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January 9, 2013 10 CFR 2.206

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

Subject: Docket Nos. 50-361 and 50-362
Response to Friends of the Earth 10 CFR 2.206 Petition
San Onofre Nuclear Generating Station, Units 2 and 3

Dear Sir or Madam:

On June 18, 2012, Friends of the Earth (FOE) submitted a petition to intervene and request for hearing to the Commission claiming, among other things, that Southern California Edison Company (SCE) violated 10 CFR 50.59 when it replaced the San Onofre Nuclear Generating Station (SONGS) steam generators in 2010 and 2011 without a license amendment for certain design changes. In a November 8, 2012 decision (CLI-12-20), the Commission denied FOE's request for hearing regarding the alleged 10 CFR 50.59 violation, and referred that portion of the request to the Executive Director for Operations for consideration as a 10 CFR 2.206 Petition.

SCE provides the attached response to this 10 CFR 2.206 Petition and requests that the Nuclear Regulatory Commission (NRC) deny the Petition in its entirety. The NRC has already evaluated SCE's 10 CFR 50.59 evaluations for the replacement steam generators, and FOE's Petition identifies no significant new information. Additionally, nothing identified by FOE constitutes a violation of 10 CFR 50.59. Therefore, under the NRC's guidelines for petitions under 10 CFR 2.206, the Petition should be denied.

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

Sincerely,

Enclosure: Southern California Edison Company's Response to Friends of the Earth 10 CFR 2.206 Petition

cc: B. J. Benney, NRC SONGS Petition Manager
R. W. Borchardt, NRC Executive Director for Operations
E. E. Collins, Regional Administrator, NRC Region IV
R. Hall, NRC Project Manager, San Onofre Units 2 and 3
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ENCLOSURE

Southern California Edison Company's Response to Friends of the Earth 10 CFR 2.206 Petition

I. PURPOSE

On June 18, 2012, Friends of the Earth (FOE) submitted a petition to intervene and request for hearing (Petition) claiming, among other things, that Southern California Edison Company (SCE) violated 10 CFR 50.59 when it replaced the San Onofre Nuclear Generating Station (SONGS) steam generators (SGs) without a license amendment for certain design changes. FOE's Petition attached a declaration from Arnold Gundersen (Gundersen Declaration), dated May 31, 2012, that addresses in part the 50.59 allegations.

In a decision on November 8, 2012, the Commission decided to treat that claim as a petition under 10 CFR 2.206. SCE is providing the following response to FOE's 2.206 Petition. As demonstrated in this response, the design changes associated with the replacement steam generators (RSGs) did not require a license amendment pursuant to 10 CFR 50.59, and FOE's 2.206 Petition should be denied.

II. BACKGROUND

SONGS Units 2 and 3 are pressurized water reactors using a Combustion Engineering design. SCE procured RSGs for SONGS Units 2 and 3 in the mid-2000s from Mitsubishi Heavy Industries (MHI) and received and installed the RSGs in 2009-2011. The RSGs included a number of design changes relative to SONGS' original steam generators (OSGs).

On June 27, 2008, SCE submitted a license amendment request (LAR) under 10 CFR 50.90 for certain technical specification changes associated with the RSGs.[1] The LAR explained that "[t]he proposed changes reflect revised SG inspection and repair criteria and revised peak containment post-accident pressure resulting from installation of the replacement SGs."[2] The NRC published a *Federal Register* notice on the LAR on September 23, 2008, including a proposed determination that the amendment involves no significant hazards consideration.[3] The NRC approved the LAR on June 25, 2009.[4]

SCE also performed 10 CFR 50.59 evaluations for the design changes associated with the RSGs and concluded that those changes did not require a license amendment. These evaluations consisted of an initial screening of the activities and changes resulting from the RSGs, and then a full 50.59 evaluation was performed for the issues identified by the initial screening. Those issues are documented in SCE's periodic Facility Change Report submitted

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1 Letter from J. Reilly, SCE, to NRC, Amendment Application Numbers 252 and 238, Proposed Change Number NPF-10/15-583, Replacement Steam Generators, San Onofre Nuclear Generating Station, Units 2 and 3 (June 27, 2008), *available at* ADAMS Accession No. ML081830421.

2 *Id.* at 2.

3 Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations, 73 Fed. Reg. 54,862, 54,867-868 (Sept. 23, 2008).

