ATTACHMENT 3

Declaration of Dr. Victor Gilinsky

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE 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
	)	

DECLARATION OF VICTOR GILINSKY

I, Victor Gilinsky, state:

1. This affidavit, on behalf of Friends of the Earth ("FOE"), presents my views on the proposed determination by the Staff of the Nuclear Regulatory Commission ("NRC") that "no significant hazards consideration" ("NSHC") is involved in approving a license amendment to permit San Onofre Unit 2 to operate at 70 percent power. The Unit's defective steam generators preclude operation at 100 percent power; the issue is whether it is safe to do so at 70 percent. Under the NRC's rules an NSHC determination would allow the NRC to deprive the public of an opportunity for a hearing on this question in advance of plant operation. FOE seeks to participate in such a hearing and so opposes the NRC Staff's proposed determination. My view in brief, on the basis of my NRC experience, is that the NRC Staff misreads its regulations, that an NSHC determination is not permissible in this instance, and that therefore a public hearing should take place before the NRC authorizes any restart of Unit 2.

2. My professional qualifications and experience for dealing with the issues in this

case include an engineering degree and physics doctorate and two terms as an NRC commissioner. I was nominated by President Ford and re-nominated by President Carter. After leaving the NRC in 1984 I was principally involved as a consultant and expert witness in commercial and regulatory litigation over nuclear power plants. In one of the first such cases, in the late 1980s, I participated on behalf of Southern California Edison ("SCE") in its proceedings before the California Public Utilities Commission regarding the reasonableness of the construction costs for San Onofre Units 2 and 3. I subsequently participated in about twenty-five such major cases, in court trials, regulatory proceedings, and arbitrations. From 2001 to 2010 I was a consultant to the State of Nevada, which intervened in the NRC licensing proceeding over the proposed national nuclear repository at Yucca Mountain. A more detailed resume is attached to this affidavit as Exhibit 1.

3. My tenure on the NRC covered the adoption of the so-called Sholly amendment to the Atomic Energy Act. The Sholly amendment underlies the NRC's regulations, 10 CFR 50.91 and 50.92, on approval of license amendments for power reactors. The NRC sought this change in the law after the United States Court of Appeals for the District of Columbia Circuit held that the NRC could not make any power reactor licensing amendments, for which a hearing had been requested, immediately effective without a prior hearing. The NRC wanted to be able to deal simply and promptly with the many routine license amendments involving minor plant design changes of no or negligible safety significance, the replacement of a particular flow meter, for example, with a more accurate one.

4. In its 1986 analysis of a similar case to the one at hand, one involving changes to Diablo Canyon's spent fuel pool, the Ninth Circuit drew attention to the importance of the language in the Conference Committee Report accompanying the Sholly amendment, and specifically the Report's warning that the NRC Staff should not abuse the new authority. The Report warned the NRC not to apply the new authority regarding license amendments in cases where the NRC Staff required extensive investigation to resolve a power plant safety issue to its satisfaction, and instructed the NRC to resolve "borderline" cases in favor of a prior hearing. *San Luis Obispo Mothers for Peace v. US NRC*, 799 F.2d 1268 (9th Cir. 1986).

5. In short, Congress permitted NSHC determinations in routine cases that obviously had no or essentially no safety significance, but not otherwise. That is why the Sholly determination is not phrased as "no significant hazards," but as "no significant hazards *consideration*." The NSHC question is whether there is a safety issue, not whether the NRC Staff resolved the safety issue to its satisfaction.

6. As an NRC commissioner, I participated in a case that bears considerable similarity to the present one, and I then interpreted the law in the same way as I do now. In 1982 the owners of Three Mile Island Unit 1 sought to repair steam generator tubing using a new explosive technique. The NRC Staff concluded that because the new technique involved unreviewed safety issues, a license amendment was necessary. But when it came to restarting the plant the NRC Staff made an NSHC determination to allow restart in advance of a hearing.

7. In November 1983 Commissioner James Asselstine pointed out that the NRC staff made its determination after extensively analyzing the safety issue, but what the Sholly change to the law required was for "the Commission to determine whether the amendment presents any significant safety questions," not whether the NRC Staff was satisfied after its review. In January 1984, on the question of whether to concur with the NRC Staff, the Chairman and one commissioner voted to concur while Commissioner Asselstine and I opposed. In doing so I said I agreed with the NRC General Counsel's reading of the law, which was opposed to that of the Staff. *NRC Public meeting Transcript,* "Discussion/Possible Vote on TMI Steam Generators," January 10, 1984.
8. Commissioner Asselstine also observed that if one accepted the NRC Staff's interpretation of the regulations on NSHC determinations, then the NRC would never have to offer the opportunity for a hearing before authorizing any license amendments. Indeed, that is apparently the NRC Staff view. If there are any recent cases for which the Staff went along with a hearing in advance of authorizing an amendment, they are rare. Even a proposed 20 percent increase in a reactor's authorized power level—the maximum uprate approved to date and one that involved extensive safety analysis—merited an NRC Staff NSHC determination. Vermont Yankee Final Determination, March 2, 2006. (To get a sense of how much NRC has changed: in publishing the NSHC rule two decades earlier, the NRC had given an authorized power level increase as an example that *was* "likely to involve significant hazards considerations." Cited in SECY-01-0142, July 27, 2001.)

9. In the current case, the NRC adopts the SCE's NSHC analysis without a word of explanation. That analysis treats the issue as if SCE were proposing to down rate a plant with healthy steam generators. It essentially argues that since the plant is licensed for 100 percent power, how can there be a problem in running it at 70 percent?

10. The actual situation is rather different. San Onofre has defective steam generators that pose a safety problem. That is why SCE cannot get NRC approval for 100 percent operation and proposes instead to operate at 70 percent power. Resolution of the safety issues involves subtle technical questions and complex analyses of the vibrational modes of the approximately 9,000 long thin steam generator tubes through which flows water heated by the reactor's core. (The Technical Evaluation Report on the issues by the steam generators' Japanese manufacturer, Mitsubishi Heavy Industries, runs to 142 pages.)

11. SCE and NRC Staff failed to appreciate the safety significance of the technical issues when the replacement steam generators were designed. The question now is whether they have finally gained a solid understanding of the safety issues and can confidently determine at what power level it is safe to operate. That is what the hearing needs to be about.

12. Outside the hearing context the NRC acknowledges the seriousness of the San Onofre steam generator tube degradation: that is of course why the Units are not operating, why the case is spotlighted on the NRC web page, and why the plant was put into the special NRC oversight category.

