period with replacement SGs. And it has essentially the same number of damaged tubes as Unit 3 (1600 vs. 1800). We hereby attach our report *FAR OUTSIDE THE NORM: The San Onofre Nuclear Plant's Steam Generator Problems in the Context of the National Experience with Replacement Steam Generators* and include its content in these comments.

Unit 2 thus is in terrible shape. It has the same design defects as Unit 3, and similar numbers of damaged tubes. The SGs, as the ASLB decision points out, do not fall within the licensing basis for the reactor; their designs are defective; they are seriously broken. The proposed license amendment would remove a safety requirement – the requirement to be able to operate at 100% power with a margin of safety—in order to allow operation of damaged SGs that otherwise would not be permitted to operate. This clearly would increase the risk and/or consequences of accident.

Note that the Final Safety Analysis Report is based on a single tube bursting. The consequences of multiple tubes bursting, and propagating damage to other tubes, would be considerably greater than this single-tube-burst. So, to the extent the accident has already been evaluated, the likelihood and consequences would be significantly greater if this safety requirement is removed from the license and operation with damaged SGs, never contemplated before, is allowed.

This is clearly demonstrated even in SCE's own analysis. SCE predicts operating at 70% power results in an unacceptable risk of tube burst at 16 months (other analyses of their suggest even earlier). See attached SCE chart from its OA for 100% power. But they are requesting permission to run with the damaged SGs for 2 years at 70% power, without having to get NRC OK no matter how much damage is found in the SGs during that period. The license amendment requests 2 years; SCE's own analysis unacceptable risk at 1 year and 4 months; its own analysis thus indicates unacceptable increase of probability of accident.

But it should be made clear that this is not a matter left up to SCE's claims or NRCs staff's positions about those claims. The NSHC determination is based on whether there are *considerations* involving hazards, not whether there are significant hazards. In other words, is there evidence that could be put forward at an adjudicatory hearing that might result in the license amendment being denied by an ASLB due to the hazards it could create, not whether NRC staff conclude that they think the weight of the evidence in such a dispute is such as to be OK to grant the license. So, SCE claims 16 months is when they get into trouble; independent experts assert it could happen in just weeks. There are thus significant considerations of hazards, and that means a prior hearing opportunity is mandated by law.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? **YES**

The FSAR analyzed only a single tube burst. Running Unit 2 with damaged SGs creates the possibility of a new or different kind of accident, one that involves multiple tube bursts, propagating, and full failure of the SGs and core meltdown caused thereby and direct release pathway to the environment for radioactivity, particularly in case of an event that stresses the weakened SGs such as a main steam line break or earthquake.

3. Does the proposed change involve a significant reduction in a margin of safety? **YES**

Requiring the reactor to be able to operate at 100% power with significant margin of safety for tube burst is a safety requirement Unit 2 cannot now meet. It bars running with degraded SGs. Eliminating that safety requirement is eliminating a significant margin of safety.

With the existing license, the damaged SGs can't run without being repaired or replaced. Eliminating the license requirements, SCE can run with broken SGs. That significantly reduces a margin of safety.

Conclusion

NSHC requires a determination at the beginning of an analysis that there are no safety issues worth considering. It is not a determination after a review that one weighs in on one side or another of a safety controversy. It is a determination there is no significant controversy. SCE's proposed license amendments cannot meet any of the tests for NSHC, and therefore an opportunity for an adjudicatory hearing before deciding whether the license amendment should issue is mandatory.

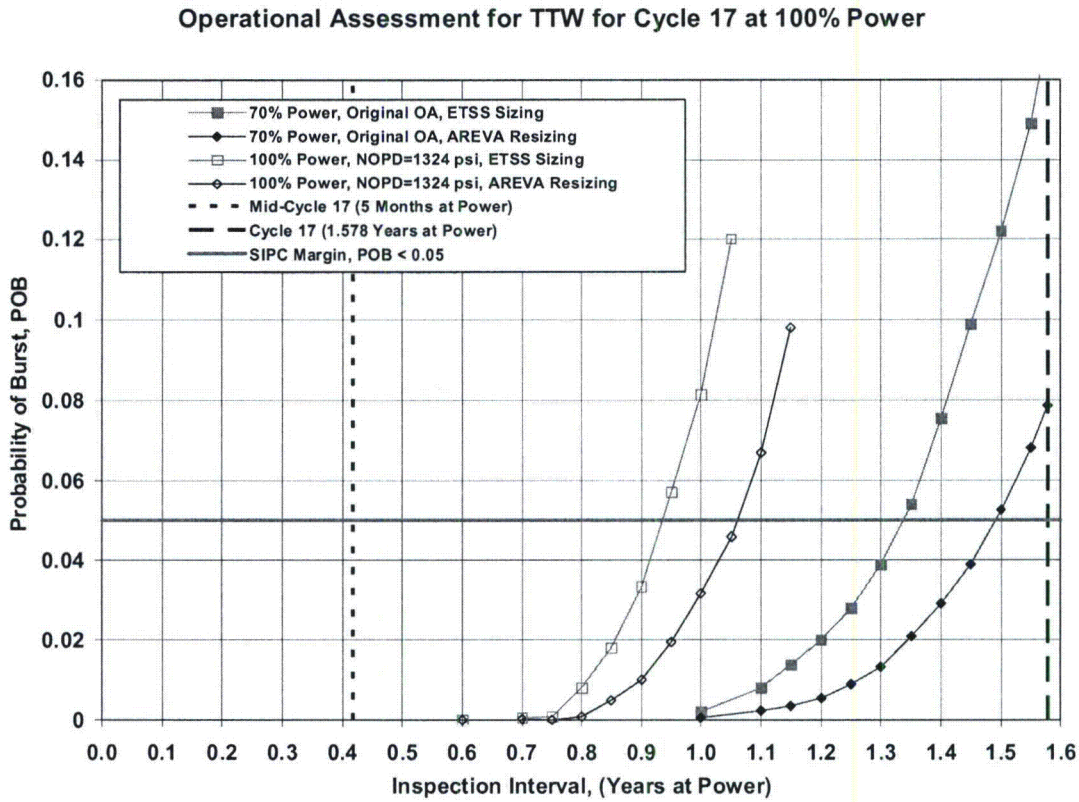

Figure I.5-1 — Probability of Burst at 70% and 100% Power Levels for TTW Growth Rate Models Based on Case 1 - ETSS 27902.2 Sizing and Case 2 - AREVA Resizing Models

LBP-13-07

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

E. Roy Hawkens, Chairman
Dr. Anthony J. Baratta
Dr. Gary S. Arnold

In the Matter of SOUTHERN CALIFORNIA EDISON CO. (San Onofre Nuclear Generating Station, Units 2 and 3)	Docket Nos. 50-361-CAL, 50-362-CAL ASLBP No. 13-924-01-CAL-BD01 May 13, 2013

MEMORANDUM AND ORDER
(Resolving Issues Referred by the Commission in CLI-12-20)

In its November 8, 2012 decision in CLI-12-20, the Commission referred to the Atomic Safety and Licensing Board Panel (ASLBP) a portion of the June 18, 2012 hearing request filed by Friends of the Earth (Petitioner) challenging aspects of a Confirmatory Action Letter (CAL) issued by the NRC to Southern California Edison Company (SCE) on March 27, 2012.[1] In particular, the Commission directed a duly constituted Licensing Board to "consider whether: (1) the [CAL] issued to SCE constitutes a de facto license amendment that would be subject to a hearing opportunity under [s]ection 189a [of the Atomic Energy Act (AEA)]; and, if so, (2) whether the petition meets the standing and contention admissibility requirements of 10 C.F.R. § 2.309." CLI-12-20, 76 NRC at __ (slip op. at 5).

For the reasons discussed below, we resolve the first issue in the affirmative, concluding that this CAL process constitutes a de facto license amendment proceeding that is subject to a hearing opportunity. Because this resolution provides Petitioner with all the relief its contention

[1] See Southern Cal. Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-12-20, 76 NRC __, __ (slip op. at 5) (Nov. 8, 2012).

seeks, the second issue referred by the Commission is moot, and the proceeding before this Board is therefore terminated.

I. FACTUAL AND PROCEDURAL BACKGROUND

A. Factual Background

The San Onofre Nuclear Generating Station (SONGS) is located near San Clemente, California.[2] SONGS Units 2 and 3 are pressurized water nuclear reactors with two steam generators per unit.[3] SCE is the licensee for SONGS Units 2 and 3. See Brabec Aff. at 3-4.

SCE's steam generators are recirculating, vertical U-tube type heat exchangers in which primary coolant is circulated inside the tubes, with heat from the primary-side coolant transferred to the secondary-side feedwater that circulates outside the tubes. This converts the feedwater into saturated steam that is used to drive a turbine-generator to create electricity. See Brabec Aff. at 4.

Steam generator tubes serve critical safety functions. For example, they are an integral part of the reactor coolant pressure boundary and thus are essential for maintaining primary system pressure and coolant inventory. They also isolate the radioactive fission products in the primary coolant from the secondary system.[4]

In September 2009, SCE shut down Unit 2 for a scheduled refueling outage and the replacement of its steam generators to resolve corrosion and other degradation issues in the original steam generators, which had been in service for nearly thirty years.[5] SCE completed

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[2] See [SCE's] Brief on Issues Referred by the Commission (Jan. 30, 2013) at 3 [hereinafter SCE's Answering Brief].

[3] See id., Att. 1, Affidavit of Richard Brabec (Jan. 30, 2013) at 3-4 [hereinafter Brabec Aff.]. SONGS Unit 1 ceased operation in 1992 and has since been decommissioned. See SCE's Answering Brief at 3.

[4] See SCE's Answering Brief, Att. 8 [SONGS] Unit 2 Return to Service Report (Oct. 3, 2012) at 14 [hereinafter Unit 2 Return to Service Report].

[5] See Brabec Aff. at 4; Unit 2 Return to Service Report at 10, 17; Letter from Ryan E. Lantz, Chief, Project Branch D, Division of Reactor Projects, US NRC, to Ross T. Ridenoure,

the Unit 2 refueling and steam generator replacement outage in April 2010, and that unit returned to full power in May 2010.[6]

In October 2010, SCE shut down Unit 3 for a scheduled refueling outage and the replacement of its steam generators, which also had been in service for nearly thirty years.[7] In February 2011, SCE completed the Unit 3 refueling and steam generator replacement outage, and that unit returned to full power in March 2011.[8]

The replacement steam generators for Units 2 and 3, which were manufactured by Mitsubishi Heavy Industries (MHI) (see Brabec Aff. at 4), differ in design from the original steam generators.[9] For example, each replacement steam generator (1) has 9,727 tubes, which is 377

Senior Vice President and Chief Nuclear Officer, SCE, NRC's [SONGS] – Unit 2 Steam Generator Replacement Project Inspection Report 05000361/2009007 (Mar. 4, 2010), Enclosure at 5 (ADAMS Accession No. ML100630838).

[6] See Letter from Ryan E. Lantz, Chief, Project Branch D, Division of Reactor Projects, US NRC, to Ross T. Ridenoure, Senior Vice President and Chief Nuclear Officer, SCE, NRC's [SONGS] – Unit 2 Steam Generator Replacement Project Inspection Report 05000361/20010008 (June 30, 2010), Enclosure at 3 (ADAMS Accession No. ML101810506).

[7] See Letter from Ryan E. Lantz, Chief, Project Branch D, Division of Reactor Projects, US NRC, to Peter Dietrich, Senior Vice President and Chief Nuclear Officer, SCE, NRC's [SONGS] – NRC Integrated Inspection Report 05000361/2010005 and 05000362/2010005 (Feb. 10, 2011), Enclosure at 7 (ADAMS Accession No. ML110420223).

[8] See Letter from Ryan E. Lantz, Chief, Project Branch D, Division of Reactor Projects, US NRC, to Peter Dietrich, Senior Vice President and Chief Nuclear Officer, SCE, NRC's [SONGS] – Unit 3 Steam Generator Replacement Project Inspection Report No. 05000362/2010009 (May 10, 2011), Enclosure at 3 (ADAMS Accession No. ML111300448).

[9] See SCE's Answering Brief, Att. 31, NRC Augmented Inspection Team [AIT] Report (July 18, 2012) at 36 [hereinafter July 18 AIT Report]; see also Opening Brief of Petitioner Friends of the Earth (Jan. 11, 2013) at 1, 3 [hereinafter Petitioner's Opening Brief]; Petitioner's Opening Brief, Att. 3, Far Outside the Norm: The San Onofre Nuclear Plant's Generator Problems in the Context of the National Experience with Replacement Steam Generators at 4 [hereinafter Hirsch Report]; Petition to Intervene and Request for Hearing by Friends of the Earth (June 18, 2012), Exh. 1, Declaration of Arnold Gundersen Supporting the Petition to Intervene by Friends of the Earth Regarding the Ongoing Failure of the Steam Generators at [SONGS] at 3 [hereinafter May 31 Gundersen Decl.].

SCE urges this Board to discount the Hirsch Report attached to Petitioner's Opening Brief because, in alleged disregard of the directive in this Board's December 7 Order, Petitioner "did not provide an affidavit to support the factual assertions in the Hirsch Report, which are

more tubes than are in the original; (2) does not have a stay cylinder supporting the tube sheet; and (3) has a broached tube design rather than an "egg crate" tube support.[10]

As discussed infra Part II.B.2, a licensee must obtain a license amendment from the NRC if a change to its facility triggers the safety standards described in 10 C.F.R. § 50.59. Despite the design differences mentioned above between the replacement and original steam generators, SCE concluded that the replacements were a like-for-like change that did not require a license amendment.[11]

On January 9, 2012, SCE shut down Unit 2 for a scheduled refueling outage and steam generator inspection.[12] On January 31, 2012, while Unit 2 was still shut down, Unit 3 operators received secondary plant system radiation alarms, diagnosed a steam generator tube leak of approximately 82 gallons per day, and shut down Unit 3 as required by plant procedures. See

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relied upon throughout [Petitioner's] Brief." SCE's Answering Brief at 14. Petitioner counters that an affidavit was not necessary to support the Hirsch Report because (1) it "uses data submitted to the NRC by utilities operating nuclear reactors with replacement steam generators to compare San Onofre to the experience of [replacement steam generators] nationally"; (2) it was "commissioned by Senator Barbara Boxer, Chair of the Senate Environment and Public Works Committee, and admitted into the Senate record in a joint hearing on September 12, 2012"; and (3) the NRC Commissioners "placed the Hirsch Report into the record of the Commission briefing on steam generator problems held on February 7, 2013, . . . at which Daniel Hirsch was invited to testify." Reply Brief of Petitioner Friends of the Earth (Feb. 13, 2013) at 27-28 [hereinafter Petitioner's Reply Brief]. In these circumstances, and given that SCE does not identify particular factual errors in the Hirsch Report, we decline SCE's suggestion to disregard that Report.

[10] See July 18 AIT Report at 36; see also May 31 Gundersen Decl. at 4-6; Petitioner's Opening Brief, Att. 2, Affidavit of Arnold Gundersen (Jan. 9, 2013) at 8-9 [hereinafter Gundersen Aff.]; Petitioner's Opening Brief, Att. 1, Corrected Affidavit of John H. Large (Jan. 22, 2013) at 11 [hereinafter Jan. 22 Large Aff.]. For a full description of the replacement steam generators, including a diagram, see Brabec Aff. at 4-5.

[11] See May 31 Gundersen Decl. at 7; Gundersen Aff. at 8. Although SCE did not seek a license amendment relating to the design differences of the steam generators, it did obtain a license amendment in 2009 for changes to certain "SONGS Technical Specifications related to steam generator tube integrity." SCE's Answering Brief at 6.

[12] See NRC Staff's Answering Brief in the [SONGS] CAL Proceeding (Jan. 30, 2013) [hereinafter NRC's Answering Brief], Att. 1, NRC Integrated Inspection Report 05000361/2012002 and 05000362/2012002 (May 8, 2012) at 18-19 [hereinafter May 8, 2012 Inspection Report].

May 8, 2012 Inspection Report at 39.

SCE's inspection of the Unit 3 steam generators revealed "extensive [tube-to-tube wear]" (SCE's Answering Brief at 9) that SCE determined "was caused by in-plane fluid elastic instability from the combination of localized high steam velocity, high steam void fraction, and insufficient contact forces between the tubes and the [anti-vibration bars]." Id. SCE states that

> more than 150 tubes of the 9,727 tubes in each [of the Unit 3 replacement steam generators] experienced [tube-to-tube wear], including more than 100 tubes in each [replacement steam generator] with wear equal to or greater than 35% of the width of the tube wall (which is the criterion in SONGS Technical Specification 5.5.2.11 for removal of the tube from service by plugging of the tube).

Id. (footnote omitted).[13]

Significantly, SCE acknowledges that "[tube-to-tube wear] due to in-plane [fluid elastic instability] had not been previously experienced in U-tube steam generators." SCE's Answering Brief at 10. SCE describes fluid elastic instability as

> a phenomenon in which the tubes vibrate with increasingly larger amplitudes due to the flow velocity exceeding the critical velocity for a tube, given its supporting conditions and thermal-hydraulic environment. [Fluid elastic instability] occurs when the amount of energy imparted on the tube by the fluid is greater than the amount of energy that the tube can dissipate back to the fluid and to the supports. During in-plane [fluid elastic instability], tubes within the same column are excited by the fluid and move with the plane of the column, resulting in tube-to-tube contact and wear of the tubes.

Id. at 9 (footnotes omitted).

With regard to Unit 2, SCE states, "[i]n contrast to the extensive [tube-to-tube wear] in Unit 3, [tube-to-tube wear in Unit 2] existed in only a single pair of tubes . . . in one of the two

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[13] As characterized by Petitioner, each Unit 3 steam generator "exhibited approximately 5,000+ indications of wear localities, with many tubes having wear indications at more than one locality and of differing degrees of wear severity, with a total of about 900 individual tubes affected in each [replacement steam generator]." Jan. 22 Large Aff. at 10. A total of 193 tubes in one steam generator and a total of 188 in the other exceeded the wall thinning threshold of 35%, above which tube plugging is mandatory. See id. "Because of the depth and length of certain of the tube wear scars, a number of tubes were subjected to in situ hydrostatic pressure testing in March 2012, [which] resulted in 8 individual tube failures, all located in one [replacement steam generator]." Id.; see also Hirsch Report at 4-5, 7-9.

. . . [steam generators]." SCE's Answering Brief at 9. One of SCE's contractors "concluded that the [tube-to-tube wear] in Unit 2 was not due to [fluid elastic instability], but instead to proximity of the tubes in question and random vibration of those tubes." Id. at 10. But other SCE analyses "assumed that [fluid elastic instability] could occur in Unit 2 at 100% power." Id. SCE attributes the difference in tube-to-tube wear between Units 2 and 3 to fabrication differences arising from allowable fabrication tolerances.[14] See id. at 10, 92; infra note 43.

On March 23, 2012, SCE submitted to the NRC Staff a "Steam Generator Return-to-Service Action Plan" and described actions it committed to take before restarting Units 2 and 3.[15] On March 26, 2012, the NRC Staff confirmed, by telephone, its understanding of the actions to which SCE had committed. See NRC Staff's Answering Brief at 3. On March 27, 2012, the NRC Staff memorialized its understanding in a CAL that confirmed the actions SCE would take prior to restarting either unit.[16]

As discussed in greater detail infra Part II.A.1, the NRC Staff uses a CAL to commence an enforcement process in which (as relevant here) a licensee agrees "to take certain actions to remove significant concerns regarding health and safety, safeguards, or the environment."[17] In

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[14] The extent of the tube-to-tube wear is described in the SONGS Unit 2 Return to Service Report's Steam Generator Operational Assessment for Tube-to-Tube Wear. See SCE's Answering Brief, Att. 12, SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear [hereinafter Assessment for Tube-to-Tube Wear]; see also Jan. 22 Large Aff. at 10-11; Hirsch Report at 4-6, 8-10.

[15] See SCE's Answering Brief, Att. 7, Docket Nos. 50-361 and 50-362 Steam Generator Return-to-Service Action Plan [SONGS] (Mar. 23, 2012) [hereinafter Mar. 23, 2012 Return-to-Service Plan].