4 Letter from J. Hall, NRC, to R. Ridenoure, SCE, San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendments Re: Technical Specification Changes in Support of Steam Generator Replacement (TAC Nos. MD9160 and MD9161) (June 25, 2009), *available at* ADAMS Accession No. ML091670298.

to the NRC according to 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2).[5] The Facility Change Report describes the 50.59 evaluations as follows:

> This activity replaces the design disclosure documentation and reference documentation for the San Onofre Nuclear Generating Station (SONGS) Unit 2 Original Steam Generators (OSGs) with that for the Replacement Steam Generators (RSGs) and performs functional testing of the RSGs. This replacement and testing is required as a result of physically replacing the OSGs with RSGs. Having the OSGs replaced with the RSGs will improve the efficiency and reliability of Unit 2 by replacing a large number of plugged or otherwise degraded heat transfer tubes in each OSG with new tubes made from thermally-treated Alloy 690, which is less susceptible to degradation than the mill annealed Alloy 600 material used for OSG heat transfer tubing. Replacement of the steam generators is a replacement in-kind in terms of an overall fit, form, and function with no, or minimal, permanent modifications to the plant [Systems, Structures, and Components] (SSCs).[6]

Based on the 50.59 evaluations, the Facility Change Report stated that SCE concluded that the changes could be made without prior NRC approval.[7]

The NRC has performed a number of inspections of the RSG project and the associated 50.59 evaluations. These inspections are documented in Inspection Report 2009-007 for the Unit 2 RSG, Inspection Report 2010-009 for the Unit 3 RSG, Inspection Report 2012-007 for the Augmented Inspection Team (AIT) inspection following the recent steam generator degradation issues, and Inspection Report 2012-010 for an AIT follow-up inspection. None of these inspections has concluded that SCE should have sought license amendments for the design changes in the RSGs.

SCE replaced the Unit 2 steam generators in January 2010 and the Unit 3 steam generators in January 2011. On January 31, 2012, SCE identified a leak in a tube in one of the Unit 3 RSGs. This leak was well below allowable limits in the technical specifications, and presented no hazard to the public health and safety. Pursuant to established procedures, SCE shut down Unit 3. At the time Unit 2 was already shutdown and undergoing a refueling outage.

III. REGULATORY STANDARDS

A. 10 CFR 2.206

10 CFR 2.206 provides the regulatory mechanism for requests for action. Section 2.206(a) states in part that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper."

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5 Letter from R. St. Onge, SCE, to NRC, Facility Change Report, San Onofre Nuclear Generating Station Units 2 and 3 and the Independent Spent Fuel Storage Installation (June 8, 2011), *available at* ADAMS Accession No. ML11161A158.

6 *Id.*, Enclosure 1A, at 4.

7 *Id.*

The NRC's Management Directive (MD) for reviewing 10 CFR 2.206 petitions is provided in MD 8.11, "Review Process for 10 CFR 2.206 Petitions," which was most recently revised on October 25, 2000. MD 8.11 provides Handbook 8.11, which details the procedures for staff review and disposition of 2.206 petitions.

MD 8.11 explains that the NRC will reject a petition under certain circumstances, including:

> The petitioner raises issues that have already been the subject of NRC staff review and evaluation either on that facility, other similar facilities, or on a generic basis, for which a resolution has been achieved, the issues have been resolved, and the resolution is applicable to the facility in question. This would include requests to reconsider or reopen a previous enforcement action (including a decision not to initiate an enforcement action) or a director's decision. These requests will not be treated as a 2.206 petition unless they present significant new information.[8]

MD 8.11 also explains that the NRC, if appropriate, will request the licensee to provide a voluntary response to the NRC on the issues raised in a 2.206 petition, but "[t]he licensee may voluntarily submit information relative to the petition, even if the NRC staff has not requested any such information."[9]

B. 10 CFR 50.59

The applicable provisions in 10 CFR 50.59 state as follows:

> (a) Definitions for the purposes of this section:
>
> (1) *Change* means a modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.
>
> (2) *Departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses* means:
> (i) Changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or
> (ii) Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.
>
> (3) *Facility as described in the final safety analysis report (as updated)* means:
> (i) The structures, systems, and components (SSC) that are described in the final safety analysis report (FSAR) (as updated),
> (ii) The design and performance requirements for such SSCs described in the FSAR (as updated), and

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8 Handbook 8.11, Part III.C.2.b.
9 *Id.* Part IV.A.2.a.

(iii) The evaluations or methods of evaluation included in the FSAR (as updated) for such SSCs which demonstrate that their intended function(s) will be accomplished.

. . .

(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 updated), 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.

(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);
(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.

The NRC has established guidance for implementing 10 CFR 50.59 in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments."[10] In Regulatory Guide 1.187, the NRC staff endorsed Revision 1 of NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," and concluded that NEI 96-07 "provides methods that are

[10] Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (Nov. 2000).

acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59."[11] Therefore, NEI 96-07 provides approved guidance for evaluating whether 10 CFR 50.59 allows changes to the facility or procedures and conduct of tests or experiments without prior NRC approval.

As discussed in Section 1.3 of NEI 96-07, changes are evaluated under 10 CFR 50.59 using a multi-step process.[12] First, a licensee must determine that a proposed change is safe and effective through appropriate engineering and technical evaluations. The 10 CFR 50.59 process is then applied to determine if a license amendment is required prior to implementation. Thus, the 50.59 process is a licensing process to determine whether a change requires a license amendment from the NRC; it is not a process to determine whether the change is safe. Instead, the safety determination is made before the 50.59 process is initiated.

