UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE NRC STAFF

In the Matter of	)	
	)	Docket ID NRC-2013-0070
	)	
SOUTHERN CALIFORNIA EDISON CO.	)	
	)	
(San Onofre Nuclear Generating Station,	)	
Units 2 and 3)	)	May 16, 2013
	)	

FRIENDS OF THE EARTH'S AND NATURAL RESOURCES DEFENSE COUNCIL'S COMMENTS IN OPPOSITION TO PROPOSED *NO SIGNIFICANT HAZARDS CONSIDERATION* DETERMINATION

I. INTRODUCTION

Friends of the Earth (FoE) and Natural Resources Defense Council (NRDC) submit the following comments in opposition to the recently proposed *no significant hazards consideration* determination regarding a license amendment request that would modify the terms of San Onofre Nuclear Generating Station Unit 2's operating license. FoE and NRDC assert that there should be a hearing prior to any Nuclear Regulatory Commission (NRC) decision on the proposed license amendment. The proposed amendment would allow operation at no more than 70% Rated Thermal Power (RTP) (or 2406.6 megawatts thermal) for the duration of Cycle 17.[1,2]

To analyze the proposed *no significant hazards consideration* determination, Friends of the Earth has enlisted the assistance of four experts with substantial and relevant experience related to the issues presented by the proposed *no significant hazards consideration* finding:

[1] *Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generation Station, Unit 2*, 78 Fed. Reg. 22576 (April 16, 2013) ("Notice of License Amendment Proposal").
[2] Because Edison plans to operate Unit 2 intermittently during Cycle 17, this operational period could last between 22 and 24 months.

- Nuclear engineer and former NRC Staff, Dr. Joram Hopenfeld: During his time at the NRC, Dr. Hopenfeld's work led to the creation of a Steam Generator Action Plan to address safety issues in steam generators. He has extensive experience with steam generator tube failure.
- The Honorable Victor Gilinsky, former Commissioner of the United States Nuclear Regulatory Commission: During Dr. Gilinsky's NRC tenure Congress passed the *Sholly* amendment[3] and the Commission first interpreted and applied the amendment.
- Nuclear engineer John Large, of Large & Associates: Mr. Large is a Consulting Engineer and Chartered Engineer, who was a full-time member of the Academic Staff at Brunel University for over 25 years. Mr. Large frequently provides expert evidence on nuclear systems failures and other technical issues in the U.K. Crown and Civil Courts.
- Mr. Arnold Gundersen, a nuclear engineer: Mr. Gundersen is a former licensed nuclear reactor operator and Chief Engineer at Fairewinds Associates.

Public comment on the proposed *no significant hazards consideration* determination is made difficult by the lack of information associated with both the specific license amendment request and the Staff's proposal. In order to respond to the proposed determination, Friends of the Earth's experts reviewed studies submitted by SCE in the parallel Confirmatory Action Letter (CAL) proceeding supporting a proposal to restart Unit 2 at 70% of power.[4]

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[3] Incorporated into the Atomic Energy Act at 42 U.S.C. § 2239(a)(2)(A).
[4] SCE submitted the operational assessments reviewed here in response to a March 27, 2012 CAL. *See* Letter from Elmo E. Collins, Regional Administrator, Region IV, Nuclear Regulatory Commission, to Peter T. Dietrich, Senior Vice President & Chief Nuclear Officer, Southern California Edison, Confirmatory Action Letter – San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation, CAL 4-12- 001 (Mar. 27, 2012), available at ADAMS Accession No. ML12087A323.

Having reviewed these submissions of SCE in support of the proposal to allow operation of Unit 2 at 70% of power, and the analyses of Mr. Large, Dr. Hopenfeld, Dr. Gilinsky, Mr. Gundersen, and consistent with LBP-07-13, the May 13, 2013 opinion of the Atomic Safety & Licensing Board (ASLB),[5] discussed below, FoE and NRDC request that the proposed *no significant hazards consideration* determination should be withdrawn because (1) the Staff's proposal exceeds the authority granted to it by the *Sholly* amendment; (2) the licensee's application of the criteria under 10 C.F.R. § 50.92, as adopted by the NRC Staff, does not justify a finding of *no significant hazards consideration*; and (3) the Staff have not performed an environmental review of the proposed finding and license amendment as required by the National Environmental Policy Act (NEPA), and the proposed actions do not satisfy criteria for a categorical exemption from NEPA review, provided at 10 C.F.R. § 51.22(c)(9)(i).

FoE's and NRDC's analysis is supported by the ASLB's May 13, 2013 decision holding that the licensee's restart plan, which proposes operation in conformance with the proposed license amendment, constitutes a *de facto* license amendment proceeding. The decision is based not only on the need to revise technical specification 5.5.2.11.b.1 through the license amendment proposed in the present action, but also on the need to revise the Updated Final Safety Analysis Report (UFSAR), which currently fails to account for in-plane fluid elastic instability (FEI)—one of the main defects present in the replacement steam generators. Approving the requested license amendment—temporary operation at 70% of power—would authorize the licensee to operate the plant with an outdated and insufficient UFSAR; another reason why a finding of *no significant hazards consideration* is inappropriate in this case.

[5] *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), LBP-13-07 (May 13, 2013) ("ASLB Order"), appended to these comments as Attachment 6.

The Staff has ignored the language and legislative history of the *Sholly* Amendment as recited by the Ninth Circuit in the *Mothers for Peace* case, discussed below. As the court held, the amendment gives the NRC Staff the right to screen out trivial changes that could not possibly affect safety. If a safety issue is identified, however, then the Staff must legally conclude that a significant hazards consideration exists, and must refer the issue(s) to an Atomic Safety and Licensing Board for a hearing prior to the decision on the proposed license amendment. The court ruled that the NRC Staff should "not resolve doubtful cases with a finding of no significant hazards consideration." The court added, from the legislative history, that the NRC Staff should not "prejudge the merits" of the issues by a proposed license amendment. The Staff does not determine whether a significant hazard exists; that is for the ASLB to determine. Thus the Staff's proper role under the *Sholly* amendment is essentially ministerial: it determines whether a significant hazard *consideration* exists, and if so, asks the ASLB to determine whether the proposed amendment creates a significant hazard. Consistent with the *Mothers for Peace* case, the Staff must refer SCE's proposed license amendment to an ASLB: A *probability*, not just a *possibility*, of significant hazards permeates this case.

It follows that the Staff has disregarded its appropriate role in proposing a *no significant hazard consideration* finding in this instance. The safety issues presented here are real, not trivial, as acknowledged in the ASLB panel decision of May 13, 2013. Under applicable law, the Staff must withdraw the proposal and refer the proposed license amendment to the ASLB for an adjudicatory hearing, as requested by Friends of the Earth and NRDC in these comments, before a decision on the proposed amendment can be made.

II. FACTUAL BACKGROUND

a. The Shut Down of Units 2 and 3

On January 31, 2012, San Onofre experienced a steam generator tube rupture in Unit 3 that resulted in the release of radioactive material into the environment. The licensee, Southern California Edison Company (SCE or "Edison") also discovered excessive wear in the Unit 2 replacement steam generators. The unit was offline for a refueling outage. Subsequently, untimely degradation of the walls of many tubes was discovered in the replacement steam generators, which had been in operation for eleven months in Unit 3 and less than two years in Unit 2.

On March 23, 2012, SCE submitted a description of the steam generator problems and its commitments to address the issues at Units 2 and 3, which were formalized in a CAL to SCE on March 27, 2012.

None of the investigations conducted to date have determined the root cause of the premature and extensive tube degradation in the replacement steam generators. Lacking such understanding, SCE has not proposed any action to actually fix the problems of either Unit 2 or Unit 3. Rather, SCE has proposed a restart plan based on substantially reduced operational limits that it has asserted is safe.

b. Southern California Edison's Replacement of the Steam Generators at San Onofre Units 2 and 3

In 2010 and 2011, SCE replaced the original steam generators in Unit 2 and Unit 3, which had operated for 28 years, with ones constructed by a different manufacturer, Mitsubishi Heavy Industries (MHI). The new design differed from the original in significant ways. SCE requested MHI, for example, change the design by adding 377 more tubes, remove the stay cylinder supporting the tube sheet, and replace the "egg crate" tube support with a broached design, among other alterations.[6] SCE convinced itself that the replacement steam generators were a "like for like" replacement for the old ones and did not seek a license amendment for these changes. Thus, the impact of these changes on safe operation of the plant has not previously been evaluated by the NRC.

c. Extent of Tube Degradation in the Steam Generators in Units 2 and 3

Both units show indications of extensive tube wear after fewer than two years of operation. The tube degradation in each unit is unlike, in both mechanism and extent, tube wear in other replacement steam generators in other U.S. plants at the same stage of their useful lives:

- San Onofre Unit 2 has 1595 degraded tubes; Unit 3 has 1806;
- Unit 2 has 4721 tubal wear indications; Unit 3 has 10,284;
- Unit 2 has 510 tubes plugged after one cycle of operation of the replacement steam generators; Unit 3 has 807;
- SCE and NRC have reported that 9% of the tubes in Unit 3 steam generators have greater than 10% through-wall wear indications; in Unit 2, 12% of the tubes show such wear.

Tube wear of this magnitude after such an abbreviated period of operation is unprecedented.[7]

d. SCE's Previous Assessments of Operation at 70% of Rated Thermal Power

As part of its response in the CAL process, SCE submitted to NRC Staff numerous operational assessments by its consultants. While agreeing that the proximate cause of wear of the replacement steam generator tubes was excessive vibration, SCE and its consultants have not identified a root cause of the excessive vibration causing the premature and extensive tube wear,

[6] For a more detailed description of the changes, see Declaration of Arnold Gundersen (May 31, 2012) (Originally submitted to the NRC as an attachment to a June 18, 2012 Petition to Intervene by Friends of the Earth) at ¶¶ 22-23 and MHI Root Cause Analysis and Supplemental Technical Evaluation Report at pp. 47-48, appended to these comments as Attachment 4 and 5, respectively.

[7] *See* ASLB Order at p. 25 (citing SCE's statement to that effect).

as described in Table 6-1 of SCE's Unit 2 Return to Service Report. In fact, Edison's own consultants disagree with one another on the mechanistic cause of the tube wear. SCE's response to the CAL includes an analysis of tube-to-tube wear and argues that the cause of such wear is FEI. However, this response does not identify the root cause that produced the FEI or acknowledge other thermal hydraulic forces at work in the steam generators. Without knowing the root cause, as declarant nuclear engineer John Large asserts, it is not possible to determine whether the steam generators can be safely operated in their current condition.

SCE's response to the CAL includes a proposal to restart Unit 2 at no more than 70% power for 150 cumulative days, at which time SCE promises to shut down the reactor and inspect the tube wear. The current proposed license amendment is required because Edison has failed to demonstrate to the NRC that it can meet the terms of the existing license requiring a demonstration of tube structural integrity at 100% of power. This point alone makes it impossible for the NRC to reach a determination that Edison's proposed license amendment presents *no significant hazards consideration*.

Edison's response to the CAL is nearly identical to the license amendment request in the present instance: to modify Unit 2's license to limit maximum power for operation at 70% for Cycle 17. SCE hired AREVA NP, Westinghouse Electric Company LLC, and Intertek/APTECH to provide operational assessments (OAs) of this proposal. MHI, the manufacturer of the replacement steam generators, also examined the unprecedented tube wear and present condition of the tubes.

These assessments, which are included in SCE's response to the CAL, not only demonstrate clearly that there are significant hazards to be considered before ruling on the license amendment request, but they also suffer from important omissions. The studies focus on tube-to-tube wear as the threat to tube rupture, incorrectly assuming that this mode of wear will outpace all other wear modes. They do not analyze the potential safety effects of further degradation of the tubes in Unit 2 that are vibrating against the retainer bars and tube restraint structures; nor do the OAs address extent and impact of metal fatigue on the damaged tubes' structural integrity. The OAs point to different mechanical interactions resulting from FEI and random fluid excitation sources as the causes of the tube degradation, but none determined the root cause of the in-plane tube motion excitation forces, which appear to be unique to San Onofre's replacement steam generators.

In addition to failing to identify the root cause of the tube degradation or to recognize the different modes of wear, SCE and its consultants also failed to agree on the projected length of time before a tube burst may occur, even by their own inadequate analysis. Estimates vary from six months to sixteen months of operation at 70% RTP, indicating that the underlying risk analysis is fundamentally flawed.[8] In other words, SCE cannot say with confidence that a tube burst is unlikely within the time frame of Cycle 17, which is 22-24 months, at 70% of power. What both assessments say is that the tubes will deteriorate at a pace that will cause steam generator failure, in the best-case scenario within 16 months, and the worst case 6 months—a mere month more than the period SCE proposes to run the plant, were its license amendment to be approved.

The consultants' estimates are remarkable for two reasons. First, neither projects the unit can be run safely, even at reduced power, for more than 16 months—even though the original expectation of SCE and the designers was that they would last for three decades or more. Second, the two estimates differ by a factor of nearly three. The fact that each of the consultants

[8] Declaration of Mr. John Large, May 16, 2013 ("Large Decl.") at ¶ 8.5.14, appended to these comments as Attachment 2.

relied upon by SCE project significantly different periods of time before reaching and surpassing this safety threshold "shows that the underlying data and methodology of the predictions is fundamentally flawed."[9] In view of this "uncertainty and unreliability" Mr. Large concludes that "little assurance can be placed with SCE's confidence that its Cycle 17 . . . will pass without encountering a significant increase in the risk of tube failure."[10]

Moreover, the estimates by SCE's consultants do not account for the fact that "a certain percentage of steam generator tubes have used up their entire or a large fraction of their allowable fatigue life during cycle 16."[11] Fatigued tubes present a more significant risk than tubes degraded by stress corrosion cracking because tube failure caused by fretting fatigue will result a sudden burst and "proceed rapidly to its maximum as it happened a North Anna (NRC Bulletin 88-02)."[12]

These facts demonstrate the absurdity of the Staff's proposal to conclude that operation of San Onofre as described in the proposed license amendment presents *no significant hazards consideration*. SCE has received assessments on the issue of various tube wear modes by AREVA, the other consultants, and MHI, but SCE did not include important aspects of these assessments in its response to the CAL. For example, SCE have chosen not to emphasize or explain an analysis by MHI, which found that tube wear from contact between the tubes and anti-vibrations bars in Unit 2's replacement steam generators arose in areas of the tube bundle where FEI was inactive, suggesting that the wear was caused by turbulent flow forces that may persist even at the proposed power level of 70% intended to suppress the FEI. In light of these facts, the NRC cannot properly find that *no significant hazards consideration* is raised by

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9 Large Decl. at ¶ 8.5.14.
10 Large Decl. at ¶ 8.5.15.
11 Declaration of Dr. Joram Hopenfeld, May 16, 2013 ("Hopenfeld Decl.") at p. 7.
12 Hopenfeld Decl. at p. 8.

amending the license to allow the unit to be restarted. What is clear is that the proposed amendment *does* present significant hazards consideration that require airing in a public adjudicatory hearing *before* the license amendment can be granted and the unit can be allowed to operate again.

