Joosten, Sandy

From: Vinod Arora [vinnie48in@gmail.com]
Sent: Tuesday, May 14, 2013 3:01 PM
To: CHAIRMAN Resource; CMRSVINICKI Resource; CMRAPOSTOLAKIS Resource; CMRMAGWOOD Resource; CMROSTENDORFF Resource
Subject: San Onofre Sad Saga Continued - NRC/SCE/MHI/SCE Experts/CPUC and Public Awareness Series based on Expert San Onofre Insider Information

Dear Honorable NRC Chairman and NRC Commissioners,

Reference: Nuclear Regulatory Commission [Docket No. 50-361; NRC-2013-00701, Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generating Station, Unit 2]

Yesterday's decision by Atomic Safety and Licensing Board is exactly what the people of Southern California have been calling for the last 15 many months. We want to express our gratitude for this decision, and at the same time, let you know what we expect to happen as this process moves forward in a Democratic Society.
1. No reversal of this decision by NRC Commission. That would be perceived seen as a breach of your oath of office and your charter "protecting people and the environment."
2. For this adjudicated public hearing under oath to happen here in Southern California.

3. Clarity and transparency in this process.

4. Withholding of Useless Proprietary Information is hurting SCE and its Vendors - It makes Public more suspicious of wrongdoing by SCE, its Vendors and NRC.

5. It seems clear that MHI does not have the technology, skills, research, facilities and manpower to rebuild the SONGS Degraded Replacement Steam Generators.

Subject: Review of SONGS 10CFR50.59 and 50.92 Evaluations - SCE Designed and MHI Fabricated 21st Century Safest & Innovative Replacement Steam Generators

Preface: Of particular concern with SONGS Unit 2 restart at reduced power are undetermined and unexamined amount of incubating circumferential cracks located in tubes next to each other caused by fluid-induced random vibrations, high cycle thermal fatigue and in-plane fluid elastic instability. When one circumferentially cracked tube ruptures, the additional stresses can cause multiple or cascading tube ruptures, which can result in a nuclear meltdown. In addition, though the Unit 3 steam generators failed more catastrophically, it appears that there is a much larger pool of tubes out of alignment and in direct contact with support plates in Unit 2. SCE, MHI, AREVA, Intertek, Westinghouse and NRC Staff are ignoring these cracks in their analyses. The difference in management of Steam Generator Tube Rupture between Finland and USA is, that no primary coolant (liquid and steam) release to the environment is allowed in Finland, while in USA, primary steam releases are not forbidden for profits to conduct risky experiments with people's lives. This situation is unique to San Onofre Steam Generator and the Potential Extent of Condition does not affect any other MHI and US Steam Generators.

Conclusions: For SCE to restart "Defectively-Designed and Degraded Unit 2", in accordance with ASLB's decision today, a full 50.90 License Amendment with trial like public hearing is required, because the pending license 50.92 amendment, CAL Actions, SCE's response to NRR RAI's, SCE Unit 2 Return to Service Reports and MHI Root Cause/Technical Evaluations do not fully satisfy the requirement of the Federal Regulations. SCE prepared a defective 50.59 Replacement Steam Generators (RSGs) evaluation and directed MHI not to inform

NRC of the RSGs design deficiencies. NRC region IV and AIT Team did a very poor job of the review of the SCE prepared defective 50.59 evaluation and defended SCE by blaming all the mistakes on the MHI. Now from review of the press reports, one is likely to conclude that NRC Commission and NRR are still leaning towards approving SCE's permission to Restart Unit 2 in violation of the President of The United States, US Congress, Federal Regulations, NRC ASLB Board and against the safety interests of 8.4 Million Southern Californians.

Background: There are hundreds of operating steam generators in the world, which have prevented in-plane fluid elastic instability by keeping the void fractions below 98.5% (Ref. AREVA Operational Assessment data for 5 steam generators, NUREG-1841, NRC Approved Power Uprate Applications, etc.) by operating at steam pressures above 900 psi and steam generator circulation ratios above 4. MHI Root Causes states, "SCE/MHI AVB Design Team recognized that the design for the SONGS RSGs resulted in higher steam quality (void fraction) than previous designs and had considered making changes to the design to reduce the void fraction (e.g., using a larger downcomer, using larger flow slot design for the tube support plates, and even removing a TSP)." So, we assume, that Edison Engineers must have foreseen the impact the problem of high void fractions on increased tube vibrations and refused to make the changes, because it could have impeded the ability to justify the RSG design under the provisions of 10 C.F.R. §50.59, delayed the construction schedule, increased the costs and reduced the profit margins. Increasing the circulation ratios meant reducing the void fractions by increasing the steam pressures, reducing pressure losses, reducing moisture content and less thermal output from the generator. High void fractions cause higher tube vibrations, fluid elastic instability and tube-to-tube wear. MHI/SCE AVB Team missed the boat on Academic Research Papers (2003 through 2006), NUREG-1841 Industry Bench Marking (World's largest CE replacement steam generators installed in 2002 and partly owned by SCE) and ignored the well-established elementary principles of physics, SG tube vibrations, nucleate boiling, heat transfer, void fractions and circulation ratios by refusing to lower the RSG void fractions. The Original Combustion Engineering Steam Generators operated at 900 psi and a void fraction of 96.1%. That is why these steam generators did not suffer fluid elastic instability in 28 years of operation. Increasing the heat transfer area by 11%, addition of 377 new tubes (4% heat transfer area), the average length of heated tubes by 50 inches (Equivalent addition of 650 tubes or 7% heat transfer area), the steam generator thermal output by 24 MWt to make more profits and refusal to reduce the void fractions was a joint decision, which we assume, was known by members of the MHI/SCE AVB Team and SCE Management, which included the Edison Engineers.