The 50.59 process utilizes three basic steps:[13]

1) **Applicability and Screening:** Determine if a 10 CFR 50.59 evaluation is required.

2) **Evaluation:** Apply the eight evaluation criteria of 10 CFR 50.59(c)(2) to determine if a license amendment must be obtained from the NRC.

3) **Documentation and reporting:** Document and report to the NRC activities implemented under 10 CFR 50.59.

Section 4.1 of NEI 96-07 further addresses the applicability of 10 CFR 50.59. Section 4.1.1 explains that 10 CFR 50.59 is applicable to tests or experiments not described in the Updated Final Safety Analysis Report (UFSAR) and to changes to the facility or procedures as described in the UFSAR, with a few exceptions.[14] One exception is that a change to the technical specifications must be made using the license amendment process in 10 CFR 50.90.[15] Another exception is that changes to the facility or procedures that are controlled by other more specific requirements and criteria established by regulation (*e.g.*, 10 CFR 50.54(q) change requirements for emergency plans) are excluded from the scope of 10 CFR 50.59.[16] If 10 CFR 50.59 does not apply to an issue, then no 50.59 evaluation is required for that issue.

Section 4.2 of NEI 96-07 further explains that, once it has been determined that 10 CFR 50.59 is applicable to a proposed activity, a screening should be performed to determine whether a 50.59 evaluation is required.[17] If the screening determines that an activity is (1) a change to the facility or procedures as described in the UFSAR or (2) a test or experiment not described in the UFSAR, then a 50.59 evaluation is needed.[18] In making the first determination, a design modification may be screened out if it does not affect:

- a design function of an SSC;

- a method of performing or controlling the design function; or

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11 *Id.* at 2; NEI 96-07, Guidelines for 10 CFR 50.59 Evaluations (Rev. 1, Nov. 2000).
12 NEI 96-07 at 4.
13 *Id.*
14 *Id.* at 23.
15 *Id.*; 10 CFR 50.59(c)(1)(i).
16 NEI 96-07 at 23; 10 CFR 50.59(c)(4).
17 NEI 96-07 at 29.
18 *Id.*

- an evaluation that demonstrates that intended design functions will be accomplished.[19]

Additionally, Section 4.2.1 of NEI 96-07 further explains that changes that have neutral or positive effects on these three areas may be screened out, because only adverse changes have the potential to increase the likelihood of malfunctions, increase consequences, create new accidents or otherwise meet the 10 CFR 50.59 evaluation criteria.

If there is a change in the method of evaluation as described in the UFSAR, then a 50.59 evaluation is required. However, a change in the method of evaluation does not require a license amendment if:

- the results of the new analysis are conservative or essentially the same; or
- the new method has been approved by the NRC for the intended application.[20]

IV. EVALUATION OF FOE PETITION

A. Issues Raised by FOE Already Have Been Reviewed and Evaluated by NRC

As noted above, NRC guidance in MD 8.11 states that a 10 CFR 2.206 petition will be rejected if the petition raises issues that already have been evaluated by the NRC, and there is no significant new information.[21] Because the NRC already has reviewed and evaluated the issues raised by FOE that are part of the 10 CFR 2.206 Petition, and FOE has not provided any significant new information, the Petition should be rejected.

The NRC staff already has reviewed whether the RSGs were appropriately evaluated under 10 CFR 50.59. This included reviews that took place at the time of the Units 2 and 3 steam generator replacements. More recently, the NRC staff reviewed the 50.59 evaluations for the RSGs in detail as part of the AIT inspection. The staff's review of the 50.59 evaluations for the RSGs is documented in detail in the July 18, 2012 AIT Report.[22] The AIT Report explains that the NRC staff had conducted inspections regarding the Units 2 and 3 RSGs, which "included a review of selected portions of modifications associated with the replacement steam generators to determine if the changes were done in accordance with 10 CFR 50.59."[23] The AIT Report states that the results of the RSG inspections are documented in Inspection Reports 2009-007 and 2010-009.[24] These inspection reports do not identify any violations related to 10 CFR 50.59 evaluations.[25]

The AIT Report stated that "[t]he team determined that the licensee's evaluation for changes in the updated final safety analysis report's design methodologies for the replacement steam

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19 *Id.*

20 *Id.* at 14-15.

21 Handbook 8.11, Part III.C.2.b.

22 NRC Inspection Report 05000361/2012007 and 05000362/2012007, San Onofre Nuclear Generating Station – NRC Augmented Inspection Team Report (July 18, 2012) (AIT Report), *available at* ADAMS Accession No. ML12188A748.

23 *Id.* at 4-5.