13. In evaluating the NSHC determination it is important to understand the full

safety significance of a breach in the steam generators. Although the NRC's safety doctrine stresses multiple barriers between the radioactive reactor core and the external environment, in fact, the steam generator tubing, about an acre of thin metal, less than $1/10^{th}$ of an inch thick, is the single sure barrier between the hot pressurized water flowing from the reactor core, and the outside environment. On the steam side of the steam generator there are large safety valves that lift if the steam line pressure is too high. These valves sometimes stick open. If this were to happen during the course of a severe accident involving core melting accompanied by failure of many steam generator tubes, there would be no way to contain the radioactive release. The possibilities for such extreme accidents were not considered during the original licensing of US nuclear plants. But since the 2011 Fukushima accident it has become clear that the regulatory system has to take them into account.

14. Fukushima also demonstrated that up to now the NRC has omitted in its analyses the most important consequences and costs of severe accidents. The NRC has only counted the possible radiation doses that individuals might receive after an accident. But we have learned from Fukushima that the most important consequences are the costs that attach to evacuation of large populations--the disruption to lives and communities and the essentially permanent loss of large land areas. So far as I can tell these costs are still not considered in any NRC license safety reviews.

15. Under Section 50.92 of the NRC regulations, the NRC Staff deems a proposed amendment to involve "no significant hazards considerations" if it meets a three-

part test: that the proposed amendment does not "(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (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." In my opinion the proposed San Onofre Unit 2 amendment does not qualify for an NSHC determination under any of these three criteria in the sense that Congress intended for these to be interpreted.

16. The first thing to say about the application of 50.92 is that it requires calculations of accident probabilities. The methodology the NRC Staff uses to do this, so-called probabilistic risk assessment, while useful for certain purposes, is not reliable for estimating overall accident probabilities. The complex calculations involve many questionable assumptions about the interconnections of individual subsystems. This puts any proposed amendment involving a hardware change that requires sophisticated technical safety analysis at least into the "borderline" category that Congress warned did not qualify for an NSHC determination.

17. Regarding (1), that the proposed amendment does not "involve a significant increase in the probability or consequences of an accident previously evaluated," it is clear that a current calculation of the consequences of an accident, including the previously omitted costs of large-scale evacuation, would show a significant increase in accident consequences from what was previously calculated.

18. Regarding (2), the original accident analyses conducted in connection with the plant's licensing examined the consequences of the failure of a very limited number of steam generator tubes. The flawed design of San Onofre's

replacement steam generators raised the possibility of failures beyond those previously analyzed during licensing. That was NRC's concern and that is why it required SCE and its vendor to carry out sophisticated analyses of the vibrational modes of the steam generator tubing and the possibilities for inducing wear in the tube walls.

19. Criterion (3) asks whether the amendment would permit a significant reduction in the previously approved margin of safety: We know the NRC Staff is concerned that operation of the plant at 100 percent power may significantly reduce the safety margin at that level. Whether that safety reduction is eliminated at 70 percent power is a complex question whose resolution involves technical analyses and professional judgment.

20. To arrive at its answer the NRC had to require considerable effort on the part of the SCE, and to expend considerable effort itself, to reach into the merits of the issue. This makes this exactly the kind of case Congress warned the NRC about when it wrote (in the Sholly amendment Conference Committee Report, as cited by the Ninth Circuit's 1986 decision in San Luis Obispo Mothers for Peace) that the NSHC standard "should not require the NRC staff to prejudge the merits of the issues raised by the proposed license amendment." But that is exactly what the NRC Staff has done.

21. In addition to the law, the public interest argues strongly for holding a hearing on a safety-significant amendment in advance of a licensing decision. In my experience, the working level Staff reviewers are generally competent and responsible, but they are also under pressure to meet agency schedules. They do

a much better job when they know they will have to defend the soundness of their technical conclusions before impartial judges in a public hearing on which a licensing decision depends. A hearing after the fact is a pale substitute.

22. The special circumstances of this case provide an additional reason for conducting a public hearing in advance of any licensing decision. As mentioned earlier, the NRC Staff failed in its first chance to review adequately the steam generator design when SCE proposed replacing the original one with a different design. The NRC Staff is now in the position of judging the significance of that failure and how much can be recovered from the earlier mistakes. It would be only human if this earlier experience affected the NRC Staff's objectivity. In this situation it would be especially useful for the NRC Staff's conclusions to be tested before an impartial adjudicatory panel, and for that panel to hear the views of outside experts, before these conclusions are put into effect.

23. In sum, in this case involving important safety issues regarding operating with defective steam generators, both the law and public interest call for an NRC hearing in advance of any decision about plant operation.

I declare, under penalty of perjury, that the foregoing information is true, accurate, and correct. Executed on May 14, 2013, in Los Angeles, CA.

Victor Gilinsky

RANDOLPH EDELMAN
Commission # 1863472
Notary Public - California
Los Angeles County
My Comm. Expires Sep 30, 2013

Randolph Edelman
SEE ATTACHED CERTIFICATE

ACKNOWLEDGMENT

State of California
County of LOS ANGELES)

On MAY 14, 2013 before me, RANDOLPH EDELMAN, NOTARY PUBLIC
(insert name and title of the officer)

personally appeared VICTOR GILINSKY,
who proved to me on the basis of satisfactory evidence to be the person~~(s)~~ whose name~~(s)~~ is/~~are~~ subscribed to the within instrument and acknowledged to me that he/~~she/they~~ executed the same in his/~~her/their~~ authorized capacity~~(ies)~~, and that by his/~~her/their~~ signature~~(s)~~ on the instrument the person~~(s)~~, or the entity upon behalf of which the person~~(s)~~ acted, executed the instrument.

I certify under PENALTY OF PERJURY under the laws of the State of California that the foregoing paragraph is true and correct.

WITNESS my hand and official seal.

