UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	
SOUTHERN CALIFORNIA EDISON COMPANY	Docket Nos. 50-361-CAL & 50-362-CAL
(San Onofre Nuclear Generating Station, Units 2 and 3)	ASLBP No. 13-924-01-CAL-BD01
	January 9, 2013

GUNDERSEN AFFIDAVIT

I, __Arnold Gundersen__, being duly sworn, state:
(Print Name)

1. My name is Arnold Gundersen and I reside at 125 Northshore Drive, Burlington, Vermont.

2. My CV is attached. I have both Bachelor's and Master's degree in nuclear engineering. I was an Atomic Energy Commission Fellow, a Licensed Reactor Operator, and I hold one nuclear plant patent.

3. My pertinent experience related to the Steam Generator matters being considered by this ASLB Proceedings include but are not limited to:

   3.1. As the Senior Vice President of Inspection Services, I was responsible for a group of approximately 200 personnel performing ASME III and ASME XI non-destructive piping inspections at nuclear plants throughout the United States. These personnel used inspection techniques identical to those used on the San Onofre tube inspections.

   3.2. As the Senior Vice President of Engineering Services, I was responsible for the development of the first ever modern steam generator nozzle dams that were sold to approximately 40 nuclear reactors in the US and Asia. Dams of a similar design are in use in San Onofre's Replacement Steam Generators (RSG).

4. Friends of the Earth (FoE) has retained me to provide my expert opinion on several Factual Issues that this Atomic Safety Licensing Board directed FoE to consider.

Issue #1: Does the Final Safety Analysis Report (FSAR) analyze a steam generator (S/G) tube failure event?

5. Yes, the FSAR does address a steam generator tube failure event.

6. One specific example of where steam generator tube integrity is addressed is in the San Onofre Technical Specifications[1] that are part of the FSAR.

   6.1. Specifically, page 505 of the Technical Specifications has as a Limiting Condition of Operation "Steam Generator tube integrity shall be maintained":

   > 5.5.2.11
   > Steam Generator (SG) Program (continued)
   > b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
   > Structural integrity performance criterion: All in- service 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. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials.

   6.2. A second example where steam generator integrity is addressed on page 510 of the San Onofre Technical Specifications that states that the limiting design basis accident is a "double ended rupture of a single tube":

   > The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to …the leakage rate associated with a double-ended rupture of a single tube.

7. Eight replacement steam generator tubes failed their pressure tests in 2012 and more than 1,000 others have been plugged.

8. Therefore, a review of the evidence makes it clear that the San Onofre Replacement Steam Generator tube damage discovered in 2012 was so severe and extensive that both reactors have been operating in violation of their NRC FSAR license design basis as defined in their Technical Specifications.

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[1] http://pbadupws.nrc.gov/docs/ML1125/ML11251A100.pdf

9. The Main Steam Line Break with radiological leakage through the steam generator tubes is one of the bounding conditions in emergency plan evaluation and the extent of steam generator tube failures directly impacts the FSAR analysis.
10. The Replacement Steam Generator (RSG) modifications at San Onofre increased both the likelihood of equipment failure and the radiological consequence of such failure and therefore directly affect the FSAR Current Design Basis.

11. In a Pressurized Water Reactor (PWR), the Containment barrier includes the steam generator tube sheet and the steam generator tubes. Edison modified the San Onofre Units 2 and 3 tube sheets by removing the "stay cylinder" from the original Combustion Engineering design and modified the tubes by adding 377 additional tubes to each RSG. Therefore, by taking this action, Edison chose to modify the San Onofre containment design by installing the radically different Replacement Steam Generators.

12. General Design Criteria 50 of 10 C.F.R. § 50 Appendix A (*Containment design basis*.) states: *"This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators…. (2) the limited experience and experimental data available for defining accident phenomena and containment responses…."*

13. The rapid and extraordinarily severe wear that resulted in the 2012 failures of all of Edison's San Onofre Replacement Steam Generators was the result of Edison's 2005 decision to radically change the RSG design and to claim that the Part 50.59 licensing process did not apply. These unlicensed unapproved design changes to the containment boundary violated General Design Criteria (GDC) 50 and therefore the FSAR must be amended to reflect Edison's significant modifications.