24 *Id.*

25 NRC Inspection Report 05000361/2009007, San Onofre Nuclear Generating Station – Unit 2 Steam Generator Replacement Project (Mar. 4, 2010), *available at* ADAMS Accession No. ML100630838; NRC Inspection Report 05000362/2010009, San Onofre Nuclear Generating Station – Unit 3 Steam Generator Replacement Project (May 10, 2011), *available at* ADAMS Accession No. ML111300448.

generators was consistent with SONGS procedures for implementation of 10 CFR 50.59 requirements."[26] The staff explained that SCE determined as part of its 50.59 screening evaluation that the proposed activity did not adversely affect a design function, or the method of performing or controlling a design function described in the UFSAR; did not change a procedure in a manner that adversely affected how an UFSAR design function is performed or controlled; and did not involve a test or experiment not described in the UFSAR.[27] This included review of the following UFSAR design functions as part of the screening:

- Steam Generator Design Functions
- Reactor Coolant System Structural Integrity
- Emergency Core Cooling System Performance
- Non-Loss of Coolant Accident Transients
- Containment Pressure-Temperature Analysis
- Low Temperature Overpressure Protection
- Reactor Protection System, Engineered Safety Features Actuation System, Core Operating Limit Supervisory System, and Core Protection Calculations
- Nuclear Steam Supply System Performance
- Non-Safety Related Control Systems Performance[28]

As detailed in the AIT Report, the 50.59 screening process identified three methods of analysis described in the UFSAR that were affected by the RSG project and required further evaluation against the 50.59 criteria: (1) Seismic Analysis of Reactor Vessel Internals; (2) Reactor Coolant System Structural Integrity; and (3) Tube Wall Thinning Analysis.[29]

Based on this review, the AIT Report concluded: "The team determined that no significant differences existed in the design requirements of Unit 2 and Unit 3 replacement steam generators. Based on the updated final safety analysis report description of the original steam generators, the team determined that the steam generators major design changes were reviewed in accordance with the 10 CFR 50.59 requirements."[30]

The AIT Report also documented the Office of Nuclear Reactor Regulation (NRR) review of the 50.59 evaluation for the RSGs.[31] This review was detailed, and included the following scope:

> The NRR technical specialist reviewed all of the design changes associated with the replacement steam generators to determine whether the changes to the facility or procedures, as described in the updated final safety analysis report, had been reviewed and

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26 AIT Report at 33-34.
27 *Id.* at 34.
28 *Id.*
29 *Id.* at 34-35.
30 *Id.* at 36.
31 *Id.* at 63-65.

> documented in accordance with 10 CFR 50.59 requirements. The technical specialist reviewed the various information used by SCE to review the changes being made to the replacement steam generators, including calculations, analyses, design change documentation, procedures, the updated final safety analysis report, the technical specifications, and plant drawings. The evaluation process used by the technical specialist included determining if the design changes to the replacement steam generators were a change to the facility or procedures as described in the updated final safety analysis report or a test or experiment not described in the updated final safety analysis report. The technical specialist also verified that safety issues related to the changes were resolved. The technical specialist compared the safety evaluations and supporting documents to the guidance and methods provided in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the 10 CFR 50.59 evaluations.[32]

This review resulted in one unresolved item (URI 05000362/2012007-10), "Evaluation of Departure of Method of Evaluation for 10 CFR 50.59 Processes."[33] This unresolved item, however, was closed during a November 9, 2012 AIT follow-up report.[34] Although the AIT follow-up report identified a minor violation regarding 10 CFR 50.59(d)(1), the NRC concluded that the issues related to the unresolved item did not require a license amendment.[35]

The AIT members reviewed a wide variety of documents related to the RSGs as part of its inspection, including calculations, design basis documents, design change notifications/supplier deviation requests, drawings, engineering reports, modifications, nuclear notifications, procedures, vendor documents, and other miscellaneous documents.[36] Among these, the AIT members specifically reviewed SCE's 50.59 screening and evaluation for the RSGs.[37]

In summary, the NRC performed a detailed review of the 50.59 process utilized for the SONGS RSGs, and did not identify any design changes that required a license amendment (other than the LAR approved by the NRC on June 25, 2009). The NRC's review was very broad and encompassed all of the design changes associated with the RSGs. The AIT Report concluded: "Based on the updated final safety analysis report description of the original steam generators, the steam generators major design changes were appropriately reviewed in accordance with the 10 CFR 50.59."[38] Therefore, the NRC's evaluation of the 50.59 process for the RSGs encompasses all of the issues raised by FOE in the 2.206 Petition. FOE does not identify any significant new information. Consistent with MD 8.11, the Petition should be rejected because it raises issues that already have been evaluated by the NRC.

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32 *Id.* at 63-64.

33 *Id.* at 65.

34 NRC Inspection Report 05000361/2012010 and 05000362/2012010, San Onofre Nuclear Generating Station – NRC Augmented Inspection Team Follow-up Report (Nov. 9, 2012), *available at* ADAMS Accession No. ML12318A342.

35 *Id.* at 22-26.

36 AIT Report, Attachment 1.

37 *Id.*, Attachment 1, at 14.

38 *Id.* at ii.