[16] See SCE's Answering Brief, Att. 3, [CAL] -- [SONGS], Units 2 and 3, Commitments to Address Steam Generator Tube Degradation (Mar. 27, 2012) [hereinafter CAL].

[17] SCE Answering Brief, Att. 13, NRC Enforcement Policy (June 7, 2012) at 68 [hereinafter NRC Enforcement Policy]. The NRC Enforcement Manual describes a CAL as follows:

> [Confirmatory Action Letters (CALs)] are flexible and valuable tools available to the staff to resolve licensee issues in a timely and efficient manner, e.g., when an order is warranted to address a specific issue, a CAL is a suitable instrument to confirm initial, agreed upon, short-term actions covering the interval period prior

the instant case, the March 27, 2012 CAL provides, inter alia, that (1) SCE will take specified investigatory and corrective actions and provide information to the NRC Staff as prescribed in the CAL; and (2) SCE may not restart Units 2 and 3 until the NRC Staff has completed its review of SCE's Restart Reports and has authorized such restarts. See CAL at 2.

B. Procedural Background

On June 18, 2012, Petitioner submitted a hearing request to the Commission arising out of the Staff's issuance of the CAL.[18] Petitioner (1) requested that the Commission recognize that the CAL process for the start up of Units 2 and 3 is a de facto license amendment proceeding requiring an adjudicatory hearing (see Petition to Intervene at 2), and (2) proffered the following contention: "Petitioner contends that [SONGS] cannot be allowed to restart without a license amendment and attendant adjudicatory public hearing as required by 10 C.F.R. § 2.309, in which Petitioner and other members of the public may participate." Id. at 16.[19]

On July 13, 2012, SCE and the NRC Staff filed answers opposing Petitioner's hearing request.[20] Petitioner filed a reply to those answers on July 20, 2012.[21]

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to the actual issuance of the order.

SCE's Answering Brief, Att. 14, [NRC] Enforcement Manual (rev. 7, Oct. 1, 2010) at 3-30 [hereinafter NRC Enforcement Manual].

[18] See Petition to Intervene and Request for Hearing by Friends of the Earth (June 18, 2012) [hereinafter Petition to Intervene].

[19] Petitioner also advanced two other claims in its hearing request that are not relevant to this proceeding. See infra note 24. In the meantime, on June 27, 2012, the National Resources Defense Council (NRDC) filed a response in support of Petitioner's hearing request. See NRDC's Response in Support of FOE Petition to Intervene, San Onofre Units 2 and 3 (June 27, 2012).

[20] See [SCE's] Answer Opposing Friends of the Earth Hearing Request and the [NRDC] Response Regarding [SONGS] Unit 2 and 3 (July 13, 2012); NRC Staff's Answer to Petition to Intervene and Request for Hearing by Friends of the Earth on the Restart of the San Onofre Reactors (July 13, 2012).

[21] See Reply to SCE's and NRC Staff's Answer to Petition to Intervene and Request for

Meanwhile, consistent with its commitment in the CAL, on October 3, 2012, SCE submitted a CAL response to the NRC Staff entitled "Unit 2 Return to Service Report."[22] In that Report, SCE represented that it had taken the following corrective actions for Unit 2 and would impose the following operational limits to prevent loss of tube integrity in the steam generators due to tube-to-tube wear:

> * SCE will administratively limit Unit 2 to 70% reactor power prior to a mid-cycle inspection outage. . . . This administrative limit is temporary and may change based upon the results of inspections, further analysis and long-term corrective actions.
>
> * SCE has plugged the tubes adjacent to the retainer bars, plugged the two tubes with [tube-to-tube wear] in Unit 2, plugged the tubes with wear that exceeds the 35% through-wall criterion in SONGS Technical Specifications, and preventively plugged additional tubes in Unit 2 based on wear characteristics in Unit 3 tubes and actual wear patterns in Unit 2 (those tubes are in approximately the same region that experienced [fluid elastic instability] in Unit 3 at 100% power). . . . [A]bout 3% of the total number of tubes in each of the [steam generators] in Unit 2 have been plugged.
>
> * SCE will shut down for a mid-cycle steam generator tube inspection outage within 150 cumulative days of operation at or above 15% power.

SCE's Answering Brief at 10-11.[23]

On November 8, 2012, the Commission issued a decision on Petitioner's hearing request. As relevant here, the Commission referred to the ASLBP that portion of the request in which Petitioner argued that "the [CAL] issued to SCE, including the process for resolving the issues raised in the [CAL], constitutes a de facto license amendment proceeding." CLI-12-20, 76 NRC at __ (slip op. at 4). The Commission thus directed a duly constituted Licensing Board

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Hearing by Friends of the Earth (July 20, 2012).

[22] See SCE's Answering Brief, Att. 4, Docket No. 50-361, [CAL] – Actions to Address Steam Generator Tube Degradation [SONGS], Unit 2 (Oct. 3, 2012) [hereinafter SCE's Unit 2 Restart Plan].

[23] SCE has not yet submitted a Unit 3 Return to Service Report (see SCE's Answering Brief at 11), and it represents that "its CAL response and restart actions for Unit 3 . . . may be quite different than those for Unit 2 because the [tube-to-tube wear] in Unit 3 is far more extensive and severe than in Unit 2." Id. at 21.

to "consider whether: (1) the [CAL] issued to SCE constitutes a de facto license amendment that would be subject to a hearing opportunity under [s]ection 189a [of the Atomic Energy Act]; and, if so, (2) whether the petition meets the standing and contention admissibility requirements of 10 C.F.R. § 2.309." Id. at 5.[24]

Following its establishment on November 19, 2012,[25] this Licensing Board held a conference call on December 3, 2012 to discuss the procedural path forward, including a briefing schedule.[26] Petitioner filed its opening brief with attachments on January 11, 2013 (see Petitioner's Opening Brief); SCE and the NRC Staff each filed an answering brief with attachments on January 30, 2013 (see SCE's Answering Brief; NRC Staff's Answering Brief); and Petitioner filed its reply brief on February 13, 2013. See Petitioner's Reply Brief.[27]

On March 22, 2013, this Board held an oral argument in the ASLBP's Rockville Hearing Room on the issues referred by the Commission.[28]

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[24] As mentioned supra note 19, in its hearing request, Petitioner also advanced two additional claims, asserting that (1) SCE violated 10 C.F.R. § 50.59 insofar as it replaced the steam generators in Units 2 and 3 without seeking a license amendment; and (2) the Commission should exercise its inherent supervisory authority to initiate a discretionary adjudicatory hearing. See Petition to Intervene at 2. The Commission (1) referred Petitioner's section 50.59 claim to the NRC Executive Director for Operations for consideration as a petition under 10 C.F.R. § 2.206 (see CLI-12-20, 76 NRC at __ (slip op. at 4)); and (2) denied, without prejudice, Petitioner's request that the Commission initiate a discretionary adjudicatory hearing. See id. at 5.

[25] See Southern Cal. Edison Co., Establishment of Atomic Safety and Licensing Board, 77 Fed. Reg. 70,487 (Nov. 26, 2012).

[26] See Licensing Board Order (Scheduling Conference Call) (Nov. 26, 2012) (unpublished). This Board's subsequent procedural directives are contained in the following orders: Licensing Board Order (Conference Call Summary and Directive Relating to Briefing) (Dec. 7, 2012) (unpublished); Licensing Board Order (Granting in Part and Denying in Part Petitioner's Motion for Clarification and Extension) (Dec. 20, 2012) (unpublished).

[27] Additionally, NRDC filed an amicus brief in support of Petitioner (see [NRDC's] Amicus Response in Support of Friends of the Earth (Jan. 18, 2013)), and Nuclear Energy Institute (NEI) filed an amicus brief in support of SCE and the NRC Staff. See Amicus Curiae Brief of [NEI] in Response to the NRC [ASLBP's] Briefing Order (Jan. 30, 2013).

[28] See Official Transcript of Proceedings (Mar. 22, 2013) [hereinafter Tr.]. The oral argument was web streamed for the benefit of individuals who were unable to attend. See

II. ANALYSIS

In Part II.A, we define the scope of the de facto license amendment issue referred by the Commission, concluding that -- based on the nature of the CAL process and the language in CLI-12-20 -- the Commission tasked us with determining whether any aspect of this CAL process, including a close-out of the CAL for Unit 2 that results in a plant start-up pursuant to SCE's Unit 2 Return to Service Plan, would constitute a de facto license amendment proceeding.[29] In Part II.B, we discuss the legal standards that will guide us in resolving this issue. In Part II.C, we apply the governing legal standards to the facts of this case, and we conclude that this CAL process constitutes a de facto license amendment proceeding that triggers the hearing requirements in section 189a of the AEA. Finally, in Part II.D, we consider the second issue referred by the Commission -- i.e., whether Petitioner has standing and has submitted an admissible contention. We conclude that, because our resolution of the first issue

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Licensing Board Order (Format for Oral Argument) (Mar. 12, 2013) at 2 (unpublished).

During oral argument, SCE announced that it was "considering filing a voluntary license amendment request with a no significan[t] hazards consideration as the most expeditious method to resolve the issue raised by [Request for Additional Information] 32." See Tr. at 10. Subsequently, on April 8 and 9, 2013, respectively, SCE filed (1) a License Amendment Request for Unit 2; and (2) Supplement 1 to the License Amendment Request for Unit 2. See Docket No. 50-361, Amendment Application Number 263, Steam Generator Program, [SONGS], Unit 2 (Apr. 8, 2013); Docket No. 50-361, Supplement 1 to Amendment Application Number 263, Steam Generator Program, [SONGS], Unit 2 (Apr. 9, 2013). On April 11, 2013, the NRC Staff filed a copy of a "Notice of Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," which it had forwarded the previous day to the Office of the Federal Register for publication. See [SONGS], Unit 2 – Notice of Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination, and Opportunity for Hearing (TAC No. MF1379) (Apr. 11, 2013). No party has filed a motion suggesting that this new development materially affects this proceeding, nor do we discern such an effect, because SCE's license amendment request for Unit 2 does not fully resolve the referred issue for Unit 2 (see infra note 48), much less for Unit 3.

[29] In this decision, we focus principally on Unit 2, because SCE has not yet submitted a "Unit 3 Return to Service Report." However, because SCE concedes that the tube-to-tube wear in Unit 3 is "far more extensive and severe" than in Unit 2 (see SCE's Answering Brief at 21), our conclusion on the first referred issue (infra Part II.C) would perforce apply to Unit 3 if SCE sought to restart it without a license amendment.

grants Petitioner all the relief that its contention seeks, the second issue referred by the Commission is now moot.

A. The Scope of the *De Facto* License Amendment Issue Referred to this Board

SCE and the Petitioner disagree sharply about the scope of the first issue referred to this Board. The Commission "direct[ed] the Board to consider whether . . . the [CAL] issued to SCE constitutes a de facto license amendment that would be subject to a hearing opportunity under section 189a [of the AEA]."[30] CLI-12-20, 76 NRC at __ (slip op. at 5). SCE argues that, consistent with the above language, this Board should cabin its review to "the provisions in the [March 27, 2012 letter] itself, without recourse to SCE's CAL response or its restart actions." SCE's Answering Brief at 20. The NRC Staff agrees with SCE's narrow view of the issue. See NRC Staff's Answering Brief at 48-49.

On the other hand, Petitioner argues that the Commission referred a broader issue to this Board. Petitioner claims that the Commission viewed the CAL as a process, not as a discrete letter, and it therefore directed this Board to resolve whether any aspect of the CAL process, including a close-out of the CAL that results in a plant start-up pursuant to SCE's Unit 2 Return to Service Plan, would constitute a de facto license amendment proceeding. See Petitioner's Opening Brief at 6. This conclusion, argues Petitioner, is compelled by (1) the nature of the CAL process; (2) the plain language in CLI-12-20; and (3) common sense. See Petitioner's Opening Brief at 9-10; Tr. at 23-24. We agree with Petitioner.

1. The Nature of the CAL Process Supports Petitioner's Interpretation Regarding the Scope of the Referred Issue

SCE and the NRC Staff argue that the first issue requires us to limit our review to the four corners of the March 27, 2012 confirmatory action letter and determine whether that letter, viewed in isolation, constitutes a de facto license amendment. This argument ignores that,

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[30] The hearing opportunity mandated by section 189a of the AEA is discussed infra Part II.B.1.

although a "confirmatory action letter" can be referred to as a "CAL," the NRC Enforcement Manual also considers the term "CAL" to be a "process." See NRC Enforcement Manual at 3-32.

As described in the NRC Enforcement Manual and as explained by the NRC Staff, the CAL process involves (1) the identification of a significant concern regarding health and safety, safeguards, or the environment; (2) the NRC Staff's issuance of a specific CAL; (3) a licensee responding by taking action and/or providing information as prescribed in the CAL; and (4) when the circumstances that prompted the NRC to issue the CAL have been addressed, the closing out of the CAL.[31] See NRC Staff's Answering Brief at 31; NRC Enforcement Manual at 3-29 to 3-36; see also NRC Enforcement Policy at 68.

In the instant case, the NRC Staff's use of the CAL process serves, inter alia, to confirm SCE's "[v]oluntary . . . suspension of licensed activities" and its "agreement to NRC approval prior to resumption of licensed activities." NRC Enforcement Manual at 3-30. The March 27, 2012 letter thus states that the CAL will remain in effect until the NRC Staff (1) completes its review of SCE's tests, assessments and evaluations, corrective actions, and proposed protocol

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[31] The Enforcement Manual describes the process for closing out a CAL as follows:

> 3.5.7 Closing Out CALs
>
> A. A CAL may or may not require follow-up inspection to verify completion of the specified licensee actions. Whether the staff believes that an inspection is necessary to close a CAL will be determined on a case-by-case basis and will depend on the circumstances of the case.
>
> B. The issuing office (i.e., region, NRR, NMSS, FSME, NRO or NSIR) will issue documentation formally closing out the CAL.
>
> C. Correspondence closing out a CAL should be sent to the same person/address as the CAL; however, verbal notification, in advance of written correspondence, may be sufficient to permit plant restart or resumption of affected licensee activities.

NRC Enforcement Manual at 3-35 to 3-36.

of inspections and/or operational limits; and (2) concludes that the SONGS Units 2 and 3 can be operated without undue risk to public health and safety, and the environment. See CAL at 2, 3.

On October 3, 2012, SCE informed the NRC Staff that it had completed the actions prescribed in the March 27, 2012 letter for the restart of Unit 2, and it provided detailed information regarding fulfillment of those actions in a document entitled "Unit 2 Return to Service Report." See Unit 2 Return to Service Report.

The NRC Staff has not yet closed out the CAL for Unit 2, because it continues to review SCE's "Unit 2 Return to Service Report." Incident to that review, to date, the NRC Staff has issued over 70 Requests for Additional Information (RAIs) to SCE, while SCE has submitted 8 voluminous responses.[32]

In short, the CAL process for Units 2 and 3 is a protracted and evolving process. It will culminate in a close-out that will permit plant restart if the NRC Staff concludes such action can be accomplished without undue risk to public health and safety, and the environment.

This Board cannot determine whether that process constitutes a de facto license amendment proceeding by looking solely at the March 27, 2012 document that set this lengthy and complex process in motion. Rather, our resolution of that issue must be informed by considering the entire process and the documents generated incident to that process.

We recognize that Licensing Boards are not empowered "to supervise or direct NRC Staff regulatory reviews." Duke Energy Corp. (Catawba Nuclear Station, Units 1 and 2), CLI-04-6, 59 NRC 62, 74 (2004). Our resolution of the referred issue will not violate that rule. We do not presume to supervise or to direct the NRC Staff in the performance of its CAL duties, including its review of the adequacy and safety of SCE's restart plan; rather, the scope of our authority is limited to adjudicating the issue referred by the Commission -- i.e., whether this CAL process constitutes a de facto license amendment proceeding.

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[32] The NRC Staff issued RAIs to SCE on December 26, 2012 (RAIs 1-32), March 18, 2013 (RAIs 33-67), and March 15, 2013 (RAIs 68-72). See SCE's Eighth Notification of Responses to RAIs (Apr. 23, 2013).

The NRC Staff nevertheless argues that the CAL process "does not involve issuing [a license] amendment. Instead, closing out a CAL would 'permit plant restart or resumption of affected licensee activities.'" NRC Staff's Answering Brief at 32 (quoting NRC Enforcement Manual at 3-36). "If the licensee or Staff determined a license amendment was required," argues the NRC Staff, "that would be done separately from the CAL close-out process." NRC Staff's Answering Brief at 32 n.157.

The short answer to this argument is that "it is the *substance* of the NRC action that determines entitlement to a section 189a hearing, *not* the particular label the NRC chooses to assign to its action." Citizens Awareness Network, Inc. v. NRC, 59 F.3d 284, 295 (1st Cir. 1995). Consistent with the Commission's directive in CLI-12-20, it is this Board's responsibility to scrutinize the substance of *this CAL process* to determine whether it constitutes a de facto license amendment proceeding. To resolve that issue, our inquiry must extend to determining whether the Unit 2 Return to Service Report, in which SCE seeks a CAL close-out that "permit[s a] plant restart" (NRC Enforcement Manual at 3-36), constitutes a de facto license amendment proceeding that triggers a hearing opportunity under section 189a of the AEA.

2. The Language in the Commission's Referral Order Supports Petitioner's Interpretation Regarding the Scope of the Referred Issue

The above conclusion is compelled by the plain language in the Commission's referral order. The Commission explicitly stated that Petitioner "contend[ed] that the [CAL] issued to SCE, including the process for resolving the issues raised in the [CAL], constitutes a de facto license amendment proceeding" (CLI-12-20, 76 NRC at __) (slip op. at 4)), and it was "this portion of the petition" that the Commission referred to the ASLBP for resolution. Id. at 4-5.

Insofar as the Commission referred a de facto license amendment claim that "*includ[ed a challenge to] the process for resolving the issues raised in the [CAL]*" (CLI-12-20, 76 NRC at __ (slip op. at 4) (emphasis added)), we conclude that the referred issue requires us to determine whether this process, in which SCE seeks a CAL close-out resulting in a plant restart,

constitutes a de facto license amendment proceeding.

It is true that there can be no actual license amendment until (and unless) it is issued by the NRC Staff. See 10 C.F.R. § 50.92. It might therefore be argued that this Board should refrain from resolving the de facto license amendment issue until the Staff completes the CAL process by, for example, authorizing the start up of Units 2 and 3.

This we decline to do for three reasons. First and foremost, we see no indication in CLI-12-20 that the Commission intended this Board to stay its hand until the Staff has taken final action in the CAL process. Second, if the hearing provision in section 189a of the AEA is to serve its intended purpose, the parties in interest should be afforded a meaningful opportunity to request a hearing *before* the NRC Staff takes final action that could result in authorizing SCE to operate in a manner that is beyond the ambit of its existing license. Cf. Citizens Awareness Network, Inc., 59 F.3d at 294-95 ("[I]f section 189a is to serve its intended purpose, surely it contemplates that parties in interest be afforded a meaningful opportunity to request a hearing *before* the Commission *retroactively* reinvents the terms of an extant license by voiding its implicit limitations on the licensee's conduct."). Third, all the parties urge this Board to resolve the referred issue without awaiting final Staff action. See Tr. at 59 (SCE), 27 (Petitioner), 112 (NRC Staff). To do otherwise could result in years of delay. See Tr. at 59 (SCE advises that, in its estimation, the CAL close-out for Unit 3 is "not imminent" and is not likely to occur for several years).

3. Common Sense Supports Petitioner's Interpretation Regarding the Scope of the Referred Issue

Common sense also supports the conclusion that the Commission did not intend this Board to limit its review to the four corners of the March 27, 2012 confirmatory action letter. Otherwise, it would have resolved the issue itself, concluding -- without difficulty -- that this austere four-page document, viewed in isolation at the incipient stage of the CAL process, does *not* constitute a de facto license amendment.