14. General Design Criteria 16 of 10 C.F.R. § 50 Appendix A (Containment design) states: *"Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."*

15. The degraded condition of the tubes in the RSGs at San Onofre make it clear that Edison had violated GDC 16 and that Edison's modifications to the containment boundary must undergo the rigorous review of a formal FSAR license amendment process including the requisite public hearings.

16. In my opinion, San Onofre's RSG modifications violated both GDC 16 and GDC 50 and created an unanalyzed accident the significance of which was not considered in its Final Safety Analysis Report.

17. In order to determine whether the consequences or severity of accidents analyzed in the Final Safety Analysis Report (FSAR) may be affected by any proposed change activity, the NRC regulations require that plant design changes be implemented through the 10 C.F.R. § 50.59 process. This process is used to evaluate whether any changes to plant design or operation require prior NRC approval.

18. The nuclear industry realized that the FSAR itself might lack sufficient details on proposed changes; therefore, the nuclear trade organization Nuclear Energy Institute (NEI) developed a set of specific guidelines for utilities and energy companies to follow in order to account for deficiencies in the each FSAR. The NRC approved the use of the NEI process.

19. One of the cornerstones to the NEI guidelines is determining if the proposed changes might have an adverse impact on plant safety. Adverse safety consequence is the driving factor for requesting NRC approval of a 50.59 change, not merely the "like-for-like" changes claimed by Edison.

20. While the NRC Augmented Inspection Team (AIT) briefly described how Edison addressed its 50.59 requirements, the evidence shows that Edison did not comply with the NEI guidelines for implementing 50.59.

21. Published reports indicate that the strategic decision made by Edison that the 50.59 process would not be applied to the RSGs was made by corporate officials *before* any engineering personnel had actually performed the 50.59 engineering analysis. Consequently, Edison made a management decision to claim that the 50.59 process did not apply and therefore San Onofre was not required to seek NRC approval for the proposed changes at San Onofre Units 2 and 3. The Edison decision to ignore the 50.59 process for San Onofre's steam generators, enabled to avoid modification of its FSAR commitments as well as avoid analysis on steam generator performance and accidents

22. Proper operation of a steam generator is a major safety issue for each PWR. In addition to providing the containment barrier to radioactivity and producing steam, the steam generator has many other important safety functions. Therefore any RSG design changes clearly have potential safety consequences that are acknowledged in the FSAR. Consequently, any design and/or fabrication change made to the steam generator must be thoroughly evaluated for its safety implications.

    22.1. The RSG is the major component in the plant that contributes to safety during transients and accidents.

    22.2. The RSG provides the driving force for natural circulation and it facilitates heat removal from the reactor core during a wide range of loss of coolant accidents.

23. The NRC has acknowledged the fact that Edison employed a new methodology not reviewed or recognized in the FSAR to calculate the heat transfer, velocities, levels and water/steam distribution on the secondary side of both the Unit 2 and Unit 3 Replacement Steam Generators. And to date, the NRC has released no findings regarding the full impact of Edison's unreviewed and undocumented changes to its FSAR as a result of such radical design and fabrication changes to San Onofre's RSGs.

24. The overall performance of the Original Steam Generators was based upon a one dimensional computer code known as CRIB described in the FSAR, while the design and performance of the RSGs was based upon an unreviewed and un-benchmarked three dimensional code known as FIT-III which is not described in the FSAR.

25. Knowing the standards applied and benchmarked for the RSG computer codes CRIB and FIT-III is critical information in the FSAR because the RSG computer code determines the thermal hydraulic performance during normal and accident conditions.

26. The AIT report indicated that the change to the FIT-III evaluation methodology was not discussed as part of Edison's 50.59 screening because the details of thermal hydraulic models used for the design of the OSG were not discussed in the original FSAR.

27. It should have been obvious to Edison that FIT-III has not been benchmarked and had not been previously used in licensing procedures showing that the use of FIT-III might have an adverse effect on the FSAR safety analysis thus necessitating the entire license amendment review and public hearing process.

28. As noted by the AIT, Edison approved the use of FIT-III code even though the code was not benchmarked nor identified as acceptable in the FSAR. Consequently, Edison operated San Onofre without knowing the uncertainties in the Replacement Steam Generators' performance characteristics. Predicted liquid levels, pressure drops, vibrations, and temperatures at both Units 2 and 3 were all subject to unknown uncertainties during both normal and abnormal operations.