III. **COMMENTS**

We note at the outset that the record for this proposed *no significant hazards consideration* determination is perplexingly thin. The docket contains only the Federal Register notice of the proposed finding (along with a few comments from citizens). To address the significant hazards consideration involved in the proposal to operate the damaged replacement steam generators at 70% of power, FoE's experts were required to review the technical analyses in the public record in another proceeding (*i.e.*, the operational assessments provided by SCE in response to the March 27, 2012 Confirmatory Action Letter).

The Staff has not placed any analysis of the § 50.92 factors into the record.[13] They have ignored the fact that these steam generators are so badly damaged that the licensee has not proposed restarting Unit 3 and concedes that the damaging forces will continue to degrade the steam generator tubes in Unit 2 to the point of failure.

In reality, the Staff's proposal ignores the fact that what is at stake is the licensing of badly damaged steam generators that Edison concedes will continue to be further damaged by operation. It fails even to attempt to explain how operating Unit 2 can pass the rigid standards of 10 C.F.R. § 50.92 and it ignores the *Sholly* amendment by treating as routine the safe operation of a plant that even the Staff admits is not safe to run at full power.

[13] *See* Docket ID No. NRC–2013– 0070.

In a related proceeding, an ASLB convened by the Commission recently determined that SCE's proposal to restart San Onofre Unit 2 at 70% of power on an experimental basis is a *de facto* license amendment proceeding, which requires "rigorous NRC Staff review appropriate for a licensing action."[14] The ASLB found that SCE's proposal would allow Unit 2 to operate outside the current licensing basis of the plant, not only because a maximum operating level of 70% of power does not comply with Technical Specification 5.5.2.11.b.1, but also because restarting the steam generators in their current degraded condition is outside the bounds of the safety analyses that form the licensing basis for the plant (the UFSAR). Having found that "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,' "[15] the Board held that prior to restart SCE is required to submit a license amendment that properly updates the FSAR to include a full assessment of the effects of in-plane fluid elastic instability:[16]

> 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 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.[17]

The Staff's proposal to find that the license amendment request presents *no significant hazards consideration* would authorize SCE to restart Unit 2 at 70% of power without updating

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[14] ASLB Order at 32.
[15] ASLB Order at 34, n. 54.
[16] ASLB Order at 32.
[17] ASLB Order at 32, 33 (internal citations omitted).

the FSAR. The proposal, if made final, would thus contravene the ASLB's order. More fundamentally, the three impartial experts who constitute the panel have confirmed the views of FoE's experts that the proposed license amendment and restart plan is an experiment that raises significant safety issues in all three of the regulatory criteria that must be satisfied in order to make a finding of *no significant hazards consideration*.

For these reasons, the proposed *no significant hazards consideration* determination should be withdrawn. Moreover, (1) the ASLB's conclusions mirror the analyses by FoE's experts that the proposed license amendment fails to meet *any* of the criteria required for a *no significant hazards consideration* finding, and (2) the Staff's proposed determination violates the terms of the *Sholly* Amendment.

a. The Proposed Finding of *No Significant Hazards Consideration* Exceeds the Authority of the NRC Staff

Section 189a of the Atomic Energy Act (AEA) requires that, if requested, a public hearing must be held prior to the issuance of any license or license amendment before an Atomic Safety and Licensing Board.[18] The "Sholly" amendment, 42 U.S.C. § 2239(a)(2)(A), provides a limited exception to this general rule. The NRC staff may issue a license amendment before a hearing *only if* it finds the license amendment raises *no significant hazards consideration*. The relevant regulations are found at 10 C.F.R. § 50.92(c), which we describe in detail below. The legislative history of the *Sholly* amendment makes clear that it is limited to only the most routine license amendments, which may be granted prior to the hearing guaranteed by the AEA.[19]

Under NRC regulation 10 C.F.R. § 50.92, the NRC Staff may not determine that a proposed license amendment raises *no significant hazards consideration*, and thus must refer the

[18] 42 U.S.C. § 2239(a)(1)(A).
[19] *See* Declaration of Victor Gilinsky, May 16, 2013 ("Gilinsky Decl.") at ¶ 5, appended to these comments as Attachment 3 ("Congress permitted [*no significant hazards consideration*] determinations in routine cases that obviously had no or essentially no safety significance, but not otherwise.").

matter to an ASLB for resolution *before* the license amendment may be issued, whenever a proposed license amendment will:

> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated;
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
> (3) Involve a significant reduction in a margin of safety.

The proposal to approve SCE's license amendment application is fundamentally inconsistent with the purpose of the *Sholly* amendment to the Atomic Energy Act, which is the authority for 10 C.F.R. § 50.92. The *Sholly* exception to the rule that proposed license amendments should not be approved prior to a public hearing before an ASLB was intended to be a narrow one, to be used to avoid delay for routine amendments no one would suggest posed significant hazards considerations, such as replacing a gauge.[20]

The three criteria of 10 C.F.R. § 50.92 are therefore to be read narrowly. Unless the three conditions are met unequivocally, the NRC should grant a hearing before an ASLB prior to deciding whether to approve a license amendment. In short, if a proposed license amendment presents a significant hazards *consideration*—that is, if a comment identifies an issue involving significant hazards—then the matter must be referred to an ASLB for resolution before the proposed license amendment may be considered.

SCE's proposed license amendment could not be further from the kind of change Congress sought to exempt from a prior hearing through the *Sholly* amendment: as demonstrated by FoE's experts and the recent opinion of the ASLB, the prospect of restarting San Onofre Unit 2 with damaged and unrepaired steam generators presents significant new and ill-understood safety risks, not routine changes to technical specifications such as updated inspection routines or new gaskets or gauges.

[20] Gilinksy Decl. at ¶ 3.

The proposed *no significant hazards consideration* determination in this case must be withdrawn because, rather than determining whether a significant hazards *consideration* was present requiring referral to an ASLB for a hearing prior to a decision on the license amendment, the NRC Staff preempted the Atomic Safety and Licensing Board, eschewing its proper role as regulator and instead assuming the judicial role allocated to the ASLB under the AEA, including the *Sholly* amendment. Rather than identify and refer the obvious significant hazards considerations involved in SCE's application for a license amendment, the NRC Staff simply adopted the licensee's evaluation of the merits of the license amendment request, apparently without question.

This is exactly what the Ninth Circuit held invalid in *San Luis Obispo Mothers for Peace*.[21] Under the *Mothers for Peace* ruling, the purpose of the *no significant hazards consideration* determination by the Staff is simply to identify whether there are new or increased risk *considerations* that should be reviewed by an ASLB before the proposed license amendment can be issued. Whether the new or increased risks are acceptable is a decision for the ASLB, to be decided in a hearing held *prior to* deciding whether to approve the proposed license amendment.

Moreover, the ASLB's recent decision, noted above, on the risks presented by SCE's proposed temporary operation at 70% of power found that,

> 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, placing them beyond the envelope of experience with U-tube steam generators…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.[22]

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[21] *San Luis Obispo Mothers for Peace v. U.S. Nuclear Regulatory Commission*, 799 F.2d 1268 (9th Cir. 1986).
[22] ASLB Order at p. 25.

In these circumstances, a finding of *no significant hazards consideration* is wholly inappropriate and exceeds the authority of the NRC under the AEA.

Thus, the NRC should convene an ASLB *prior to making a decision whether to issue the license amendment* to examine the significant safety issues posed by SCE's proposed license amendment to allow operation of the damaged replacement steam generators at 70% power for Cycle 17 (22-24 months). Such an adjudicatory hearing would provide reassurance to the people of Southern California and would be consistent with the Commission's announced policy of transparency. The current attempts to misuse 10 C.F.R. § 50.92 to exclude public participation can only exacerbate public distrust for the NRC and of the safety of the San Onofre plant, whatever decision is ultimately made.

b. The Proposed License Amendment Presents New and/or Increased Risks That Endanger Public Health and Safety

If it actually considered the criteria of 10 C.F.R. § 50.92, the Staff could not determine that the proposed license amendment for San Onofre entails *no significant hazards consideration*. If the proposed change in the license fails to meet any one of the three criteria in 10 C.F.R. § 50.92, the NRC must withdraw the proposed *no significant hazards consideration* determination. As demonstrated in the technical analyses appended to these comments, and by the ASLB's recent decision on San Onofre, the proposed amendment does not satisfy *any* of the three criteria.

To assess whether the change proposed by SCE creates a significant hazards consideration, the appropriate comparison is between the operation of the unit with undamaged steam generators as assumed in SCE's current license, on the one hand, and the operation at 70% of power with damaged steam generators that Edison now proposes. SCE's rationale for concluding that *no significant hazards consideration* is presented is apparently based on comparing operation with undamaged tubes at 100% and 70%, completely ignoring the current highly-damaged state of the steam generators in Unit 2. As Dr. Hopenfeld states, SCE's evaluation of the § 50.92 criteria "is based on the presumption that change in power level can be discussed without giving any considerations to the physical conditions of the tubes before and after the change."[23]

The NRC cannot, despite its best efforts, ignore the events of the past 16.5 months. Major defects causing unprecedented tube wear have been discovered in the replacement steam generators at San Onofre, and while the mechanical force that inflicted the wear has been identified as primarily in-plane FEI, neither the NRC nor the licensee has yet determined the root cause of the FEI, let alone a remedy for it. Instead, SCE and the NRC propose to simply restart Unit 2 and operate it at reduced power for one cycle as an experiment to see whether the plant can be run longer at that reduced rate. One could not possibly conclude that such a proposal does not at least raise "significant hazards" considerations that require further scrutiny in a hearing to decide whether the additional risk of exposing Californians to radiation are acceptable. The Staff simply ignores the fact that Unit 2's replacement steam generators have already demonstrated design flaws in components and systems critical to the safety of San Onofre Unit 2.

i. The Proposed Finding of *No Significant Hazards Consideration* Should Be Withdrawn Because the Proposed License Amendment Would Involve a Significant Increase in the Probability or Consequence of an Accident Previously Evaluated.

Staff addresses the first criterion of 10 C.F.R. § 50.92 by simply restating SCE's analysis, which concludes that the proposed license amendment would not involve a significant increase in the probability or consequence of an accident previously evaluated "because there is no

[23] Hopenfeld Decl. at p. 6.

adverse effect on plant operations or plant conditions."[24] SCE relies on its response to Requests for Additional Information (RAIs) 11-14 as its basis for this assertion.[25]

SCE, however, fails to make the appropriate comparison when applying this first criterion. SCE's apparent position is that the relevant comparison is between operation of fully functional undamaged steam generators originally licensed to run 100% of power and operation of those same steam generators at 70% of power. That characterization is incorrect. The proper consideration is whether operating at 70% of power with *defective, damaged, and unrepaired* steam generators involves a significant increase in the probability or consequence of an accident previously evaluated, as compared to the risk of operating at 100% of power with fully functional, undamaged steam generators. In this context, operating the steam generators in their present condition at 70% of power creates a significant increase in the probability of a release of radioactivity and in the consequences — exposure of potentially millions of people to increased radioactivity.

The three impartial experts who wrote the ASLB's recent decision on San Onofre found that operating the replacement steam generators at 70% would significantly increase the probability and consequences of a previously analyzed accident.[26] For example, the replacement steam generators can no longer meet 10 C.F.R. Part 50, App. A – General Design Criterion (GDC) 14 (Reactor Coolant Pressure Boundary), which requires "an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." SCE's own tube-to-tube wear assessment, as the ASLB order notes, shows that "one unstable tube can drive its neighbor into instability through repeated impact events."[27] Given this condition, there is no

[24] Notice of License Amendment Proposal at p. 22577.
[25] Notice of License Amendment Proposal at p. 22577.
[26] ASLB Order at p. 27.
[27] ASLB Order at p. 27.

longer "an extremely low probability" of the kind of tube failure GDC 14 is meant to guard against.

Nuclear engineers Mr. Large and Dr. Hopenfeld show in the attached declarations that the proposed amendment would involve a significant increase in the probability or consequence of an accident previously evaluated. Mr. Large explains that the excitation forces present in the steam generators exist due to pressure and temperature conditions that will not be affected by reducing the power from 100% to 70%.[28] Thus, contrary to the assertions of SCE, operating Unit 2 at 70% of power during Cycle 17 would not reduce the forces exerted on the tubes during Cycle 16 that caused the unprecedented rapid tube wear and deterioration.[29]

Both of SCE's operational assessments agree that the damage will continue at an unprecedented pace, differing only between 6 months and 16 months as the remaining life-expectancy of the Unit. Even at 70% of power, large numbers of tubes in the replacement steam generators will continue to wear and degrade and, as a consequence, significantly increase the probability of tube rupture.

Dr. Hopenfeld asserts that the probability and consequences of a previously considered accident are significantly increased because, in addition to the fact that operating at 70% of power will not reduce the excitation forces that cause tube wear, SCE also failed to take into account metal fatigue caused by fretting, which is brought on by the FEI-induced vibration. Tubes in Unit 2's steam generators used up a large fraction, if not all, of their allowable "fatigue life" during the last cycle of operation, Cycle 16.[30] Dr. Hopenfeld asserts:

> The number of tubes which are susceptible to rupture by fatigue during a given accident scenario must be known if one is required to predict accident consequences. Until this is done the present pressure based burst performance criteria cannot be used as a reliable

[28] Large Decl. at ¶ 8.5.3.
[29] Large Decl. at ¶ 8.5.5.
[30] Hopenfeld Decl. at p. 7.