However, by referring the issue to the ASLBP, and by acknowledging that Petitioner's claim "include[ed] the *process* for resolving the issues raised in the [CAL]" (CLI-12-20, 76 NRC at __ (slip op. at 4) (emphasis added)), it may fairly be concluded that the Commission intended a Licensing Board to examine the entire CAL process, and to determine whether any aspect of that process -- including a close-out of the CAL that results in a plant start up pursuant to SCE's Unit 2 Return to Service Plan -- constitutes a de facto license amendment proceeding.[33]

SCE advances a policy reason in support of its argument that this Board should focus exclusively on the March 27, 2012 CAL and conclude that it is not a de facto license amendment. Namely, to do otherwise may discourage licensees in the future from agreeing to a CAL, thus (1) diminishing the NRC Staff's use of this important regulatory tool in the future; and (2) undermining the Staff's discretion to select the enforcement action that best fits the factual circumstances. See SCE Brief at 20-23.

This argument lacks merit. First, whether a CAL process constitutes a de facto license amendment proceeding is a highly fact-specific question, and there is no reason to believe that this Board's resolution of this fact-specific issue in this exceptionally unusual case will influence other licensees when they are considering whether to agree to a CAL. Second, "unreviewed Board rulings do not constitute precedent or binding law" (Baltimore Gas & Elec. Co. (Calvert Cliffs Nuclear Power Plant, Units 1 and 2), CLI-98-25, 48 NRC 325, 343 n.3 (1998)), which fortifies our conclusion that our resolution of the referred issue in this unique case will not impact the decision-making process of other licensees when they are considering whether to agree to a CAL. Finally, and dispositively, SCE's policy argument cannot trump the Commission's directive in CLI-12-20 that a Licensing Board examine this CAL process and determine whether it

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[33] We thus agree with the NRC Staff's assertion (see NRC Staff's Answering Brief at 35) that if we were to limit our review to the March 27, 2012 letter, we would conclude that this document, viewed in isolation, is not a de facto license amendment. In our judgment, however, the Commission eschewed such a facile analytic approach by referring Petitioner's claim to the ASLBP, "including the process for resolving the issues raised in the CAL." CLI-12-20, 76 NRC at __ (slip op. at 4).

constitutes a de facto license amendment proceeding.

B. Legal Standards That Address License Amendments

1. Relevant Statutory Provisions Related to License Amendments

It is imperative that the terms of a reactor operating license be clear and unambiguous, and also that a licensee scrupulously adhere to those terms, because section 101 of the AEA makes it "unlawful . . . for any person within the United States to . . . use . . . any utilization . . . facility except under and in accordance with a license issued by the Commission." 42 U.S.C. § 2131.[34]

Section 182a of the AEA addresses what must be included in a reactor operating license. It states that such licenses must include "technical specifications" that include, inter alia, "the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material . . . will provide adequate protection to the health and safety of the public." 42 U.S.C. § 2232(a).[35]

The Commission is empowered to issue an order amending any license as it deems necessary to "effectuate the provisions of [the AEA]" (42 U.S.C. § 2233) -- that is, to "promote the common defense and security or to protect health or to minimize danger to life or property." Id. § 2201; see also id. § 2237. Additionally, the Commission "may at any time . . . before the expiration of the license, require further written statements [from the licensee] to determine whether . . . a license should be modified." Id. § 2232(a).

Finally, section 189a of the AEA states that "[i]n any proceeding under [the AEA], for the

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[34] A "utilization facility" includes a commercial nuclear power reactor. See 10 C.F.R. § 50.2.

[35] "The AEA, however, leaves it up to the Commission to determine, and prescribe by rule or regulation, what additional information should be included in technical specifications to ensure public health and safety and the common defense and security." Dominion Nuclear Connecticut, Inc. (Millstone Nuclear Power Station, Units 2 and 3), CLI-01-24, 54 NRC 349, 351 (2001).

. . . amending of any license . . ., the Commission shall grant a hearing upon the request of any person whose interest may be affected by the proceeding, and shall admit any such person as a party to such proceeding." 42 U.S.C. § 2239(a)(1)(A).

2. Relevant Regulatory Provisions Related to License Amendments

10 C.F.R. §§ 50.90 to 50.92 provide the applicable process when a licensee wishes to request a license amendment. Specifically, section 50.90 authorizes applications to amend existing operating licenses; section 50.91 provides for notice and comment regarding license amendment applications, as well as consultation with the State in which the facility is located; and section 50.92 provides the standard considered by the NRC when determining whether to issue an amendment.

Section 50.59 establishes standards for a licensee to request a license amendment before it may make "changes in the facility as described in the [updated] final safety analysis report [UFSAR[36]], make changes in the procedures as described in the [UFSAR], and conduct tests or experiments not described in the [UFSAR]." 10 C.F.R. § 50.59(c)(1). Section 50.59 states that a licensee need not request a license amendment pursuant to section 50.90 if "(i) A change to the technical specifications incorporated in the license is not required, and (ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section." Id. § 50.59(c)(1)(i)-(ii).

Restated, a licensee *must* request a license amendment if the proposed action requires that existing technical specifications be changed (see 10 C.F.R. § 50.59(c)(1)(i)),[37] or if a

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[36] A final safety analysis report (FSAR) is part of the application for an operating license, and it contains "a description of the facility; the design bases and limits on operation; and the safety analysis for the structures, systems, and components (SSC) and of the facility as a whole." Changes, Tests, and Experiments: Proposed Rule, 63 Fed. Reg. 56,098, 56,099 (Oct. 21, 1998). "When a plant is licensed, the NRC states in its Safety Evaluation Report (SER) why it found each FSAR analysis acceptable." Id. Licensees must periodically update their FSARs to reflect changes to the facility "so that the [updated FSAR (UFSAR)] remains a complete and accurate description and analysis of the facility." Id.

[37] Because changes to technical specifications require a license amendment, the

change, test, or experiment satisfies any of the eight criteria in section 50.59(c)(2). See id. § 50.59(c)(1)(ii). The section 50.59(c)(2) criteria require a licensee to seek a license amendment if the proposed change, test, or experiment would

> (i) Result in more than a minimal increase in the frequency of occurrence of any accident previously evaluated in the [UFSAR];
>
> (ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the [UFSAR];
>
> (iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the [UFSAR];
>
> (iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the [UFSAR];
>
> (v) Create a possibility for an accident of a different type than any previously evaluated in the [UFSAR];
>
> (vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the [UFSAR];
>
> (vii) Result in a design basis limit for a fission product barrier as described in the [UFSAR] being exceeded or altered; or
>
> (viii) Result in a departure from a method of evaluation described in the [UFSAR] used in establishing the design bases or in the safety analyses.

Id. § 50.59(c)(2).[38]

Commission has instructed that technical specifications should be limited to "'those plant conditions most important to safety.'" Millstone, CLI-01-24, 54 NRC at 360 (quoting Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, 58 Fed. Reg. 39,132, 39,135 (July 22, 1993)). Thus, technical specifications "should be reserved for those reactor operation 'conditions or limitations . . . necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health or safety.'" Id. at 361 (quoting Technical Specifications, Final Rule, 60 Fed. Reg. 36,953, 36,957 (July 19, 1995)). See also 10 C.F.R. § 50.36 (identifying criteria to be used in determining what items must be included in technical specifications).

[38] The term "design bases" to which section 50.59(c)(2)(vii) and (viii) refer is defined as follows:

> Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for a design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or

Finally, 10 C.F.R. § 2.105 implements the hearing opportunity provision for license amendment procedures that is mandated by section 189a of the AEA, and Subpart C of 10 C.F.R. Part 2 contains the general rules governing hearing requests and subsequent hearing-related activities.

In sum, Congress has commanded that licensees may not, under penalty of law, deviate from the terms of their reactor operating licenses. See 42 U.S.C. § 2131. If a licensee is unable to operate a reactor in strict accordance with its license, it must seek authorization from the NRC for a license amendment (10 C.F.R. §§ 50.59, 50.90 to 50.92), which is a process that triggers a right to request an adjudicatory hearing by persons whose interests may be affected by the proceeding. See 42 U.S.C. § 2239(a)(1)(A); 10 C.F.R. § 2.105.

3. *De Facto* License Amendments

As shown above, amending a license is, by design, a carefully considered process that is closely regulated by the NRC and in which "any person whose interest may be affected" is entitled to request a hearing. 42 U.S.C. § 2239(a)(1)(A). As discussed below, however, there have been occasions when the NRC has taken action that effectively constituted a license amendment, but it failed to recognize that its actions effectively amended the license.

In other words, there have been occasions when the NRC has -- without formally amending a license and without providing the public with the opportunity for a hearing as required by section 189a of the AEA -- authorized activity by the licensee that was incompatible with the statutory requirement that the facility operate "in accordance with" its existing operating license. 42 U.S.C. § 2131. Such NRC action is characterized as a de facto license amendment. According to Petitioner, this CAL process is a de facto license amendment

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> (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

10 C.F.R. § 50.2.

proceeding because SCE seeks effectively to amend its license via the CAL process.

Specifically, Petitioner argued to the Commission that "the [CAL] issued to SCE, including the process for resolving the issues raised in the [CAL], constitutes a de facto license amendment proceeding within the hearing provision of section 189a of the AEA, and therefore an adjudicatory hearing is required." CLI-12-20, 76 NRC at __ (slip. op at 4). The Commission referred that claim to the ASLBP for resolution. Id.

Determining whether the CAL process constitutes a de facto license amendment proceeding "is a highly fact-specific question." NRC Staff's Answering Brief at 10. Case law, however, provides a straight-forward analytic framework for assessing the relevant facts. For example, in Cleveland Elec. Illum. Co. (Perry Nuclear Power Plant), CLI-96-13, 44 NRC 315 (1996), the Commission considered whether the NRC Staff's decision to authorize changes to a material specimen withdrawal schedule was a de facto license amendment. Examining decisions from the U.S. Courts of Appeals for the First Circuit and the District of Columbia Circuit, the Commission distilled the following factors that are material to determining whether NRC actions constitute a de facto license amendment:

> In evaluating whether challenged NRC authorizations effected license amendments within the meaning of section 189a, courts repeatedly have considered the same key factors: did the challenged approval grant the licensee any "greater operating authority," or otherwise "alter the original terms of a license"? If so, hearing rights likely were implicated. For example, in Citizens Awareness Network, Inc. v. NRC, 59 F.3d 284, 295 (1st Cir. 1995) (CAN), . . . the court found that the challenged NRC approval "undeniably *supplement[ed]*" the original license. The agency had permitted the licensee to dismantle major structural components, an activity that the court found unauthorized by the original license and agency rules. Similarly, in another case [San Luis Obispo Mothers for Peace v. NRC, 751 F.2d 1287 (D.C. Cir. 1984) (SLO)], where the NRC Staff extended the duration of a low-power license, a reviewing court viewed the Staff approval to be a license amendment changing a term of the license, and therefore triggering an opportunity for a hearing under section 189a.

44 NRC at 326-27 (footnotes omitted). Guided by CAN and SLO, the Commission in Perry considered whether the Staff's action (1) "alter[ed] the . . . license," or (2) "permit[ted] the licensee to operate 'in any greater capacity' than [the original license prescribes]." Id. After

examining the relevant terms and technical specifications in the license, the Commission resolved both inquiries in the negative.[39]

As illustrated in the Perry case, a de facto license amendment claim typically involves a tribunal "looking backward" to determine whether action already taken by the NRC Staff effectively constituted a license amendment. Here, however, consistent with the Commission's referral order, we are tasked with looking at an ongoing CAL process to determine whether that process constitutes a de facto license amendment proceeding. See supra Part II.A. To resolve that issue, this Board must determine whether the requested change in authority to operate Unit 2 sought by SCE pursuant to the CAL process is strictly "in accordance with" the terms and technical specifications in its existing license. 42 U.S.C. § 2131.

In other words, this Board must consider the following connate factors: whether SCE's start-up request, if granted, would permit SCE to operate (1) in a manner that deviates from a technical specification in its existing license; (2) beyond the ambit, or outside the restrictions, of its existing license; or (3) in a manner that is neither delineated nor reasonably encompassed

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[39] For additional pronouncements on standards employed by tribunals in the context of considering de facto license amendment issues, see, e.g., Perry, CLI-96-13, 44 NRC at 319 ("Because technical specifications are an integral part of an operating license, changes to technical specifications require a license amendment."); id. at 320 (the UFSAR "can be modified without a license amendment, so long as the modifications do not involve a change to the technical specifications or an unreviewed safety question"); CAN, 59 F.3d at 294 ("[B]y its nature a license is presumptively an *exclusive* -- not an inclusive -- regulatory device. . . . Regulated conduct which is neither delineated, nor reasonably encompassed within delineated categories of authorized conduct, presumptively remains unlicensed."); id. at 295 (NRC's actions constitute de facto license amendment when they authorize licensee to "engage in [activities] beyond the ambit of [its] original license"); Massachusetts v. NRC, 878 F.2d 1516, 1520-21 (1st Cir. 1989) (NRC's actions in requiring 47 improvements, granting an exemption from emergency drills, and lifting a license suspension did not require a license amendment, because the licensee can "operate[] in accordance with its unaltered license" and need not be "exempted . . . from following a specific license requirement"); In re Three Mile Island Alert, Inc., 771 F.2d 720, 729 (3d Cir. 1985) (NRC's lifting of license suspension and authorizing restart under stipulated restrictions was not a license amendment because "nothing in this record . . . indicates . . . that license amendments are necessary to permit the licensee to operate in accordance with the restrictions which have been imposed"), cert. denied, 475 U.S. 1082 (1986).

within the prescriptive terms of its existing license. See supra note 39 and accompanying text.[40]

In assessing the referred issue, this Board can refer to 10 C.F.R. § 50.59, which -- as discussed supra Part II.B.2 -- identifies situations where a licensee *must* request a license amendment. In our view, reference to the criteria in section 50.59 is eminently appropriate here, because the ultimate question before this Board is whether SCE's request that the Staff close out the CAL by permitting a plant restart constitutes a de facto license amendment proceeding that triggers a hearing opportunity under section 189a of the AEA. To resolve this question, we must look at SCE's Unit 2 Return to Service Plan to determine whether SCE is seeking authority from the NRC Staff to deviate from a technical specification or to otherwise operate in a manner that is beyond the ambit, or inconsistent with the prescriptive terms, of its existing license. Section 50.59 establishes standards that may guide this Board in resolving that issue.

Contrary to arguments advanced by the NRC Staff (see NRC Staff Answer at 43-47; Tr. at 140), the fact that section 50.59 is designed for a licensee to determine whether it must seek a license amendment ab initio poses no impediment to this Board referring to those same regulatory standards as guides in determining whether this CAL process constitutes a de facto license amendment proceeding. The standards in section 50.59 -- which establish when a "licensee shall obtain a license amendment" (10 C.F.R. § 50.59(c)(2)) -- have the imprimatur of the Commission and therefore, a fortiori, are appropriate guides for determining whether SCE's Unit 2 Return to Service Plan requires a license amendment, thereby converting the CAL process into a de facto license amendment proceeding.

Our use of section 50.59 as a tool in resolving the referred issue is to be distinguished from scrutinizing the actual actions taken by SCE under section 50.59. The latter is prohibited

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[40] At the March 22, 2013 oral argument, counsel for the NRC Staff was asked whether the need for a license amendment is limited to circumstances that involve an *increase* in licensing authority, or whether a license amendment would also be required where, for example, the Staff were to change the licensing authority by decreasing the maximum operating thermal power for a nuclear reactor. Counsel responded that a license amendment would be required for both situations. See Tr. at 130.

by case law, which establishes that "[a] member of the public may challenge an action taken under 10 C.F.R. § 50.59 only by means of a petition under 10 C.F.R. § 2.206." Yankee Atomic Elec. Co. (Yankee Nuclear Power Station), CLI-94-3, 39 NRC 95, 101 n.7 (1994). Contrary to the NRC Staff's assertion (see NRC Staff Answer at 44-49; Tr. at 141), any reference we might make to section 50.59 will not run afoul of this rule, because the issue presented here is not a challenge to SCE's previous actions taken under section 50.59.[41] Rather, the Commission directed us to determine whether this CAL process constitutes a de facto license amendment proceeding. To resolve this issue, it is manifestly appropriate for this Board to consider, and to be guided by, all relevant analytic tools, including -- if warranted -- the standards in section 50.59. Cf. Tr. at 31-32, 59-60 (SCE and Petitioner both agree that this Board can properly refer to section 50.59 for purposes of resolving whether this CAL process constitutes a de facto license amendment proceeding).

C. This CAL Process Constitutes a *De Facto* License Amendment Proceeding

We turn now to the first of the two issues referred by the Commission: whether this CAL process for the start up of SONGS Unit 2 constitutes a de facto license amendment proceeding.[42] As discussed supra Part II.B.3, to constitute a de facto license amendment proceeding, this CAL process must involve proposed actions by SCE that, if authorized, would allow SCE to deviate from a technical specification or otherwise operate Unit 2 in a manner that is inconsistent with existing licensing requirements or restrictions. We conclude that this CAL process constitutes a de facto license amendment proceeding for the following three independent reasons:

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[41] Indeed, it is impossible on the present record -- as a legal and factual matter -- for Petitioner to challenge, or for this Board to review, SCE's section 50.59 analysis for the Unit 2 Return to Service Plan because a copy of SCE's analysis has not even been filed with this Board.

[42] As stated supra note 29, although our analysis focuses on Unit 2, it would necessarily apply to Unit 3 if SCE sought to restart it without a license amendment.

(1) The restart of Unit 2 would grant SCE authority to operate without the ability to comply with all applicable technical specifications;

(2) The restart of Unit 2 would allow SCE to operate beyond the scope of its existing license; and

(3) SCE's Unit 2 Return to Service Plan includes a test or experiment that meets the criteria in 10 C.F.R. § 50.59 that require a license amendment.

Below, we provide a factual backdrop for our analysis, after which we discuss each of the above reasons in turn.

The unprecedented extent of tube wear and failures that SCE experienced in the SONGS Unit 3 replacement steam generators reveal that these steam generators have serious design and operational issues (see SCE's Answering Brief at 10; supra Part I.A), placing them beyond the envelope of experience with U-tube steam generators. SCE's investigation into the cause of the multiple tube leaks indicates that the design is prone to tube-to-tube wear caused by in-plane fluid elastic instability, which "had not been previously experienced in U-tube steam generators." SCE's Answering Brief at 10.

As mentioned supra Part I.A, fluid elastic instability results from the combination of localized high steam velocity, high steam void fraction, and insufficient contact forces between the tubes and the anti-vibration bars. The fluid elastic instability caused vibration of steam generator tubes in the in-plane direction resulting in rapid, localized tube wear. See SCE's Unit 2 Restart Plan at 2; Assessment for Tube-to-Tube Wear at 15.

"In contrast to the extensive [tube-to-tube wear] in Unit 3, [tube-to-tube wear in Unit 2] existed in only a single pair of tubes . . . in one of the two [replacement steam generators]." SCE's Answering Brief at 9. Although the Unit 2 steam generators did not experience the accelerated and extensive tube-to-tube wear suffered in the Unit 3 steam generators, they nevertheless are the identical design as those in Unit 3 and they operate under similar conditions. See SCE's Answering Brief, Att. 18, SONGS UFSAR Excerpt at 5.4-20 [hereinafter SONGS UFSAR]; Brabec Aff. at 4-6, 18.

SCE claims that the fact that steam generator tube-to-tube wear was significantly less in Unit 2 than in Unit 3 is attributable to the differences in meeting fabrication tolerances. See SCE's Answering Brief at 10, 92. Fabrication tolerances permit small differences between components designed to the same specifications, and SCE attributes the large difference in steam generator operational performance to very small differences in their construction.[43]

More precisely, SCE asserts that the difference in steam generator tube wear between Unit 3 and Unit 2 is due in large part to differences in contact between the steam generator tubes and the anti-vibration bars arising from differences in meeting fabrication tolerances. SCE explains the role played by anti-vibration bars in preventing in-plane vibrations as follows: "The effect of flat bar supports with small clearance is to act as apparent nodal points for flow-induced tube response. They not only prevent out-of-plane mode as expected but also in-plane modes." Assessment for Tube-to-Tube Wear at 17.