29. In my opinion, by approving the use of an un-benchmarked and untested design tool like FIT-III, Edison did not did not meet the requirements expected from a nuclear licensee. Use of an un-benchmarked computer code that is not included in the FSAR protocol demands a formal FSAR license amendment process including the requisite public hearings.

30. The AIT makes no reference to a NRC review or lack of review of the requisite 50.59 screening evaluation or whether the NEI criteria involving safety significance were included in Edison's analysis.

31. Design changes of the magnitude created by Edison to the San Onofre RSGs should have triggered a Request for Additional Information from the NRC. No RAI was issued by the NRC, because Edison never notified the NRC of the significant modifications its San Onofre operating license.

32. The AIT reported that FIT-III predictions differed considerably in comparison to an Electric Power Research Institute developed code named ATHOS. FIT-III predicted lower flow velocities and void fractions that were not conservative compared to ATHOS. The AIT Report neglected an analysis of the root cause of the critical differences between FIT-III and ATHOS, and the negative impact such lax calculational modeling had on the design, fabrication, and successful operation of the San Onofre RSGs. Had Edison sought the required FSAR license amendment, comparisons between FIT-III and ATHOS would have been identified six years ago.

33. The AIT did not address the possibility that the lack of conservatism in FIT-III predictions, in addition to causing tube vibrations, could also result in non-conservative predictions of the behavior of the steam generator pressure vessel and associated main steam piping during accident conditions that are required to be analyzed in the FSAR.

34. The AIT noted that the non-conservatisms in FIT-III are a contributor to the failure by Edison to adequately calculate the San Onofre RSG tube vibrations.

34.1. But equally important, the AIT failed to address that FIT-III could also create non-conservative predictions of the behavior of the steam generator pressure vessel and associated main steam piping during accident conditions that are required to be analyzed in the FSAR.

34.2. Such a conclusion implies that damage to the steam generator pressure vessel itself, and not just the tubes, might have occurred at San Onofre and remains unanalyzed by either Edison or the NRC.

35. The probability of an accident exceeding the plant's Current Design Basis is increased by the radically different Edison Replacement Steam Generators.

35.1. For example, uncertainties in predicting the thermal hydraulic performance of the steam generator nozzle may lead to stratification and early fatigue failures in the steam generator itself or associated main steam piping.

35.2. Hence, the operational risks involved in operating the San Onofre RSGs have created a licensing condition that should have been addressed as part of an FSAR license amendment and hearing process.

36. It is my professional opinion that Edison should have applied for the 50.59 process so that the FSAR license amendment evaluation and public hearings would have occurred six years ago, prior to creating an accident scenario and facing losses that by the end of this process will easily total more than $1 Billion.

37. The seriousness of the licensing and safety impact of the damaged RSGs at San Onofre cannot be overstated or underestimated.

37.1. Any Design Basis Accident (DBA) as defined in the FSAR needs to be accurately modeled in order to protect public health and safety.

37.2. The FSAR's DBA analysis including the extent of tube leakage in the event of a Main Steam Line Break significantly impacts the design and implementation of Emergency Evacuation Plans.

38. In the event of a steam line break accident in the San Onofre Replacement Steam Generators with the degraded condition of the tubes, an accident would have occurred that is more severe than any design basis accident scenario previously analyzed by Edison in the FSAR.

39. Such a DBA steam line break accident would render the San Onofre emergency plan totally inadequate and most likely cause an evacuation of a large portion of Southern California.

40. Edison dramatically increased the radiation risk to the public as a result of San Onofre with Replacement Steam Generators that were extremely flawed beginning with their original design. The fact that 8 tubes failed the pressure tests in Unit 3 indicates that those tubes would have failed during a main steam line break (MSLB).

41. It is uncertain if a reactor operator would have been able to shut the plant down without melting the core. A simultaneous rupture of 8 tubes would have caused a primary to secondary leak of radioactive coolant of about 5000-6000 gallons per minute. This leakage would have begun to drain the nuclear core as well as releasing radioactive primary coolant to the atmosphere.

42. The ability of a reactor operator to control the water level in the affected steam generator with this high leakage rate and keep the nuclear reactor core cooled has never been analyzed or tested. An accident of this magnitude is outside ANY reactor's Current Design Basis (CDB).

43. The evidence presented by Edison and the NRC AIT shows that the real reason San Onofre had to plug 1300 tubes (and not just the eight that failed the pressure test) was that the San Onofre units were operating outside their Current Design Basis as defined in the FSAR and were in an unanalyzed, unlicensed condition.