> indicator of risk. As a result it must be conservatively concluded that allowing Unit 2 to operate at any power level would significantly increase the consequences of the accidents, which were evaluated by SCE and were described in the UFSAR.[31]

SCE and its consultants have inspected the steam generators for tube surface wear and tube wall thickness but have failed to account for metal fatigue, which cannot be discerned by inspection. Technical Specification Task Force (TSTF) 449 requires SCE to evaluate additional loads on the tubes that could contribute to burst or collapse, even if they cannot be physically measured.[32] SCE's analysis ignores the increased probability or consequences of an accident contributed to by metal fatigue in the tubes of the steam generators.

Tube fatigue increases the probability of an accident. It also increases the consequences, because tube failure owing to metal fatigue happens more suddenly than failure owing to stress corrosion cracking (SCC). A tube failure from fatigue, such as that experienced at the North Anna Generating Station Unit 1 on July 15, 1987, occurs suddenly and quickly.[33] In the event of a main steam line break, for example, accompanied by the rupture of five or more fatigue-weakened tubes, the operator's inability to control the loss of coolant rapidly enough would lead to a significant increase in the probability of uncovering the core, with major increases in the consequences of a previously evaluated accident, including the exposure of millions of Californians to radiation.[34]

Dr. Hopenfeld therefore concludes that restarting the plant for another cycle would place Unit 2 outside of the bounds of accidents evaluated in the updated final safety analysis (UFSAR) report by significantly increasing the probability and consequences of a main steam line break

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31 Hopenfeld Decl. at p. 8.
32 Hopenfeld Decl. at p. 7-8.
33 Hopenfeld Decl. at p. 8.
34 Hopenfeld Decl. at p. 9, 33.

(MSLB) accident.[35] Similarly, Mr. Large found that a single tube burst caused by an MSLB that damages the fuel core could result in severe consequences beyond those considered in the UFSAR.[36]

The NRC's proposed finding of *no significant hazards consideration* addresses none of the issues identified by Friends of the Earth's experts, as summarized above. Thus, the proposed finding must be withdrawn and a hearing on the proposed license amendment held before a decision is made on the proposal.

ii. The Proposed Finding of *No Significant Hazards Consideration* Should Be Withdrawn Because the Proposed License Amendment Would Involve the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated.

> *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[,] ...[an assumption that is] demonstrably unjustified for the replacement steam generators.*[37]
>
> *--- ASLB Opinion, May 13, 2013*

The NRC's regulations do not allow the Staff to make a *no significant hazards consideration* determination if it finds that the proposed license amendment would create the *possibility* of a new or different kind of accident not previously evaluated. The Staff restates in the Federal Register notice proposing the license amendment the licensee's position that "the proposed changes do not require a change in any plant systems, structures, or components or the method of operating the plant other than to reduce power for the duration of Cycle 17. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated."[38]

[35] Hopenfeld Decl. at p. 32.
[36] Large Decl. at ¶ 8.5.17.
[37] ASLB Order at pp. 31-32.
[38] Notice of License Amendment Proposal at p. 22577.

Edison's "therefore" is misplaced: the conclusion of the second sentence does not follow from the statement in the first. The premise of "no change" that SCE relies on for this conclusion, however, is erroneous because it ignores the change that shut the plant down more than a year ago: that an abnormally high amount of tube wear has occurred in the replacement steam generators, and, in particular, the unprecedented fretting fatigue caused by massive FEI and its impact on the steam generator tubes.

First, the UFSAR does not consider the possibility of accidents caused by tube wear from in-plane FEI because it is based on an assumption that in-plane FEI will not occur. UFSAR section 5.4.2.3.1.3, which analyzes steam generator tube integrity, is therefore inadequate and demonstrates that operation at 70% of power presents new and different kinds of accidents from those previously evaluated.

The ASLB agrees. In its recent opinion on SCE's proposed restart plan under the CAL, the ASLB found that operating the replacement steam generators in their current degraded condition is a test or experiment as described under 10 C.F.R. § 50.59(C)(2).[39] By definition then, the proposed license amendment cannot possibly meet the second criterion for a *no significant hazards consideration* determination. Operating at 70% for any length of time with the replacement steam generators in their current condition is an experiment, the outcome of which has not been analyzed in the UFSAR.

Second, the UFSAR currently considers only accidents resulting from excessive pressure loads, not fretting fatigue. During Cycle 16, the tubes in Unit 2's steam generators experienced fretting previously not experienced in the history of any U.S. steam generator.[40] To allow the operation of the steam generators without repairs would, because of this unanalyzed fatigue,

[39] ASLB Order at p. 33.
[40] ASLB Order at p. 25.

create the possibility of a new or different kind of accident from any accident previously evaluated.[41]

Accidents caused by fretting fatigue are different from accidents caused by stress corrosion cracking (SCC). As described above, unlike SCC, metal fatigue is difficult to detect through in-service inspections, and near or at the end of a tube's fatigue life cracking propagates much more quickly than SCC.[42] There is no available data correlating field measurements to leakage from fatigued tubes during a design-basis accident.[43] Thus, any safety analysis that is based on fatigue failures relates to a new and previously unanalyzed accident.[44] SCE has yet to perform an analysis of probable accidents owing to fretting fatigue failures, which it must do before the proposed license amendment could possibly satisfy the second criterion of 10 C.F.R. § 50.92.

Specifically, Dr. Hopenfeld discusses five possible accident scenarios owing to fretting fatigue not considered by the existing UFSAR.[45] In other words, the risk of these accidents arises from the fact that the tubes have already been substantially fatigued and will experience further fatigue at 70% operation:

1. Fretting fatigue rupture of a tube in the free span with a relief valve stuck open or a broken header;
2. Unplanned closure of an isolation valve, increasing steam pressure abruptly, causing rupture of tubes on the border of exhausting their fatigue life;
3. Seismically-induced ruptures of both plugged and unplugged tubes near the end of their fatigue life;

[41] Large Decl. at ¶ 8.6.26; Hopenfeld Decl. at p. 27-34.
[42] Hopenfeld Decl. at p. 29.
[43] Hopenfeld Decl. at p. 4.
[44] Hopenfeld Decl. at p. 27-34.
[45] Hopenfeld Decl. at p. 30-32.

4. Severe accident causing rupture of tubes near the end of their fatigue life; and
5. Main steam line break accident: in situ tests of tube integrity show only the tendency of tubes to leak on the basis of loss-of-wall-material or weakening by stress corrosion cracks. Fatigue failure would cause propagating circumferential cracks.[46]

Little data is available to assess the safety risks presented by these accidents due to the unprecedented and unique nature and extent of the damage to the tubes in Unit 2's steam generators.[47] Dr. Hopenfeld calculates that of the nearly 1100 tubes susceptible to fatigue failure, the probability of only 5 tubes rupturing during Cycle 17 exceeds the NRC's safety goals by a factor of 5.[48] Thus, the proposed license amendment involves serious risks that SCE and the NRC have not considered, precluding a finding of no significant safety hazards consideration. The risks associated with fretting fatigue are serious, and must be evaluated under the TSTF 449.

Mr. Large also raises a number of considerations not taken into account by the Staff in its *no significant hazards consideration* determination. While Mr. Large's technical analysis is presented in detail at section 8.6 of his attached declaration, the key points are summarized here.

Foremost, Mr. Large emphasizes a critical omission in SCE's analysis: SCE did not adequately consider—despite the evidence of extensive damage to literally hundreds of tubes—the possibility of a multiple tube failure, which would greatly exceed the design basis accident of a single tube burst. When evaluated against the current condition of the steam generators in Unit 2, Mr. Large details a number of situations with the potential for multiple tube failure that were ignored by SCE.

The first of these situations is a scenario in which one of the restraining structures (the anti-vibration bars, or "AVBs"), some of which are already significantly worn, physically detach,

[46] Hopenfeld Decl. at pp. 30-32.
[47] Hopenfeld Decl. at p. 33.
[48] Hopenfeld Decl. at p. 33.

damaging tubes in the surrounding area.[49] Since the conditions for such a potential AVB "break up" are possible (including a scenario in which seismically induced loading on the tube bundle could detach a worn-through AVB component),[50] SCE is required to consider the possibility of a worn section of an AVB detaching under various accident scenarios, thereby leading to a multiple tube failure.[51] SCE, however, failed to do so.[52] Notably, this includes SCE's failure to evaluate the seismic loading of the overall tube bundle, taking into account the degraded and defective tubes and components.[53]

Mr. Large also describes a second accident scenario ignored by SCE in which both pressurized and plugged tubes failed locally, dislodging shrapnel into the tube bundle and thereby creating a pathway for a multiple tube failure.[54] Mr. Large notes that various mechanisms exist that could lead to this result, including tube surface damage and flaws—scarring that is already present in the tubes but which SCE has not taken into consideration in the UFSAR.[55] In short, SCE has not accounted for the effect of known mechanisms, such as this scarring, in its analysis of whether the proposed amendment would exceed the allowable stress limits in place to ensure tube integrity.[56]

As Mr. Large explains, when a tube is subject to certain stresses such as exist here, it is subject to two types of fatigue[57] (one of which, fretting fatigue, is discussed at length in the Hopenfeld Declaration). A situation in which a number of tubes have high levels of fatigue is

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[49] Large Decl. at ¶¶ 8.6.14–8.6.17 (describing a number of situations that could detach portions of a worn AVB and the potential effects of an unrestrained object within the tube bundle).
[50] Large Decl. at 8.6.16 (stating "[t]here are a number of situations that could challenge and possibly physically detach sections of such a worn down AVB, including seismically induced loading on the tube bundle, the immediate aftermath of a LOCA, and, quite possibly, the dynamic fluid forces triggered by a MSLB").
[51] Large Decl. at ¶ 8.6.17.
[52] Large Decl. at ¶ 8.6.33.
[53] Large Decl. at ¶ 8.6.34 (noting that, moreover, SCE may in fact be required to undertake a seismic response evaluation for the entire replacement steam generator assembly).
[54] Large Decl. at ¶ 8.6.18.
[55] Large Decl. at ¶¶ 8.6.19–8.6.22.
[56] Large Decl. at ¶ 8.6.22.
[57] Large Decl. at ¶ 8.6.24.

more likely to result in multiple tube failure, particularly in the event that fatigue-weakened tubes come into contact with either shrapnel from a single burst tube or the severed tube itself.[58] Having failed to address *even the issue of fatigue*, SCE could not have evaluated, as it must, the effect of fatigue on a new or different type of accident involving multiple tube failures.

Last, and significant for the purpose of evaluating the proposed license amendment, fatigue can run its course to failure within a single operation cycle,[59] underscoring the importance of taking this factor into account in accident scenarios.

At base, the fundamental point here is that the damage to the tubes and tube restraint components that occurred during the previous operating cycle at San Onofre Unit 2 was so substantial that the response of these structural components to both normal—as well as possibly adverse—operating conditions have not been accounted for, either in the original design accident cases, nor in the analyses SCE relies upon to justify restarting Unit 2 at 70% of power. Accordingly, SCE's analysis cannot purport to demonstrate that running the plant at 70% power will not involve the possibility of a new or different kind of accident from the types considered previously. Soberingly, it is precisely this type of accident, such as, for example, a multiple tube failure, that would result in the most severe consequences for public health and safety.[60]

[58] Large Decl. at ¶ 8.6.25.
[59] Large Decl. at ¶ 8.6.26.
[60] Large Decl. at ¶ 9.1 (stating that "it is quite feasible that failure of a few defective tubes could trigger a major nuclear plant malfunction that, in itself, provokes the bursting of more degraded or defective tubes creating a very significant radiological release via a primary containment bypass. Also, there is the possibility that a major plant malfunction, such as a MSLB, could rapidly result in failure of multiple tubes already weakened in a degraded or defective condition").

iii. **<u>The Proposed Finding of *No Significant Hazards Consideration* Should Be Withdrawn Because the Proposed License Amendment Would Involve a Significant Reduction in a Margin of Safety.</u>**

The assessment in this [Hopenfeld's] report does not support SCE's position that operation of Unit 2 for five months at 70% power will not affect safety. It is shown that SCE conclusions are not conservative. ***Operation of Unit 2 even for one month at any power level would present a safety risk***.[61]

--- *Dr. Joram Hopenfeld*

NRC's regulations at 10 C.F.R. § 50.92 prevent the Staff from making a finding of *no significant hazards consideration* where the proposed amendment would involve a significant reduction in a margin of safety. As an initial matter, the ASLB's decision raises a number of serious safety considerations that are evidence that the Staff's position on the *no significant hazards consideration* is indefensible. SCE's optimistic Operational Assessment estimates of the margins of safety of operation at 70% of power are not justified by experience, as the ASLB pointed out:

> 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. . . .[T]here 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-plant [fluid elastic instability] ha[s] not been previously experienced in U-tube steam generators."[62]

The ASLB further held that the in-plane vibrations caused by FEI were never considered in the UFSAR.[63] The analyses in the UFSAR provide the basis for operating the plant within an acceptable margin of safety. Restarting a reactor unit with known defects caused by mechanisms (*e.g.*, in-plane FEI) that were not analyzed in the UFSAR thus significantly decreases the margin of safety provided for by the UFSAR.

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[61] Hopenfeld Decl. at p. 10.
[62] ASLB Order at p. 34, n.54, quoting Edison Answering Brief at 10.
[63] ASLB opinion at p. 31.

FoE's experts agree that SCE and the Staff cannot show that SCE's license amendment proposal would maintain the required margin of safety in the current license. Dr. Hopenfeld, for example, concludes that operating Unit 2 at 70% of power for Cycle 17 would not be in compliance with ASME code, as required by 10 C.F.R. § 50.55(a), because many of the tubes in Unit 2's steam generators have exhausted their fatigue life.[64] An increased risk of a MSLB accident is an obvious example of the significant reduction in the margin of safety posed by the license amendment request, since such an accident would cause the largest leakage from the fatigued tubes.[65]

According to Dr. Hopenfeld's analysis, the proposed license amendment would increase the Large Early Release Frequency (LERF) of radiation escaping to the environment to a level five times greater than the Commission's stated safety goals.[66] A five-fold increase in risk with potential for large-scale human exposure and the evacuation of southern California is undoubtedly a "significant reduction in the margin of safety."

Mr. Large similarly rejects SCE's conclusion that the proposed amendment would not involve a significant reduction in a margin of safety on the grounds that when it was originally determined, the safety margin[67] required by the NRC assumed that the functionality of the replacement steam generators complied with the design specifications.[68] The fact that they do not is now evident. Critically, the import of this is that "any detriment arising from a design omission or design shortcoming," such as those discussed above, "would not have been included

[64] Hopenfeld Decl. at p. 9.
[65] Hopenfeld Decl. at p. 9.
[66] Hopenfeld Decl. at p. 9.
[67] Large Decl. at ¶ 8.7.2.
[68] Large Decl. at ¶ 8.7.4.

for in the safety margin"[69]—meaning that the safety margin that exists now has been substantially eroded by the defective tube conditions.