But "[w]ear at [anti-vibration bar] locations will degrade in-plane support effectiveness over time." Assessment for Tube-to-Tube Wear at 104. Such degradation can be caused "by a combination of turbulence and out-of-plane fluid-elastic excitation." Id. at 15. As contact is lost between the tube and the bar, the restraining effect of the anti-vibration bars in the in-plane direction decreases. These decreases, when combined with certain thermal hydraulic conditions, allow in-plane vibration and tube-to-tube wear to develop over time at locations

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[43] Manufacturing of components is never perfectly exact. Thus, if the nominal design specifies a required distance between adjacent steam generator tubes, it will also specify how closely the manufacturer must come to that required distance. This permitted variance from the design is referred to as the fabrication tolerance. See SONGS Unit 2 Return to Service Report, Att. 6 – App. D, Operational Assessment of Wear Indications in the U-bend Region of San Onofre Unit 2 Replacement Steam Generators at 100-02 (ADAMS Accession No. ML12285A269, which is entitled "Attachment 6: Appendix A: Estimate of FEI-Induced TTW Rates" on ADAMS, but also contains Appendix D, starting on page 78 of 209 of the ADAMS portable document format (PDF) version). Ironically, SCE indicates that the steam generators for Unit 3 were built more closely to design specifications than those in Unit 2, and it maintains that this greater manufacturing precision rendered the Unit 3 steam generators more susceptible to in-plane tube vibration. See SCE's Answering Brief at 92; accord Unit 2 Return to Service Report at 36.

where it previously had not occurred. See id. at 104; SONGS Unit 2 Return to Service Report, Att. 6 – App. B, SONGS U2C17 Generator Operational Assessment for Tube-to-Tube Wear at 21 (ADAMS Accession No. ML12285A268). Moreover, tube-to-tube wear "due to in-plane fluid elastic instability is a unique degradation mechanism because one unstable tube can drive its neighbor into instability through repeated impact events." Assessment for Tube-to-Tube Wear at 18. It is thus possible for in-plane instability to develop in a single tube and propagate to a larger number of tubes in the vicinity.

Wear of steam generator tubes is of critical importance to evaluations performed in the FSAR, because the tubes are part of the reactor coolant pressure boundary, and assurance of their integrity is required by General Design Criterion 14.[44] Numerous analyses are grounded on the assumed integrity of steam generator tubes, and technical specifications exist to assure their integrity.[45] Any new phenomenon that could negatively impact tube integrity can affect, and possibly negate, assumptions used in FSAR analyses.

SCE and its contractors have evaluated the in-plane tube-to-tube wear due to fluid elastic instability and have developed a theory to explain its occurrence and to predict how it can be avoided. As a result of comparing the thermal hydraulic conditions in the SONGS replacement steam generators with those of other steam generators, SCE concluded that the likelihood of fluid elastic instability will decrease if the steam quality in the steam generators is reduced (i.e., if the moisture content of the steam is increased). See Unit 2 Return to Service Report at 37. SCE determined that a reduced steam quality results in greater "damping" within the steam generators, which decreases the potential for fluid elastic instability. See id.

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[44] 10 C.F.R. Part 50, App. A – General Design Criteria for Nuclear Power Plants, Criterion 14, states: "Reactor Coolant Pressure Boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

[45] See, e.g., SCE's Answering Brief, Att. 9, SONGS Technical Specification 5.5.2.11, Steam Generator Program [hereinafter SONGS Unit 2 Technical Specifications].

SCE provided the following explanation regarding the relation between steam quality and damping, and the effect of damping on fluid elastic instability:

> Damping is the result of energy dissipation and delays the onset of [fluid elastic instability]. Damping is greater for a tube surrounded by liquid compared to a tube surrounded by gas. Since quality describes the mass fraction of a vapor in a two-phase mixture, it provides insight into the fluid condition surrounding the tube. A higher steam quality correlates with dryer conditions and provides less damping. Conversely, lower steam quality correlates with wetter conditions resulting in more damping, which decreases the potential for [fluid elastic instability].

Unit 2 Return to Service Report at 38.

When compared to steam generators at other plants that do not experience fluid elastic instability, SCE calculated that the steam quality in the SONGS replacement steam generators was higher when operated at 100% power. On the other hand, when SONGS steam generators were operated at 70% power, steam quality was in the same range as those steam generators that did not experience fluid elastic instability. See Assessment for Tube-to-Tube Wear, Figures 4-3 and 5-1.

SCE concluded that limiting the power generated at SONGS Unit 2 to 70% would reduce steam quality and hydrodynamic pressure to values that would eliminate the thermal hydraulic conditions that cause fluid elastic instability and associated tube-to-tube wear in the SONGS Unit 2 steam generators. See SCE's Unit 2 Restart Plan at 3; Unit 2 Return to Service Report at 37.[46]

SCE's most recent assessment indicates that, after operating for less than two years (i.e., 20.6 months), tube integrity for the Unit 2 steam generators can be guaranteed only for another eleven months of operation at 100% power. See SCE's Fifth Notification of Responses to RAIs, Enc. 1, Docket No. 50-361, Operational Assessment for 100% Power Case Regarding

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[46] See also Transcript of Briefing Before Commission on Steam Generator Tube Degradation (Feb. 7, 2013) at 48 (MHI agrees that a reduction to 70% power would improve the thermal hydraulic condition in the steam generators by reducing the steam quality and bringing it into a range seen in other steam generators manufactured by MHI).

[CAL] Response (TAC No. ME9727) [SONGS], Unit 2 (Mar. 14, 2013) [hereinafter SCE's Fifth Notification of Responses to RAIs].

Against the above backdrop, we explain below why we conclude that this CAL process is a de facto license amendment proceeding.

1. Under SCE's Return to Service Plan, Unit 2 Cannot be Operated "Over the Full Range Of Normal Operating Conditions" Up to 100% Power, Which is Inconsistent with a Technical Specification and Therefore Requires a License Amendment

SONGS Unit 2 Technical Specification 5.5.2.11b.1 requires that "[a]ll inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents."[47] Under its current license, SCE is authorized to operate Unit 2 up to 3,438 megawatts thermal, which is defined as 100% power. See SCE's Answering Brief, Att. 19, SONGS Operating License 226 at 3.

In its Unit 2 Return to Service Report, SCE proposes administratively to limit Unit 2 to 70% reactor power prior to a mid-cycle inspection outage. See SCE's Unit 2 Restart Plan at 3. Based on its analyses, asserts SCE, a 70% power-level limit will provide adequate margin to preclude the onset of in-plane fluid elastic instability and excessive tube wear. See id.

If, pursuant to the CAL process, the NRC Staff were to authorize SCE to operate Unit 2 at a power limit not to exceed 70%, this condition would result in a deviation from the technical specification requirement that tube integrity be maintained over the "full range of normal operation conditions" up to 100%. Such a deviation from a technical specification requires a license amendment, thus converting this CAL process to a de facto license amendment proceeding.[48]

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[47] See NRC's Answering Brief, Att. 8, Docket No. 50-361, SONGS Unit 2 Facility Operating License No. NPF-10 Excerpts at 5.0-14.

[48] In SCE's license amendment request for Unit 2 (see supra note 28), SCE seeks the following licensing revisions:

2. Unit 2 Cannot Operate Within the Scope of its Operating License,[49] Which Requires that the License Must be Amended

SONGS Unit 2 is currently licensed to operate anywhere in the normal power range from 0% to 100% power with steam generators that meet the original design specifications. The original steam generators in SONGS Unit 2 (and Unit 3) were replaced without a license amendment arising from design differences, which SCE claims was in compliance with 10 C.F.R. § 50.59. See Tr. at 79-81. As discussed in greater detail supra Part II.B.2, section 50.59 permits changes with respect to components (i.e., steam generators) without a license amendment under prescribed conditions that assure the replacement components are sufficiently similar to the original so that safety requirements are maintained or improved. See 10 C.F.R. § 50.59(c)(2).

The replacement steam generators in SONGS Unit 3, however, unexpectedly demonstrated significant in-plane vibrations due to fluid elastic instability. The vibrations were

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> The proposed amendment requests that Technical Specification 5.5.2.11.b.1 be revised to add a footnote to require that compliance with the steam generator structural integrity performance criterion (SIPC) be demonstrated up to 70% Rated Thermal Power (2406.6 megawatts thermal) and that Facility Operating License Condition 2.C(1) "Maximum Power Level" be revised to add a footnote to restrict operation of SONGS Unit 2 to no more than 70% Rated Thermal Power for the SONGS Unit 2, Cycle 17.

Docket No. 50-361, Amendment Application Number 263, Steam Generator Program [SONGS], Unit 2 (Apr. 5, 2013) at 1. Although SCE's license amendment request addresses the first reason underlying our conclusion that this CAL process constitutes a de facto license amendment proceeding, it does not address the alternative reasons underlying our conclusion (see infra Parts II.C.2 and II.C.3) and it, thus, does not fully address, much less moot, the first issue referred by the Commission.

[49] Although the term "scope of an operating license" does not have a regulatory definition, it is a useful concept in the instant context, because the Court of Appeals for the First Circuit has held that actions by the NRC Staff constitute a de facto license amendment when they authorize a licensee to "engage in [activities] beyond the ambit [i.e., scope] of [its] original license." CAN, 59 F.3d at 295; accord Perry, CLI-96-13, 44 NRC at 327. As described by the Commission, an operating license reflects a specific facility-design basis, a safety analysis documented in an FSAR, facility-specific technical specification, and NRC regulations. See 63 Fed. Reg. 56,098, 56,099-100. These factors comprise the scope of an operating license as we use the term in this Memorandum and Order.

severe enough to cause tube-to-tube contact resulting in accelerated wear of the tube wall and premature wall failure. See Assessment for Tube-to-Tube Wear at 18. This phenomenon has never before been seen in a U-tube steam generator (see SCE's Answering Brief at 10), which supports a conclusion that the replacement steam generators differ in significant respects from the originals. Because the Unit 3 steam generators are identical in design to the Unit 2 steam generators (see SONGS UFSAR at 5.4-20; Brabec Aff. at 4-6, 18), we conclude that the latter steam generators likewise differ in significant respects from the originals.

Concerning the FSAR analysis of steam generator tube integrity, SCE states that "[t]he original analysis was fine if we had simply received steam generators that met our specifications" (i.e., were like-for-like replacements), but "[w]hat we had is a degraded or nonconforming condition in our steam generators where they did not perform per the procurement specifications." See Tr. at 98. The extent to which the replacement steam generators failed to perform per the procurement specifications is graphically illustrated by the fact that the original steam generators lasted about twenty-eight years, whereas SCE's most recent operational assessment indicates that, after less than two years of operation (i.e., 20.6 months), tube integrity for Unit 2 steam generators can be guaranteed only for another eleven months of operation at 100% power. See SCE's Fifth Notification of Responses to RAIs.

Significantly, the UFSAR for the original steam generators for SONGS Units 2 and 3 excluded the possibility of in-plane vibrations caused by fluid elastic instability when evaluating the conditions necessary to maintain steam generator tube integrity. In this regard, the UFSAR states:

> The steam generator was designed to ensure that critical vibration frequencies are well out of the range expected during normal operation and during abnormal conditions. The tubing and tubing supports are designed and fabricated with considerations given to both secondary side flow-induced vibration and reactor coolant pump-induced vibrations.

SONGS UFSAR at 5.4-21;[50] see also id. at 5.4-23 to 5.4-26 (analysis in section 5.4.2.3.1.3 evaluating conditions necessary to maintain tube integrity in the original steam generators based on the assumption that vibrations caused by in-plane fluid elastic instability will not occur).

However, the UFSAR assumption for the original steam generators that in-plane vibrations caused by fluid elastic instability were excluded by design is demonstrably unjustified for the replacement steam generators. This renders inadequate the UFSAR section 5.4.2.3.1.3 analysis of steam generator tube integrity, which places the replacement steam generators outside the scope of the operating license.[51]

We conclude that until the tube degradation mechanism is fully understood, until reasonable assurance of safe operation of the replacement steam generators is demonstrated, and until there has been a rigorous NRC Staff review appropriate for a licensing action, the operation of Unit 2 would be outside the scope of its operating license because the replacement steam generator design must be considered to be inconsistent with the steam generator design specifications assumed in the FSAR and supporting analysis. In short, the start-up of Unit 2 pursuant to the CAL process would transform that process into a de facto license amendment

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[50] The reference in the UFSAR to "critical vibration frequencies" and "secondary side flow-induced vibration" subsume the in-plane vibrations caused by fluid elastic instability experienced in the SONGS replacement steam generators. See generally SONGS Unit 2 Return to Service Report, Att. 6 – App. D, Operational Assessment of Wear Indications in the U-bend Region of San Onofre Unit 2 Replacement Steam Generators at 10-12 (ADAMS Accession No. ML12285A269, which is entitled "Attachment 6: Appendix A: Estimate of FEI-Induced TTW Rates" on ADAMS, but also contains Appendix D, starting on page 78 of 209 of the ADAMS portable document format (PDF) version); cf. SCE's Answering Brief, Att. 5, MHI Document L5-04GA564 Tube Wear of Unit-3 RSG Technical Evaluation Report at 11 (MHI states that incident to the design of the SONGS replacement steam generators, "only out-of-plane vibration of the [steam generator] U-tubes was evaluated").

[51] The purpose of the UFSAR section 5.4.2.3.1.3 analysis is to verify that General Design Criterion 14 -- which concerns maintaining integrity of the reactor coolant pressure boundary (see supra note 44) -- is satisfied. We now know that General Design Criterion 14 cannot be satisfied for the steam generator tubes without an analysis of in-plane fluid elastic instability.

proceeding by allowing steam generator operation with a tube degradation mechanism not considered in the FSAR – i.e., in-plane vibrations due to fluid elastic instability.[52]

3. A Unit 2 Start-Up Pursuant to SCE's Return to Service Report Would Result in SCE Conducting a Test or Experiment Pursuant to 10 C.F.R. § 50.59(c)(2)(viii), Which Requires a License Amendment

In Part II.B.3 supra, we determined that we may use the standards in section 50.59 -- which establish when a "licensee shall obtain a license amendment" (10 C.F.R. § 50.59(c)(2)) -- as guidance to determine whether implementation of SCE's Unit 2 Return to Service Report requires a license amendment. As relevant here, section 50.59 requires a licensee to seek a license amendment before implementing a "test or experiment" that will "[r]esult in a departure from a method of evaluation described in the [UFSAR] used in establishing the design basis or in the safety analysis." 10 C.F.R. § 50.59(c)(2)(viii). Guided by that provision, we conclude that the authority to operate sought by SCE in its Unit 2 Return to Service Report is such a "test or experiment" that requires a license amendment and, thus, transforms this CAL process into a de facto license amendment proceeding.[53]

SCE's analysis of the cause of the excessive tube wear and the measures it proposes to implement to preclude such wear are based on a theory as applied to U-tube steam generators,

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[52] The required change to the current FSAR analysis is that it must be augmented with a vibration analysis to assure that steam generator tubes do not fail prematurely due to tube-to-tube wear and that tubes are thus able to satisfy their design bases. As the Commission has explained, a licensee must seek a license amendment "at the point in time [when] the revised method [in the FSAR] becomes the means used for purposes of satisfying FSAR safety analysis or design bases." Changes, Tests, and Experiments: Final Rule, 64 Fed. Reg. 53,582, 53,598 (Oct. 4, 1999).

[53] Although Petitioner's briefs rely heavily on 10 C.F.R. § 50.59 in support of its argument that this CAL process constitutes a de facto license amendment proceeding (see, e.g., Petitioner's Brief at 19-23), they do not specifically reference section 50.59(c)(2)(viii). We do not view this omission as a waiver, however, because Petitioner's brief included an argument based on the rationale in section 50.59(c)(2)(viii). See Petitioner's Brief at 13; Large Affidavit at 5; see also Tr. at 42-44. Indeed, SCE understood Petitioner to be advancing such an argument, as evidenced by the fact that SCE endeavored to rebut it. See SCE's Answering Brief, App. A, Examples of Mischaracterizations in the FOE Brief, Affidavits, and NRDC Brief at 118-19.

although that theory is not yet supported by actual experience.[54] SCE nevertheless proposes to implement the following sequence of steps incident to the start-up and operation of Unit 2: (1) Unit 2 will be operated at 70% power for a limited duration; (2) this duration will be selected in such a manner that if the calculations are wrong, tube-to-tube wear will likely not progress far enough to cause any tube failures; (3) Unit 2 will then be shut down; and (4) 100% of the steam generator tubes will be inspected, and the inspection results can be compared to current wear data to determine the wear rate and provide confirmation vel non of the theoretical analysis. See SCE's Answering Brief at 10-11.

The above steps satisfy the regulatory definition of "tests or experiments not described in the [UFSAR,]" which constitute "any activity where any structure, system, or component is utilized or controlled in a manner which is either: (i) [o]utside the reference bounds of the design bases as described in the [UFSAR] or (ii) [i]nconsistent with the analyses or descriptions in the [UFSAR]." 10 C.F.R. § 50.59(a)(6). Because the phenomenon of in-plane fluid elastic instability had not previously been observed in U-tube steam generators, and because tube

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[54] As evidenced by the following, SCE's prediction that accelerated tube wear will be precluded by plant operations limited to 70% power is grounded on theory that is not yet supported by actual experience. First, SCE's Steam Generator Operational Assessment for Tube-to-Tube Wear by Areva states that "[i]n-plane modes that have never been observed to be unstable even though the computed fluid-elastic stability margins are well below 1." Assessment for Tube-to-Tube Wear at 16. In other words, in-plane vibrations due to fluid elastic instability have not occurred even though the theory predicts in-plane instability. Second, regarding the tests conducted by Westinghouse, which developed the criteria for in-plane vibrations used for the Unit 2 steam generators, SCE states that the "[in-plane] instability was never observed in any of [the] square-pitch U-bend tests despite early attempts to force its occurrence without any [anti-vibration bar] support for flows up to three times the [out-of-plane] instability threshold." SONGS Unit 2 Return to Service Report, Att. 6 – App. D, Operational Assessment of Wear Indications in the U-bend Region of San Onofre Unit 2 Replacement Steam Generators at 14 (ADAMS Accession No. ML12285A269, which is entitled "Attachment 6: Appendix A: Estimate of FEI-Induced TTW Rates" on ADAMS, but also contains Appendix D, starting on page 78 of 209 of the ADAMS portable document format (PDF) version). Additionally, SCE states that in subsequent tests using triangular arrays, "[a]s was the case for square array patterns, no in-plane instability was observed in these tests even for U-bend tubes with no supports above the top tube support plate." Id. In short, there is a dearth of applicable experiential data available for in-plane vibrational motion, because, as conceded by SCE, "tube-to-tube wear due to in-plane [fluid elastic instability] ha[s] not been previously experienced in U-tube steam generators." SCE's Answering Brief at 10.

failures based on that phenomenon had not been envisioned, the FSAR did not include an analysis or description of it. See supra note 50 and accompanying text. Accordingly, any operation of Unit 2 that might result in in-plane vibrations due to fluid elastic instability is "[i]nconsistent with the analyses or descriptions in the UFSAR" (10 C.F.R. § 50.59(a)(6)), which, in turn, is the type of "test or experiment" that triggers the obligation under section 50.59(c)(2)(viii) to seek a license amendment.[55]

According to SCE, even if the sequence of start-up and operational steps in its Unit 2 Return to Service Report are viewed as tests or experiments that result in a "substantial change in an analysis" in the UFSAR, such a change "does not per se require a license amendment under 10 C.F.R. § 50.59." SCE's Answering Brief at 83. For example, "[i]f the analytical method is not described in the UFSAR," states SCE, "a change to that method does not require [a license amendment pursuant to section 50.59]." Id. "Furthermore, only changes to the 'method of evaluation' are covered by 10 C.F.R. § 50.59(c)(2)(viii). Changes to inputs to methods of evaluation are not covered by this provision" and, hence, do not trigger the requirement of seeking a license amendment. Id.

In other words, SCE claims that the standard in section 50.59(c)(2)(viii) has not been triggered because the tests or experiments embodied in its Unit 2 Return to Service Report are not inconsistent with the analysis or descriptions in the UFSAR. We disagree.