44. Not only have Edison's modifications to the RSGs increased the severity of an accident, but also the Replacement Steam Generator modifications have increased the likelihood of a main steam line break. Even the NRC's AIT concluded that the probability of a MSLB was double what it had been with the OSG's.

45. In my opinion, thermal stratification and changes in the outlet steam flow from the Replacement Steam Generators would have induced stresses in the main steam piping that would likely increase the probability of a MSLB even beyond the NRC's conclusion.

46. Therefore, both the probability and the consequences of an accident have increased beyond those in the FSAR and the plant's Current Design Basis as a result of Edison's replacement team generator modifications.

47. The evidence clearly shows that Edison has been operating outside the design basis of its Final Safety Analysis

48. The modifications to the Replacement Steam Generators at San Onofre and the fact that eight tubes failed critical pressure tests significantly raises the potential for radiation bypassing the containment during severe accidents such as a main steam line break accident (MSLB), station blackout (SBO) and anticipated transients without scram (ATWS) events. This situation violates General Design Criteria 16 and 50 and thus triggers the commencement of a formal FSAR license amendment process including the requisite public hearings.

ISSUE # 2 Figure 4-3 in the report entitled "Operational Envelope for Large U-bend Steam Generators, SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear" [hereinafter Tube-to-Tube Report] compares the bulk velocity ratio and void fraction ratio to several successfully operating large S/Gs, and it notes that "[a]t 100% power, the thermal-hydraulic conditions in the u-bend region of the SONGS replacement [S/Gs] exceed the past successful operational envelope for U-bend nuclear [S/Gs] based on presently available data." Tube-to- Tube Report at 17. How similar to the SONGS S/Gs are these other S/Gs? Do the other steam generators, for example, use alloy 670 tubes and have similar spacing, similar support structures, etc.?

49. The Combustion Engineering (CE) designed original steam generators (OSG) are not at all similar to the Mitsubishi RSGs, nor are the Mitsubishi RSGs similar to any other steam generators with which Edison is attempting to make a comparison.

    49.1. No other Replacement Steam Generator design in the country has been modified in the extreme manner that those at San Onofre Units 2 and 3 have been altered.

    49.2. Combustion Engineering built the OSG's at San Onofre. Because CE used only two steam generators, these OSG's were very large and had a tight tube pitch. To assure proper water flow the OSG's had egg crate tube support plates with a region at the center with no tubes and no heat load where a "stay cylinder" was located.

    49.3. Mitsubishi Heavy Industry, the fabricator of the Replacement Steam Generators (RSG), is a Westinghouse licensee and is not prepared to manufacture the tight tube pitch and the egg crate tube supports of the San Onofre RSG design.

50. Edison instructed Mitsubishi to replace the OSG egg crate design with broached tubes and to remove the OSG stay cylinder to add additional tubes to an area where there formally was no heat load. Edison also instructed Mitsubishi to add many other modifications to the RSG that are simply too numerous to list in this affidavit.

51. To the best of my knowledge and belief, no other steam generator in the nation is as large as those at San Onofre with broached tube supports, a tight Combustion Engineering tube pitch, and no stay cylinder. Therefore, comparing San Onofre to "several other successfully operating large S/G's" is simply not a valid engineering or scientific comparison.

52. My professional experience shows that the actual root cause of the steam generator tube degradation is the 2005 strategic decision by Edison to remove the stay cylinder, change the tube sheet, change the tube support structures and add an additional 400 tubes in the Replacement Steam Generator design while still claiming that this significant design modification was a "like-for-like" replacement. These changes have created Replacement Steam Generators unlike any other in the nation.

53. Adding almost 400 additional tubes to the central location where the stay cylinder had been previously located increased the heat load where it was already the highest.

54. At the same time, Edison removed the egg crate tube supports and replaced them with broached tube supports that reduced cooling flow.

55. These three changes (additional tubes, removal of stay cylinder and egg crate removal) caused a unique and unanalyzed heat load to the interior of the Replacement Steam Generators that will continue to cause the tubes to vibrate and fail even after some have been plugged.

56. The center section of the original San Onofre steam generators contained a key structural element called a "stay cylinder" and no steam generator tubes. In 2005 or early 2006, Edison made a management decision to eliminate this vital support pillar and add additional tubes in its place.