This deficiency, which reduces the safety margin by an unknown degree, is further exacerbated by any additional processes created by the design defects, such as, for example, the fretting fatigue discussed by Dr. Hopenfeld. Thus, as Mr. Large states, the "particular processes arising from such a omission or shortfall, in this case the occurrence of fretting fatigue at the AVB-to-tube contact point and its potential to substantially reduce the plain fatigue life of individual tubes, would also not have been included for in the safety margin."[70]

In sum, the safety margin critically does not take into account the current condition of the plant, specifically, the effect that operating with numerous, severely damaged tubes has on the margin of safety assumed to be in place. In other words, the safety margin is not nearly conservative enough, given the condition of the plant. The second critical point the Staff missed is that the safety margin—overly optimistic to begin with—is now being further reduced, according to FoE's expert, "in ways and to an extent that cannot be precisely defined,"[71] as operating the plant at 70% versus 100% will not reduce the forces acting to degrade the tubes.[72]

Last, regarding stress analyses, MHI's analysis, performed for SCE, of stress on the tubes in the replacement steam generators is deficient in a number of ways that significantly reduce the margin of safety of the proposed change. For example, MHI used a finite element model to calculate the stress to which the tubes were subjected and concluded based on this model that the

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[69] Large Decl. at ¶ 8.7.4.
[70] Large Decl. at ¶ 8.7.5.
[71] Large Decl. at ¶ 8.7.8.
[72] Large Decl. at ¶ 8.5.3 (stating "The driving force, so to speak, for single tube failure is the differential pressure acting across the tube wall at the operating temperature. Operating at the proposed 70% RTP will not result in any significant change in the tube differential pressure and the peak tube wall temperature, so the tubes will be subject to the much same forces (radial stress) and tube material strength response (ie the yield stress weighted in account of temperature) as experienced at 100% RTP.").

tubes would not fail from fatigue.[73] MHI's analysis was based on erroneous assumptions, however. When corrected, MHI's model would predict tube failure from fatigue because the stress on the tubes exceeds the ASME Endurance Limit.[74]

Taken together, these analyses by FoE's experts show that the proposed amendment would involve a significant reduction in the margin of safety of Unit 2.

iv. <u>Summary</u>

In order to issue a finding of *no significant hazards consideration*s, the NRC Staff bears the burden of showing that the hazards considerations raised by Friends of the Earth's experts in these comments and by the ASLB's recent decision in the CAL proceeding are insignificant. The Staff cannot make that showing, and consequently the proposed finding must be withdrawn and a hearing on the proposed license amendment held by an ASLB before the amendment may be approved by the NRC.

c. National Environmental Policy Act

The proposed license amendment should not be considered prior to a public hearing because the proposal presents a significant hazards consideration. The National Environmental Policy Act of 1969 (NEPA), 42 U.S.C. § 4321 *et seq.*, requires NRC Staff in such circumstances to at least prepare an Environmental Assessment (EA), which the Staff has not yet done.

NEPA requires federal agencies such as the NRC to examine and report on the environmental consequences of their actions. NEPA is an "essentially procedural" statute intended to ensure "fully informed and well considered" decisionmaking.[75] Under NEPA, each

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[73] Hopenfeld Decl. at p. 11.
[74] Hopenfeld Decl. at p. 11-14; 20, Figure 7.
[75] *Vermont Yankee Nuclear Power Corp. v. NRDC*, 435 U.S. 519, 558 (1978).

federal agency must prepare an Environmental Impact Statement ("EIS") before taking a "major Federal action[] significantly affecting the quality of the human environment."[76]

An agency can avoid preparing an EIS, however, if it conducts an Environmental Assessment ("EA") and makes a Finding of No Significant Impact ("FONSI").[77] Specifically, no EIS is required if the agency conducts an EA and issues a FONSI sufficiently explaining why the proposed action will not have a significant environmental impact.[78] However, in deciding whether to prepare an EIS, the agency must 1) "accurately identif[y] the relevant environmental concern," 2) take a "hard look at the problem in preparing its EA," 3) make a "convincing case for its finding of no significant impact," and 4) show that even if a significant impact will occur, "changes or safeguards in the project sufficiently reduce the impact to a minimum."[79] An agency's decision not to prepare an EIS must be set aside if it is "arbitrary, capricious, an abuse of discretion, or otherwise not in accordance with law."[80]

The Federal Register notice is silent as to the application of NEPA to this case. One can only conclude that the Staff is relying on the categorical exemption from the procedural requirements of the NEPA, as described in NRC's regulations at 10 C.F.R. § 51.22(c)(9), available when the Staff makes a finding of *no significant hazards consideration*. However, as FoE and NRDC demonstrate in these comments, the Staff cannot make such a finding in this instance.

At the very least, an EA and subsequent FONSI must be completed because the proposed amendment would allow steam generators with a severe and dangerous level of wear to operate

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[76] 42 U.S.C. § 4332(2)(C).
[77] *See Sierra Club v. Dep't of Transp.*, 753 F.2d 120, 127 (D.C. Cir. 1985); *see also Theodore Roosevelt Conservation P'ship v. Salazar*, 616 F.3d 497, 503–04 (D.C. Cir. 2010) (explaining NEPA procedures).
[78] *Dept. of Transportation v. Public Citizen*, 541 U.S. 752, 757–58 (2004).
[79] *Taxpayers of Michigan Against Casinos v. Norton*, 433 F.3d 852, 861 (D.C. Cir. 2006) (internal quotation omitted).
[80] *Public Citizen*, 541 U.S. at 763 (quoting 5 U.S.C. § 706(2)(A)).

without repair. Since the leak of radioactive steam in January 2012 resulting from rapid wear in the steam generator tubes, the licensee has proposed no actions to prevent the conditions that caused the leak. The proposed license amendment therefore poses great potential risk to the environment, as shown by the analyses of FoE's experts and the recent ASLB decision, and thus requires the NRC to follow the procedures under NEPA to address that risk.

IV. CONCLUSION

For the foregoing reasons, the Staff's proposed finding of *no significant hazards consideration* should be withdrawn and the significant hazards consideration instead referred to an ASLB, with an attendant public adjudicatory hearing held prior to a decision on SCE's proposed license amendment. As the ASLB recently held with respect to San Onofre Unit 2:

> 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.[81]

There is simply no basis for a *no significant hazards consideration* determination in the case of the proposed license amendment for San Onofre Unit 2.

Respectfully submitted,
/Signed (electronically) by Richard Ayres/

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[81] ASLB Order at p. 32 (emphasis supplied).

/**<u>Signed (electronically) by Geoffrey H. Fettus/</u>**

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Dated in Washington, D.C.
this 16th day of May 2013

<u>Attachments</u>
1. Declaration of Dr. Joram Hoppenfeld
2. Declaration of John Large
3. Declaration of Dr. Victor Gilinksy
4. Declaration of Arnold Gundersen, in Support of the June 18, 2012 Petition to Intervene by Friends of the Earth Regarding the Ongoing Failure of the Steam Generators at the San Onofre Nuclear Generating Station
5. MHI Root Cause Analysis and Supplemental Technical Evaluation Report (Selected Excerpts)
6. *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), LBP-13-07 (May 13, 2013)

ATTACHMENT 1

Declaration of Dr. Joram Hopenfeld

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE NRC STAFF

In the Matter of	)	
	)	Docket ID NRC-2013-0070
	)	
SOUTHERN CALIFORNIA EDISON CO.	)	
	)	
(San Onofre Nuclear Generating Station,	)	
Units 2 and 3)	)	May 16, 2013
	)	

<u>DECLARATION OF DR. JORAM HOPENFELD</u>

CONTENTS

Qualification of Dr. Hopenfeld to Assess the Southern California Edison Response to 10 CFR 50.92

While employed by the Nuclear Regulatory Commission, NRC, Dr. Hopenfeld's research included a focus on steam generator tube degradation. Consequently the NRC launched a Steam Generator Action Plan, SGAP, to address the various safety issues raised by Hopenfeld in a series of documents from 1992, known as the DPO and GSI 163. On September 2007 the NRC issued a new performance technical requirement specifications, TS, to reduce the risk from accident induced and normal operations tube ruptures. This action essentially closed the DPO and GSI 163, as discussed at the May 7, 2009 Advisory Committee on Reactor Safeguards (ACRS) meeting. During the fifteen year review Dr. Hopenfeld made numerous presentations to the Atomic Safety Licensing Board (ASLB) and the ACRS on various steam generator related issues.

- Steam Generator Degradation Monitoring.
- Erosion/Corrosion, FAC (relevant to the feed ring failure at SONGS (1992)
- Safety Consequences of Steam Generator Tube Failures,
- Iodine transport and Spiking,
- POD of crack detection by Eddy Current,
- Metal Fatigue from Thermal Transients (PWRs and BWRS)
- Vibrations in BWR dryers.
- Managed a major International program, MB-2 (US, UK, EPRI) on steam generator performance during design basis accidents.
- Conducted sensitivity studies with the RELAP computer code on operator's ability to keep the SG inventory at mid level as a function of the number ruptured tubes.
- Conducted studies on jet erosion as a potential for leakage increase during SG accidents.
- Conducted numerical studies on SG tube ruptures during severe accidents
- Designed, fabricated and field-tested instrumentation for a very harsh vibration environment.
- Holds several patents on methods for monitoring wall thinning
- Managed the development of acoustic leak detection system for LMFBR steam generators.
- Testified before Congressman DeFazio regarding steam generator degradation at the Trojan Nuclear reactor.

SUMMARY

Southern California Edison (SCE) requested the approval of the Nuclear Regulatory Commission (NRC) for a change in Technical Specification (TS) 5.5.2.11.b1 to allow operation of San Onofre reactor Unit 2 during Cycle 17 at power levels up to 70% of Rated Thermal Power. To obtain approval, SCE claimed that it has demonstrated that the change would not involve any significant hazards, as required by 10CFR 50.92. The assessment in this declaration for Friends of the Earth demonstrates that that SCE has in fact not met the standards prescribed in 10CFR 50.92 which require a "no" answer to three questions. The NRC 10CFR 50.92 states,

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?
3. Does the proposed change involve a significant reduction in the margin of safety?

SCE's justification for providing the three negative answers is based solely on a brief fatigue assessment by the replacement steam generator manufacture Mitsubishi Heavy Industries (MHI Ref 1), which showed that the vibration-produced stresses were too low to cause fatigue failures. SCE endorsed these findings despite the fact that MHI, (a) relied on data which was inconsistent with the visual observations of tube degradation and (b) disregarded American Society of Mechanical Engineers (ASME) code requirement to account for variation in code data and field conditions. Equally important is the fact that in their effort in trying to justify the restart of San Onofre reactor Unit 2 there is no indication that SCE utilized the large amount of data generated by the "lessons learned" from the vibration fatigue tube failures at North Anna (1987), Mihama (1991) and Indian Point (2001).

SCE answered "no" to the three 50.92 questions but only by disregarding fatigue damage to existing tubes and industry guidelines of how to evaluate tube integrity under multiple loads. The SCE analysis is based on showing that San Onofre reactor Unit 2 will operate safety because tube rupture is only controlled by tube wall thickness and the tube differential pressure, ΔP. This declaration shows that the controlling factors of tube rupture are more complex when a significant fraction of tube fatigue life has already been incurred and in addition to ΔP loads the tube is subjected to cyclic loads from flow-induced vibration. Under these conditions, the determination of the margin of safety, solely on the basis of ΔP, is invalid and significantly non-conservative.

The assessment herein includes a discussion of potential radiation release from tube ruptures for five design basis accidents and one severe accident. Because of the unprecedented and unforeseen damage to 1806 tubes during one cycle of operation, there is no data that one can use to reliably calculate the consequences of tube failure risks in such accidents. This declaration demonstrates the high degree of technical uncertainties and lack of robustness in the "no" answers provided by SCE.

The analysis in this declaration indicates that a Main Steam Line Break (MSLB) would result in the most significant large early radiation release (LERF) because of the potential for many tubes to rupture and the high probability for human errors. Events which occur more frequently than MSLB exposing the tubes to relatively lower stress such as unplanned valve opening or closing or earthquakes have a lower probability for human error but are more difficult to analyze. Considerable effort would be required to ensure that the safety risk from such events is significantly lower than the safety risk from MSLBs.

If as few as 1% of the degraded tubes in one steam generator, operating for six months, fail during an MSLB, the result is an LERF of 5x10-5 /yr which exceeds the Commission safety goals by a factor of 5.

My assessment leads me to the conclusion that the proposed SCE TS change:

- Represents a new accident with high risk significance
- Would create a new accident previously not evaluated and,
- Would involve a significant reduction in the margin of safety.

Therefore my answer to each of the three questions is yes.

NO HAZARD CONSIDERATIONS – 10 CFR 50.92

Introduction

A determination of No Significant Hazard must provide assurance that the San Onofre Nuclear Generating Station (SONGS) licensing base (CLB) will be maintained between Steam Generator (SG) inspections during future operation over 18 months so-called cycle 17. However, SCE has failed to demonstrate that the modification of SONGS Technical Specification (TS) which will allow a change from 100% power to 70% power represents an added assurance of the functionality and integrity of SG tubes. As discussed below such a change entails a significant reduction in the margin of safety.

SCE answers no to all three 10 CFR 50.92 questions. Their answers are based on the presumption that a change in power level can be discussed without giving any considerations to the physical conditions of the tubes before and after the change. SCE is mistaken in believing that tube integrity is a function of the power level alone and independent of the actual degree of tube degradation. As discussed in Appendix A, large numbers of both plugged and unplugged tubes have exceeded their allowable fatigue life. This loss of tube integrity significantly affects primary to secondary leakage during design basis accidents and consequently increases the Large Early Release Frequency (LERF).

SCE disregarded the affects of fatigue damage on tube degradation by claiming that the stresses were too low to cause tube fatigue. The analysis in this declaration leads to a different conclusion: the vibration during cycle 16 resulted in sufficiently large cyclic stresses to cause fatigue damage to a significant number of tubes.