The General Design Criteria in Appendix A of 10 C.F.R. Part 50 establish minimum requirements for the principal design criteria for water-cooled nuclear reactor plants. And as discussed supra note 44, General Design Criterion 14 refers to the reactor coolant boundary and includes steam generator tubes.

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[55] The test or experiment proposed by SCE that must be the subject of a license amendment is required (1) to validate the vibration analysis that will become part of the FSAR (see supra note 52); and (2) to assure the steam generator tubes do not fail prematurely due to tube-to-tube wear and, thus, are able to satisfy their design bases. See id. (quoting 64 Fed. Reg. at 53,598).

Section 5.4.2.3.1 of the SONGS FSAR analyzes the maintenance of steam generator tube integrity. Subsection 5.4.2.3.1.3.A describes the "Degraded Tube Evaluation." Its methodology essentially consists of calculating the maximum thinning for which tube integrity can be assured.[56] Additionally, an inspection program, defined in Technical Specification 5.5.2.11, assures that tubes are removed from service before they reach maximum wall thinning.[57]

SCE's experience with SONGS Unit 3 forcefully demonstrates that the current analysis used to support the maintenance of steam generator tube integrity is inadequate for the replacement steam generators. More specifically, the current analysis underlying tube inspections to prevent maximum thinning is inadequate to assure tube integrity in light of the accelerated wear mechanism that might occur in this type of steam generator, and that did occur in the Unit 3 steam generators.

Without question, the current analysis described in the FSAR failed to achieve its intended purpose, and it must therefore be changed. We view this change as sufficiently significant to trigger the license amendment requirement of section 50.59(c)(2)(viii), because it is "[i]nconsistent with the analyses or descriptions in the [UFSAR]." 10 C.F.R. § 50.59(a)(6)(ii). Indeed, this change is a radical deviation from the prior analysis and description in the UFSAR, because without this change, tube integrity cannot be assured for the SONGS steam generators.

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In sum, we conclude that SCE's Unit 2 Restart Plan, if implemented, would (1) grant SCE authority to operate without the ability to comply with all technical specifications; (2) grant SCE authority to operate beyond the scope of its existing license; and (3) grant SCE authority to

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56 See SONGS UFSAR at 5.4-24, section 5.4.2.3.1.3.A.

57 See SONGS Unit 2 Technical Specification, section 5.5.2.11.

operate its replacement steam generators in a manner that constitutes a test or experiment that meets the criteria in 10 C.F.R. § 50.59(c)(2)(viii) for seeking a license amendment. For these three independent reasons, this CAL process constitutes a de facto license amendment proceeding that is subject to a hearing opportunity under section 189a of the AEA.

D. Because Our Resolution of the First Referred Issue Grants Petitioner All the Relief Its Contention Seeks, the Second Issue Referred by the Commission Is Moot

The second issue referred to this Licensing Board is whether Petitioner "meets the standing and contention admissibility requirements of 10 C.F.R. § 2.309." CLI-12-20, 76 NRC at __ (slip op. at 5).[58] In its contention, Petitioner claims that "SONGS cannot be allowed to restart without a license amendment and attendant adjudicatory public hearing as required by 10 C.F.R. § 2.309, in which Petitioner and other members of the public may participate." Petition to Intervene at 16.

In the course of resolving the first issue referred by the Commission (supra Part II.C), we concluded that this CAL process constitutes a de facto license amendment proceeding that is subject to a hearing opportunity. As Petitioner conceded during oral argument (see Tr. at 29), such a conclusion grants all the relief sought in its contention. Petitioner's contention, therefore, is moot.

Were we to adjudicate either (1) the admissibility of a moot contention, or (2) the standing of a petitioner who sought to adjudicate a moot contention, we would be issuing an advisory opinion in derogation of Commission precedent. This we decline to do. See U.S. Dep't of Energy (High-Level Waste Repository), CLI-08-21, 68 NRC 351, 352 (2008); accord

[58] SCE urged this Board to resolve the standing and contention admissibility issues before considering the de facto license amendment issue. See Tr. at 63-65. The NRC Staff and Petitioner disagreed (see Tr. at 138 (NRC Staff); Tr. at 150 (Petitioner)), arguing that SCE's suggested approach was inconsistent with the Commission's unequivocal directive "to consider whether: (1) the [CAL] . . . constitutes a de facto license amendment that would be subject to a hearing opportunity . . . ; and, if so, (2) whether the petition meets the standing and contention admissibility requirements." CLI-12-20, 76 NRC at __ (slip op. at 5). We agree with the NRC Staff and Petitioner that SCE's suggested approach is at odds with the Commission's clearly expressed instruction in CLI-12-20.

Texas Utilities Generating Co. (Comanche Peak Steam Elec. Station), ALAB-714, 17 NRC 86, 94 (1983).[59]

III. CONCLUSION

For the foregoing reasons, we resolve the first issue referred by the Commission in the affirmative, concluding that the CAL process for SONGS Units 2 and 3 constitutes a de facto license amendment proceeding that is subject to a hearing opportunity under section 189a of the AEA. Our resolution of the first issue grants Petitioner the relief it seeks in its contention; namely, the opportunity for an adjudicatory hearing incident to the license amendment proceedings for the restart of Units 2 and 3. Petitioner's contention is thus moot, which renders moot the second issue referred by the Commission. The proceeding before this Board is therefore terminated.

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[59] "It is well established that, absent compelling reasons, the Commission adheres to the 'case' or 'controversy' doctrine in its adjudicatory proceedings." Hydro Resources, Inc. (P.O. Box 777, Crownpoint, New Mexico 87313), LBP-05-17, 62 NRC 77, 91 (2005) (citing Texas Utilities Elec. Co. (Comanche Peak Steam Elec. Station), CLI-93-10, 37 NRC 192, 200 n.28 (1993)). Pursuant to this doctrine, a justiciable controversy must involve parties who raise questions "presented in an adversary context and in a form historically viewed as capable of resolution through the judicial process." Flast v. Cohen, 392 U.S. 83, 95 (1968). When -- as is the case here -- a petitioner obtains the relief it is seeking before the admissibility of its contention is resolved, the admissibility vel non of the contention is no longer justiciable, because it no longer presents a live controversy involving a true clash of interests that is susceptible to meaningful adjudicative relief. Cf. Moore v. Charlotte-Mecklenburg Bd. of Ed., 402 U.S. 47, 48 (1971) (per curiam) (dismissing appeal for lack of live controversy where both litigants desired the same result); David B. Kuhl (Denial of Senior Reactor Operator License), LBP-09-14, 70 NRC 193, 195-96 (2009) (dismissing hearing request as moot where petitioner's claim was not susceptible to meaningful adjudicative relief).

If a party wishes to appeal this decision, it must file a petition for review with the Commission within 25 days after service of this decision. See 10 C.F.R. § 2.341(b)(1). Unless otherwise authorized by law, a party to an NRC adjudicatory proceeding must seek Commission review before seeking judicial review of an agency action. See id.

It is so ORDERED.

THE ATOMIC SAFETY
AND LICENSING BOARD

/RA/

E. Roy Hawkens, Chairman
ADMINISTRATIVE JUDGE

/RA/

Dr. Anthony J. Baratta
ADMINISTRATIVE JUDGE

/RA/

Dr. Gary S. Arnold
ADMINISTRATIVE JUDGE

Issued at Rockville, Maryland
this 13th day of May 2013.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
SOUTHERN CALIFORNIA EDISON CO.	)	
	)	
	)	Docket Nos. 50-361-CAL
(San Onofre Nuclear Generating Station -	)	50-362-CAL
Units 2 and 3)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing **MEMORANDUM AND ORDER (Resolving Issues Referred by the Commission in CLI-12-20) – LBP-13-07** have been served upon the following persons by Electronic Information Exchange.

Office of Commission Appellate
Adjudication
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: ocaamail@nrc.gov

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Mail Stop - T-3 F23
Washington, DC 20555-0001

E. Roy Hawkens
Chief Administrative Judge
E-mail: roy.hawkens@nrc.gov

Anthony J. Baratta
Administrative Judge
Email: anthony.baratta@nrc.gov

Gary S. Arnold
Administrative Judge
Email: gary.arnold@nrc.gov

U.S. Nuclear Regulatory Commission
Office of the Secretary of the Commission
Mail Stop O-16C1
Washington, DC 20555-0001

Hearing Docket
E-mail: hearingdocket@nrc.gov

Southern California Edison Company
Douglas Porter, Esq.
Director and Managing Attorney
Generation Policy and Resources
Law Department
2244 Walnut Grove Ave., GO1, Q3B, 335C
Rosemead, CA 91770
Email: douglas.porter@sce.com

Counsel for Licensee
Morgan, Lewis & Bockius, LLP
1111 Pennsylvania, Ave. N.W.
Washington, D.C. 20004
Paul M. Bessette, Esq.
Kathryn M. Sutton, Esq.
Stephen J. Burdick, Esq.
Steven P. Frantz, Esq.
William E. Baer, Jr.
Mary Freeze, Legal Secretary
Lena M. Long, Legal Secretary
E-mail: pbessette@morganlewis.com
sburdick@morganlewis.com
ksutton@morganlewis.com
wbaer@morganlewis.com
sfrantz@morganlewis.com
mfreeze@morganlewis.com
llong@morganlewis.com

San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362-CAL
MEMORANDUM AND ORDER (Resolving Issues Referred by the Commission in CLI-12-20) - LBP-13-07

U.S. Nuclear Regulatory Commission
Office of the General Counsel
Mail Stop - O-15 D21
Washington, DC 20555-0001
Edward Williamson, Esq.
David Roth, Esq.
Catherine Kanatas, Esq.
David Cylkowski, Esq.
Jeremy Wachutka, Esq.
Email: edward.williamson@nrc.gov
david.roth@nrc.gov
catherine.kanatas@nrc.gov
david.cylkowski@nrc.gov
jeremy.wachutka@nrc.gov

OGG Mail Center: ogcmailcenter@nrc.gov

Friends of the Earth
Ayres Law Group
1707 L St., NW
Suite 850
Washington, D.C. 20036
Richard E. Ayres, Esq.
Jessica L. Olson, Esq.
Kristin L. Hines, Esq.
Email: ayresr@ayreslawgroup.com
olsonj@ayreslawgroup.com
hinesk@ayreslawgroup.com

Natural Resources Defense Council
Geoffrey H. Fettus, Esq.
1152 15th Street, NW
Suite 300
Washington, DC 20005
Email: gfettus@nrdc.org

[Original signed by Herald M. Speiser]
Office of the Secretary of the Commission

Dated at Rockville, Maryland
this 13th day of May, 2013

FAR OUTSIDE THE NORM: The San Onofre Nuclear Plant's Steam Generator Problems in the Context of the National Experience with Replacement Steam Generators

by
Daniel Hirsch
and
Dorah Shuey

with a Foreword
by
Dale Bridenbaugh

September 12, 2012

http://www.committeetobridgethegap.org

EXECUTIVE SUMMARY

Southern California Edison (SCE) and the Nuclear Regulatory Commission (NRC) have suggested that the problems experienced in the steam generators of the two San Onofre reactors are fundamentally different and that Unit 2's difficulties are merely "settling in" wear normal for new replacement steam generators. No data have been provided to date by SCE or NRC to support these claims, yet SCE has suggested that for these reasons it expects to request permission to restart Unit 2 and run it at somewhat reduced power, without repairing or replacing the damaged devices.

This report assembles national data from inspections of similar replacement steam generators after one cycle of operation. The conclusion is that both San Onofre Unit 2 and Unit 3 have experienced damage greatly in excess of the typical reactor:

- **The median number of steam generator tubes nationally showing wear after one cycle of operation is—FOUR. San Onofre Unit 2 had 1595 damaged tubes, approximately 400 times the median; San Onofre Unit 3 had 1806.**

- **The median number of indications of wear on steam generator tubes nationally after one cycle of operation is—FOUR. San Onofre Unit 2 had 4721, greater than a thousand times more. San Onofre Unit 3 had 10,284.**

- **The median number of steam generator tubes that were plugged after one cycle of operation is—ZERO. San Onofre Unit 2 had 510; Unit 3 had 807.**

Additionally, the replacement steam generators at San Onofre Unit 2 and 3 suffer from the same fundamental design errors. Indeed, the number of damaged tubes in each unit is approximately the same.

The conclusion is clear: San Onofre Unit 2 and Unit 3 are both very ill nuclear plants. Unit 3's fever is slightly higher, but both are in serious trouble. What they are experiencing is not just normal wear due to "settling in" purportedly experienced with similar replacement steam generators. They are far, far outside the norm of national experience. And Unit 2 cannot be said to be acceptable for restart, any more than Unit 3. Unit 2 has hundreds of times more bad tubes and a thousand times more indications of wear on those tubes than the typical reactor in the country with a new steam generator, and nearly five times as many plugged tubes as the rest of the replacement steam generators, over a comparable operating period, in the country combined. Restarting either San Onofre reactor with crippled steam generators that have not been repaired or replaced would be a questionable undertaking at best.

FOREWORD

SAN ONOFRE NUCLEAR GENERATING STATION REPLACEMENT STEAM GENERATOR PROBLEMS

by
DALE BRIDENBAUGH
NUCLEAR ENGINEER, RETIRED

As a retired professional nuclear engineer and long time citizen of California, I have followed the recent experience of the San Onofre Nuclear Generating Station with great interest. I am particularly troubled by the extent and causes of the early failures of tubes in the replacement steam generators at both of the San Onofre units (Units 2 and 3) that have not yet been thoroughly explained and reported. As this report makes clear, the conflicting failure data thus far made available by the San Onofre operating utility and the Nuclear Regulatory Commission, along with the lack of specificity detailing the mode(s) of failure, lend little credibility to Southern California Edison's claims that the large number of damaged steam generator tubes and indications of wear on the tubes are in fact completely understood. The data assembled in this report call into question assertions that the San Onofre damage is due primarily to normal "settling in" found commonly in other new replacement steam generators and that no immediate corrective action is needed before the restart of Unit 2.

As dramatically shown in Figures 3, 4, and 5 of this report, the San Onofre experience after only two or less years of operation with replacement steam generators lies far outside the bounds of normality when compared to the experience of other nuclear units with such replaced components. Steam generators, and more specifically the tube boundaries, play a critical role in assuring plant safety and the containment of possible radioactive releases. In spite of Edison's attempt to assert a different level of risk between Units 2 and 3, it seems clear that similar design and failure challenges are present in both units and that future operation of either unit has not been technically justified. It is my opinion that measures necessary for the future safe operation of either of these unit have not been adequately put forth at this time, and that operation with or without reduced power of Unit 2 should not be authorized.

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THE SAN ONOFRE NUCLEAR PLANT'S STEAM GENERATOR PROBLEMS IN THE CONTEXT OF THE NATIONAL EXPERIENCE WITH REPLACEMENT STEAM GENERATORS

Introduction

On January 31, 2012, a steam generator tube in Unit 3 of the San Onofre Nuclear Generating Station burst, leading to a shutdown of the reactor. Shortly thereafter, it was revealed that a previously scheduled inspection of Unit 2, which was down for refueling, had identified hundreds of damaged tubes in that reactor. Subsequent inspections of both units revealed approximately 3,400 tubes were showing indications of wear.

This was surprising because the steam generators in both units were virtually new. Unit 3's steam generators were about a year old, and Unit 2's were approximately two years old. Yet they were showing extensive wear.

Since then, further inspections have revealed serious problems with the steam generators in both units. 1317 tubes at San Onofre have been plugged to date, far more than have been plugged over a similar period of operation in all replacement steam generators in the country combined.

Southern California Edison, which operates San Onofre, has recently conceded that Unit 3 will not be operating anytime soon, if ever, and that the long-term viability of the plant as a whole is now in question.[1] However, the utility continues to suggest it may in the near future request approval from the Nuclear Regulatory Commission to restart Unit 2, even though its steam generators have been neither repaired nor replaced.

Underlying this anticipated action are two assertions: (1) that the problems in Unit 2 and Unit 3 are dramatically different, and (2) that the extent of the wear seen in Unit 2 is nothing out of the ordinary and commonly seen in similar new replacement steam generators, just a routine "settling in" phenomenon that stops soon after installation. The analysis that follows examines those two claims.

What Steam Generators Do and Why Their Proper Functioning is Important

Steam generators are critical components of Pressurized Water Reactors (PWRs) and their failure could lead to serious consequences. In a PWR, the primary coolant is kept under high enough pressure that it remains liquid at temperatures above the normal boiling point. That primary coolant, which picks up significant radioactivity from the nuclear fuel, must transfer its heat to a secondary coolant, which then becomes steam to turn turbines to generate electricity. The steam generators transfer heat from the primary to the secondary coolant and produce steam.

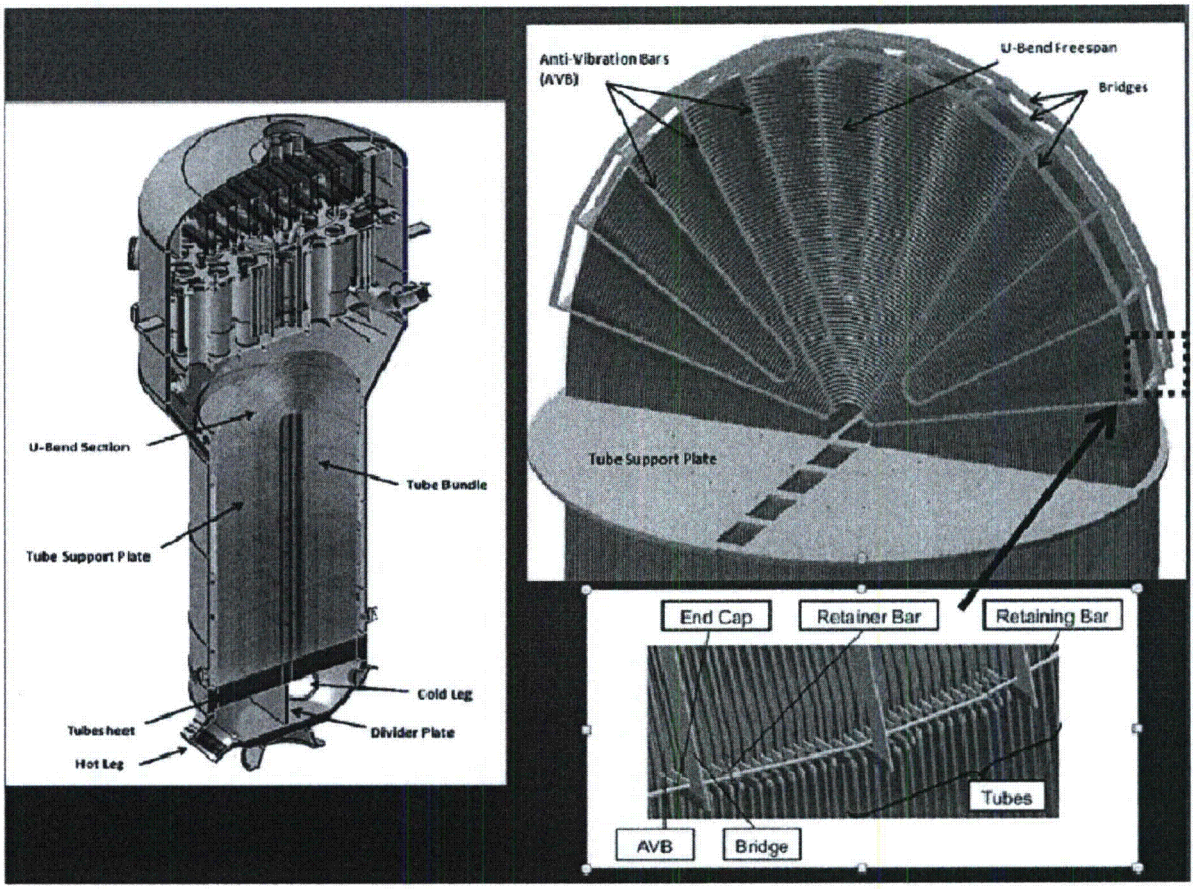
A steam generator is composed of a large number of very thin tubes through which the hot (both thermally and radioactively) primary coolant flows, transferring its heat to secondary coolant on the outside of the tubes. Significantly, while the steam generators are inside the containment structure, the large concrete dome designed to contain radioactivity in case of an accident, the secondary coolant loop/steam line travels outside the containment to run the turbines and generate power.