B. Issues Raised by FOE Do Not Require a License Amendment Per 10 CFR 50.59

Even had the NRC not already reviewed the 50.59 process utilized by SCE for the SONGS RSGs, the issues raised by FOE do not require a license amendment under 10 CFR 50.59. These include allegations regarding design changes, changes in computer codes, and nonconformances in the RSGs. These categories are each discussed below.

1. Design Changes

FOE argues that SCE violated 10 CFR 50.59 because it made "major changes" in the steam generators and the RSGs "differ significantly" from the OSGs.[39] FOE claims that design changes "such as removal of the stay cylinder, replacement of the egg crate tube support with a broached plate tube support, or the thickening of the tube sheet" are "major design changes" and therefore required a license amendment.[40] That is not the correct standard under 10 CFR 50.59 for evaluating whether a license amendment is needed. The fact that the design is changed, or even that a design change is "major," does not automatically result in the need for a license amendment.[41]

As discussed above, design changes may be screened out under 10 CFR 50.59 if the changes do not adversely affect a design function.[42] As explained in UFSAR Section 5.1, the RSGs have two design functions: (1) Function as a part of the reactor coolant pressure boundary (RCPB) as a barrier to the release of fission products; and (2) Transfer the heat generated in the reactor from the reactor coolant system into the secondary system.

The design changes identified by FOE did not have an adverse effect on the heat transfer function of the RSGs, and FOE does not allege anything to the contrary. Additionally, as shown on the table provided in Appendix 1, the design changes identified by FOE did not have an adverse effect on the RCPB function of the RSGs. Because the design changes identified by FOE did not have an adverse effect on any design function, the changes were properly screened out, and there was no violation of 10 CFR 50.59. The adverse condition that later resulted in a tube leak was a deficiency associated with the design and was not known at the time the 50.59 evaluation was performed.

Mr. Gundersen provided a table that purports to show that certain design changes would increase the probability or consequences of an accident, create the possibility of a new accident, or have other effects.[43] The design changes addressed in the table include removing the stay cylinder, changing the tubesheet, adding tubes, changing the tube support, adding a flow restrictor, additional water volume, and a feedwater distribution ring.[44] Mr. Gundersen,

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39 Petition at 2-3, 17.

40 *Id.* at 2 n.2.

41 FOE further alleges that "[t]he NRC failed to follow its own regulations, in particular 10 C.F.R. § 50.59, which require a formal licensing proceeding be convened and a license amendment granted before changes can be made to the facility that affect the final safety analysis." *Id.* at 18. This too is the wrong standard. A change to the UFSAR or analyses found therein does not automatically require a license amendment. Instead, such changes require a license amendment only if they meet one of the eight criteria in 10 CFR 50.59(c)(2).

42 NEI 96-07 at 29.

43 Gundersen Declaration at 9-10; Petition at 18-19.

44 Gundersen Declaration at 9.

however, provides no explanation or information to support those claims in the table. Therefore, the table does not provide a basis for a 50.59 violation. These design changes are further addressed in the table in Appendix 1.

In some places, Mr. Gundersen appears to be taking the position that *any* change in the UFSAR requires a license amendment under 50.59. That position is inconsistent with 10 CFR 50.59 and the associated NRC-endorsed guidance. As discussed above, a license amendment is needed only if (1) the design change adversely affects a design function; and (2) the impact on the design function satisfies one of the eight criteria in 10 CFR 50.59(c)(2). This is not the case for the design changes associated with the RSGs.

Moreover, the AIT Report evaluated and rejected the very issues raised by FOE regarding design changes. The AIT Report stated:

> With regard to the major design changes between the original and replacement steam generators, the updated final safety analysis report did not specify how the original steam generators relied on special design features such as the stay cylinder, tubesheet, tube support plates, or the shape of the tubes to perform the intended safety functions. . . .
>
> Consistent with [NEI 96-07], SCE's 50.59 screening evaluated the differences in subcomponents between the original steam generators and replacement steam generators as to whether the differences adversely affected the design function (reactor coolant pressure boundary) of the steam generators. The replacement steam generators were designed and fabricated in accordance with quality assurance requirements, and 10 CFR 50.59 does not require the licensee to presume deficiencies in the design or fabrication. . . .
>
> The team determined that no significant differences existed in the design requirements of Unit 2 and Unit 3 replacement steam generators. Based on the updated final safety analysis report description of the original steam generators, the team determined that the steam generators major design changes were reviewed in accordance with the 10 CFR 50.59 requirements.[45]

In summary, SCE's 50.59 activities appropriately evaluated whether any design changes associated with the RSGs required a license amendment. This conclusion is consistent with the AIT Report, and FOE has not demonstrated that any of the design changes (if properly implemented) would have adversely affected a design function.

In addition, the design changes associated with the replacement steam generators are similar to design changes made in replacement steam generators for other plants, and the changes in those plants have not required a license amendment. As stated in a June 11, 2012 letter from NRC Chairman Jaczko to Senator Barbara Boxer:

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[45] AIT Report at 36.