Another important factor that must be considered in comparing the change of operating Unit 2 from 100% to 70% power is the unknown behaviour of the tubes at the lower power level. Even if vibrations due to fluid elastic instability were significantly reduced at the beginning of the cycle it is uncertain that this will remain so through the five months of operations. Tubes with low natural frequencies may continue to wear due to fluid turbulence. The resultant increase in clearance between the AVB support and the tube could lead to an increase in the intensity of the impacts between these two components. This could lead to an abrupt failure even for those tubes whose fatigue life has not been used up during cycle 16, i.e their cumulative usage factor was less than one (CUF < 1).

It is for these reasons that my answers are in the affirmative to all three 10CFR 50.92 questions as discussed below.

2. Answers to 10CFR 50.92 Questions

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: Yes

In comparing the change proposed by SCE, one must compare the change in San Onofre reactor Unit 2 at the beginning of cycle 16 to its proposed cycle 17 operations. Consideration must be given to both power levels and the degree of tube degradation, not just the power level as SCE have done. The operation of Unit 2 which entered service in in 2011 (cycle 16) at 100% power must be compared to how it would operate if permitted to restart in cycle 17 at 70% power level with a large number of defective tubes.

The proposed change would significantly affect the probability of accident initiators because a certain percentage of steam generator tubes have used up their entire or a large fraction of their allowable fatigue life during cycle 16. For this reason the operation of San Onofre reactor Unit 2 during cycle 17 will fall outside the bounds of the accidents that were evaluated in the existing SCE Updated Final Safety Analysis Report (UFSAR). While the proposed change does not affect the design of SG or its method of operation, it does increase adversely the consequences of Design Basis Accidents (DBAs), i.e., main steam line break (MSLB) and tube rupture (SGTR). The SONGS Technical Specifications, TS 5.5.2.11, require that SONGS provide the NRC during every outage an assessment, CM, with respect to tube structural integrity, accident induced leakage, and operational leakage. As discussed by SCE UFSAR Sections 3.2 and 5.2.9, the entire CM assessment is based on "operating experience with SG tube degradation mechanism that result in tube leakage". Likewise, SCE's determination of Core Damage Frequencies (CDF) is also based on leakage methodology which was derived from tubes that were degraded by stress corrosion cracking. As discussed below, these results are not applicable to the 70% power operation with fatigued tubes. A comparison of operation at 70% power with fatigued damaged tubes versus operation at 100% power with undamaged tube must consider fatigue damage. SCE is wrong in claiming that the change from 100% power to 70% only changes the power level without any potential adverse safety consequences.

In discussing its "no" response to 10CFR 50.92 question 1, SCE did not explain why industry guidance on how to ensure tube integrity was not included in its submission. These guidelines, issued by the Technical Specification Task Force, TSTF 449, specify that primary/secondary pressure differential ΔP loads alone are not sufficient to ensure integrity when other loads are also present. Specifically,

"additional loading conditions associated with the design basis accident or combination of accidents in accordance with the design and licensing base shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse."

In some accidents cyclic loads may be controlling tube rupture in others, ΔP loads may act in tandem depending on the degree to which the tube wall thickness was already reduced by wear. Therefore, the mean stress level of any tube must be considered together with superimposed cyclic stresses. SCE treated cyclic loads as if they have never occurred at San Onofre reactor Unit 2.

As discussed in Appendix A, the predominant degradation mechanism at San Onofre reactor Unit 2 during cycle 16 for an unquantified fraction of the tubes is fretting fatigue. Fretting fatigue would result in a larger and faster leakage rate from a tube rupture than the leakage from a tube that was degraded by cracks due to Stress Corrosion Cracking, (SCC) or wall thinning by erosion alone. The existing leakage performance criteria are based on the latter.

For those tubes in San Onofre reactor Unit 2 where the Stability Ratio (SR) was relatively low, (less than 0.4,) tube rupture is expected to be controlled by burst pressure. In this case present performance criteria are applicable. During operational (non-Loss-Coolant Accidents, LOCAs) and accident transients (LOCAs) cracked tube can be expected to result in a slow progressing leakage, in contrast when fretting fatigue is the cause of tube failure the leakage would occur suddenly and proceed rapidly to its maximum as happened at North Anna (Ref 2).

To evaluate the effect of existing defects in San Onofre reactor Unit 2 on the consequences of a given accident one must identify first the fraction of the tubes that were damaged predominantly by fatigue and the fraction of tubes that were damaged by wall thinning alone. This must take into account that high cycle vibration fatigue does not lend itself to in-service detection. Tube fatigue life is almost entirely spent in the incubation period and once the crack is formed failure would follow quickly.

To comply with industry guidelines TST- 449, Rev 4, degradation of each tube must be assessed simultaneously in terms of both its existing fretting damage (wall thinning) and its local SR. The number of tubes which are susceptible to rupture by fatigue during a given accident scenario must be known if one is required to predict accident consequences. Until this is done the present pressure based burst performance criteria cannot be used as a reliable indicator of risk. As a result, it must be conservatively concluded that allowing San Onofre reactor Unit 2 to operate at any power level would significantly increase the consequences of the accidents, which were evaluated by SCE and were described in the UFSAR.

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

Response: Yes

The proposed change will introduce significant changes to postulated accidents resulting from tube degradation. Appendix B discusses how primary to secondary leakage from fatigue-ruptured tubes differs from leakage that resulted from tubes that failed from excessive loads. Since existing safety studies are based solely on changes in ΔP this represents a new type of accident.

The analysis in Appendix A demonstrates that some tubes will enter service in Cycle 17 with no fatigue life left. The leakage from these tubes will not be affected by changes in ΔP. The rupture of these tubes would depend on the intensity of cyclic stresses that varies with the stability ratio, SR. The SR and the ΔP are independent variables: different driving forces govern their respective changes during a given accident. For this reason the operation of San Onofre reactor Unit 2 with fatigued tubes creates a new and different kind of an accident.

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

Yes

Because many tubes have exhausted their fatigue life the proposed change would not be in compliance with the ASME code as required by 10 C.F.R. 50.55a, Codes and Standards. Since the SG tubes form a barrier between the radioactive fission products in the primary water and the secondary system, loss of fatigue life reduces the safety function of the SG.

Initial assessment in Appendix B suggests that the MSLB accident would be the most damaging accident from the standpoint of causing the largest primary/secondary leakage from fatigued tubes in comparison to more frequently accidents with a lesser damage potential.

Based on a probability of E-4/yr (one in every 10,000 years) that a steam a line break would occur outside containment, the Large Early Release Frequency, LERF of radiation escaping to the environment due to the reactor core becoming exposed is also E-4 /year because, as discussed in Appendix B, no credit can be given to the operators that they would terminate the accident before depleting the reactor water storage tank, RWST. This represents an LERF of 5 E-5 /yr (one in every 20,000 years) for six months operation. This is an increase by a factor of 5 over NRC goals as considered in NRC Probability Risk Assessments.(Ref 11)

Assuming that the ΔP will not be large enough to rupture tubes, and no leakage from fatigued tubes, SCE calculated a change in LERF of 4E-6/yr (NRC AIT Report July 18 2012)

SCE results are not realistic and therefore the answer to question 3 must be yes.

Conclusions

SCE analysis is based on the assumption that any increase of radioactive primary water during hypothetical accidents would be controlled by burst pressure, ΔP. This assumption is flawed because some tubes at San Onofre reactor Unit 2 have already used up their fatigue life. In this case leakage increase would be controlled by the intensity of vibration-induced stresses, not ΔP. The large and sudden fatigue tube ruptures at North Anna, Mihama and Indian Point occurred when the ΔP was essentially constant but tubes failed because they exhausted their fatigue life due to intense vibration similar to those that occurred at SONGS.

In spite of its 170,000 inspections to understand the tube wear problem, SCE has not even mentioned the possibility that a fraction of the tubes had sustained fatigue damage. Nor did SCE discuss the uncertainties and errors in the MHI fatigue analysis and how they could affect their "no" answers.

The assessment in this report does not support SCE's position that operation of San Onofre reactor Unit 2 for five months at 70% power will not affect safety. It is shown, that SCE conclusions are not founded on science and equally important are not conservative.

Operation of San Onofre reactor Unit 2 even for one month at any power level would present a significant safety risk.

I Joram Hopenfeld declare, under penalty of perjury, that the foregoing information and facts are true and correct to the best of my knowledge and belief, and that the opinions expressed herein are based on my independent and best professional and personal judgment.

Joram Hopenfeld, May 14, 2013

Joram Hopenfeld

APPENDIX A- FATIGUE ANALYSIS

Introduction

When an SG tube is in contact with its support either the Anti Vibration Bar (AVB) or the plate (TSP), the two contacting surfaces are damaged by fretting. As discussed in detail by the Electric Power Research Institute (EPRI Ref 4) and Volchock (Ref 5) fretting damage occurs due to a combination of the sliding motion between the surfaces and the impacts when an external cyclic load is superimposed on the sliding motion. The sliding motion alone produces fretting wear while the cyclic impact produces fretting fatigue, the synergy of the two significantly reduces fatigue life and enhances wear. This synergy is especially important when wear predictions are mostly based on empirical parameters from laboratory tests. Even though the EPRI report indicates that wear and fatigue are controlled by different mechanisms and their respective equations are different, SCE, in calculating tube wall thinning, applied equations for wear without considering the differences between wear fatigue and fretting wear. Numerous considerations must be given in extrapolating lab data on fretting wear to SONGS conditions. Since SCE did not perform any similitude studies on wear rates, SCE projection of tube wear for cycle 17 are unreliable.

This appendix was divided into two parts, the first part shows why SCE analysis lead them to believe that the tubes at SONGS did not suffer fatigue damage. The second part, Part 2, is a rebuttal to SCE and MHI's contention that tube fatigue damage during cycle 16 can be ignored.

PART 1 –MHI ANALYSIS.

As describe in Attachment 4 MHI Document L5-04GA564, MHI used a finite element model (FE), to calculate that the tubes were subjected to a stress of 4.2 Ksi (P 16-2), which is smaller than the endurance limit stress of 13.6 ksi (P 16-2). Consequently, MHI concluded that the tubes would not fail from fatigue even if they were subjected to infinite number of stress cycles, (P16-13).

The MHI results are based on two erroneous assumptions. When these assumptions are corrected the opposite conclusion is reached. The issue is not with the FE, rather it is how the FE results were adjusted to account for high stresses at surface discontinuities.

a. Stress Concentration

It is a well-established fact that geometrical discontinuities such as sharp corners introduce high local stresses, which act as a site for crack initiation. A common engineering practice is to fillet or chamfer sharp corners to reduce stress concentrations and increase fatigue life. Any conceivable discontinuity has been considered and the results have been published in numerous publications to guide designers in selecting the particular fillet for a given application. MHI used a design chart, Figure 2, for a tube in pure tension to determine the stress concentration factor Kt. Assuming an undisclosed value for the fillet radius and the value of the parameter (t) MHI concluded that Kt was less than 1.5 when t/r =1.33. These numbers indicate that MHI used a value of t/h that exceed unity. Had MHI assumed smaller values for t/h, and a smaller radius, Kt would have exceeded 1.5 because Kt is sensitive to the assumed geometry of the fillet. MHI selected an arbitrary geometry, which is not valid, and for this reason they only obtained an unrealistically low value for Kt.

Fig 2 is intended for applications when one is trying to minimize stress concentration. Visual examination of the contact between the AVB plates and the tubes do not suggest that the relative motion resulted in geometry with minimum stress concentration. On the contrary, as shown in Figures 3 and 4, the method in which the AVB interacted with the tubes allows for a formation of sharp corners at the intersection of the plate with the tube. MHI's own discussion is not consistent with their application of Chart 5 in Fig. 2. The observation that the "tube and the AVB are worn into each other" and the fact that the AVB plate has sharp corners suggest that Chart 3.5 does not apply to observed wear pattern.

The model shown in Figure 5, represents more closely the wall thinned geometry than the one used by MHI in selecting the stress concentration factor. Since Figure 2 does not provide data for fillets with very small radius, it is necessary to consider a similar geometry giving Kt values for small radii. In Figure 6 (a special case of Fig 2, di = 0), Kt is plotted for very small radiuses for bending - Kt values in tension are similar). Using a reported wear of 35%TW. Kt is calculated as follows:

t = (1 – %TW) T = (1-0.35) 0.043 = 0.028in

d/D = D/D-2t = 0.750 /0.750 – 0.056 = 0.750/0.694 = 1.08

Kt = 5 when r= 0.0014

for Kt =5, r = 0.002 (0 .694) = 0.0014

(Theoretically the chamfer radius of a sharp corner is zero, and therefore Kt will tend to be very large for a finite but small radius of 0.0015 which is close to describing a sharp corner, Kt exceeds 5.)

b. Loss of Wall Thickness (wall thinning)

The effective wall thickness Teff of the geometry in Figure 4 can be expressed as:

$T_{eff} = t \times (2\theta)/360 + T \times (360 - 2\theta)360$

$\theta = 2 \ \cos^{-1} (d/2 + t)/(d/2 + T)$

For a 35% and 70 % tube wear, θ equals 44.6 and 44.1 degrees respectively, the corresponding effective tube thickness equals 0.0360 and 0.0345 and respectively.

c. Surface Finish

Fatigue life, and therefore the endurance limit, is strongly affected by surface finish. Figure 9 show that fatigue life can be considerably reduced by abrasions.

The data (Edison Attachment 6- Appendix D pages 130 -131) indicates that the fretted tube surfaces do not maintain their original surface finish instead they are severely scarred. Such scars are sites for the formation of micro cracks.

Bounding calculations would require that the ASME design stress used by MHI (13.6 ksi) be lowered to account for surface finish. It is not clear however that the introduction of both a stress concentration and surface finish correction simultaneously would not be overly conservative. Since no data was found in the literature where both a sharp corner and adjacent rough surface, a surface finish correction was not included in the present assessment. In that sense, the application of a concentration factor of 5 together with curve C of Figure 1 may not be conservative.

d. Corrected MHI stress.

Corrected stress = MHI stress multiplied by concentration correction factor K, multiplied by thickness correction factor Tc, = 4.2Kx(Tc)

K = actual stress concentration factor / MHI concentration factor = 5/1.5 = 3.33

1/Tc = Decrease in wall thickness /original wall thickness = 0.036/0.043 for beginning of cycle 0.0345/0.043 at the end of cycle assuming the same wear rate.