57. In the original steam generator design, there was no heat input in this central area of the steam generator, because there were no tubes to add the heat. When Edison added almost 400 tubes (4% of the tubes) to the center of the tube bundle in the San Onofre Replacement Steam Generators, Edison effectively increased the power distribution to the center of the steam generator.

58. This radical and unanalyzed design change moved 4% of the heat to the inside of the tube bundle while reducing the heat by 4% to the outside of the tube bundle.

59. Adding this heat to the center of the bundle was then exacerbated by removing the egg crate tube supports and replacing them with a broached tube support plate design that further reduced flow to the center of the steam generator.

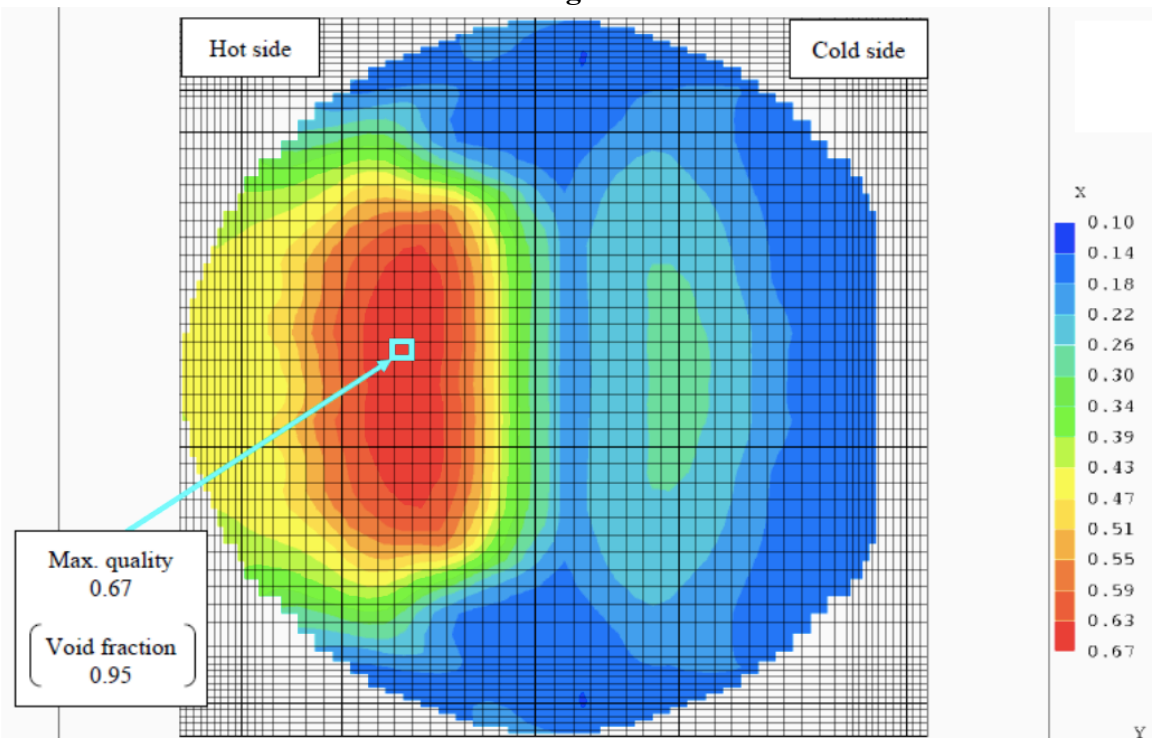
60. As the NRC confirmed in its AIT report, a large steam void has developed near where the additional tubes were added in the Replacement Steam Generators (called fluid elastic instability) that allows many types of excess vibrations to occur.

61. Fairewinds review of Figure 1 below from Edison's Condition Report clearly shows that the location within the steam generators where the steam "fluid elastic instability" has developed is precisely the region where the extra heat created by the 400 new tubes would create an excess of steam and various vibrational modes.

62. While 4% may seem like a small change, it is not. Each San Onofre reactor generates a total thermal output of approximately 3400 megawatts of heat. If one mathematically converts 4% of 3400 megawatts of heat, it equals 135 megawatts, or to illustrate it differently: 180,000 horsepower of thermal heat that was transferred from the outside of the tube bundles to the center.

63. Contour Of Steam Quality[2]

Figure 1



64. This data shows that a significant quantity of additional heat has been transferred to an area that previously had no tubes. That heat must be removed from this central area, yet Edison also reduced flow by replacing the egg crate supports with broached supports.