NRC regulations at 10 CFR 50.59 and associated guidance in Regulatory Guide 1.187 include criteria for a licensee to determine when a license amendment is required for proposed changes to a facility. Historically, RSGs have been evaluated against these criteria and no license amendment was required.

2. Changes in Computer Codes

FOE also alleges that "[t]he computer code MHI used for design validation simply was not capable of analyzing the reactor design at San Onofre; rather, the code was qualified only for Westinghouse generators, which are not similar to CE generators."[46]

Changes in computer codes, however, do not require a 50.59 evaluation unless the code is different than that specified in the UFSAR. As stated in 10 CFR 50.59(c)(2)(viii), a change in a code requires a license amendment only if it represents a "departure from a method of evaluation *described in the FSAR (as updated)*" (emphasis added). The codes mentioned by FOE are not described in the UFSAR. Therefore, changes in those codes did not require a 50.59 evaluation, and did not involve any violation of 50.59.

This conclusion is consistent with the NRC staff's evaluation of this issue as described in the AIT Report:

> The team noted that a key methodology for the design of the replacement steam generators was the thermal-hydraulic code used to model the flow conditions in the steam generators. Mitsubishi's FIT-III thermal-hydraulic code was accepted by SCE for the design of the replacement steam generators. The team noted that the updated final safety analysis report did not describe the thermal-hydraulic code used for the design of the original steam generators and therefore the use of the FIT-III thermal-hydraulic code did not constitute a change in methodology or a change in an element of a methodology described in the updated final safety analysis report. The updated final safety analysis report did describe the computer code CRIB as the code used to analyze overall steam generator performance. As described in the updated final safety analysis report, CRIB was used to establish the recirculation ratio and fluid mass inventories as a function of power level in the original steam generators.[47]

As a result of its recent evaluations, SCE has determined that MHI's thermal-hydraulic analysis code did not predict the fluid elastic instability that occurred in the RSGs. That concern, however, was not known during the design and manufacturing of the RSGs. Therefore, those concerns could not have been a basis for a license amendment and do not provide any basis for an allegation that SCE violated 50.59 in 2009-2011.

In summary, SCE's 50.59 activities appropriately evaluated whether any changes in computer codes associated with the RSGs required a license amendment. This conclusion is consistent with the AIT Report, and FOE has not demonstrated that a change in codes represents a departure from a method of evaluation described in the UFSAR.

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46 Petition at 21; Gundersen Declaration at 12.[47] AIT Report at 35-36.

47 AIT Report at 35-36.

3. Nonconformances in the RSGs

FOE further refers to the leak in one of the RSGs and the unexpected tube wear as a basis for its allegation that SCE violated 10 CFR 50.59, claiming that the design changes resulted in risks not considered in the UFSAR.[48] The leak and unexpected tube wear are nonconforming conditions. If the RSGs had been designed and manufactured in accordance with the procurement specification, the leak and tube wear would not have occurred. These nonconformances were not known at the time the 50.59 evaluations were performed. Therefore, the nonconformances do not indicate any violation of 10 CFR 50.59 or any need for a license amendment in 2009-2011.

As explained above, SCE evaluated all of the changes related to the RSGs, and the changes met the screening criteria for not performing a 50.59 evaluation, or a 50.59 evaluation determined that no license amendment was necessary. Later-identified errors in an evaluation or nonconformances do not mean that an earlier 50.59 evaluation, such as SCE's 50.59 analysis for the RSGs, was deficient or that a license amendment should have been obtained. And FOE has not demonstrated otherwise in their 10 CFR 2.206 petition.

This concept is further discussed in Section 4.4 of NEI 96-07. As that section indicates, a 50.59 evaluation is not needed for a nonconforming condition, unless the licensee proposes to change its licensing basis to accept the nonconforming condition. Thus, 10 CFR 50.59 does not apply to nonconforming conditions unless the licensee accepts such conditions; instead, nonconforming conditions are addressed through a licensee's corrective action program pursuant to Appendix B to 10 CFR Part 50.

4. Other Claims by FOE

Many of FOE's allegations pertain to the current safety of the RSGs. Those claims are being dealt with under the subject of NRC staff inspections and evaluations, and do not pertain to whether SCE violated 10 CFR 50.59 in 2009-2011.

V. CONCLUSIONS

The NRC already has evaluated SCE's 10 CFR 50.59 evaluations for the RSGs, and FOE's Petition identifies no significant new information. Additionally, nothing identified by FOE warrants a license amendment under 10 CFR 50.59. Therefore, under the NRC's guidelines for petitions under 10 CFR 2.206, the Petition should be denied.

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[48] Petition at 3, 18, 20.