Tc = 1.19 to 1.25

Increase in stress= 4.2x 3.33 x 1.19 to 4.2x 3.33 x 1.25 = 16.7 to 17.5

Actual increase over the endurance limit = 16.7/13.6 to 17.5/13.6 = 1.22 to 1.29

e. Conclusions

The impact of correcting the MHI calculations is demonstrated in Figure 7: it is self-explanatory. It should be noted that the stresses that ruptured the tubes at Mihama and North Anna (Ref 5), about 7.4 to 8.7ksi and 4 to 9ksi respectively was within the error band indicated in the Figure. The ruptures at Mihama and North Anna occurred due to support plate/ tube interaction not due to AVB/tube interaction.

APPENDIX A GRAPHS/CHARTS

1. Purpose
The purpose of this document is to show that the stress of the tube in SONGS RSG due to in-plane vibration is under the fatigue limit.

2. Conclusions
The stress on the tube due to in-plane vibration is 4.2ksi and is under fatigue limit (13.6ksi). The tube has structural integrity for the stress due to in-plane vibration from the view point of fatigue evaluation.

3. Assumptions and Open Items
The tube deforms in-plane until contacting with the outer next tube in Row direction due to in-plane vibration.
The stress due to in-plane vibration is high cycle fatigue

4. Acceptance Criteria
The fatigue limit is 13.6ksi according to the following design fatigue curve.

Fig. I-9.2.2 1998 SECTION III, DIVISION 1 — APPENDICES

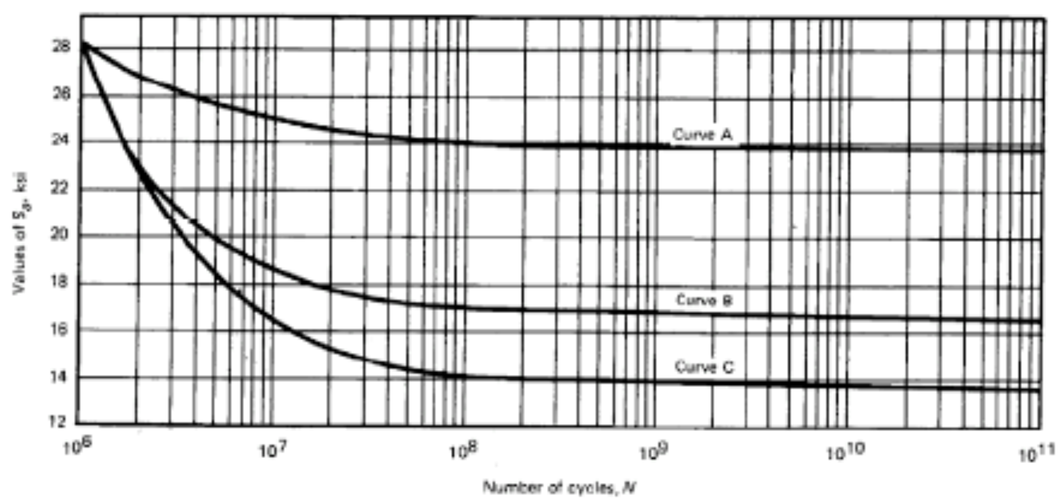


Figure 4-1 Design Fatigue Curve for Tube

1-Figure-1 Fatigue data used by MHI to determine tube fatigue life. The cycle independent line represents the endurance limit, MHI used an endurance limit of 13.6ksi. Attachment 4, P 16-2. data for smooth specimen.

CHARTS **157**

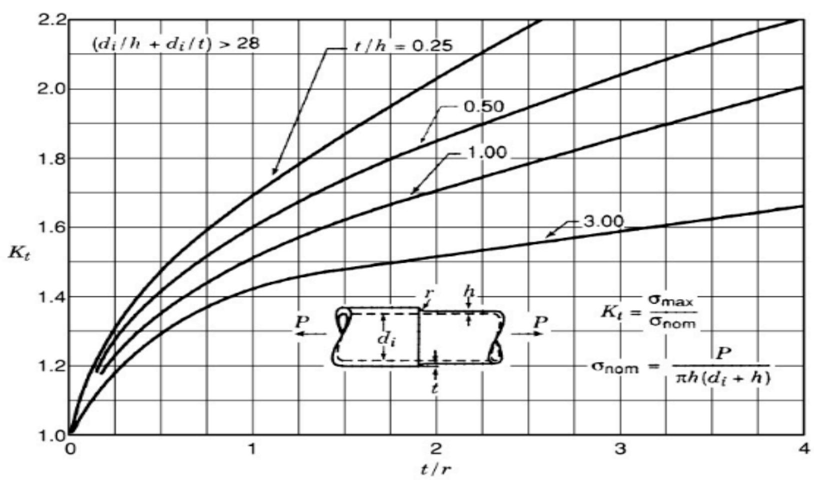


Chart 3.5 Stress concentration factors K_t for a tube in tension with fillet (Lee and Ades 1956; ESDU 1981).

Figure 2 - Stress concentration factors used by MHI for calculating maximum tube stress, Attachment 4 P.16-2. Source: W.D. Pilkey, Peterson's Stress Concentrations Factors, John Wiley and Sons 1997.

3.2 Wear Pattern-2（Local Wear on Tube Surface）

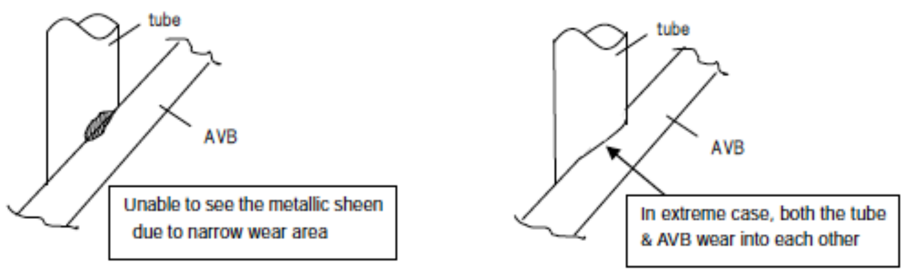
Characteristics

① Local wear occurs on the tube but the wear surface is not exposed (cannot be seen)

② Unable to determine if wear occurs on tube or AVB or both

③ Unable to determine the direction of motion or vibration

④ An extreme interpretation is that both tube and AVB are worn into each other.



(a) Case 1 (b) Case 2

Fig-2 Wear Pattern 2

Fig. 3 - Wear due to AVB/tube Interaction - Attachment 4. It should be noted that both the impact and the sliding motions play a part in the tube/AVB interaction. These factors reduce tube strength because of material loss but also because of loss of fatigue strength.

Non-proprietary Version

(P.10-20)

Document No.L5-04GA564(9)

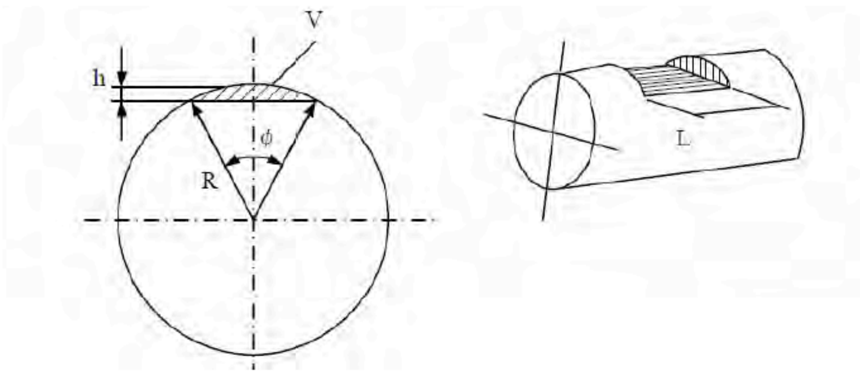


Fig. 6-4 Wear shape of tube at the contact point with AVB

Figure 4 – MHI description of wear shape at tube/ AVB contact point. Attachment 4. P. 10-20

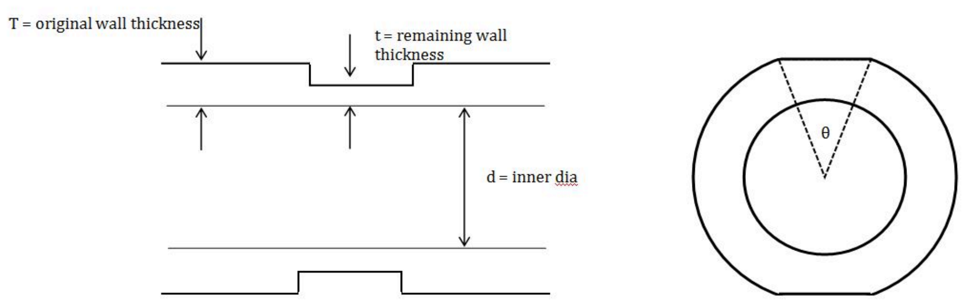


Figure 5 - Schematic for determining a stress concentration factor Kt and reduced wall thickness of a tube due to double sided wear to a thickness (t) over an arc defined by θ

166 SHOULDER FILLETS

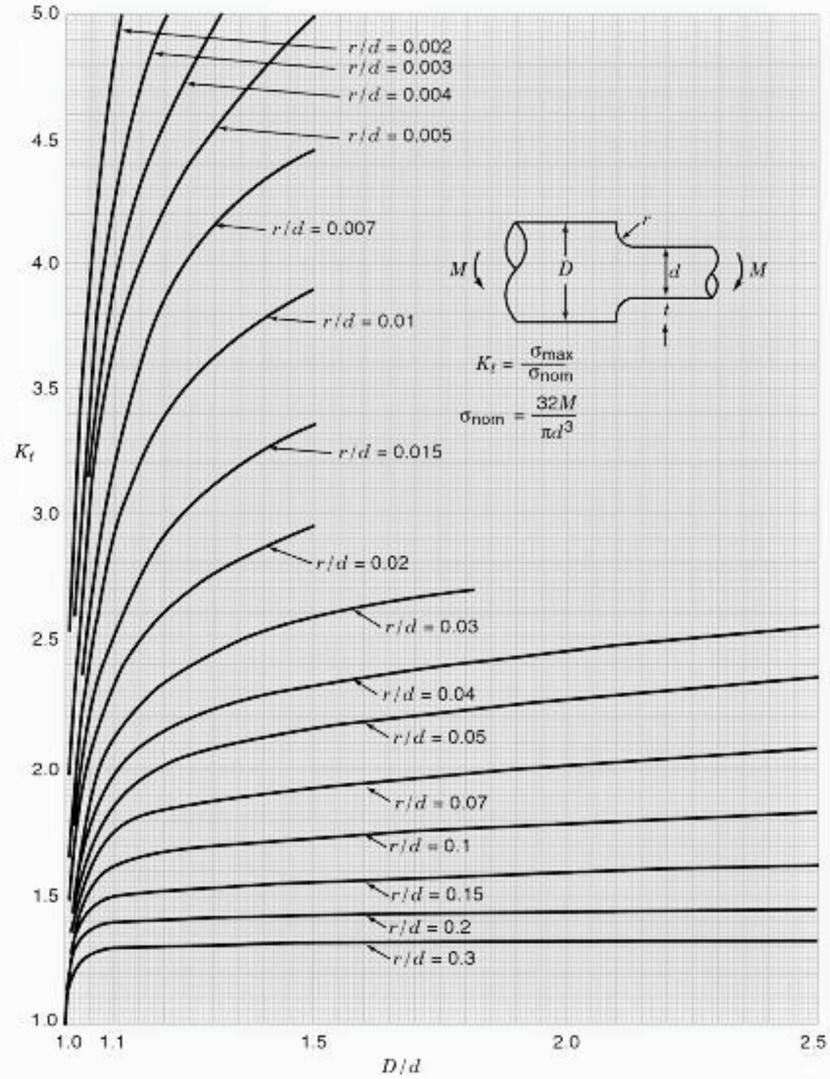


Chart 3.11 Stress concentration factors K_t for bending of a stepped bar of circular cross section with a shoulder fillet (based on photoelastic tests of Leven and Hartman 1951; Wilson and White 1973). This chart serves to supplement Chart 3.10.

Figure 6 – Stress concentration used in the present analysis. W.D. Pilkey, Peterson's Stress Concentrations Factors, John Wiley and Sons 1997. (Similar Kt values in tension -Peterson's Chart 3.4, and for internally pressurized vessel Chart 3.6 for a small radius)

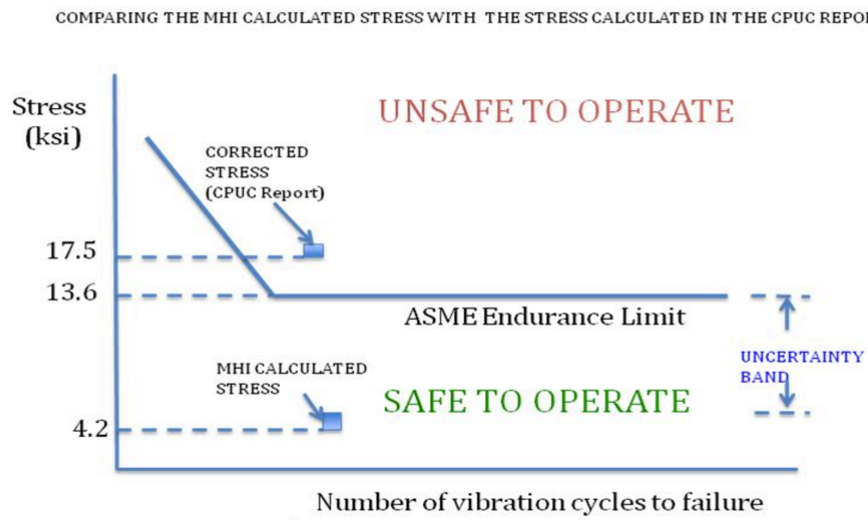


Figure 7 – Effect on potential fatigue damage by correcting the MHI calculation

PART 2 -Rebuttal to SCE/MHI Assessment of Hopenfeld Fatigue Calculations

Appendix A was attached to Dr. Hopenfeld Testimony to the CPUC and has been in the public domain since March 29, 2013. In reply to questions from ABC Channel 10 in San Diego, SCE and MHI responded on April 25th as follows:

SCE

"Hopenfeld's fatigue analysis concerning in-plane tube vibration is significantly flawed in that it applies an unreasonably high stress concentration factor based on solid body geometry rather than the more realistic stress concentration factors for a cylindrical geometry applicable to the SONGS steam generator tubes."