RANDOLPH EDELMAN
Commission # 1863472
Notary Public - California
Los Angeles County
My Comm. Expires Sep 30, 2013

Signature Randolph Edelman (Seal)

OPTIONAL INFORMATION

DESCRIPTION OF THE ATTACHED DOCUMENT

DECLARATION OF
(Title or description of attached document)

VICTOR GILINSKY
(Title or description of attached document continued)

Number of Pages 9 Document Date 05-142013

(Additional information)

VICTOR GILINSKY

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Professional Activities:

1984-present	Independent consultant to energy firms and governmental, commercial, and non-profit organizations, primarily on matters related to nuclear energy
1975-1984	Commissioner U.S. Nuclear Regulatory Commission Washington, DC
1973-1975	Head, Physical Sciences Department, and Director, Applied Science and Technology Program The Rand Corporation Santa Monica, CA
1972-1973	Assistant Director for Policy and Program Review, Office of Planning and Analysis U.S. Atomic Energy Commission Washington, DC
1961-1971	Physicist The Rand Corporation
Education:	Ph.D., Physics, California Institute of Technology, 1961 Bach. Engineering Physics, Cornell University, 1956
Awards:	California Institute of Technology Distinguished Alumni Award, 1982
Member:	American Physical Society Institute of Electrical and Electronics Engineers

ATTACHMENT 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)

Docket No. 50-361 and 50-362
FRIENDS OF THE EARTH
EXHIBIT ONE

UNITED STATES OF AMERICA BEFORE THE NUCLEAR REGULATORY COMMISSION

In the matter of

	)	May 31, 2012
Southern California Edison Company	)	Docket No. 50-361 and 50-362
	)	
San Onofre Nuclear Generating Station	)	

DECLARATION OF ARNOLD GUNDERSEN SUPPORTING THE PETITION TO INTERVENE BY FRIENDS OF THE EARTH REGARDING THE ONGOING FAILURE OF THE STEAM GENERATORS AT THE SAN ONOFRE NUCLEAR GENERATING STATION

I, Arnold Gundersen, declare under penalty of perjury under the laws of the United States of America that the following is true and correct, and executed this 31st day of May 2012:

1. My name is Arnold Gundersen. I am sui juris. I am over the age of 18-years-old.

2. As Chief Engineer for Fairewinds Associates, I have been retained by Friends of the Earth to provide expert services in connection with the above captioned matter regarding the ongoing failure and deterioration of the steam generators at San Onofre Nuclear Generating Station.

3. I earned my Bachelor Degree in Nuclear Engineering from Rensselaer Polytechnic Institute (RPI) cum laude. I earned my Master Degree in Nuclear Engineering from RPI via an Atomic Energy Commission Fellowship. Cooling tower operation and cooling tower plume theory were my area of study for my Master Degree in Nuclear Engineering.

4. I began my career as a reactor operator and instructor in 1971 and progressed to the position of Senior Vice President for a nuclear licensee prior to becoming a nuclear engineering consultant and expert witness. My Curriculum Vitae is attached as Exhibit 2.

5. I have testified before the Nuclear Regulatory Commission (NRC) Atomic Safety and Licensing Board (ASLB) and Advisory Committee on Reactor Safeguards (ACRS), the State of Vermont Public Service Board, the State of Vermont Environmental Court, the Florida Public Service Commission, the State of New York Department of Environmental Conservation, and in Federal Court.

6. I am an author of the first edition of the Department of Energy (DOE) Decommissioning Handbook, and the book entitled *Fukushima Daiichi: The Truth And The Way Forward*, Shueisha Publishing, 2012-2-17, Japan.

7. I have more than 40-years of professional nuclear experience *including and not limited to*: Cooling Tower Operation, Cooling Tower Plumes, Consumptive Water Loss, Nuclear Plant Operation, Nuclear Management, Nuclear Safety Assessments, Reliability Engineering, In-service Inspection, Criticality Analysis, Licensing, Engineering Management, Thermohydraulics, Radioactive Waste Processes, Decommissioning, Waste Disposal, Structural Engineering Assessments, Nuclear Fuel Rack Design and Manufacturing, Nuclear Equipment Design and Manufacturing, Prudency Defense, Employee Awareness Programs, Public Relations, Contract Administration, Technical Patents, Archival Storage and Document Control, Source Term Reconstruction, Dose Assessment, Whistleblower Protection, and NRC Regulations and Enforcement.

8. I have personal knowledge of the facts contained in this Declaration; and I am qualified to testify in support of this Petition. I have previously testified to the Advisory Committee on Reactor Safeguards and the NRC's 2.206 Petition Review Board.

OVERVIEW AND SCOPE OF THE PROCEEDING

9. My declaration is intended to support Friends of the Earth's Petition Concerning the Steam Generators at San Onofre Nuclear Generating Station.

SAN ONOFRE NUCLEAR REACTOR BACKGROUND

10. Originally designed and built by Combustion Engineering (CE), San Onofre's nuclear steam generators are a very unique design that is radically different from all other Pressurized Water Reactor (PWR) designs. Southern California Edison (Edison) decided to replace each San Onofre steam generator due to tube deterioration and degradation that slowly evolved during each Unit's 25-years of operation.

11. Documents reviewed show that the four replacement steam generator specifications are identical to each other and they were purchased together under a single contract with Mitsubishi Heavy Industries (MHI). However, rather than simply rebuild the steam generators to their original design specifications, Edison decided to extensively modify the original San Onofre steam generator design. Furthermore, none of the design modifications were necessary for operation of either San Onofre Unit 2 or 3.

ISSUES OF REACTORS

12. It now appears that after new Steam Generators were installed at San Onofre Unit 2 and Unit 3, the new tubes began to seriously degrade very quickly. Technicians first detected the unanticipated problems of significant wear in the tubes during the Unit 2 refueling outage in January 2012.

13. The wear-rate for these steam generator tubes is extraordinary because tube thickness has been reduced by as much as 30 percent in less than two years. While Unit 2 was shutdown for refueling, San Onofre Unit 3 was operating at full power when it experienced a complete perforation of one steam generator tube that allowed highly radioactive water from inside the reactor to mix with the non-radioactive water that turns the turbine.

14. As a consequence, an uncontrolled release of radiation into the environment ensued, and San Onofre Unit 3 was also forced to shut down due to steam generator failure.

RISKS POSED

15. The San Onofre reactors have significant problems because their newly installed steam generators have extensive degradation and are unable to perform their design function of containing the radioactive water in the facility. Steam generator tube degradation, like that which San Onofre is experiencing, causes a significant nuclear safety risk by substantially increasing the likelihood of an accident that releases radioactivity into the environment.

16. Unfortunately, a leak or disintegration of one or more tubes would cause the radioactive water to escape the containment. Because there is a 1,000-pound-per-square-inch (psi) pressure difference between the high-pressure radioactive side of the tubes and the lower pressure steam that then leaves the containment, a leak will inevitably release radioactivity to the environment.

17. Gross failure of one or more of the steam generator tubes could create a nuclear design basis accident and cause the nuclear reactor core to lose a portion of its cooling water. However, the unique concern of degraded steam generator tubes is that uncontrolled radiation releases from a tube break do not remain inside the containment building and instead leak out of the facility and into public areas via atmospheric dump valves and steam generator blowdown.

18. If a steam line break accident were to occur, the depressurization of the steam generator caused by the steam line break coupled with the lack of water at the top of the steam generators would cause cascading tube failures, involving hundreds of tubes. The cascading tube failures would pop like popcorn and cause excessive offsite radiation exposures.