APPENDIX 1

FOE Allegation	Analysis
The Petition states: "The key fabrication change in the new generators was the decision to add almost 400 tubes to each steam generator, increasing the total number of tubes by more than 4%." Petition at 17. The Gundersen Declaration states that "[t]he 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." Gundersen Declaration at 4-5.	The number of tubes is not specified in the UFSAR, and therefore a change in the number of tubes did not involve a change in the UFSAR. SCE's 50.59 Screens evaluated the addition of tubes to the steam generators and determined that the increase did not adversely affect the function of the tubes as described in the UFSAR, and therefore did not require a license amendment. There are other plants that have more tubes per steam generator than SONGS Units 2 and 3. The SONGS RSGs have 9727 tubes each. In comparison, each of the original and replacement steam generators for one other Combustion Engineering (CE) facility, has more than 11,000 tubes, and each of the replacement steam generators for another CE facility has more than 10,000 tubes.
The Petition refers to "removing the stay cylinder, which functioned as a support pillar to the tubesheet into which the U-tubes are inserted." Petition at 17; 19-20. The Gundersen Declaration states that "[t]he 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." Gundersen Declaration at 5.	The UFSAR did not specify that the OSGs relied on the stay cylinder to perform the intended safety functions. SCE's 50.59 Screens evaluated the removal of the stay cylinder and determined that removal of the stay cylinder in conjunction with other changes did not adversely affect the functions of the steam generators and therefore did not require a license amendment. As part of the analysis for the RSGs, MHI evaluated the stresses in the RSGs and determined that no part was stressed beyond the allowable limits in the ASME Code. SCE has evaluated whether the change from the stay cylinder to a divider plate led to the tube damage. SCE has determined that this modification was not a causal factor in the tube-to-tube wear.

FOE Allegation	Analysis
The Petition refers to "thickening the tubesheet to compensate structurally for the removal of the stay cylinder." Petition at 17. The Gundersen Declaration states that "[b]ecause 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." Gundersen Declaration at 5.	FOE has not alleged that the thicker tubesheet adversely affected any design function of the RSGs. Therefore, this statement does not provide a basis for alleging that SCE violated 10 CFR 50.59. The change in the tubesheet did not require a license amendment. The increase in thickness of the tubesheet did not adversely affect the functions of the steam generators.
The Petition refers to "reducing the volume of water in the steam generator." Petition at 17. The Gundersen Declaration states that "[t]his change in the tubesheet thickness meant yet another design change by reducing the volume of water in the steam generator." Gundersen Declaration at 5-6.	FOE does not allege that the change in volume had any adverse effect on a function of the RSGs or any other system. The primary side volume for each RSG is slightly greater than that of each OSG due to a larger internal volume of the tube bundle. The secondary side masses of the RSGs and OSGs are approximately the same at full-load operating conditions. SCE's 50.59 Screens evaluated the change in the primary side volume and determined that it had an insignificant impact on the safety analyses.
The Petition refers to "changing the flow pattern." Petition at 17. The Gundersen Declaration states that "[t]his change in the tubesheet thickness meant yet another design change by . . . changing the flow pattern." Gundersen Declaration at 5-6.	A change in a flow pattern does not, in and of itself, require a license amendment under 10 CFR 50.59. FOE does not allege that the change in the flow pattern had any impact on a safety function. At the time the RSGs were designed, MHI evaluated the flow patterns and determined that fluid elastic instability (FEI) would not occur. The experience with SONGS Unit 3 indicates that that conclusion was not correct, and that the RSGs do not conform to the procurement specification. However, that was not known at the time, and therefore provides no basis for a claim that SCE violated 10 CFR 50.59.

FOE Allegation	Analysis
The Petition refers to "reducing the inspection access area below the tubesheet." Petition at 17. The Gundersen Declaration states that "[t]his change in the tubesheet thickness meant yet another design change by . . . also reducing the inspection access area beneath the tubesheet that is required to fit personnel and equipment for tube inspection." Gundersen Declaration at 5-6.	This change pertains to access for inspections. This change is not relevant to the design functions of the RSGs to transfer heat and act as part of the RCPB. The RSG design meets the 10 CFR 50 Appendix A General Design Criteria for inspection access.
The Petition states: "These design modifications altered the structural loads on the tubesheet, a critical safety consideration as the tubesheet serves as the key barrier keeping radiation inside the containment." Petition at 17. The Gundersen Declaration states that "[c]hanging 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." Gundersen Declaration at 6.	As part of the analysis for the RSGs, MHI evaluated the stresses in the RSGs and determined that no part was stressed beyond the allowable limits in the ASME Code. Therefore, these modifications did not affect the function of the tubesheet with respect to the RCPB.
The Petition states: "Adding tubes also required increasing the nuclear reactor core flow, on which the original design basis safety calculations for cooling the reactor are based." Petition at 17-18. The Gundersen Declaration states that "[f]abricating more tubes increased nuclear reactor core flow, which was unacceptable because it changed the original design basis safety calculations for cooling the reactor." Gundersen Declaration at 6.	FOE does not allege that this design change adversely affected any design function of the RSGs or any other system. The change in the primary coolant flow rate as a result of the RSGs was minimal; an increase from 198,000 gpm to 209,880 gpm. This is approximately a 5% difference. The difference in heat transfer rate was even less; an increase from 5.819×10^9 Btu/hr to 5.900×10^9 Btu/hr, or approximately a 1% difference. As discussed below, this increase in flow did not adversely affect the safety analysis of other systems and components. SCE's 50.59 Screens evaluated the impact of the RSG changes on the reactor coolant system (RCS). SCE concluded that the functional and performance requirements for the RCS were met and that the RCS will continue to perform its design functions with the RSGs installed.