Therefore, the steam generators are critical because they are the primary coolant boundary that cannot be permitted to be breached significantly. Such a breach could both release radioactivity via a pathway to the outside environment and result in a loss of cooling to the reactor core, leading in some circumstances, if there are other failures, to a potential meltdown. The steam generator tubes must be very thin, in order to effectively transfer heat, and simultaneously very strong, so as to assure they do not burst and cause a loss of reactor cooling and release of radioactivity. Damage to the tubes can thus be problematic. The NRC has described their importance:[ii]

> The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system; the SG tubes are also relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity, as necessary, to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage severe accidents.

Figure 1 below shows a schematic view of the San Onofre replacement steam generators.

Figure 1 San Onofre Replacement Steam Generator Schematic



Source: NRC[iii]

The tubes are in an inverted U shape: in the upper part of the steam generator, the tubes bend to return downward again. There are four key parts of the steam generators for the present discussion: the tube support plates, through which the tubes run; the anti-vibration bars (AVBs), designed to reduce vibration; the retainer bars, which help retain the AVBs; and the U-Bend Freespan, where the tubes bend near the top of the steam generator and have no immediate support.

There thus are at least four locations where steam generator tubes can get damaged: they can rub against the tube support plates, the AVBs, the retainer bars, or against each other in the U-Bend Freespan.[iv] Damage has occurred in the new steam generators at San Onofre at all four locations.

What Happened at San Onofre

The original steam generators for San Onofre Units 2 and 3 were supposed to last for forty years, the design life of the reactors. (Unit 1, a Westinghouse design system, was shut down long ago due in part to extensive steam generator tube degradation.[v]) Therefore, the containment structures were not built with a pre-engineered way to get the old steam generators out and the replacement ones in. The original steam generators, manufactured by Combustion Engineering, began failing earlier than anticipated, and within about twenty years of operation, SCE began planning to replace them.

Mitsubishi Heavy Industries was chosen to construct the new steam generators. It took nearly four years to fabricate the Unit 2 steam generators, and nearly six years for Unit 3's.[vi] They then had to be shipped from Japan and installed. This required cutting large openings into the containment structures, something generally to be avoided both from a cost standpoint and because of the importance of not risking reducing the integrity of the structures designed to prevent release of radioactivity into the environment in case of an accident.

At Edison's request, Mitsubishi made numerous changes to the design of the steam generators compared to those originally at San Onofre, such as using a different tube alloy, Inconel 690, and adding hundreds of more tubes. Yet, by asserting that it was making a "like for like" change, SCE bypassed the normal requirement to apply for a license amendment, which would have entailed a higher degree of scrutiny by the NRC and the opportunity for the public to request an evidentiary hearing. This turned out to be a fateful decision, because it appears possible that the greater degree of review that would have been required with a full license amendment application might have detected the problems that the design changes caused and that have since crippled San Onofre.

Regardless, the changes made from the original design resulted in the replacement steam generators failing within a year or two of installation. Subsequent reviews by NRC and SCE determined that computer modeling errors by Mitsubishi resulted in actual steam flows in parts of the steam generators being four times higher than originally estimated by Mitsubishi, leading to "fluid elastic instability," vibration, and damage to the tubes. *This fundamental problem exists for both Unit 2 and 3.*

Extensive Damage In Units 2 and 3

It has taken considerable effort to get SCE and NRC to disclose fully the number of damaged tubes and the magnitude of their wear. In early February, an NRC spokesman told the news media that 80% of the 9727 tubes in one of the two steam generators in Unit 2 had been inspected, with the following results: Two of the tubes showed more than 30% wall thinning, 69 had 20% thinning and more than 800 had 10% thinning.[vii] *Thus, as of early February, about 11% of the tubes inspected in Unit 2 had 10% or more through-wall wear, after just two years of operation.* This is significant because the full-power plugging limit is 8%, meaning that at the end of forty years of operation of steam generators, one isn't supposed to plug more than 8% of the tubes because of damage and still be able to run at full power. In just two years, therefore,

San Onofre Unit 2 has suffered damage that normally takes decades.

Repeated requests for the complete data based on inspection of the remaining tubes in Units 2 and 3 were denied for several months. Then, after being pressed for updated figures by the author at a public meeting called by the NRC on June 18 to discuss its Augmented Inspection Team (AIT) review, a senior SCE executive stated:[viii]

> We will get you the specific numbers—I will share the percentages with you tonight... On Unit 3, 9% of the tubes in the Unit 3 steam generators -- so 19,454 tubes in the steam generators, 9% of them showed wear of greater than 10% through-wall indications, 9%. On Unit 2, 12% of the tubes showed wear greater than 10% through-wall indication.

Note that the percentage provided by the SCE official for Unit 2 matches fairly closely with the figures given by NRC in early February when 80% of the tubes in only one of the two steam generators in that Unit had been inspected. After giving the above percentages, the SCE spokesman stated, "Compared to other steam generators in the industry, those numbers by themselves are not alarming. What is alarming and the reason we are here tonight is the unexpected tube-to-tube wear." He went on to assert that problems are far worse in Unit 3 than Unit 2, because there are hundreds of tubes in Unit 3 showing tube-to-tube wear but only two in Unit 2.

Those statements, and others by SCE and NRC, assert that it is only the tube-to-tube wear that is of concern and that the amount of wear other than tube-to-tube wear is comparable to what is generally seen in other replacement steam generators in the industry. This report evaluates those assertions and assesses whether the severity of the problems with the San Onofre steam generators is in line with typical experience nationally.

Weeks passed without the actual tube wear numbers being provided for San Onofre. It took intervention by staff of the Senate Committee on Environment and Public Works before the data were finally posted on the NRC website. The data are critical and can be found below. Table 1 provides data for both steam generators in Unit 2 of the San Onofre Nuclear Generating Station (SONGS Unit 2). Table 2 provides the data for the two steam generators in Unit 3.

Table 1

SONGS Unit 2 Steam Generators
Wear Depths Summary

Steam Generator SG2E88 (Through-Wall Wear)	Anti-Vibration Bar	Tube Support Plate	Tube-to-Tube Wear	Retainer Bar	Foreign Object	Total Indications	Tubes with Indications (out of 9727 total per SG)
≥ 50%	0	0	0	1	0	1	1
35 - 49%	2	0	0	1	0	3	3
20 - 34%	86	0	0	0	2	86	74
10 - 19%	705	108	0	0	0	813	406
< 10%	964	117	0	0	0	1081	600
TOTAL	1757	225	0	2	2	1984	734*

Steam Generator SG2E89 (Through-Wall Wear)	Anti-Vibration Bar	Tube Support Plate	Tube-to-Tube Wear	Retainer Bar	Foreign Object	Total Indications	Tubes with Indications (out of 9727 total per SG)
≥ 50%	0	0	0	1	0	1	1
35 - 49%	0	0	0	1	0	1	1
20 - 34%	78	1	0	3	0	82	67
10 - 19%	1014	85	2	0	0	1101	496
< 10%	1499	53	0	0	0	1552	768
TOTAL	2591	139	2	5	0	2737	861*

* This value is the number of tubes with wear indications of any depth and at any location. Since many tubes have indications in more than one depth and location, the total number of tubes is less than the total number of indiations.

Source: NRC[ix]

Table 2

SONGS Unit 3 Steam Generators
Wear Depths Summary

Steam Generator SG3E88 (Through-Wall Wear)	Anti-Vibration Bar	Tube Support Plate	Tube-to-Tube Wear	Retainer Bar	Foreign Object	Total Indications	Tubes with Indications (out of 9727 total per SG)
≥ 50%	0	117	48	0	0	165	74
35 - 49%	3	217	116	2	0	338	119
20 - 34%	156	506	134	1	0	797	197
10 - 19%	1380	542	98	0	0	2020	554
< 10%	1818	55	11	0	0	1884	817
TOTAL	3357	1437	407	3	0	5204	919*

Steam Generator SG3E89 (Through-Wall Wear)	Anti-Vibration Bar	Tube Support Plate	Tube-to-Tube Wear	Retainer Bar	Foreign Object	Total Indications	Tubes with Indications (out of 9727 total per SG)
≥ 50%	0	91	26	0	0	117	60
35 - 49%	0	252	102	1	0	355	128
20 - 34%	45	487	215	0	0	747	175
10 - 19%	940	590	72	0	0	1602	450
< 10%	2164	94	1	0	0	2259	838
TOTAL	3149	1514	416	1	0	5080	887*

* This value is the number of tubes with wear indications at any depth and at any location. Since many tubes have indications in more than one depth and locations, the total number of tubes is less than the total number of indications.

Source: NRC[x]

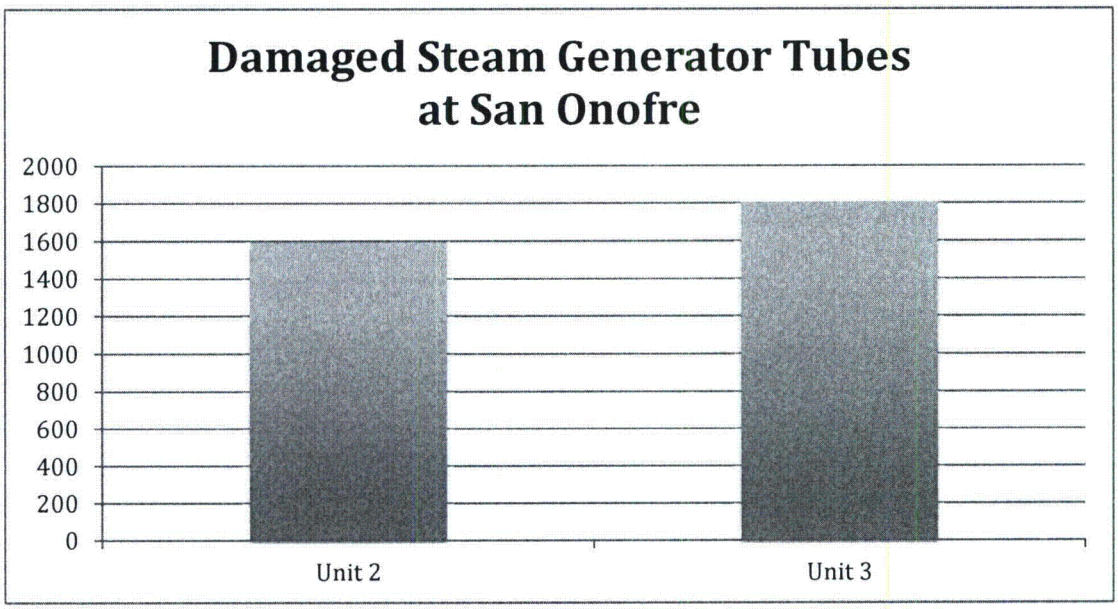
Note that the data tables do not comport with either the numbers given by the either the spokesman for NRC in early February or the spokesman for SCE in June. Whereas NRC indicated in February that, with only 80% of the tubes inspected in one of the 2 steam generators in Unit 2 as of that time, nearly 900 tubes with wear 10% or greater had been detected, the tables NRC posted months thereafter show neither steam generator in Unit 2, after inspection of 100% of the tubes, with more than 565 tubes with wear 10% or greater. And the NRC tables assert at most about 5% of the tubes in Unit 2 had wear of 10% or greater, whereas SCE had said the figure was 12%.

Efforts to have NRC clarify which of the three sets of data—NRC's summary from early February, SCE's from June, or the tables posted on NRC's website in July—is correct, and describe what is the cause of the discrepancies, have been unsuccessful to date. NRC personnel responsible for the San Onofre investigation indicated they do not know.[xi] For the purposes of this analysis, the NRC data tables above are employed, resulting in the use of the smallest estimate of damaged tubes. Should either the earlier NRC or SCE summaries be more accurate than the data tables used here, the disparity with the national experience with replacement steam generators would be even greater than shown in the discussion that follows.

Steam Generator Tube Damage is Not Dramatically Different Between San Onofre Units 2 and 3

The data tables posted by NRC show similar numbers of damaged tubes in the two units. Unit 2 has 1,595 tubes with wear, Unit 3 has1,806.

Figure 2



Additionally, as will be seen in Table 3 and Figure 5, the number of steam generator tubes that have had to be plugged in each reactor is in the same approximate range: 510 in Unit 2 and 807 in Unit 3. As this report shows, these numbers are dramatically higher than the national experience. Each San Onofre unit has had to plug many times more tubes than all reactors with new steam generators in the country, over a comparable operational period, combined.

Unit 3 has a somewhat greater number of wear indications than Unit 2 (i.e., tubes showing wear on more than one location per tube) and more tubes in the higher ranges of through-wall wear. And Unit 3 has hundreds of indications of through-wall wear due to tube-to-tube rubbing whereas Unit 2 has only two.

However, tube-to-tube wear represents less than 10% of the wear indications in Unit 3. The great majority of tubes that are in trouble in either unit are experiencing tube-to-AVB wear or tube-to-tube-support-plate wear. And both reactors are faced with thousands of such wear indications.

The focus by SCE and NRC on tube-to-tube wear and the effort to thus distinguish Unit 2 from Unit 3 is misplaced. By far, the majority of tubes showing wear are evidencing it from other kinds of wear and exist in large numbers in both units.

Furthermore, and most critically, both Unit 2 and 3 suffer from the same fundamental design defect. The computer model employed by Mitsubishi, coupled with the design changes inherent in the steam generators in both San Onofre reactors, resulted in considerably higher steam flows than predicted, causing vibrations resulting in rubbing and damage to the sensitive, very thin tubes.[xii] The same fundamental problem is crippling the steam generators in both reactors.

The Steam Generator Tube Wear at San Onofre Is Far Worse Than the National Experience

The NRC's AIT report dismissed all but the tube-to-tube wear (which is primarily in Unit 3) and four wear indications at retainer bars in Unit 2 as common in new steam generators. The report stated that, with those exceptions, "*the wear indications found are similar to those found at other replacement steam generators after one cycle of operation*."[xiii] (emphasis added)

However, at other times NRC has stated the opposite. For example, the *Los Angeles Times* quoted an NRC spokesman on July 14: "Other large steam generators have exhibited wear after one cycle of operation which resulted in tube plugging...but not to the extent seen on San Onofre steam generators." Another NRC spokesperson was quoted as saying, "It is accurate to say San Onofre's demonstrated wear is unprecedented for the length of time the steam generators were used."[xiv]

Also, SCE has made assertions similar to the statement in the NRC AIT report. In a July press statement about the release of the tube wear tables, for example, SCE stated, "The majority of this wear is related to support structures. *The nature of the support structure wear is not unusual in new steam generators and is part of the equipment settling in.*"[xv] (emphasis added)

So where does the truth lie? How does San Onofre compare to the national experience with new replacement steam generators?

Efforts to get NRC to provide data supporting the claim in its AIT report have not been successful. NRC staff in Region IV responsible for the San Onofre steam generator investigation stated that they believed the number of wear indications in Unit 2 was comparable to other similar steam generators. When asked for the basis for that belief, they said they had no data but had heard it anecdotally.[xvi] Obviously, a matter important for determining whether San Onofre Unit 2 should be permitted to restart should be based on more than an anecdote.

NRC regional staff indicated they would attempt to get supporting data on the national experience from NRC headquarters. NRC headquarters staff reported NRC had not compiled any such data.[xvii] This report, in the following sections, assembles and evaluates available data on replacement steam generator tube wear and describes where San Onofre falls within that national experience.

The Only Similar Replacement Steam Generators—at Fort Calhoun—Had NO Damaged Tubes

The claim has been made that San Onofre experience is comparable to that of reactors with similar replacement steam generators. However, the only similar steam generator in the country is found at the Fort Calhoun reactor; it has the only Mitsubishi steam generators in the U.S. outside of San Onofre. The number of steam generator tubes showing any wear at Fort Calhoun after one cycle of operation: zero. The number of wear indications: zero. The number of tubes that had to be plugged due to operation: zero.

San Onofre Unit 2, by contrast, has 1,595 damaged tubes, with 4,721 wear indications, and 510 tubes plugged. That is obviously not anywhere in the range of what the only similar steam generators in the country experienced. Furthermore, an assessment of the experience of replacement steam generators of other designs yields a similar disparity, as shown below.

As of 2002, the Majority of Replacement Steam Generators Had NO Damaged Tubes

How does San Onofre compare with the experience with replacement steam generators (RSGs) more generally? A January 2002 article in *Nuclear Engineering International*, entitled "Replacement Steam Generators," answers that question:

> Of the 30 RSGs now in operation, 26 have received 100% eddy current inspection during in service inspection. Of these, 12 have experienced limited fretting wear. The other 14 RSGs have no evidence of any wear. ECT [Eddy Current Testing] indications have resulted in 23 plugged tubes out of a total population of 176,282 in the 26 inspected SGs.

Thus, when the article was written, the majority of replacement steam generators showed "no evidence of any wear." The remaining minority showed limited wear—so limited, that a total of only 23 tubes had to be plugged out of 176,282 tubes in the 26 inspected steam generators. Unit 2 of San Onofre, the reactor asserted to be far healthier than Unit 3, had plugged more than twenty times as many tubes as the 26 replacement steam generators considered in that 2002 review, combined.

Analysis of Most Current National Replacement Recirculating Steam Generator Tube Wear Data Shows San Onofre Is Far Outside the Norm

Perhaps it could be argued that the data from the 2002 article are old and more recent replacement steam generators are having more trouble than was identified a decade ago. NRC staff, in stating that the agency has no compiled data on national experience with replacement steam generators, indicated that data for each individual plant should be found in each plant's first In-Service Inspection (ISI) report submitted to the NRC after installation of the replacement steam generators. The analysis that follows is based on reviewing the data from those ISI reports and numerous related documents for replacement recirculating steam generators that are available to the public through NRC's Agencywide Documents Access and Management System (ADAMS).

NRC staff provided a list of all replacement steam generators in the country and identified which, like San Onofre, are of the recirculating type and use Inconel 690 alloy tubes, and which few (a small minority) are once-through designs or use Inconel 600.[xviii] This analysis compiles the data for all recirculating replacement steam generators using Inconel 690 in the U.S., going back to ones installed around 1998 (data for earlier years are not available in the NRC's ADAMS database.) The results are striking, and are summarized in Table 3 and Figures 3 through 5 below. In short, the damage experienced by the replacement steam generators in both San Onofre reactors is far out of the norm of other comparable nuclear plants, even when taking into account the minor variation in the number of steam generator tubes at each plant.*

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* SCE has attempted to compare its steam generator experience to St. Lucie 2, in order to assert that what is happening at San Onofre is typical for new replacement steam generators and is simply a "settling in" process common to them. These assertions are clearly misplaced. St. Lucie 2's steam generators are having great trouble, and as the data show, not in any fashion the norm. Indeed, St. Lucie 1 had only 17 damaged tubes at its first ISI. The serious problems at St. Lucie 2 have resulted in its operators having to conduct a root cause analysis which concluded that "the root cause was that the U-tubes were not effectively supported during SG [steam generator] manufacture, which caused the tubes to sag into the AVBs and led to slight AVB deformation that closed the tube-to-AVB gap at specific locations. This exacerbated tube wear in those locations."[xix] NRC's Advisory Committee on Reactor Safety concluded that the St. Lucie 2 tube wear is "different than the form of degradation reported to have occurred at San Onofre. There are a number of design differences between the SGs installed at San Onofre and those at St Lucie 2."[xx] Thus the problems at St. Lucie 2 are not standard "settling in" but due to a serious manufacturing error and unrelated to San Onofre's problems. Even with all the troubles St. Lucie 2 has, it had to plug only 14 tubes, compared to the hundreds plugged at San Onofre.