CASCADING DESIGN CHANGES AS BASIC CAUSE

19. A cascading series of deliberate design changes likely caused the tube failures and tube degradation.

20. The key fabrication change supplanted to the San Onofre steam generators by the Edison/MHI team increased the total number of tubes in each steam generator by almost 400

tubes to more than 104 percent of each generator's original design. Each Original Steam Generator contained 9350 tubes while the Replacement Steam Generators each contain 9727 tubes.

21. Fairewinds believes it was this management decision to increase the number of tubes that lead in turn to a series of cascading design changes that created the serious problems San Onofre is experiencing in 2012.

22. The original San Onofre steam generator contained a tubesheet, which is a metal disc approximately 13-feet in diameter and slightly less than two feet thick, located near the bottom of the steam generator. Due to the already extremely large size of the CE steam generators, this tubesheet is one of the largest tubesheets ever fabricated after which 18,700 holes (9,350 in-hot/9,350 out-cold) were then drilled. This metallic disk serves as an anchor into which both sides of the U-tubes are inserted. Not only is the tubesheet extraordinarily heavy, but also there can be a pressure difference of approximately 2,000 pounds per square inch (psi) between the radioactive water on one side and non-radioactive water on the other.

23. In order to support the enormous tubesheet metallic disk, the original steam generator design at San Onofre contained a 'stay cylinder' in the center of the tubesheet that is a support pillar designed to relieve the weight in the middle of the tubesheet.

23.1. When Edison decided to cram in additional steam generator tubes, the fabrication technique created by Edison/MHI for the San Onofre steam generators necessitated the removal of the 'stay cylinder' so that more tube holes could be drilled through the tubesheet. The Edison/MHI decision to add additional tubes and replace this key support pillar was part of the cascading fabrication changes that caused additional stresses and steam generator failure.

23.2. Removing the stay cylinder required additional cascading fabrication changes. Because the tubesheet was no longer supported in the center by the stay cylinder, Edison/MHI required the fabrication of a thicker tubesheet so that it could bear the additional stress without a stay cylinder. This change in the tubesheet thickness meant

yet another design change by reducing the volume of water in the steam generator and changing the flow pattern and also reducing the inspection access area beneath the tubesheet that is required to fit personnel and equipment for tube inspection.

23.3. Changing the structural loads on the tubesheet have not only affected the reliability of the steam generators but also should have raised a serious safety concern because the tubesheet is the key barrier keeping radiation inside the containment. Should the tubesheet fail, radiation within the reactor would bypass the containment and pass directly into the environment. Due to the installation of the ‘stay cylinder’ in the original San Onofre steam generator configuration, a tubesheet failure and subsequent radiation release is considered to be beyond the calculations for a design basis accident at San Onofre. Yet Edison chose to challenge this critical safety barrier and licensing parameter by removing the “stay cylinder” in order to install more, unnecessary tubes.

23.4. Fabricating more tubes increased nuclear reactor core flow, which was unacceptable because it changed the original design basis safety calculations for cooling the reactor. For that reason Edison welded a flow-restricting ring into the steam generator nozzle in order to reduce the flow of cooling water back into the reactor to the original design parameters, which also changes the flow distribution to the tubes. Thus significant operational changes were also made to the radioactive side of the steam generator as a result of Edison’s addition of more steam generator tubes.

23.5. All of these changes necessitated even more fabrication changes within the steam generator. For example, more tubes meant that the tube supports had to be modified in an attempt to avoid the increased vibration caused by the flow changes induced by the Edison/MHI fabrication changes. The feedwater distribution ring inside the steam generator was also dramatically modified in order to avoid a serious flow induced water hammer.

SIGNIFICANCE OF DESIGN MODIFICATIONS ON SAFETY

24. The requirements for the process by which nuclear power plant operators and licensees may make changes to their facilities and procedures as delineated in the safety analysis report and without prior NRC approval are limited by specific regulations detailed in The Nuclear Regulatory Commission's *10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, Section 50.59, Changes, Tests and Experiments*.

25. The implementing procedures for the 10 CFR 50.59 regulations have eight criteria that are important for nuclear power plant safety. (These eight criteria are provided in Table 1, footnote A below.)

26. These implementing procedures created for 10 CFR. 50.59 require that the license be amended unless none of these eight criteria are triggered by any change made by Edison at San Onofre. If a single criterion is met, then the regulation requires that the licensee pursue a license amendment process.

27. By claiming that the steam generator replacements were a *like-for-like* design and fabrication, Edison avoided the more rigorous license amendment process. From the evidence reviewed, it appears that the NRC accepted Edison's statement and documents without further independent analysis. In the analysis detailed below, Fairewinds identified 39 separate safety issues that failed to meet the NRC 50.59 criteria. Any one of these 39 separate safety issues should have triggered the license amendment review process by which the NRC would have been notified of the proposed significant design and fabrication changes.

28. As the NRC guidelines state:

> "(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as 1.187-A-1updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to § 50.90 only if: (i)A change to the technical specifications

incorporated in the license is not required, and **(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section."[1]** [Emphasis Added]

29. In its previous reports, Fairewinds identified at least eight modifications to the original steam generators at San Onofre.

30. Table 1 below was designed to compare the eight major design modifications that Fairewinds identified in its analysis with the eight criteria the NRC applies to the license review process in order to determine whether or not a new license amendment process is required.

31. The major design changes are located at the top of the table, and the NRC Criteria are listed in the left hand column of table. The term SSC stands for Systems, Structures and Components. A green *No* means that the *like-for-like* criteria were indeed met and that no license amendment was required. A red *Yes* means that Edison should have applied for a license amendment.

32. Table 1 shows that 7 out of 8 of the major design changes to the original steam generators meet a total of 39 of the NRC's 50.59 criteria requiring amendment to the license.