Edison Steam Generator Expert stated in an International Engineering News Magazine in 2012, "The contract for design, fabrication and delivery of the RSGs was awarded to Mitsubishi Heavy Industries Ltd. (MHI). As specified, the RSGs were supposed to be a replacement in-kind for the OSGs in terms of form, fit and function. At the same time, however, the RSG specification included many new requirements derived from both industry and SONGS operating experience, and the requirement to use the best and most suitable materials of construction. These requirements were aimed at improving the RSG longevity, reliability, performance and maintainability. Also, the specification called for very tight fabrication tolerances of the components and sub-assemblies, especially the tubesheet and the tube U-bend support structure. In addition, SONGS steam generators are one of the largest in the industry, which called for innovative design solutions and improved fabrication processes when working on the RSGs. Conceivably, the MHI and Edison project teams faced many tough challenges throughout the entire project in the design, manufacturing and QC areas, when striving to meet the specification requirements. Both teams jointly tackled all these challenges in an effective and timely manner. At the end, MHI delivered the RSGs, which incorporated all the latest improvements found throughout the industry, as well as innovative solutions specific to the SONGS RSGs. In Unit 2, the RSGs were installed and tested in 2009/10 and in Unit 3 in 2010/11. The RSG post-installation test results met or exceeded the test acceptance criteria for all specified test parameters, thus properly rewarding the effort put into their fabrication."

A. Review of SONGS Replacement Steam Generators (RSGs) 10CFR50.59 Evaluation

SCE Engineer states in RSGs 10CFR50,59 documents, "Having the OSGs replaced with the RSGs will improve efficiency and reliability of Units 2 & 3 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 or components (SSCs). Each RSG is designed to produce 7.588E6 lbs/hr (vs. 7.414E6 lbs/hr for OSGs) of 833 psi (vs. 900 psi for OSGs) saturated steam with void fraction of > 98.5% (vs. 96.1% for OSGs) moisture content when supplied with feedwater at 442°F.

A.1 - The major physical differences between the RSGs and OSGs are as follows:

1. The RSGs have a greater number of tubes (9,727 vs. 9,350) and a larger heat transfer surface area than the OSGs (116,100 ft2 vs. ~ 105,000 ft2). The average length of the heated RSG tube is approximately 50 inches more than the average length of the heated OSG tube.
2. The RSG reactor coolant volume is greater than the OSG volume (2003 ft3 vs. 1895 ft3).

3.The RSG tube wall thickness is less than the wall thickness of the OSG tubes (0.0429 in. vs. 0.048 in.)- Low Tube In-plane Frequency.

4. The RSG tubes are Alloy 690 (thermally-treated) while the OSG tubes are Alloy 600 (mill-annealed).

5. The RSG feedwater ring is fabricated from erosion-corrosion resistant Cr-Mo alloy steel with Alloy 690 TT fittings, whereas the OSG feedwater ring is made of carbon steel (with the exception of the flow distribution box).

6. All RSG tubes are U-bend shape, whereas the OSG tubes have both U-bend shape (inner rows of the tube bundle) and square-bend shape (outer rows of the tube bundle).

7. The RSG channel head has a flat bottom, thicker divider plate, as compared to the OSGs, and no stay cylinder.

8. The RSG tube supports consist of 7 broached tube support plates in the straight-leg region and anti-vibration bars in the U-bend region, while the OSG tube supports consist of the egg-crate type supports in the straight-leg region and batwings and vertical strips in the U-bend region.

A.2 - Design Function(s) and/or Method(s) of Evaluation: The design functions of steam generators are to:

1. Function as a part of the reactor coolant pressure boundary (RCPB).

2. Transfer heat between the RCS and main steam system.

3. Remove heat from the RCS to achieve and maintain safe shutdown following design basis accidents (except for a large break LOCA) and other UFSAR-described events.

A.3 - The design functions of the steam generator tubes and tube supports are to:

1. Limit tube flow-induced vibration and reactor coolant pump-induced vibration to acceptable levels during normal operating conditions.

2. Withstand blowdown forces from severance of a steam nozzle and ensure that ASME Code allowable stress limits are met.

3. Maintain acceptable ASME Code stress levels under design basis accident conditions (i.e., to prevent a tube rupture concurrent with other accidents, and to prevent multiple tube ruptures during a postulated single steam generator tube rupture event), and

4. Function as a part of the RCPB.

A.4 - State if the proposed activity changes an SSC in a manner that adversely affects the UFSAR/DSAR design function(s) or has an adverse affect on the method of performing or controlling UFSAR/DSAR design function(s).