Table 3

Nuclear Plant	# of Wear Indications	# of Damaged Tubes	# of Tubes Plugged	Total Tubes
South Texas 1	0	0	0	31,540
South Texas 2	0	0	0	30,340
Kewaunee	0	0	0	7,184
Shearon Harris	0	0	0	18,921
Ft. Calhoun	0	0	0	10,400
Farley 1	0	0	0	10,776
Farley 2	0	0	0	10,776
Diablo Canyon 1	1	1	0	17,776
Diablo Canyon 2	1	1	0	17,776
Comanche Peak 1	1	1	0	22,128
Braidwood 1	1	1	1	26,532
Beaver Valley 1	2	1	1	10,776
ANO 2	3	3	0	21,274
Palo Verde 1	4	4	0	25,160
Watts Bar 1	9	6	7	20,512
Sequoyah 1	11	11	11	19,932
St. Lucie 1	19	17	11	17,046
Palo Verde 2	81	48	15	25,160
Prairie Island	104	67	6	9,736
Palo Verde 3	140	68	4	25,160
Calvert Cliffs 1	189	166	0	16,942
Calvert Cliffs 2	200	170	29	16,942
Callaway	214	36	0	22,144
Salem 2	1,567	591	10	20,192
San Onofre 2	**4,721**	**1,595**	**510**	19,454
St. Lucie 2	5,994	2,174	14	17,998
San Onofre 3	**10,284**	**1806**	**807**	19,454

Figure 3

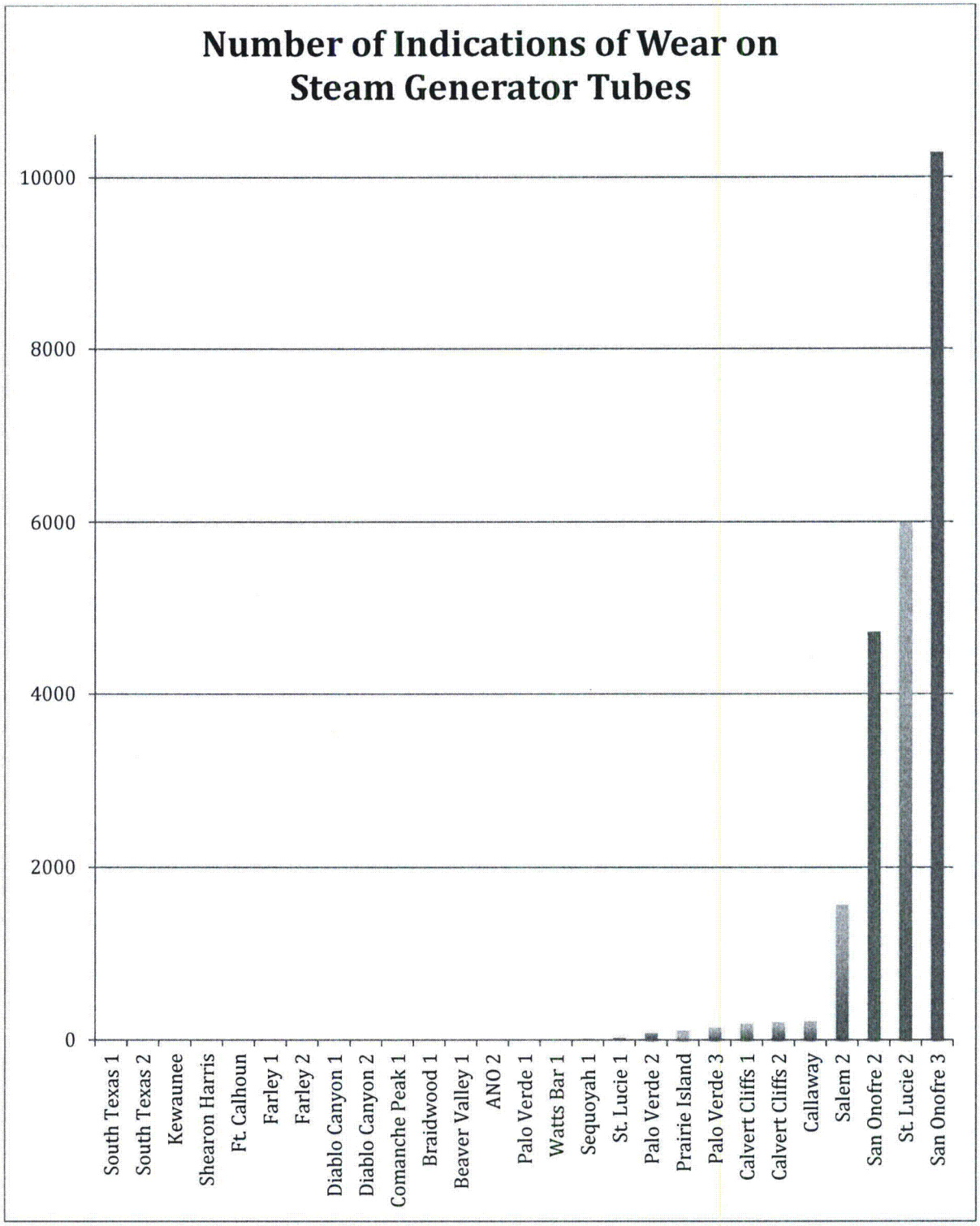

Number of Indications of Wear on Steam Generator Tubes
10000
8000
6000
4000
2000
0
South Texas 1
South Texas 2
Kewaunee
Shearon Harris
Ft. Calhoun
Farley 1
Farley 2
Diablo Canyon 1
Diablo Canyon 2
Comanche Peak 1
Braidwood 1
Beaver Valley 1
ANO 2
Palo Verde 1
Watts Bar 1
Sequoyah 1
St. Lucie 1
Palo Verde 2
Prairie Island
Palo Verde 3
Calvert Cliffs 1
Calvert Cliffs 2
Callaway
Salem 2
San Onofre 2
St. Lucie 2
San Onofre 3

Figure 4

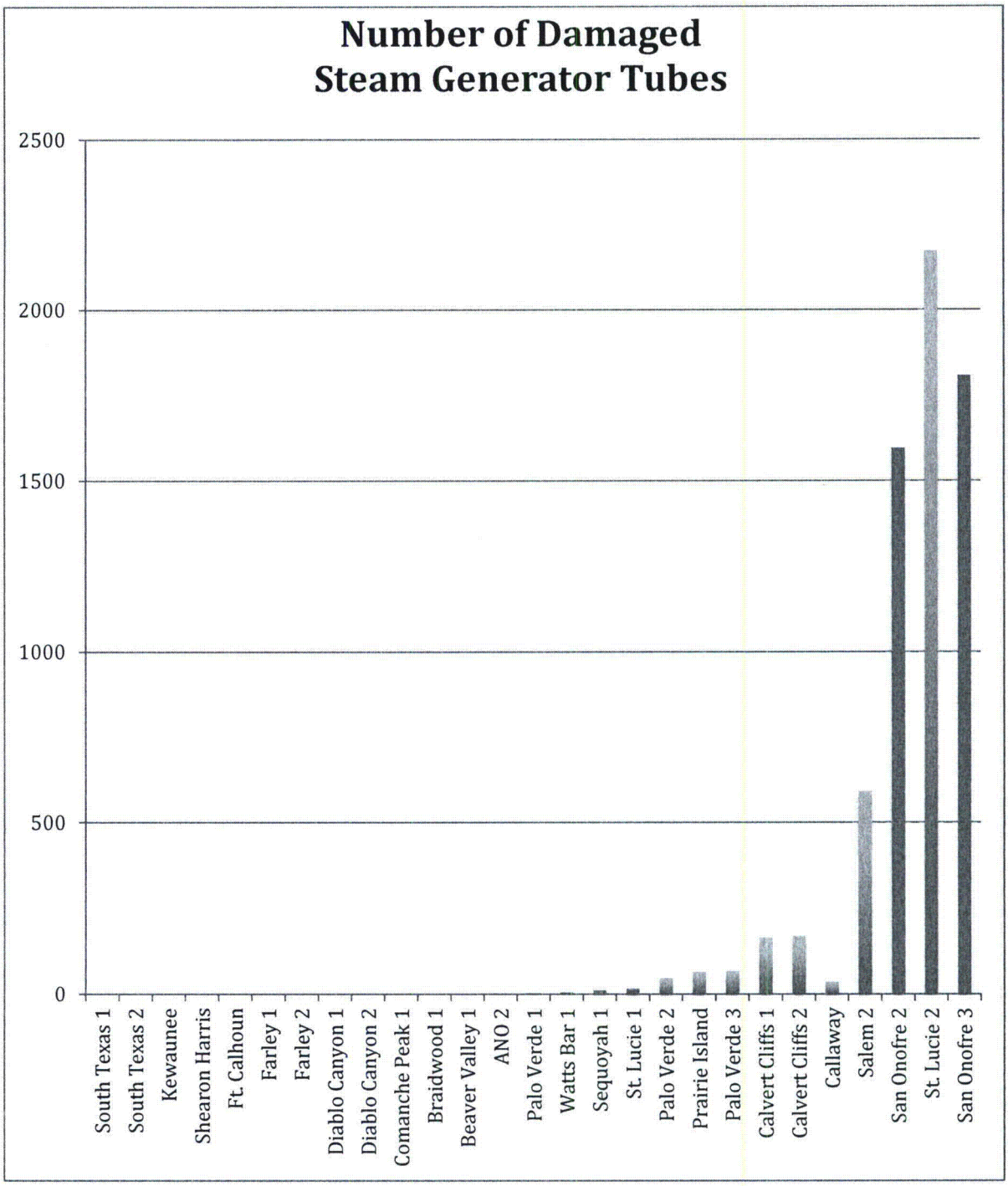
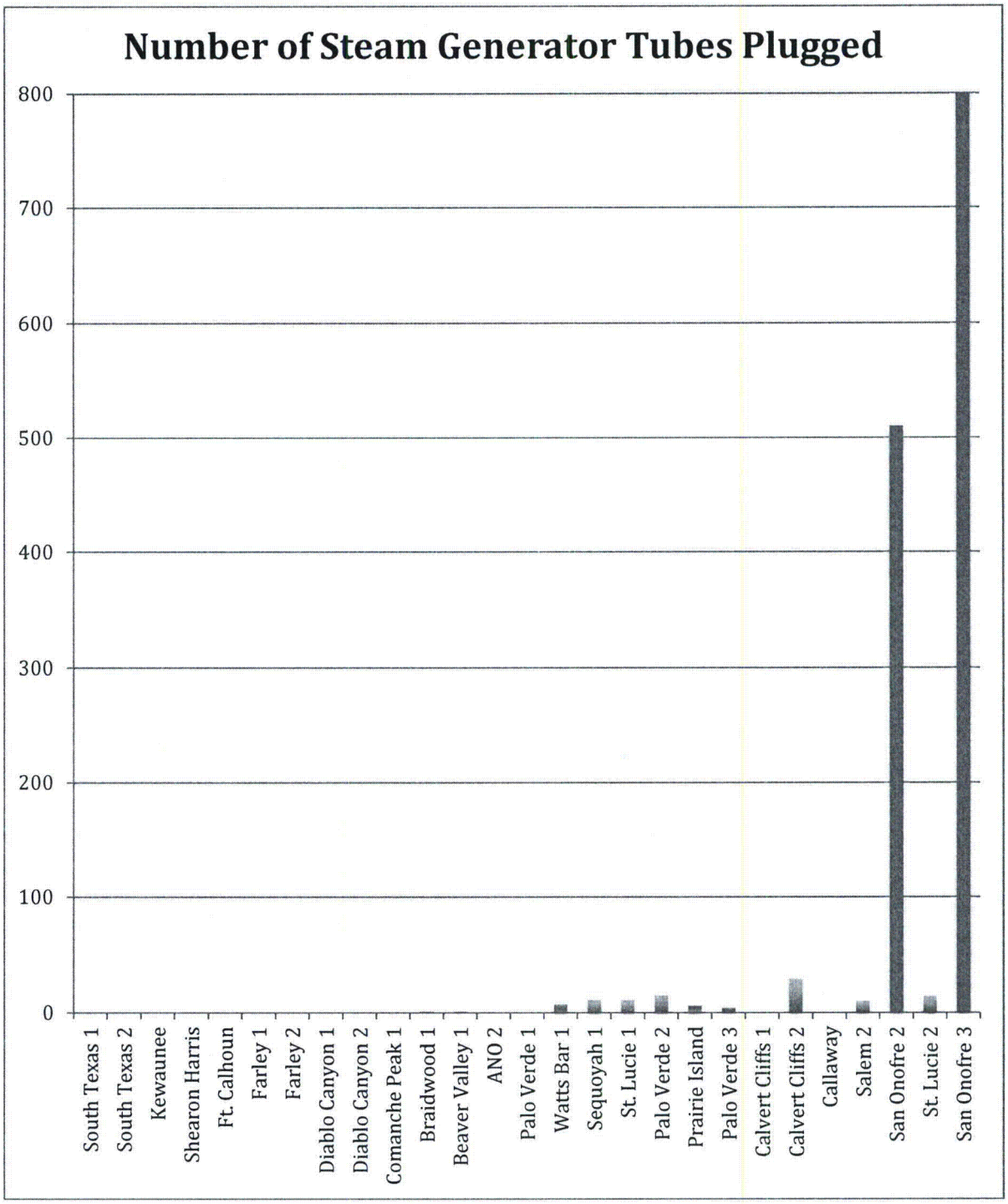

Number of Damaged
Steam Generator Tubes
2500
2000
1500
1000
500
0
South Texas 1
South Texas 2
Kewaunee
Shearon Harris
Ft. Calhoun
Farley 1
Farley 2
Diablo Canyon 1
Diablo Canyon 2
Comanche Peak 1
Braidwood 1
Beaver Valley 1
ANO 2
Palo Verde 1
Watts Bar 1
Sequoyah 1
St. Lucie 1
Palo Verde 2
Prairie Island
Palo Verde 3
Calvert Cliffs 1
Calvert Cliffs 2
Callaway
Salem 2
San Onofre 2
St. Lucie 2
San Onofre 3

Figure 5


Number of Steam Generator Tubes Plugged
800
700
600
500
400
300
200
100
0
South Texas 1
South Texas 2
Kewaunee
Shearon Harris
Ft. Calhoun
Farley 1
Farley 2
Diablo Canyon 1
Diablo Canyon 2
Comanche Peak 1
Braidwood 1
Beaver Valley 1
ANO 2
Palo Verde 1
Watts Bar 1
Sequoyah 1
St. Lucie 1
Palo Verde 2
Prairie Island
Palo Verde 3
Calvert Cliffs 1
Calvert Cliffs 2
Callaway
Salem 2
San Onofre 2
St. Lucie 2
San Onofre 3

The Damage at Both San Onofre Units Greatly Exceeds That at Typical Reactors

The data for replacement recirculating steam generators nationally indicate:

- **The median number of steam generator tubes showing wear after one cycle of operation nationally is—FOUR. San Onofre Unit 2 had 1595 damaged tubes, approximately 400 times the median; San Onofre Unit 3 had 1806.**

- **The median number of wear indications on steam generator tubes after one cycle of operation is—FOUR. San Onofre Unit 2 had 4721, greater than a thousand times more. San Onofre Unit 3 had 10,284.**

- **The median number of steam generator tubes that were plugged after one cycle of operation is—ZERO. San Onofre Unit 2 had 510; Unit 3 had 807.**[xxi]

CONCLUSION

The conclusion is clear: San Onofre Unit 2 and Unit 3 are both very ill nuclear plants. Unit 3's fever is slightly higher, but both are in serious trouble. What they are experiencing is not just normal wear due to "settling in" purportedly experienced with similar replacement steam generators. They are far, far outside the norm of national experience. And Unit 2 cannot be said to be acceptable for restart, any more than Unit 3. Unit 2 has hundreds of times more bad tubes and a thousand times more indications of wear on those tubes than the typical reactor in the country with a new steam generator, and nearly five times as many plugged tubes as the rest of the replacement steam generators, over a comparable operating period, in the country combined. Restarting either San Onofre reactor with crippled steam generators that have not been repaired or replaced would be a questionable undertaking at best.

ENDNOTES

[i] Edison International, "Southern California Edison Announces Intent to Downsize Staffing at San Onofre Nuclear Generating Station," August 20, 2012, http://www.edison.com/pressroom/pr.asp?id=7986, last accessed September 9, 2012.

[ii] NRC Draft Regulatory Guide DG-1074, *Steam Generator Tube Integrity,* Dec 1998, ML003739223.

[iii] http://www.nrc.gov/info-finder/reactor/San Onofre/San Onofre-steam-generator-internal-diagram.pdf last accessed September 9, 2012.

[iv] There is a fifth potential damage mechanism, damage by foreign object (i.e., loose parts). Only two tubes at SAN ONOFRE showed this type of damage.

[v] Kenneth Karwoski, Leslie Miller, and Nadiyah Morgan, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Regulatory Perspective on Steam Generator Tube Operating Experience," in *Ageing Issues in Nuclear Power Plants* (undated).

[vi] NRC AIT report, p. 3-4.

[vii] Esmeralda Bermudez, "San Onofre nuclear power plant incidents draw attention: A radiation leak, the discovery of tube damage and a worker falling into a reactor pool all happened within days of one another," *Los Angeles Times,* February 5, 2012. See also the *Wall Street Journal,*. Rebecca Smith, "Worn Pipes Shut California Reactors," February 4, 2012.

[viii] The NRC meeting was videotaped and the answers by the senior SCE executive to questions about the tube data can be viewed at http://www.youtube.com/watch?v=VPxDYWa0b8o and http://www.youtube.com/watch?v=8tCQWeEauHo. last accessed 9/6/12.
The question asked was for the number of tubes in each SAN ONOFRE Unit that had greater than 10% through-wall wear, and also the total number of tubes showing any indication of wear. SCE provided data about the former.

[ix] http://www.nrc.gov/info-finder/reactor/songs/songs-unit-2-steam-generator-tube-wear-data.pdf last accessed September 9, 2012

[x] http://www.nrc.gov/info-finder/reactor/songs/songs-unit-3-steam-generator-tube-wear-data.pdf last accessed September 9, 2012.

[xi] email, Hirsch to Elmo Collins, July 13, 2012; response July 17; telcon with Tom Blount, Ryan Lantz, Michael Bloodgood, July 18.

[xii] NRC AIT report, pp. i, 46-56

[xiii] NRC AIT report, p. 10

[xiv] *North County Times*, "SAN ONOFRE: Rate of tube wear at nuke plant 'unprecedented,' NRC says," April 4, 2012

[xv] SCE press release, "Southern California Edison Releases Steam Generator Tube Wear Data," July 13, 2012

[xvi] Conference call July 18, *supra.*

[xvii] emails from Tom Blount, June 17, 2012; from Ryan Lantz, July 31; from Kenneth Karwoski, August 7.

[xviii] August 7 and 9, 2012, emails from Ryan Lantz, Chief, Reactor Projects Branch D, NRC Region IV.

[xix] Advisory Committee for Reactor Safeguards, NRC, July 23, 2012, letter to R.W. Borchardt, Executive Director for Operations, NRC, "SUBJECT: Final Safety Evaluation Report Associated with the Florida Power and Light St. Lucie, Unit 2, License Amendment Request for an Extended Power Uprate," p. 3

[xx] *ibid.*, p. 4

[xxi] Arnie Gundersen, in a July 2012 report, "San Onofre's Steam Generators: Significantly Worse Than All Others Nationwide," previously pointed out the high number of plugged tubes at San Onofre compared to plugging rates nationally, based on data in a 2006 NRC report. SCE tried to dismiss the significance of those findings by saying the data were old and that many tubes plugged at San Onofre were plugged preventively. The present study examines more current data, finding the same trend for plugged tubes, but also determines that this is not due to preventive plugging, since the number of damaged tubes and wear indications on tubes at San Onofre far exceeds the national median.

APPENDIX A

PLANT-BY-PLANT DESCRIPTIONS OF REPLACEMENT RECIRCULATING STEAM GENERATOR TUBE WEAR EXPERIENCE

A) PLANT-BY-PLANT DESCRIPTIONS OF REPLACEMENT RECIRCULATING STEAM GENERATOR TUBE WEAR EXPERIENCE

Arkansas Nuclear One, Unit 2: 2 replacement steam generators installed in 2000. 0 tubes plugged during first InService Inspection (ISI) of the steam generator tubes after installation, 1 tube plugged prior to service.