[1] See, 1.187-A-1, ttp://pbadupws.nrc.gov/docs/ML0037/ML003759710.pdf

Table 1
Steam Generator Design Changes Identified By Fairewinds Compared With The NRC's Like-For-Like Criteria

50:59 Criteria (A)	(B) Remove stay cylinder	Change tube sheet	Tube alloy change	Add tubes	Change tube support	Add flow restrictor	Additional water volume	Feed water distribution ring
i – Accident Frequency Increase	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	No	No	No
ii – Increase in SSC Malfunction occurrence	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	No	No	No
iii - Accident consequent increase	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	Yes (2)	Yes (2,5,6)	No
iv - Increase in SSC consequence of malfunction	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	Yes (2)	Yes (2,5,6)	No
v - Create unanalysed accident	Yes (1)	Yes (1)	No	No	No	Yes (2)	Yes (2,5,6)	Yes (3,7,8)
vi – Create new malfunction	Yes (1)	Yes (1)	No	No	Yes (3,8)	Yes (2)	No	Yes (3,7,8)
vii – Alter fission product barrier	Yes (1)	Yes (1)	No	Yes (3)	No	No	No	No
viii – Change design basis evaluation method	Yes (2)	Yes (2)	No	Yes (2)	Yes (2,8)	Yes (2)	Yes (2,5,6)	No

Table Footnotes

A - The criteria listed in the left column in the table above refers to the criteria as laid out in the NRC Guidelines[2] which states as follows:

"(2) A licensee shall obtain a license amendment pursuant to § 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);

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[2] See, 1.187-A-1, ibid, ttp://pbadupws.nrc.gov/docs/ML0037/ML003759710.pdf

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);
(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);
(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or
(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses."

B – The horizontal axis contains a list of design changes made by Edison and whether they meet or have not met the criteria as set out in 10 CFR 50.59.

1 – The Steam Generator Replacement Project modified the tube sheets and stay cylinder that are a containment barrier – The NRC was not informed nor did it specifically approve these changes to the containment barrier as they were apparently not addressed under Edison's analysis for the 10 CFR 50.59 process;
2 – The Mitsubishi thermo hydraulic code is inadequate to assess flow inside the Steam Generators that dramatically affect the ability to cool the nuclear reactor core in the event of an accident;
3 – The Steam Generator Replacement Project increases the consequences of a steam line break accident;
4 – The Steam Generator Replacement Project has already proven to increase the frequency of tube failure;
5 – The Steam Generator Replacement Project changed the volume of primary coolant because more tubes were added, which changes the Final Safety Analysis Report;
6 – The Steam Generator Replacement Project changed the flow rate of primary coolant, which changes the Final Safety Analysis Report;
7 – The Steam Generator Replacement Project changed the potential for water hammer. Given that the Mitsubishi thermo hydraulic code is inadequate, the potential for water hammer is increased;
8 – The Steam Generator Replacement Project created steam binding at top of steam generator. The steam generator is designed to remove heat in the event of an accident and its role has been compromised.

The Actual Steam Generator Problem Causing Vibration

33. As water moves vertically up in a steam generator, the water content reduces as more steam is created. When the volume of steam is much greater than water then the flow resistance of the water/steam mixture passing through the tube supports accounts for one third of the total resistance at the top of the steam generator. Therefore to avoid vibration at the top of the tubes, Mitsubishi needed to specifically analyze the type of tube support to use in this unique application.

34. The flow resistance of the Mitsubishi broached plate is *much higher* than that of the original Combustion Engineering egg crate design because the tubes are so tightly packed in the original CE San Onofre steam generators. By reviewing the documents thus far produced, it appears that due to Mitsubishi's fabrication experience with broached plates, both Edison and Mitsubishi missed this key difference in the design and fabrication of the new San Onofre steam generators.

35. Not only is Mitsubishi unfamiliar with the tightly packed CE design, but also Edison's engineers created so many untested variables to the new fabrication that this new design had a significantly increased risk of failure. As a result of the very tight pitch to diameter ratios used in the original CE steam generators, Mitsubishi fabricated a broached plate design that allows almost no water to reach the top of the steam generator.

36. The maximum quality of the water/steam mixture at the top of the steam generator in the U-Bend region should be approximately 40 to 50 percent, i.e. half water and half steam. With the Mitsubishi design the top of the U-tubes are almost dry in some regions.[3] Without liquid in the mixture, there is no damping against vibration, and therefore a severe fluid-elastic instability developed.

37. In response to the Edison/Mitsubishi steam generator changes, the top of the new steam generator is starved for water therefore making tube vibration inevitable. Furthermore, the problem appears to be exacerbated by Mitsubishi's three-dimensional thermal-hydraulic analysis determining how the steam and water mix at the top of the tubes that has been benchmarked against the Westinghouse but not the Combustion Engineering design.

38. The real problem in the replacement steam generators at San Onofre is that too much steam and too little water is causing the tubes to vibrate violently in the U-bend region. The tubes are quickly wearing themselves thin enough to completely fail pressure tests. Even if the new tubes are actively not leaking or have not ruptured, the tubes in the Mitsubishi fabrication are at risk of bursting in a main steam line accident scenario and spewing radiation into the air.

RAMIFICATIONS OF AN INADEQUATE NRC REVIEW

39. Edison's strategic goal was to avoid the process of license amendment according to the January 2012 article in *Nuclear Engineering International NEI Magazine.*[4] Had Edison

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[3] With the Mitsubishi design the top of the U-tubes are almost dry in some regions. Fairewinds research and four independent industry experts, who wish to remain anonymous, substantiate this statement.

[4] Improving Like-For-Like Replacement Steam Generators by Boguslaw Olech of Southern California Edison and

notified the NRC that the new steam generators at San Onofre were not a *like-for-like* replacement, a more thorough review through the license amendment process would have been required. Given that scenario, it is likely that the requisite and thorough NRC review would have identified the design and fabrication inadequacies that appear to have caused the San Onofre steam generator tube failures.

40. More specifically, Fairewinds believes that the NRC would have identified the inadequacy of the Mitsubishi Heavy Industry computer code applied to validate the tube design and vibration pattern prior to fabrication. Mitsubishi's computer code was simply not capable of analyzing Combustion Engineering (CE) designs like San Onofre and was only qualified for Westinghouse designs that are not similar to the original CE steam generator design. In NRC licensing jargon, the Mitsubishi design codes were not benchmarked for the CE Design.

41. While Mitsubishi Heavy Industry has been supplying steam generators for many years in Japan, it did so under a specific license from Westinghouse for Westinghouse nuclear reactors. Although Mitsubishi made several incremental changes to the Westinghouse design, such as switching to alloy 690 tubing and the use of stainless steel broached plate tube supports, Mitsubishi has had very little experience with the tight tube pitch and the egg crate design used in the original CE design for San Onofre.