65. These design changes by Edison created too many steam bubbles that are causing various vibration modes and degradation in all four steam generators.

66. The data reviewed shows that the decision by Edison, to add almost 400 tubes to the center of the four Replacement Steam Generators, changes flow patterns by removing the stay cylinder. This decision by Edison also reduces flow by removing the egg crate tube supports and created the excess heat that is the causative factor in the fluid elastic instability in the Replacement Steam Generators at San Onofre.

67. No other steam generator in the United States was ever modified in a similar fashion and therefore comparisons to other steam generators at other reactors is not relevant or applicable.

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[2] *Condition Report: 201836127*, Revision 0, 5/7/2012, Figure 2: Contour of steam quality at the height of the maximum quality in U-bend region for Thot = 598"F (Figure 8.1-2 (a) in Reference [2]), Page 74.

Issue #3: Figure 5-1 in the Tube-to-Tube Report compares the same parameters as in Figure 4-3, but for operation at 70% power. It appears from Figure 5-1 that the bulk fluid velocity for SONGS is at the high end of the experiential range. Given the likely differences between the SONGS generators and those cited in the discussion, can one conclude that operation at 70% power is conservative?

68. The request by Edison to operate San Onofre Unit 2 at 70% power is not a conservative decision.

69. To focus on Fluid Elastic Instabilities and tube-to-tube interactions is to miss the significant problems with the defective San Onofre Replacement Steam Generators.

70. Fluid Elastic Instability (FEI) causes the tubes to vibrate abruptly at large amplitudes, so it would be imperative that the velocity is maintained below the critical values that create dynamic instabilities. Both the NRC's AIT and Edison's Cause Report neglect the criticality of accurate predictions in the relationship between power and local velocities would be required to restart Unit 2.

71. However, Vortex Induced Vibrations (VIV) and Turbulence Induced Vibration (TIV) might be created if San Onofre Unit 2 were allowed to operate at reduced power, and once again, the NRC and Edison have neglected to review and acknowledge these scenarios.

72. Significant tube damage from fatigue and wear during relatively long periods of operation can cause FEI, VIV, and TIV. Therefore the restart of San Onofre Unit 2 should not be considered because Edison and the NRC reviewed and addressed these issues in their pro-forma reviews.

73. Additionally, properly scaled physical mockups of the San Onofre Replacement Steam Generators, not inadequate computer simulations, are needed and must be required to accurately assess tube wear and vibrational risk created by the possible operation of Unit 2.

74. Computer codes cannot operate and be assessed with out a full-scale mockup prepared by which to provide benchmarks for the computer codes. Once a complete assessment of full-scale mockups is completed, then the computer codes should have the capability to predict local heat transfer rates, pressure drops, void fraction, and velocities.

75. Focusing on measuring and plugging tubes that have become thinner as a result of internal vibrations does not verify San Onofre's RSGs. Edison is attempting to avoid the serious and necessary scientific analysis that would determine which unplugged tubes have become cracked from vibrations and yet are not deemed thin enough to require plugging.

76. Thus, prior to considering the restart of San Onofre Unit 2 at reduced power, Edison and the NRC must also prove to the public that the undetected cracks, which may have been already produced, will not suddenly fail during an unanticipated swing in reactor

conditions (called an operational transient in the nuclear industry) and/or a design basis accident (DBA) that the plant must be built to withstand.

77. Restart of San Onofre Unit 2 should not be considered unless both Edison and the NRC are able to clearly demonstrate that the relationship between plant power and tube vibration is well understood and that FEI, VIV, or TIV will not add to tube wear and create additional safety risks.

Issue #4: Section 8.0 in the Tube-to-Tube Report states that "[t]he desired margin is a projected maximum stability ratio of 0.75 with 0.95 probability at 50% confidence over the next inspection interval of 5 months." Tube-to-Tube Report at 104. Does a confidence level of 50% meet the reasonable assurance requirement in the regulations?

78. In my opinion, a confidence level of 50% does not provide reasonable assurance of anything related to nuclear safety.

Issue #5: Throughout the Tube-to-Tube Report, the term "operational assessment" is used. How is the term "operational assessment" different than or the same as the terms "test" and "experiment" used in 10 C.F.R. § 50.59?