MHI

MHI did analyze the potential for fatigue failure of the RSG tubes under operating conditions and determined that fatigue was not a credible tube failure mechanism because the stresses sustained by the tubes due to in-plane vibration are well below the stresses that would cause fatigue failure. The analysis that supports this conclusion is contained in Appendix 16 to the "Tube wear of Unit-3 RSG - Technical Evaluation Report." It should be noted that the technical reviews and analysis, both by the NRC and industry experts, have not mentioned fatigue failure of the tubing."

Since SCE approved MHI fatigue analysis as specified in the original SCE design document, the reply to the above separate statements will be consolidated.

REPLY

SCE/MHI calculations are based on ASME data that has not been corrected for the conditions that represent the tube surface following fretting after 18 months of operations. The S-N fatigue data was obtained by testing a number of polished solid specimens and the lines represent mean stress limits. It is a common practice of a user of the ASME code to make conservative correction when that data is applied to field conditions which are drastically different than those in the code. When applying the data to tube surfaces that have experienced fretting, Curve C, which was used by MHI, must be lowered to reflect the increase in surface roughness due to fretting. As shown in Figure 9 a change of surface roughness from 0.05 microns to 2.67 microns reduces the fatigue life by a factor of 8.8.

In their report (Appendix 16), SCE/MHI stated that because the AVB and the tubes are imbedded in each other the condition of the surface cannot be seen. Given that the respective surfaces have been sliding and impacting each other it is difficult to imagine how SCE/MHI concluded that such motion would produce polished surfaces. One must conclude that SCE/MHI disregarded the intent of the ASME code by not adjusting the Curve C stress to account for surface roughness.

Comparison of the actual ASME curve with those that were reproduced by SCE/MHI shows that SCE/MHI incorrectly labeled the data to indicate that it was generated for tubes and was limited to operation below 800 F. The data would not be applicable to severe accidents, which were discussed above. It is not clear why SCE/MHI mislabeled the ASME figure to indicate that it was generated for tubes.

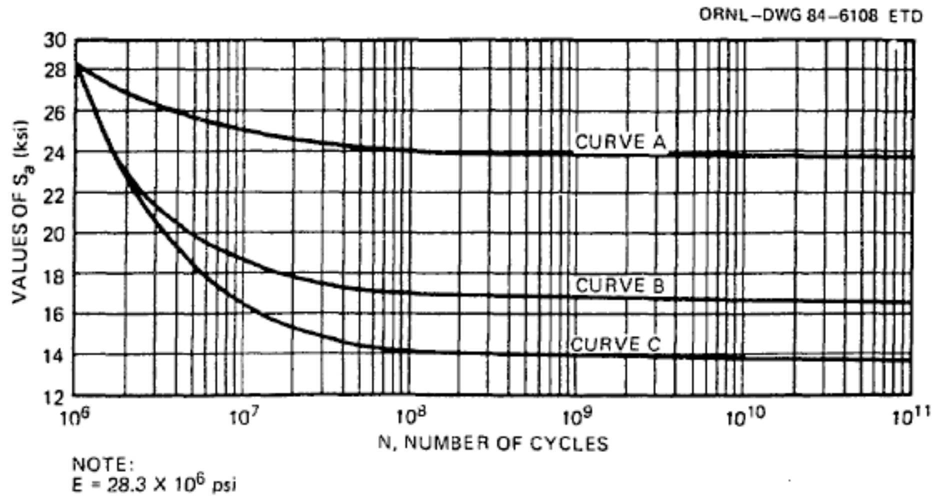


Fig. 1. ASME Code Sect. III high-cycle design fatigue curves for austenitic steels, nickel-chromium-iron alloy, nickel-iron-chromium alloy, and nickel-copper alloy for temperatures not exceeding 800°F (from Ref. 22).

Figure 8 - Showing the same ASME data that was shown in Figure 1 but with the original correct caption. This the data was not for SONGS steam generator tubes.

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Table 12-3 Fatigue life of SAE 3130 steel specimens tested under completely reversed stress at 655 MPa†

Type of finish	Surface roughness, μm	Median fatigue life, cycles
Lathe-formed	2.67	24,000
Partly hand-polished	0.15	91,000
Hand-polished	0.13	137,000
Ground	0.18	217,000
Ground and polished	0.05	234,000
Superfinished	0.18	212,000

† P. G. Fluck, *Am. Soc. Test. Mater. Proc.*, vol. 51, pp. 584–592, 1951.

Figure 9- Effect of surface roughness on Fatigue life

2. Incorrect selection of the Stress Intensity Factor

The ASME curves are used only to calculate average stresses only. At least 100 years of experience has been accumulated to show that sharp surface discontinuities introduce high local stress concentrations where crack are initiated. The ASME code requires that the average stress of a component be multiplied by the appropriate stress intensity factor. Because of the importance of local stresses on fatigue life, hundreds publication are available for smooth discontinuities and thereby reducing local stress. The concept

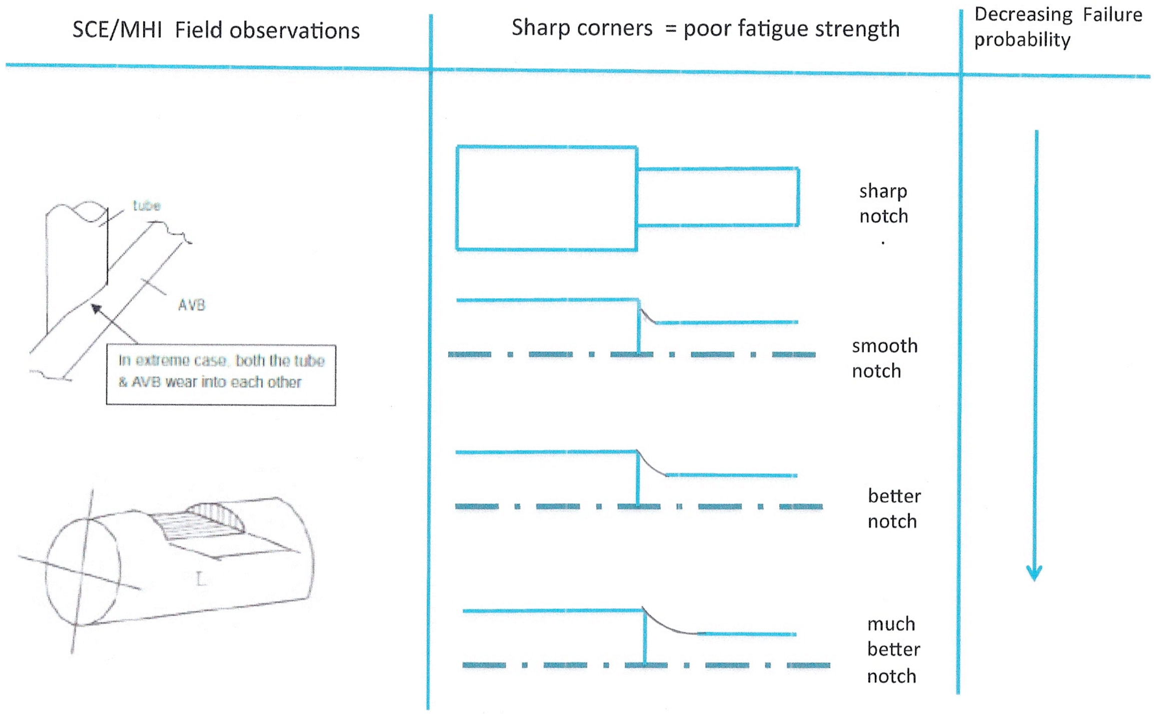

Figure 3 –IGNORING FIELD DATA SCE/MHI SELECTED GEOMETRY WITH GOOD FATIGUE LIFE

FIGURE 10– This has been duplicated to show why SCE /MHI concluded that the "vibration are well below the stresses that would cause fatigue failure" as illustrated in Figure 7.

Sharp corners lead to a poor fatigue strength while smooth corners or a gradual transition reduces stress concentrations thereby improving fatigue strength. The most common source for stress concentration factors are the Peterson's charts which are available for numerous different geometries. As shown in Fig. 2of Appendix A, SCE/MHI had to select a fillet radius in order to calculate the stress concentration factor. If one selects the radius arbitrarily, you can get any number he wishes. SCE/MHI used radius that they have redacted, however an examination of their calculated stress concentration factor,(K t) clearly indicates that they selected a relatively a smooth fillet (large radius) and that SCE/MHI did not select a sharp notch. Since SCE/ MHI stated that the interface between the AVB and the tube is not visible, and their interpretation of the contact surface geometry shows a 90 degree corner, it is impossible to conceive how could they justify using a large radius fillet. The Peterson charts were designed to minimize stress concentrations, when the AVBs impact the tubes they do not follow fracture mechanics guidance to avoid formation of sharp notches.

Figure 10 above illustrates schematically how fatigue life is improved as the notch radius increases.

In the stress calculation, Par 1, I have selected a sharp notch because this is consistent with the observation that the AVB and the tube imbedded in each other through impacts.

Fatigue damage by impact loads would lead to a brittle fracture because such loads do not mitigate slip. Selection of sharp notch geometry is appropriate because such notches can lead to a brittle fracture. In contrast, a well-designed fillet would result in a ductile fracture.

Another reason why it is incorrect to select an arbitrary fillet radius with smooth surface to calculate fatigue life is the synergy between surface roughness effects and cyclic loading effects. Such synergy leads to a significant reduction in fatigue life as has been clearly demonstrated in Reference 3. Therefore even if MHI had corrected their stress intensity factor (K t) of 1.5 to account for surface roughness (Fig 9) it still would leave a large uncertainty due to synergy. This only indicates that calculations which are solely based on a sharp notch (K t =5) may not be sufficiently conservative.

As a reality check on their fatigue model, one must wonder why SCI/MHI did not compare their calculated stress of 4.2ksi at San Onofre reactor Unit 2 with the stress (7.4-8.7 ksi and 4-10 ksi) that caused the rupture at Mihama and North Anna (Ref 6) respectively. Such a comparison should be made for each affected tube on the basis of the local velocity, steam quality, tube stiffness, natural frequency, and temperature gradients across the tube wall and Δ P. SCE/MHI should show that the differences in conditions at Mihama and North Anna vs. conditions in San Onofre reactor Unit 2 account for the fact that Mihama and North Anna tube ruptures occurred at somewhat a higher stress.
The SCE/MHI statement that stress concentrations at sharp discontinuities depend on whether the component is a hollow or solid, appears to be a new discovery in fracture mechanics. It is well established that stress concentration gradients at sharp notches decrease rapidly with the distance from the notch. In other words, the crack would be initiated at the tip of the discontinuity and is practically independent of the geometry further away. As the comment to Figure 6 indicate, examination of Peterson's charts clearly demonstrates this point.

In light of the many unstated assumptions that SCE/MHI used in applying Figures 1 and 2 to the SONGS tubes, the statement that it is unrealistic to apply "stress concentration factor based on solid body geometry rather than the more realistic stress concentration factors for a cylindrical geometry applicable to the SONGS steam generator tubes." Is not appropriate. I used the solid geometry for convenience only. Extrapolation of the tube data in Figure 2 to sharp corners (r=0) would have resulted in the same stress concentration factor.

SCE/MHI appear to justify their position that fatigue failure would not occur at SONG by relying on the fact that the NRC did not raise this issue. In the light of the significant component failures in power plants from high cyclic fatigue due to thermal or hydraulic instabilities, it is puzzling that the NRC did not raise the fatigue issue. The suggestion that the fact that the NRC did not raise the fatigue issue is not a valid technical reason that supports SCE/MHI fatigue analysis. Nevertheless, ultimately it is SCE's responsibility to operate the plant safely. It is not the NRC’s responsibility.

APPENDIX B

FRETTING FATIGUE TUBE DAMAGE – NEW AND DIFFERENT FROM ANY ACCIDENT PREVIOUSLY EVALUATED AT SONGS

1. Introduction

There are two main reasons why fretting fatigue introduces a new un-analyzed accident at SONGs. The massive fretting fatigue suffered by the SONGS steam generators is unique in the history of United States SG tube degradation. Assessments of accidents, which could be induced by degraded SG tubes, were focused on the consequences of operations with tubes that were degraded by Stress Corrosion Cracking (SSC). With three exceptions, North Ana (1985), Mihama (1991) and IP B (2000) all other tube ruptures resulted from stress corrosion cracking and loose part wear as shown in Table 1 below. Fatigue failures at these three plants were limited to a single tube and unlike at SONGS the root cause was fairly well understood.

Given the fact that fatigue damage in the above three accidents was confined to one tube it is puzzling why SCE/MHI completely ignored the wide spread fatigue damage at SONGS. In comparison very extensive fatigue investigation was conducted in connection with the North Anna event, (Ref 6).

Since SCE frequently quotes the existing performance criteria the understanding of these criteria is critical in assessing the SCE conclusion it would be appropriate to briefly review the basis for the present performance criteria.

Since it became obvious in the late 1980s that steam generators would have to stay in service with SSC cracks all efforts were focused on attempting to define the safety consequences of such operations. Starting in 1991 with a series of documents that became known as Differing Professional Opinion (DPO), efforts were made to cope with various aspect of the problem. In particularly the DPO focused on improving the voltage based methodology of predicting accident leakage from eddy current voltage measurements.