REPAIR

42. San Onofre engineers should have precise maps detailing the degraded and leaking tubes as well as the exact location of the leak(s) on each tube. Such data is just one piece of critical information required in conducting a thorough root cause analysis of the problem and determining an accurate solution. Edison claims that the proximate cause of these U-tube failures at San Onofre is high vibration, and it has embarked upon a process of plugging some of these damaged tubes in hopes of quickly restarting one or both units. Fairewinds

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Tomouki Inoue of Mitsubishi Heavy Industries, Nuclear Engineering International, January 2012, page 39. This article was based on a paper published at ICAPP 2011, 2-5 May 2011, Nice, France, paper 11330. Boguslaw Olech, P.E., Southern California Edison Company, 14300 Mesa Rd., San Clemente, CA 92674, USA, Email: bob.olech@sce.com. Tomoyuki Inoue, Mitsubishi Heavy Industries Ltd. (MHt), 1-1 Wadasaki-cho 1-Chome, HyogoKu, Kobe, Japan 652 8585, Email: tomoyukiJnoue@mhi.co.jp.

believes that this damage is occurring on the outside of the tubes where they collide with each other, while access to the tubes for repair and/or plugging can only be conducted from inside the tubes. Space limitations due to the tight fit of the 9,700 tubes (19,400 holes in the tube sheet) in each steam generator have made it impossible to access the outside of the U-tubes for inspection where the wear is actually occurring.

43. Presently, the Edison approach is to plug tubes in the most heavily damaged zone of each steam generator. Plugging the tubes only eliminates the radioactive water inside the tubes, but it does not eliminate the vibration, so the plugged tubes will continue to vibrate and damage adjacent tubes.

44. If a steam line break accident were to occur, the depressurization of the steam generator caused by the steam line break coupled with the lack of water at the top of the steam generators would cause cascading tube failures, involving hundreds of tubes. The cascading tube failures would pop like popcorn and the cascading failures would cause excessive offsite radiation exposures.

45. Fairewinds investigation has found that plugging the tubes is not a sure solution, because it fails to deal with the root causes of a failed design and it relies upon the incorrectly applied Mitsubishi 3-Dimensional steam analysis to determine which tubes should be plugged. Realistically, the 3-D steam analysis is not accurate enough to apply to such important safety-related determinations. To make such mathematical risk 3-D analysis, a very large margin of error must be applied, and that has not been done. For example, if the 3-D steam analysis determines that plugging 100 tubes is a solution, then plugging ten times that number might be the appropriate solution due to the mathematical errors in the 3-D analysis being applied by Edison and Mitsubishi.

46. Fairewinds concludes that plugging the tubes will never solve the underlying problem because vibration is the result not the root cause of the steam generator problems at San Onofre. The actual problem is a variety of design changes that have caused too much steam and too little water at the top of the steam generators. Plugging tubes cannot repair these design changes that are causing the tubes to collide with each other.

OPTIONS FOR CONTINUED OPERATION

47. Complete Replacement - The ongoing plugging of the tubes will not eliminate the vibrational failure mechanism causing tube failures. Over time, the damaged tubes that are plugged will in turn damage more tubes. Therefore, Fairewinds believes that the only sure solution to this significant safety issue is to once again cut open the reactor containment and install new steam generators that replicate the original CE design.

48. Repair In Place - While technically this would be an extremely challenging repair process, it may be possible to cut the steam generators apart while still inside the containment. Such a process would take approximately 18 months to make repairs and then weld the steam generators back together again without cutting the containment open. Cutting the top off the steam generators would allow construction personnel access so that additional supports could be inserted into the U-tube region. Smaller replacement packages would fit through the existing equipment hatch and the containment would not be compromised another time. The cost for these repairs would be less than completely redesigning and manufacturing new steam generators and replacement power costs would be less.

49. Power Reduction - Reducing power does not provide a remedy for the underlying structural problems that are creating the vibration that has damaged and will continue to damage tubes deep inside the San Onofre steam generator. Edison has suggested that plugging tubes and operating at indeterminate reduced power levels for the remainder of the life of the plant may be a solution to the San Onofre tube vibration problem.

50. Unfortunately this course of action would leave San Onofre operating with a significant safety risk if the NRC were to allow the reactors to restart.

51. Operating at reduced power will not prevent previously damaged tube supports and plugged tubes from vibrating and damaging surrounding tubes and tube supports, and it will worsen the existing damage.

52. More importantly, Fairewinds concern is that operating the San Onofre reactors at a lower power and flow rate might actually create a resonant frequency within the steam generators at

which some of the tubes will vibrate as bad or worse than they did originally. Because the plugged tubes are now filled with air their weight has changed, and therefore the plugged tubes will vibrate with a different amplitude and frequency. The inaccuracies in the Edison and Mitsubishi computer code do not allow Edison and Mitsubishi to conduct a resonant frequency analysis proving that such a problem will not occur.

53. Historical evidence from other operating nuclear reactors that have attempted to mitigate vibrational damage by using power reductions rather than solving the resonant frequency issues have in fact compromised other nuclear safety related components by operating at reduced power.

53.1. In 2002 the Exelon Quad Cities Nuclear Power Plant in Illinois operated its Unit 2 reactor at reduced power in order to eliminate vibrationally induced damage causing high moisture carryover in its steam dryer. While the power reduction temporarily reduced moisture carryover, the problem reoccurred and a shutdown was ordered causing an extended unplanned outage. Vibrationally induced severe cracking was discovered in the steam dryer and repaired. Following an analysis and subsequent repairs, Exelon claimed to have rectified the Quad Cities Unit 2 problems only to be forced in 2003 to once again attempt operation at a reduced power level when vibrationally induced steam dryer moisture carryover became excessive. Following this second attempt to operate the reactor at a reduced power level, pieces of the dryer as large as a man broke off and damaged nuclear power safety related components, and a second unplanned extended outage ensued. Once again, vibration was determined to be the cause of the gross failure and another unplanned and forced outage. Finally, following years of analysis and two damaged steam dryers, Quad Cities made major piping modifications that are alleged to have eliminated harmonic frequencies, prevented further component damage, and allowed Unit 2 to eventually return to full power production.[5]

53.2. A second example of a failed attempt to reduce power to solve vibrationally induced resonance frequency problems occurred at the Susquehanna nuclear plant in

[5] http://pbadupws.nrc.gov/docs/ML0609/ML060960338.pdf

Pennsylvania. During the mid 1990s, a vibrationally induced failure in the jet pump sensing lines occurred at Susquehanna. This failure was attributed to the vane passing frequency from the recirculation pumps causing harmonic vibration of the lines. Like Quad Cities, Susquehanna attempted to implement a power reduction in order to minimize the harmonic vibrations. Unfortunately, the resonant vibration issues continued to damage systems after the power was reduced thereby forcing an unplanned outage and extensive modifications and repairs.