3 wear indications in 3 tubes identified during 1st ISI. Source: April 2002 ISI report, NRC Agencywide Documents Access and Management System (ADAMS) Accession Number ML031080421, pg 4 of PDF/pg 2 of attachment & pg 6 of PDF/pg 4 of attachment. (Note, hereafter NRC ADAMS Accession numbers will be given just by their ML #. Also note that the PDF page number is often different from the document's page number due to how pages are numbered in the cited documents). See also ML031820241, the 2003 NRC review of the licensee's ISI report.

The 2 replacement steam generators are Westinghouse model Delta 109, pg 3 of PDF/pg 1 of attachment of April 2002 tube inspection ML031080421.

The total number of tubes is not explicitly stated in those reports but it is stated that 100% of unplugged tubes were tested with the bobbin coil according to the 2003 NRC review ML031820241, pg 3 of PDF/unnumbered in report. Pg 4 of PDF/pg 2 of the April 2002 tube inspection ML031080421 states that 10,637 tubes were inspected for SG A and 10,636 were tubes were tested in SG B, which had one tube plugged by the manufacturer prior to installation, for a total of 21, 273 inspected, and 21,274 total when the pre-installation plugged tube is included.

Beaver Valley, Unit 1 in Pennsylvania: 3 replacement steam generators 2006. 1 tube plugged during first ISI after installation.

1 tube with 1 wear indication of 29%, believed to have been caused by a burr left from the manufacturing process. Source: 2007 ISI report ML080800448, see the table in pgs 4-6 of PDF, pgs 3-5 of the report, source for explanation is on pg 7 of PDF/pg 6 of attachment 1

The 3 replacement SGs are Westinghouse Model 54s, manufactured by ENSA in Spain, and containing 3,592 tubes each according to the preservice inspection report ML061990398, pg 21 of the PDF/pg 1 of Appendix 2.

Braidwood, Unit 1: 4 replacement steam generators 1998. 1 tube plugged during first ISI, 3 tubes plugged prior to service.

One tube with one wear indication as stated in the 2000 tube inspection report ML010930262, pgs 8-10 of PDF/pg 7-9 of report. The single tube with one wear indication, that was subsequently plugged, had less than 10% through wall (TW) wear according to the 2000 steam generator inspection report ML010930262, pg 10 of PDF/pg 9 of report, this tube was preventively plugged (pgs 4-5 of PDF/pgs 3-4 of report).

The 4 replacement steam generators are Babcock and Wilcox models with 6,633 tubes per generator, see pg 4 of PDF/pg 3 of report

Callaway, Unit 1 in Missouri: 4 replacement steam generators 2005
0 tubes plugged during first ISI, 1 tube plugged prior to service.

214 wear indications on 36 tubes. The greatest through wall wear was 1 indication of 13%, the least was 1%. See Table 2, Summary of Wear Indications, pg 5-11 of PDF/pg 2-8 of attachment 1of the 2007 ISI, ML 073050323.

The steam generators have 5536 tubes each, SG A had one tube plugged prior to service for a total of 5,535 inspected and operational tubes. (pg 5 of PDF/pg 2 of report).

Calvert Cliffs, Unit 1 in Maryland: 2 replacement steam generators in 2002.
0 tubes plugged.

189 wear indications on 166 tubes. The great majority had wear under 10% and only two had wear equal or greater than 20%, at 20% and 22%, according to the 2004 tube inspection report ML050610714, attachment 1, pgs 4-8 of PDF/pgs 1-5 of attachment.

Both Babcock & Wicox replacement steam generators have 8,471 tubes each. See 2005 NRC review ML051440076, pg 3 of PDF/unnumbered in document.

Calvert Cliffs, Unit 2 in Maryland: 2 replacement steam generators in 2003.
29 tubes plugged in first ISI, 3 tubes plugged prior to service.

Of the 29 tubes plugged due to the 2005 inspection, 5 had wear indications and the other 24 were plugged as a precautionary measure due to a possible loose part in an area which cannot be visually inspected. See 2005 memo of NRC-licensee conference call, ML052410150, pgs 1-2 of PDF & memo.

All told, there were 200 wear indications on 170 tubes, with the majority having wear under 10%. 8 tubes had wear 20% or greater, with the highest indication being one tube with 25% wear. See 2005 tube inspection report ML060610081, pg 4-9 of PDF/1-6 of attachment.

The replacement steam generators have 8471 tubes each, with 3 plugged prior to service, according to the cover letter to the tube inspection report ML060610081, pg 1 of PDF/pg 1of letter, and are described as Babcock & Wilcox design and manufacture in 2005 memo ML052410150, pg 1 of PDF & memo.

Comanche Peak, Unit 1 in Texas: 4 replacement steam generators in 2007.
0 tubes plugged during first ISI, 1 tube plugged during manufacture.

1 wear indication on 1 tube, depth ,10% TW. See ISI report 2008 pg 7 of PDF/pg 5 of ISI report ML090300118, pg 9 of PDF/pg 7 of report.

The steam generators are Westinghouse Model Delta 76s with 5,532 tubes per steam generator, reference steam generator tube inspection 2008 ML090300118, pg 3 of PDF/pg 1 of report.

Diablo Canyon, Unit 1 in California: 4 replacement steam generators in 2009
0 tubes plugged.

1 wear indication on 1 tube, at 5% TW. See 2010 steam generator inspection report ML111160101, pg 3,4, and 11 of PDF/pg 2,3, and 10 of enclosure. This one wear indication was the first report of AVB wear in Westinghouse model 54s, leading PG&E to inform the NRC on Oct 15,2010 (pg 4 of PDF/pg 3 of enclosure for ML111160101).

The replacement steam generators are Westinghouse Model Delta 54s and each one contains 4,444 tubes, according to the 2012 Nuclear Regulatory Commission review ML120740373, pg 2 of PDF & review and the 2010 steam generator inspection report ML111160101, pg 2 of PDF/pg 1 of report.

Diablo Canyon, Unit 2 in California: 4 replacement steam generators in 2008
0 tubes plugged during first ISI, 3 tubes plugged prior to service.

1 wear indication on 1 tube, see 2009 steam generator inspection ML101330269, pg 3 of PDF/pg 2 of enclosure.

The replacement steam generators are Westinghouse Model Delta 54s with 4,444 tubes each, according to pg 2 of PDF/pg 1 of enclosure above.

Farley, Unit 1 in Alabama: 3 replacement steam generators in 2000.
0 tubes plugged.

NO wear indications, see Fall 2001 ISI report ML020300072, pg 12 of PDF/unnumbered in report and 2002 supplemental information ML021960109, pg 4 of PDF/pg 2 of letter.

Westinghouse model 54F steam generators, 2001 inservice inspection ML020300072, pg 12 of PDF/unnumbered in report.

3,592 tubes in each of the 3 replacement steam generators, as stated in 2003 NRC review ML031110259.

Farley, Unit 2 in Alabama: 3 replacement steam generators in 2001.
0 tubes plugged.

NO wear indications. See Fall 2002 ISI report ML030300235 pg 12 of PDF/unnumbered in report, Sept/Oct 2002 inspection.

Westinghouse model 54F steam generators with 3,592 tubes per steam generator; see 2008 NRC Review ML083100232, pg 3 of PDF/unnumbered in enclosure.

Fort Calhoun in Nebraska: 2 replacement steam generators in 2006.
0 tubes plugged in first ISI, 1 tube plugged prior to service.

NO wear indications. See 2008 eddy current test ML083440629, pg 3 of PDF/pg 2 of attachment, pgs 9-11 of PDF/pgs 8-10.

Both Mitsubishi MHI-49TT-1 steam generators have 5,200 tubes each. See steam generator tube inspection review ML093000157, pg 2 of PDF/unnumbered in report.

Kewaunee in Wisconsin: 2 replacement steam generators in 2001.
0 tubes plugged.

NO wear indications. See 2003 annual report ML0460650370, pg 6 of PDF/pg 2 of report, and 2003 ISI ML032250165 pgs 156 &157 of PDF.

Westinghouse model 54Fs with 3,592 tubes in each steam generator, from April 2003 steam generator inspection ML032250165, pg 155 of PDF/pg 1 of attachment 8.

Palo Verde, Unit 1 in Arizona: 2 replacement steam generators 2005.
0 tubes plugged during first ISI, 116 tubes plugged prior to service.

4 wear indications on 4 tubes, <20% TW. See 2007 ISI report ML080090193, pg 9 of PDF/unnumbered in report, pgs 14-17 of PDF/unnumbered in report, Appendices B & C.

Palo Verde Units 1, 2, and 3 have essentially the same design for their replacement steam generators. They were all "designed by Asea Brown Boveri/Combustion Engineering (ABB/CE) (now Westinghouse) and manufactured by Ansaldo, and are considered a modified System 80 design (no specific model number)." There are 12,580 tubes for each steam generator; see ML082890538, pg 3 of PDF, pg 1 of enclosure.

Palo Verde, Unit 2 in Arizona: 2 replacement steam generators in 2003.
15 plugged during first ISI, 24 plugged prior to service.

81 wear indications on 48 tubes. See the data tables in 2005 tube ISI report ML053130156, pg 11 of PDF/unnumbered in report, Table 2 Indication Summary, pgs. 29-38 of PDF, Appendices C & D of report.

[Dents found were pre-existing before operation and not due to operational wear. According to the supplement to the steam generator report ML 060890657, pg 10 of PDF/pg 8 of enclosure, the dents were present in the preservice inspection, 100% of the dents > or equal to 0.5 volts were inspected in 2005 and none exhibited any change between the preservice inspection and the 2005 inspection. Regarding the dents that were plugged, these were plugged preventively though they hadn't changed any either, reference pg 3 of PDF/pg 1 of enclosure.]

There are 12,580 tubes per steam generator.

Palo Verde, Unit 3 in Arizona: 2 replacement steam generators in 2007.
4 tubes plugged during first ISI, 118 plugged prior to service.

140 wear indications on 68 tubes, according to Palo Verde 3 ISI report ML093310442, pg 10 of PDF/ pg 8 of report, Appendices B & C, pgs 15-22 of PDF/pgs 13-20.

Steam generators have 12,580 tubes in each. NRC review ML112060490, pg 2 of PDF/unnumbered in review.

Prairie Island, Unit 1 in Minnesota: 2 replacement steam generators in 2004.
6 tubes plugged during first ISI.

104 wear indications in 67 tubes, 2006 teleconference re: tube inspection ML061680005, pg 4 of PDF/pg 2 of report.

Framatome Model 56/19s with 4,868 tubes each, according to revision to the ISI ML101530111, pg 9 of PDF/pg 1 of enclosure 2.

Saint Lucie, Unit 1 in Florida: 2 replacement steam generators in 1997.
11 tubes plugged preventively during first ISI.

19 wear indications on 17 tubes, 1999 ISI, ML 003684169, pgs 4-6 of PDF/unnumbered in report.

Each Babcock and Wilcox advanced series pressurized water reactor steam generator has 8,523 tubes, according to 2008 NRC review ML100960626, p. 2 of PDF/unnumbered in review.

Saint Lucie, Unit 2 in Florida: 2 replacement steam generators in 2008.
14 tubes plugged during first ISI.

5,994 wear indications on 2,174 tubes. See 2009 tube inspection ML093230226, pg 13-115 of PDF/pgs 2-64 of attachment 1, pgs 2-40 of attachment 2.

Only 2 indications exceeded 30% wear, no indications over 35%; 2009 tube inspection ML093230226, pg 14 of PDF/pg 3 of Attachment 1, pg 78 of PDF/pg 3 of Attachment 2

Steam generators are Areva-NP Model 86/19TIs, 2009 tube inspection ML093230226, pg 2 of PDF/pg 1 of enclosure and have 8999 tubes each, according to the NRC review of 2009 tube inspection ML03340040, pg 2 of PDF/pg 1 of enclosure.

Salem, Unit 2 in New Jersey: 4 replacement steam generators in 2008.
10 tubes plugged during first ISI.

1,567 wear indications on 591 tubes, see 2009 steam generator tube inspection report ML101250176, pg 10 of PDF/pg 1 of attachment 3.

The steam generators are Areva Mod 61/19Ts with 5,048 tubes per steam generator, 2009 tube inspection ML101250176, pg 4 of PDF/pg 1 of attachment 1.

San Onofre 2 in California: 2 replacement steam generators in 2010.
510 tubes plugged during first ISI.

4721 wear indications on 1,595 tubes. See NRC tables in main body of report.

Mitsubishi steam generators with 9,727 tubes per generator. See Southern California Edison, "San Onofre Nuclear Generating Station Confirmatory Action Letter Fact Sheet," last updated on 6/13/2012

San Onofre 3 in California; 2 replacement steam generators in 2011.
807 tubes plugged within one year of installation (tube failure during operation led to shutdown and inspection prior to normal ISI.)

10,284 wear indications on 1806 tubes.

Mitsubishi steam generators with 9,727 tubes per generator, same as Unit 2.

Sequoyah, Unit 1 in Tennessee: 4 replacement steam generators in 2003.
11 tubes plugged during first ISI, 20 plugged prior to service.

11 wear indications on 11 tubes; see 2004 ISI report ML050550413, pg 55 of PDF/unnumbered Appendix A.

All 11 tubes plugged as a result of this inspection were preventively plugged with TW% ranging from 8-17% according to Sequoyah 1 steam generator inspection ML053050386, pg 3 of PDF/unnumbered in report.

Model 57AG steam generators by Doosan, 4,983 tubes per SG. 2006 NRC review ML060950510, p. 4 of PDF/unnumbered in review.

Shearon Harris in North Carolina: 3 replacement steam generators in 2001
0 tubes plugged during first ISI, 2 tubes plugged during manufacture.

0 wear indications, 2003 ISI ML032680868, pg 7 of PDF/unnumbered report supplemental information ML041120371 pg 4 of PDF/pg 2 of attachment, pg 7 of PDF/pg 5 of attachment, 2003 tube test ML041320496 pg 5 of PDF/pg 2 of attachment 1.

Westinghouse Model Delta 75 replacement steam generators, 6,307 tubes in each steam generator, 2003 tube test ML041320496, pg 4 of PDF/pg 1 of attachment 1, and pg 3 of PDF, pg 1 of attachment,, ML042360545.

South Texas Project, Unit 1: 4 replacement steam generators in 2000.
0 tubes plugged during first ISI, 108 tubes pre-service.

0 wear indications, see 2001 ISI ML020390361, pg 12 of PDF/pg 7 of report.

Steam generators are Westinghouse Model Delta 94s with 7,885 tubes per steam generator, pg 6 of PDF/pg 1 of above report.

South Texas Project, Unit 2: 4 replacement steam generators in 2002.
0 tubes plugged during first ISI, 6 tubes plugged pre-service.

0 wear indications, 2004 ISI ML041730355, pg 13 of PDF/pg 8 of report, pg 14 of PDF/pg 9 of report.

Steam generators are Westinghouse Delta 94s with 7,585 tubes each, see South Texas Project 2 pre-service inspection ML030710429 pg 6 of PDF/pg 1 of report

Watts Bar, Unit 1 in Tennessee: 4 replacement steam generators in 2006.
7 tubes plugged during first ISI, 2 plugged prior to service.

9 wear indications on 6 tubes. All the tubes with any wear indications were plugged preventively. One tube with a tube sheet bulge detected prior to service was also preventively plugged which is why there were 7 tubes plugged and only 6 tubes with wear indications. The TW% detected ranged from 7% to 13%, well under the plugging limit of 40% TW. Source is 2008 tube inspection ML082600068, pg 5 of PDF/pg E-3 of report, pg 6 of PDF/pg E-4 of report.

Westinghouse designed the replacement steam generators, and Doosan Heavy Industry and Construction manufactured them. There are 5,128 tubes per steam generator, supplemental information ML090960558, pgs 4 and 9 of PDF/pgs 2 and 7 of enclosure.

APPENDIX B

NOTES ON SOURCES AND METHODS

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Licensees are generally required to conduct, at the first shutdown for reactor refueling after installation of replacement steam generators, inspection of 100% of the steam generator tubes. That inspection is typically performed using eddy current testing (ECT). If signals from the ECT suggest a potential problem, frequently follow-on tests are performed to ascertain if indeed there is wear.

The licensee is required to submit to the NRC within a set period after completion a report on the results of the steam generator inspection conducted during the In-Service Inspection (ISI). NRC staff review the ISI report, and will occasionally submit requests for additional information to the licensee. Thus, the primary records related to the number of wear indications found during an ISI, the number of tubes experiencing wear, and the number of tubes plugged during the ISI, are: the ISI report itself, requests for additional information by NRC and responses thereto by the licensee, and correspondence by NRC concluding its review. When there is a significant problem identified, NRC may initiate a meeting or conference call with the licensee and a memorandum may result therefrom. Lastly, the pre-service inspection report—after installation but before operation with replacement steam generators—may also provide useful information about steam generator design and dings, dents, and manufacturing burnishing marks that pre-date operation and thus, if noted thereafter, are not due to operational wear.

Unfortunately, the ISI reports are not always entirely consistent in form and content from one licensee to another. Sometimes a summary is provided quantifying the total numbers of tubes and indications of wear that observed; other times one has to tabulate the figures by hand. Additionally, definitions are not always clear or consistent. For example, guidance from the Electric Power Research Institute (EPRI) defines wear as "the loss of tube material caused by excessive rubbing of the tube against its support structure, a loose part, or another tube," but also uses the term "degradation" as wear of greater than 20% or greater through wall (TW). ML ML080450582. NRC draft guidance on steam generator tube integrity, by contrast, defines a degraded tube as a tube showing any wear below the applicable plugging limit. ML003739223. To avoid any question, data for wear rather than degradation were relied upon for this report.

Furthermore, the raw data were reviewed to confirm, for example, that all measurable wear was in fact reported, not just wear below a threshold such as 20% TW. This was readily determinable for virtually all of the plants, as they reported wear down to a few % TW, and for those that reported zero wear, statements in the ISI or NRC communications generally made clear that this indeed meant no measurable wear.

In some cases, a few tubes were identified in the ISI reports as being involved with possible loose parts in the steam generators. Where damage to the tubes was indicated by %TW wear indications, they were generally included; where it appears that subsequent evaluation had determined no TW damage, they were not.

In some cases, tubes were plugged by the manufacturer or otherwise prior to operation. In Appendix A, tubes plugged prior to operation and tubes plugged thereafter at the time of the first ISI are both identified. Table 3 and Figures 3-5 of the main body of the report, however, are

worn tubes, i.e., those damaged by steam generator operation. The reports also generally identified dents, dings, manufacturing burnishing marks and the like that pre-dated operation. These also were not included here, as the analysis is on wear due to operations.

It is possible that ambiguities remain in the ISI reports that were not fully resolvable by reviewing associated documents such as correspondence with NRC, but it appears that they would not have any substantive effect on the fundamental conclusions of this report. One take-away suggestion from this analysis, however, is that greater uniformity and clarity in ISI reports would be helpful in analyzing national trends.

APPENDIX C

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BIBLIOGRAPHY

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APPENDIX D

ABOUT THE AUTHORS

DANIEL HIRSCH is President of the Committee to Bridge the Gap and has been associated with it since 1970. He is also a Lecturer at the University of California, Santa Cruz, where he teaches courses on Nuclear Policy and Environment Policy. He is the former Director of the Adlai E. Stevenson Program on Nuclear Policy at UCSC.

DORAH ROSEN is a Research Associate with the Committee to Bridge the Gap. She had primary responsibility for acquiring and compiling the steam generator tube data from licensee ISI reports and related documentation.

DALE BRIDENBAUGH is a retired Nuclear Engineer with forty years experience with the commercial nuclear industry. He was a nuclear engineer and manager for General Electric's Nuclear Division, spending twenty years with GE. In 1976, he and two colleagues resigned from GE and testified before Congress regarding their concerns that safety issues with the GE Mark I containment structures were being ignored. The subsequent Fukushima nuclear accident tragically proved them correct. After their resignation from GE, the three nuclear engineers formed MHB Associates, which for more than twenty years performed studies on the operation, safety and costs of nuclear plants for state agencies and foreign countries.

THE COMMITTEE TO BRIDGE THE GAP is a forty-two-year-old non-profit public policy organization focused on issues of nuclear safety, proliferation, waste disposal, and terrorism.

www.committeetobridgethegap.org

(831) 336-8003