FOE Allegation	Analysis
The Petition refers to "changes to the tube supports in an attempt to avoid increased vibration in the tubes." Petition at 18. "MHI changed to broached plate tube supports in the replacement steam generator design." Petition at 21. "Mr. Gundersen explains in his Declaration how the flow resistance of the broached plate designed by MHI is much higher than the original CE egg crate design because of the reduced spacing of the tubes in the broached plate." Petition at 21. The Gundersen Declaration states: "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." Gundersen Declaration at 10. The Gundersen Declaration also states that "Mitsubishi fabricated a broached plate design that allows almost no water to reach the top of the steam generator." Gundersen Declaration at 11.	The tube support plates (TSPs) are located in the portion of the steam generators with the straight legs of the steam generator tubes. However, the area of concern with respect to the tube vibration was in the U-bend area of the steam generators. SCE and MHI evaluated the impact of broached plates in their 50.59 reviews and concluded that they did not affect the function of the steam generators and that they provided sufficient margin against vibration. Furthermore, SCE has recently evaluated whether the change in tube support led to the tube damage, and has determined that this modification was not a causal factor in the tube-to-tube wear. Use of TSPs with broached holes has been common since at least the 1970s. Furthermore, the majority of the RSGs in the United States have used broached holes for the TSPs. In contrast, few plants (ten units in total in the United States, including the OSGs for SONGS Units 2 and 3) have used the egg crate design. Therefore, there is nothing improper in using broached hole TSPs.

FOE Allegation	Analysis
The Petition states that "both MHI and SCE missed [that the design change] has resulted in almost no water reaching the top of the steam generator, creating regions where the U-tubes are almost dry. Without liquid in the mixture, there is no damping against vibration, resulting in a severe fluid-elastic instability. A fundamental problem in the steam generator causing the vibration and, consequently, the tube wear is that there is too much steam and too little water at the top of the steam generators in the U-bend region." Petition at 21. The Gundersen Declaration states: "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." Gundersen Declaration at 11. The Gundersen Declaration also states that "[t]he 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." Gundersen Declaration at 11.	SCE and MHI did not "miss" this issue. The tube bundle flow analysis was the subject of special design review meetings between SCE and MHI. MHI provided a thermal-hydraulic analysis as part of the original design of the RSGs that showed that there would be no FEI. In retrospect, SCE and MHI have determined that MHI's thermal-hydraulic code had errors, and that it did not accurately predict the thermal-hydraulic conditions. This was not known in 2009-2011 and therefore does not provide a basis for a claim that SCE violated 10 CFR 50.59 in 2009-2011.
The Gundersen Declaration states: "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." Gundersen Declaration at 6.	FOE does not allege that this change had any adverse impact of the design functions of the RSGs. A flow restrictor was added to the steam outlet from the RSGs to reduce the amount of energy released to the containment during a postulated steam line break. It also reduces the loads on the internals of the steam generators during such an event. These are design improvements and did not require a license amendment. Additionally, a flow restrictor was added to the reactor coolant inlet for the steam generators. Its purpose is to ensure that the maximum allowable reactor coolant flowrate will not be exceeded. As a result, it also did not require a license amendment.

FOE Allegation	Analysis
The Gundersen Declaration states: "The feedwater distribution ring inside the steam generator was also dramatically modified in order to avoid a serious flow induced water hammer." Gundersen Declaration at 6.	FOE does not allege that this change had any adverse impact on the design functions of the RSGs. With respect to water hammer, the RSG feedwater distribution ring has a gooseneck design for preventing a water hammer; whereas, the OSGs did not have the gooseneck. This change represents an improvement in the design. SCE's 50.59 Screens evaluated this change and determined that it did not adversely affect the function of the RSGs and therefore did not require a license amendment.
The Gundersen Declaration states: "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. Without liquid in the mixture, there is no damping against vibration, and therefore a severe fluid-elastic instability developed." Gundersen Declaration at 11.	At the time the RSGs were designed, MHI performed analyses that demonstrated that the steam in any area of the tube bundles would be low enough to provide the required damping, and that the quality of the steam in the vast majority of the secondary side of the steam generators would be even less. Furthermore, MHI analyzed the potential for fluid elastic vibration, and determined that conditions were stable. SCE's root cause evaluation has determined that FEI did occur. However, SCE had no evidence of that beforehand. Thus, FOE's allegation provides no basis for concluding that a license amendment was needed in 2009-2011.