79. Operating the damaged San Onofre Unit 2 at reduced power is an experiment by Edison on steam generators that are unlike any other steam generators that have been designed and fabricated anywhere in the world. The term "operational assessment" is a euphemism employed by Edison to avoid meeting its regulatory requirements.

80. Edison has already acknowledged to the NRC that a research experiment, not an "operational assessment", will be performed and at San Onofre Unit 2 during its proposed five-month period of reduced power operation.

81. Unfortunately, the official transcript of the December 18 meeting between the NRC and Edison is not yet publically available, but Michael Blood of the Associated Press quotes Edison consultant Mike Short as saying research will be performed on tube vibrations when the plant operates at 70% power. Specifically, according to AP: "Short said the data collected by the system could be used in future research examining vibrations picked up by the monitors." [3]

82. I note that this pattern of avoiding the intent of the NRC's regulation by relying on euphemism and carefully parsing words is a persistent mode of operation by Edison dating back to its earliest licensing decision to knowingly avoid the rigorous 50.59 process for the Replacement Steam Generators at San Onofre.

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[3] San Onofre: Edison backpedals on claim that retooling will aid safety, Associated Press, December 18, 2012, http://www.ocregister.com/news/plant-381083-edison-unit.html

List of documents[4] used to conduct my analysis and arrive at my opinions:

1. San Onofre Technical Specifications
   http://pbadupws.nrc.gov/docs/ML1125/ML11251A100.pdf

2. General Design Criteria 50 of 10 C.F.R. § 50 Appendix A

3. General Design Criteria 16 of 10 C.F.R. § 50 Appendix A

4. Nuclear Energy Institute (NEI) 50.59 guidelines
   http://pbadupws.nrc.gov/docs/ML0037/ML003771157.pdf

5. Edison Management Strategic Decision Not To Implement 50.59: Improving Like-For-Like Replacement Steam Generators by Boguslaw Olech of Southern California Edison and Tomouki Inoue of Mitsubishi Heavy Industries, Nuclear Engineering International, January 2012, page 36-38. http://edition.pagesuite-professional.co.uk/launch.aspx?referral=other&pnum=36&refresh=K0s3a21GRq61%20&EID=af75ecb1-5b23-49be-9dd6-d806f2e9b7b5&skip=&p=36

6. NRC SAN ONOFRE REPLACEMENT STEAM GENERATOR AIT REPORT:
   http://pbadupws.nrc.gov/docs/ML1218/ML12188A748.pdf

7. STEAM GENERATOR FAILURES AT SAN ONOFRE: THE NEED FOR A THOROUGH ROOT CAUSE ANALYSIS REQUIRES NO EARLY RESTART, Fairewinds Associates, Monday, Mar 26, 2012:
   http://www.fairewinds.com/content/steam-generator-failures-san-onofre

8. SAN ONOFRE CASCADING STEAM GENERATOR FAILURES CREATED BY EDISON: IMPRUDENT DESIGN AND FABRICATION DECISIONS CAUSED LEAKS, Fairewinds Associates, Monday, Apr 9, 2012
   http://www.fairewinds.com/content/san-onofre-cascading-steam-generator-failures-created-edison

9. SAN ONOFRE'S STEAM GENERATOR FAILURES COULD HAVE BEEN PREVENTED, Fairewinds Associates, Monday, May 14, 2012
   http://www.fairewinds.com/content/san-onofre's-steam-generator-failures-could-have-been-prevented

10. SAN ONOFRE'S STEAM GENERATORS: SIGNIFICANTLY WORSE THAN ALL OTHERS NATIONWIDE, Fairewinds Associates, Tuesday, Jul 10, 2012
    http://www.fairewinds.com/content/san-onofre's-steam-generators-significantly-worse-all-others-nationwide

[4] No documents were provided by Edison, and no documents are covered in any confidentiality agreement between the parties.

11. Steam Generator Steam Quality Graph copied from Edison Condition Report: 201836127, Revision 0, 5/7/2012, Figure 2: Contour of steam quality at the height of the maximum quality in U-bend region for Thot = 598"F (Figure 8.1-2 (a) in Reference [2]), Page 74.

Arnold Gundersen, MSNE, RO
Chief Engineer, Fairewinds Associates, Inc

Arnold Gundersen
(Name)

Subscribed to and sworn before me this 10th day of January, 2013.

Notary Public:

My commission expires: 02/10/15