Plant/SG Model/ Tube Material	Date	Leak Rate (gpm)	Size	Location	Degradation Mechanism	Contributing Factors
Point Beach 1[1] W-44 600MA	2/26/75	125	2 adjacent ruptured bulges, each 20mm in length and width	Slightly above tube sheet, outer row hot leg	Wastage	Sludge pile
Surry 2[1] W-51 600MA	9/15/76	330[3]	114 mm long axial crack	U-bend apex, Row 1, Col. 7	PWSCC	Hour-glassing
Prairie Island 1 W-51 600MA	10/2/79	336[3]	38 mm long axial fishmouth crack	76 mm above tube sheet, hot leg, Row 4, Col. 1	Loose parts wear	Sludge lancing equipment left in SG
Ginna 1 W-44 600MA	1/25/82	760[3]	100 mm long axial fishmouth crack	127 mm above tube sheet, hot leg, Row 42, Col. 55 (3rd row from bundle periphery)	Loose parts wear, fretting	Baffle plate debris left in SG
Fort Calhoun CE 600 MA	5/16/84	112	32 mm long axial fishmouth crack	Top of horiz. run, between batwing supports, hot leg, Row 84, Col. 29	ODSCC	Tube deformation from corrosion of vertical batwing supports, secondary side impurities
North Anna 1[1] W-51 600MA	7/15/87	637	360° circumferential crack	Top of 7th tube support plate, cold leg, Row 9, Col. 51	High-cycle fatigue	Lack of anti-vibration bar support, denting
McGuire 1[1] W-D2 600MA	3/7/89	500	95 mm long axial crack, 9.5 mm maximum width	At the lower tube support plate, cold leg, Row 18, Col. 25	ODSCC	long shallow groove, possible contaminant
Palo Verde 2 CE-80 600 MA	3/14/93	240	65 mm long axial fishmouth opening in an 250 mm long axial crack	Freespan between upper tube supports, hot leg, Row 117, Col.144	ODSCC	tube-to-tube deposit formation, caustic secondary water chemistry
*Indian Point 2[2] W-44 600 MA	2/15/00	146	56 - 61 mm long axial crack	U-bend apex, Row 2, Col.5	PWSCC	Hour-glassing

*Except for the IP2 event, this table was reproduced from a list of SGTRs in NUREG/CR-6365. The IP2 event, although not considered a tube rupture, was included in this list for comparison.

Table 1 - Tube Ruptures in US Plants Excluding Major Tube Leaks

The DPO was resolved by the promulgation of new tube performance criteria in 2007. These criteria are strictly based on predicting the probability of tube rupture during various accidents and the related leakage from Bobbin voltage measurements. Such predictions are not applicable when the mechanism of tube rupture is fretting fatigue. Voltage based methodology of leakage predictions does not bound fretting fatigue leakage because the latter results in an instantaneous circumferential tube rupture.

The reason why the leakage from SCC cracks is fundamentally different than leakage from tubes that exceeded their allowable fatigue life can best be illustrated by considering the design basis MSLB. Thermal Hydraulic (T-H) analysis shows that the pressure differential ΔP across a tube initially increases slowly and therefore even if several tubes contained a very large number of cracks they would open slowly minimizing the primary to secondary leakage. It is only later during the event when the SG has been emptied and the emergency core cooling, ECCS kicks in that the ΔP starts increasing. At this time however, its relative value is small. In contrast a rupture of a fretting fatigued tube does not depend on ΔP, the change in the sudden increase in stress intensity. (Increase in DP would be important only if wall thinning due to fretting was reduced to below the burst thickness). During the MSLB event, vibrations triggered by forces from outside or inside the steam generator vessel would be an obvious source for increases in local stress intensities. High cyclic stress from FEI during the MSLB event would cause a small crack to rapidly propagate circumferentially to failure when the tube is near or at its allowable fatigue life. Leakage increase from the propagation of circumferential fatigue cracks was not addressed in the DPO and therefore is not included in the 2007 tube performance criteria.

The NRC AIT report states that SCE informed the NRC that "they are reviewing their calculations of the LERF (4E-6/yr) and believe that review that will likely indicate that the differential pressures generated by a steam line break would not be large enough to rupture the degraded tubes as long as operators successfully implemented their emergency procedures. If this is confirmed, the risk associated with steam line breaks will be significantly reduced." Such a conclusion would be only valid if the tubes had not been damaged by fatigue. Since this is clearly not the case, SCE hope for lowering the LERF is unrealistic.

The stress intensity during the MSLB can best discussed in terms of the Stability Ratio (SR) which is an indicator of the FEI vibration intensity. SCE calculated that the SR varies from 0.33 to 1.15 and 0.16 to 0.83 for 100% and 70% power respectively, depending on tube location. Such reduction in the SR may be significant for steady state operation but is insignificantly small compared to the increase in SR on depressurization of the SG during the MSLB accident. The corresponding increase in velocity and steam quality overshadows the reduction in these parameters by operating San Onofre reactor Unit 2 at 70 % power. Therefore, the reduction in SR has no relevance to accident analysis when the tubes entering cycle 16 have been damaged by fatigue.

A second factor that distinguishes tube failures by SCC and high cycle fretting fatigue is the difficulty of detecting the latter during in-service inspections. This is because the crack initiation phase constitutes a high fraction of the total fatigue life in high cycle fatigue, once an engineering crack has been initiated, fracture occurs abruptly when the intensity level is sufficiently high (Ref. 7, 8, 9). The DPO project invested considerable effort on improving the sensitivity of eddy current detection of SCC cracks for leakage predictions. Since comparable data for predicting leakage from fatigue induced cracks does not have any safety analysis that is based on fatigue failures one cannot use the 2007 performance criteria to ensure safety. This is a reason why the SCE safety analysis is not valid and why it must be re-evaluated in terms of fretting fatigue induced leakage instead SCC induced leakage.

Since it took more then 15 years to develop the SCC based leakage methodology and close-out the DPO (and the related GSI 163) it cannot be expected that the NRC will revise the existing performance criteria any time soon. Until that time, conservative assessments must be performed before nuclear plants with considerable fretting fatigue damage are allowed to remain in service. The SCE safety assessment is not conservative.

Therefore, before starting San Onofre reactor Unit 2 at any power level, SCE must formulate an approach that would assure that the public safety margins would not be decreased. SCE can use any method for that purpose as long as it can defend it on a technically conservative basis. The following five accident scenarios are discussed to provide further insight why operation with pristine tubes at 100% power, current licensing base (CLB), is drastically different than operation at 70% power with fatigue damage tubes.

2. Accident Scenarios

The Steam Generators in San Onofre reactor Unit 2 (SGE 88 and SGE 89) contain 482 and 563 tubes respectively, with AVB wear ranging from 10% to 34%. The two SGs also contain a total of 515-plugged tubes. These tubes act as multiple sources for leakage during normal operations and during accidents (Ref. 9). They must be considered as sources for causing accidents and sources for propagating the leakage intensity during the accident. An assessment of operations with such degraded tubes must demonstrate that at any time during normal operations and during accidents their local gap velocities, the corresponding SR and the burst pressure, will remain at sufficiently low levels to prevent leakages from exceeding acceptable levels. The following accidents are examples of accidents which must be included in such assessments.

A. Spontaneous fretting fatigue rupture of a single steam generator tube in the free span with a stuck open relief valve or a broken header

Steam Generator overfill occurs relatively frequently in PWRs, an assessment should consider that the DBA SGTR will cause the relief valve to be stuck open during this event. The resulting higher local gap velocities and the corresponding increase in the SR must not cause additional tubes, (both plugged and un-plugged) to rupture.

B. Tube Ruptures from Unplanned closing of an isolation valve.

Closing an isolation valve would lead to an increase in steam flow through the unaffected SG. The corresponding increase in gap velocity would increase the local SR causing tubes which are on the border to exhausting their fatigue life to rupture abruptly (Ref 7, 8). This accident is similar to case A above with the exception that the increase in SR is expected to take place at a slower rate.

C. Seismically –Induced Tube Rupture

Both plugged and unplugged tubes can potentially lead to large primary to secondary leakage. Plugged tubes would behave differently, firstly because they do not generate a failure signal at the steam ejectors, and secondly, because the natural frequency of a broken tube would be lower than that that of an in service tube.

Reactor experience (Ref. 9) has demonstrated that tubes that have been plugged due to wear will continue to wear and eventually break to impact and damage adjacent tubes. Material loss by wear not the mode of failure at plants was studied by EPRI. In their studies combining tube swelling with Fluid Induced Vibration (FIV) led to circumferential fatigue failure. The difference between the cases studied by EPRI and the plugged tubes at SONGS is that at SONGS some plugged tube have already suffered considerable fatigue damage prior to plugging and are prone to fatigue failure. In this regard, EPRI recommends that tubes with pre-existing circumferential cracks be evaluated using linear elastic fracture mechanics. Because some tubes at SONGS used up a significant fraction of their fatigue life they may contain micro cracks of various size. Because such cracks have not been detected at SONGS there is no indication that they do not exist. SCE did not address this issue.

EPRI did not assess the effectiveness of tube stabilization in preventing damage to adjacent tubes; neither did SCE provide any information on their criteria for selecting tubes for stabilization.

SCE conclusions that the combined forces of the differential pressure and the seismic loads would not cause any tube to burst cannot be justified when the tubes are also subjected to cyclic loads simultaneously. SCE calculation are based on the tensile strength that would cause tube rupture, a much lower stress, less than half, would be sufficient to severe tubes with cumulative fatigue usage (CUF) near unity (Ref 8)

SCE calculations are based on a non conservative model and therefore their conclusions in the FSAR (5.4.2.2.1.3) regarding the ability of degraded tubes to withstand seismic loads are not valid.

D. Station Blackout, SBO

Severe accidents are not considered design basis accidents, nevertheless when changes in system operations are contemplated those changes must not increase safety risk. The operation of San Onofre reactor Unit 2 with a large number of fatigued tubes `represents a new accident that has never been previously analyzed.' All the analysis to date was based on tube failure by creep at high temperature. The fact that the tubes were fatigued damaged demonstrates they can fail earlier due to natural flow instabilities in the steam generator. The SBO accident is briefly described below.

In this accident the primary system remains pressurized following a core becoming uncovered. In the station blackout, SBO, accident scenario after the core is uncovered the secondary sides of all four steam generators are dry while on the primary side, steam flow by natural convection from the core to the steam generators and back to the core. The high pressure, high temperature steam will cause the weakest component in the system to fail thereby depressurizing the primary side. In this regard the hot leg surge line and the SG tubes are the weakest components in the reactor coolant system. If the high hoop stress on the hot leg surge line causes it to fail, the release of the highly radioactive gases will be contained within the containment. If on the other hand, the high pressure high temperature steam opens up existing cracks in the steam generator tubes or ruptures the tubes the primary side will be depressurized, by-passing the containment and allowing the highly radioactive gases to escape directly to the environment through the SG relief valve. The above scenario, also known as the high/dry core damage sequence, represents an early containment failure, which significantly increases the large early release frequency (LERF). When the containment fails early, the release to the environment is several thousands times larger in comparison to the release when the containment is intact. Most importantly, this early release occurs prior to the evacuation of the close population and therefore may cause early health effects (prompt fatalities).

Conformance to 10 CFR 50, Appendix B Criterion 16 dictates that operation with fatigued tubes will not increase the probability that fatigued tubes will not fail before the surge line. Appendix B dictates that to maintain its licensing basis the licensees must provide measures to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, are promptly identified and corrected. Fatigued tubes definitely represent conditions which are adverse to quality.

E. Main Steam Line Break, MSLB

The fact that San Onofre reactor Unit 2 can pass the existing performance criteria from the in-situ tests results of San Onofre reactor Unit 3 provides no assurance at all that during a spontaneous MSLB accident the leakage will not exceed the DBA leakage. The in-situ tests only show that if the tubes were only exposed to tested pressure they would not leak if they maintained their wall geometry as tested. The in-situ tests were intended to determine leakage on the basis of tube weakening by actual loss of material and inclusions of stress corrosion cracks. In contrast to static pressure tests, fatigue failure due to high cycle FIV would result in a fast propagating circumferential crack at relatively low stresses (Refs 1, 5). Leakage from degraded tubes must be assessed in terms of the mechanism that has the potential to cause the largest leakage.

If SCE wants to base their calculations on a realistic accident scenario, it must first demonstrate that the wear equation that was developed for laboratory data would be applicable to a tube that experienced impact wear in the SONGS steam generators. As discussed in Appendix A, the wear equation which was used by SCE to calculate wall thickness did not properly incorporate the effects of impact wear. Secondly and more importantly, SCE must demonstrate that their burst pressure mechanism of determining leakage is conservative in comparison to the leakage that would occur during the fast MSLB depressurization.

The fast depressurization of the secondary side following an MSLB will lead to rapid increases in local gap velocity steam quality, thereby significantly increasing the stability ratio SR. The higher SR would, in term, increase the stress on the tube leading to rapid circumferential crack propagation as occurred in North Ana (Ref 2)

F. Risk Considerations

The unusually large tube damage exhibited in both steam generators at San Onofre reactor Unit 2 is unprecedented, therefore little data is available to assess the increase in safety risk that would be associated with the above five accident scenarios. Consequently accident assessments must be based on conservative assumptions. The main uncertainty that must be considered in arriving at a risk estimate is the ability of the operator to shut the reactor down in a safe manner before depleting the RW Storage Tank inventory. Operator's success would depend primarily on the unpredictable increase in leakage in an environment experiencing violent vibrations due to secondary side depressurization. Operators are not trained in simulators that can reproduce such environments. In my judgment, based on computer calculations, an operator would not be able to prevent the reactor core from being uncovered if the number of tubes failures would exceed five. Given that steam generator 89 contain at least 500 AVB tubes which have used up a significant fraction of their fatigue life and another 86 TSP tubes (Ref 10) which also lost some fatigue life, a rupture of 5 tubes out of 600 susceptible tubes as result of fatigue failure during an MSLB event is not an overly conservative assumption.

Taken the probability of a steam a line break outside containment at E-4 per year the Large Early Release Frequency, LERF of radiation escaping the environment due to the reactor core being exposed becomes 1E-4 /year or 5 E-5 /yr for six month of operation. Such an increase is by a factor of 5 higher than Commission goals as described in Reference 11. In contrast, SCE calculated a change in LERF of 4E-6/yr on the basis of that the ΔP will not be large enough to rupture tubes.

The LERF is a measure of risk, the safety goal takes into consideration that the LERF must be by an order of magnitude lower than the core damage frequency (CDF) to account for a large and early radiation release due to containment bypass.

As discussed above, when the controlling mechanism of tube rupture is cyclic stresses from FIV, tube rupture will be controlled by variations in the stability ratio SR and not by variations in Δ P. During the MSLB the SR will be drastically increased due to an increase in local velocities and steam quality.

G. Summary

The reason that SCE concluded that operation of San Onofre reactor Unit 2 at 70% power would not involve a new unanalysed accident was because SCE assumed that the tubes would enter service in cycle 17 in the same conditions as they were at the beginning of cycle 16. In addition, SCE implicitly assumed that the stability ratio would not increase during Design Basis Accidents and the burst pressure could be determined by ignoring scaling effects in fretting wear by impacts. Based on ample reactor experience and laboratory data there is no basis to accept SCE proposed no statements to CFR 50.92.

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I declare, under penalty of perjury, that the foregoing information is true, accurate, and correct. Executed on **May 15, 2013**, in Rockville, MD.

Joram Hopenfeld

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