CONCLUSION

54. In conclusion, the NRC has stated that nuclear power plants like San Onofre cannot risk compromising critical safety systems and possible radiological contamination in an effort to return to operation before a thorough root cause analysis, modifications, and subsequent repairs are adequately reviewed by the NRC and implemented. Historical evidence has proven that power reductions do not solve underlying and serious degradation problems, resonance frequency issues. Rather, power reductions can significantly increase the risk of unplanned, forced outages during times of peak demand and can cause significant risk to public health in the event of a single tube rupture or a series of ruptures if the main steam line were to break.

55. Finally, if a steam-line accident were to occur, vibrationally induced tube damage at San Onofre could cause an inordinate amount of radioactivity to be released outside of the containment system compromising public health and safety in one of the most heavily populated areas in the entire United States.

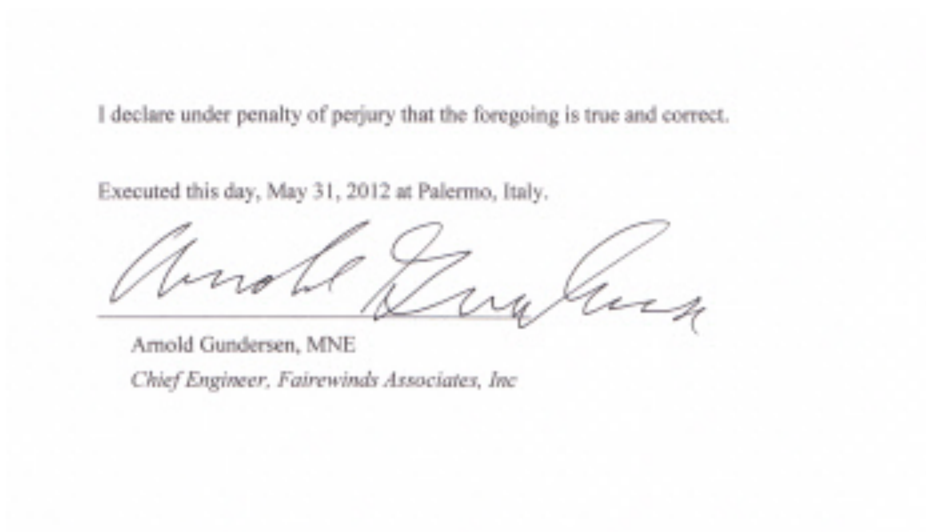
-End-

I declare that under penalty of perjury that the foregoing is true and correct to the best of my knowledge. The facts presented in this declaration are true and correct to the best of my knowledge, and the opinions expressed are based on my best professional judgment. Executed in Accord with 10 CFR 2.304 (d) and 2.326 (b),

(Electronically signed)

____

Arnold Gundersen, MENE, RO
Fairewinds Associates, Inc
Burlington, Vermont 05408
Tel: (802) 865 9955
Email: arnie@sailchamplain.net
Date: May 31, 2012


I declare under penalty of perjury that the foregoing is true and correct.

Executed this day, May 31, 2012 at Palermo, Italy.

Arnold Gundersen, MNE
Chief Engineer, Fairewinds Associates, Inc

ATTACHMENT 5

MHI Root Cause Analysis and Supplemental Technical Evaluation Report (Selected Excerpts)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

March 6, 2013

Edmund Baumgartner, Esquire
Corporate Counsel
Mitsubishi Nuclear Energy Systems, Inc.
1001 19th Street North Suite 2000
Arlington, VA 22209

SUBJECT: MITSUBISHI HEAVY INDUSTRIES – REQUEST FOR WITHHOLDING ROOT CAUSE ANALYSIS AND SUPPLEMENTAL TECHNICAL EVALUATION REPORT INFORMATION FROM PUBLIC DISCLOSURE

Dear Mr. Baumgartner:

In a February 14, 2013, letter to you, the NRC requested Mitsubishi Heavy Industries (MHI) to provide the MHI document "Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators of San Onofre Nuclear Generating Station," and a redacted version of that document. You provided the requested documents in a letter (ML13057A012) dated February 25, 2013, and requested that certain information contained within the root cause analysis (RCA) and a supplemental technical evaluation report (STER), provided as a supplement to the RCA, be withheld from public disclosure pursuant to 10 CFR 2.390. Redacted versions of the RCA and STER documents were provided as Enclosures 4 and 6 of your letter, respectively (ML13057A013 and ML13057A014).

Mitsubishi Heavy Industries stated in affidavits dated February 22, 2013, that it considered certain information within MHI's RCA and STER to be proprietary and confidential and requested that the information be withheld from public disclosure pursuant to 10 CFR 2.390. A summary of the key points in the affidavits is as follows:

1. The information has been held in confidence by MHI.

2. The information describes unique design, manufacturing, experimental, and investigative information developed by MHI and not used in the exact form by any of MHI's competitors.

3. The information was developed at significant cost to MHI.

4. The RCA is MHI's organizational and programmatic root cause analysis, which is a sensitive, internal document of the type that MHI and others in the industry do not make public, because its purpose is to set forth a critical self-appraisal, with the benefit of hindsight, containing information and analyses that are the result of candid assessments performed by MHI.

5. MHI provided the information to the NRC voluntarily in confidence.

6. The information is not available in public sources and could not be gathered readily from other publicly available information.

7. Disclosure of the information would assist competitors of MHI in their design and manufacture of nuclear plant components without incurring the costs or risks associated with the design and manufacture of the subject component.

We have carefully reviewed your original redacted documents and the information contained in your request. Additionally, we held several discussions with you regarding the redacted information in your documents. Based on these discussions, MHI made some revisions to release additional information. Subsequently, MHI provided final revised versions of Enclosures 4 and 6 via e-mail on February 28 and March 6, 2013, respectively. We have concluded that the submitted information sought to be withheld in the final revised versions contains proprietary and confidential information. Therefore, the final revised versions of the submitted information marked as proprietary will be withheld from public disclosure pursuant to 10 C.F.R. 2.390(a)(4).

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, ensure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You also should understand that the NRC may have cause to review this determination in the future if, for example, the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

/RA/

Ryan E. Lantz, Chief
SONGS Project Branch

Dockets: 50-361, 50-362
Licenses: NPF-10, NPF-15

Enclosures:
MHI's Revised Non-Proprietary RCA and STER

Internal Distribution
A. Howell, TM
J. Andersen, DTM
R. Lantz, C:SPB
G. Werner, I&AL
N. Taylor, SPE
B. Parks, PE
G. George, SRI
J. Reynoso, RI
K. Fuller, RC
V. Dricks, PAO
T. Rothschild, OGC
R. Hall, NRR
D. Broaddus, NRR
E. Roach, NRO
D. Dorman, NRR
D. Merzke, OEDO
J. Weil, OCA
A. Powell, OCA

R:_REACTORS_SONGS\2013\Review of MHI and STER MNES 2-25-13 Letter_Agreement with Revised NonProp Reviews
G:\ORA\SPB\MHI Letter and RCA and STER Enclosures ML13065A097

ADAMS: ☐ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: GEW
	☒ Publicly Available	☒ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

RIV:I&AL:SPB	RC	C:SPB	
GEWerner;dlf	KDFuller	RELantz	
/RA/	***/RA/***	***/RA/***	
3/6/13	3/6/13	3/6/13	

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San Onofre Nuclear Generating Station, Unit 2 & 3
REPLACEMENT STEAM GENERATORS

Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators of San Onofre Nuclear Generating Station

Purchase Order No.	4500024051
Specification No.	SO23-617-01R3

CONTENT		REMARKS	PURCHASER	ORDER No. / ITEM No.	DATE / REFERENCE	QUALITY ASSURANCE DEPARTMENT	
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MITSUBISHI HEAVY INDUSTRIES, LTD.

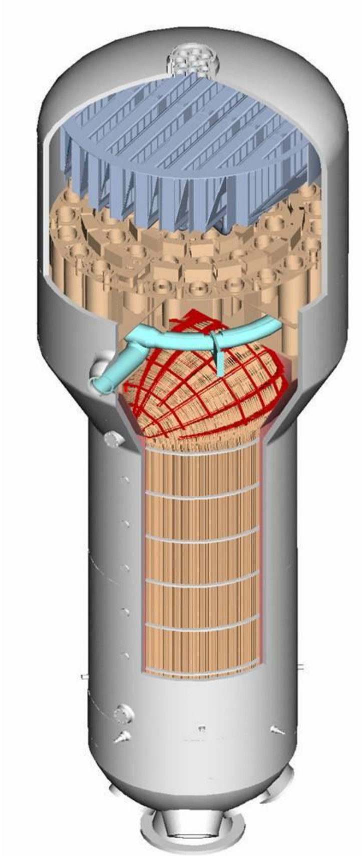
Revision History

Rev.	Summary of Changes	Date issued	Approved By	Checked by	Prepared by
0	Original Issue	See Cover Page	See Cover Page	See Cover Page	See Cover Page

Root Cause Analysis Report for tube wear

identified in the Unit 2 and Unit 3 Steam Generators of

San Onofre Nuclear Generating Station



Change analysis

For the SONGS RSGs, a change analysis was performed in two stages. The first stage compared the SONGS SG design to previous MHI SG designs for the triangular tube configuration. MHI had previously performed three steam generator designs using a triangular tube configuration. The second stage compared the SONGS RSGs to the previous SONGS SG design (Combustion Engineering type design). Only the most significant changes are included in this analysis.

The change analysis results are set out below.

(1) Differences between SONGS RSGs and previous MHI SG triangular design.--

The SONGS RSGs have:

- circulation ratio
- maximum flow velocity
- average flow velocity
- P/D ratio
- out-of-plane FEI stability ratio
- Largest U bundle radius
- Specified AVB twist
- range of G-value (tube diameter, out-of-plane)
- Highest steam quality (void fraction)
- Thinnest and longest retainer bar
- nominal tube-to-AVB gap (0.002” cold / 0.000” hot)
- variation in tube-to-AVB gap (3 sigma)

(2) Differences between SONGS RSGs and the previous SONGS OSG design. --

- Increase in tube bundle heat transfer surface area (11%)
- Increase in number of tubes (5%)

- Removal of stay cylinder
- Change from lattice bars to trefoil broached tube support plates
- Change in tube support configuration in U region
- Change from CE to MHI moisture separators
- Power level / operating temperature / tube plugging margin

(3) Identification of the changes from previous SG designs led to the recognition that the RSG design deserved close scrutiny. MHI considered the changes in the SONGS design from previous steam generator designs and compared the basic design parameters of the SONGs RSGs (e.g., heat transfer area, circulation ratio, steam pressure, etc.) with other steam generator designs. Further, as part of the development of the SONGS RSG design, MHI conducted a detailed comparison between its proposed AVB support for the tubes in the U-bend region and that of a comparison plant of similar design. A special AVB team was formed and included industry experts to conduct an extensive design review process in 2005 / 2006 to optimize the U-bend design and address the technical issues. The team concluded that the SONGS design was significantly more conservative than previous designs in addressing U-bend tube vibration and wear.
Also MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and MHI and SCE agreed not to implement them. It was concluded that the final design was optimal based on the overall RSG design requirements and constraints. These included physical and other constraints on the RSG design in order to assure compliance with the provisions of 10 C.F.R. §50.59. Thus, MHI did compare the SONGs RSG design with previous steam generator designs, and in particular did a detailed evaluation of different options of the AVB design taking into account other large steam generator designs.

ATTACHMENT 6

Southern California Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3), LBP-13-07 (May 13, 2013) ("ASLB Order")

LBP-13-07

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

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

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

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

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

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

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

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

I. FACTUAL AND PROCEDURAL BACKGROUND

A. Factual Background

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

May 8, 2012 Inspection Report at 39.

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

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

Id. (footnote omitted).[13]

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

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

Id. at 9 (footnotes omitted).

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

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

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

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

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

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

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

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

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

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

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

B. Procedural Background

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

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

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to the actual issuance of the order.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

II. ANALYSIS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

constitutes a de facto license amendment proceeding.

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

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

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

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

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

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

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

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

constitutes a de facto license amendment proceeding.

B. Legal Standards That Address License Amendments

1. Relevant Statutory Provisions Related to License Amendments

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

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

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

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

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

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

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

2. Relevant Regulatory Provisions Related to License Amendments

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

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

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

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

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

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

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

Id. § 50.59(c)(2).[38]

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

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

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

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

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

3. *De Facto* License Amendments

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

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

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

10 C.F.R. § 50.2.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Unit 2 Return to Service Report at 38.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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56 <u>See</u> SONGS UFSAR at 5.4-24, section 5.4.2.3.1.3.A.

57 <u>See</u> SONGS Unit 2 Technical Specification, section 5.5.2.11.

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

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

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

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

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

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

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

III. CONCLUSION

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

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

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

It is so ORDERED.

THE ATOMIC SAFETY
AND LICENSING BOARD

/RA/

E. Roy Hawkens, Chairman
ADMINISTRATIVE JUDGE

/RA/

Dr. Anthony J. Baratta
ADMINISTRATIVE JUDGE

/RA/

Dr. Gary S. Arnold
ADMINISTRATIVE JUDGE

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

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

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[Original signed by Herald M. Speiser]
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Dated at Rockville, Maryland
this 13th day of May, 2013