Yes. After the Unit 3 Leak, it is clear that the RSGs were designed and fabricated poorly compared with the OSGs. RSGs were not OSGs replacement in-kind in terms of design functions. OSGs lasted for 28 years and RSGs were destroyed in less than 2 years. Let us now examine the other differences between Unit 2 and Unit 3's Operational Factors, which were significant contributors to the "fluid-elastic instability" in SONGS Unit 3 and the tube-to-tube wear resulting in the tube leak.

A.4.1 - Adverse Design/Operational Factors responsible for Fluid Elastic Instability: Low steam generator pressures (SONGS RSGs range 800-850 psi, the primary cause of the onset of severe vibrations) caused high dry steam and high fluid velocities conducive for in-plane fluid elastic instability and flow-induced random vibrations, whereby U-tube bundle tubes started vibrating with very large amplitudes in the in-plane directions. Extremely hot and vibrating tubes need a little amount of water (aka damping, 1.5% water, steam-water mixture vapor fraction of 98.5%). When the void fractions exceed 98.5% and are in the range of 99.5-100%, the extremely hot and vibrating tubes cannot dissipate their energy and return to their original in-plane design position. In effect, one unstable tube drives its neighbor to instability through repeated violent and turbulent impact events, which caused tube leakage In SONGS Unit 3, 8 tube failures at MSLB test conditions, unprecedented tube-tube wear and tube-to-AVB/tube support plates wear. So in review, due to narrow tube pitch to tube diameter, low tube frequency, low tube clearances, in certain portions of the RSGs U-tubes bundle, fluid velocities exceeded the critical velocities due to extremely high steam flows (100% SONGS power conditions outside the industry NORM). These high fluid velocities cause U-tubes to vibrate with very large amplitudes in the in-plane direction and literally hit other the tubes with repeated and violent impacts. Due to lower steam operating pressures (required to generate more heat, electricity and profits) and excessive pressure drops due to high flows and velocities, steam saturation temperature dropped. This lowering of steam temperature combined with high heat flux in the hot leg side of the U-tube bundle caused steam dry-outs to form (Vapor fraction >99%) outside the tubes, known as "NO Effective Thin Tube Film Damping." Thin film damping refers to the tendency of the steam inside the generators to create a thin film of water between the RSG tubes and the support structures. That film is enough to help keep the tubes from vibrating with large amplitudes, hitting other tubes violently, and protect the Anti-Vibration Bar support structures and maintain the tube-to-AVB gaps and contact forces. These adverse conditions in SONGS at 70% power operation (RTP) with the present defective design and degraded of RSGs known as fluid elastic instability (Tube-to-Tube Wear, or TTW) can lead to rapid U-tubes failure from fatigue or tube-to-tube wear in Unit 2 due to a main steam line break. In summary, FEI is a phenomenon where due to SONGS RSGs design intended for high steam flows demand caused the tubes to vibrate with increasingly larger amplitudes due to the fluid effective flow velocity exceeding critical velocity for a given tube and its supporting conditions and a given thermal hydraulic environment. This 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. The lack of Nucleate boiling on the tube surface or absence of water is the reason for destabilizing effect on fluid-elastic stability.

A.4.2 - Unit 2 FEI Conflicting Operational Data

NRC AIT Report SG Secondary U2/3 Pressure Range 833 – 942 psi

SCE RCE SG Secondary U2/3 Pressure – 833 psi

SONGS Unit 3 RCE Team Anonymous Member – Unit 2 SG Secondary Pressure 863 - 942 psi

SONGS SG System Description Unit 2 SG Pressure Range 892 – 942 psi

Westinghouse OA SG Secondary U2/3 Pressure ~ 838 psi

SONGS Plant Daily Briefing Unit 3 Electrical Generation – 1186 MWe

SONGS Plant Daily Briefing Unit 2 Electrical Generation – 1183 MWe

A.4.3 -Unit 2 FEI Conclusions

A.4.3.1 - NRC AIT Report – Operational Differences between U2/3 – The result of the independent NRC thermal-hydraulic analysis indicated that differences in the actual operation between units and/or individual steam generators had an insignificant impact on the results and in fact, the team did not identify any changes in steam velocities or void fractions that could attribute to the differences in tube wear between the units or steam generators.

A.4.3.2 – SCE Unit 2 Restart Report Enclosure 2 Conclusions – Because of the similarities in design between the Unit 2 and 3 RSGs, it was concluded that FEI in the in-plane direction was also the cause of the TTW in Unit 2.

A.4.3.3 – SCE U2/3 FEI SONGS RCE Team Member Conclusions – FEI did not occur in Unit 2

A.4.3.4 – Westinghouse OA Conclusions: (a) An evaluation of the tube-to-tube wear reported in two tubes in SG 2E089 showed that, most likely, the wear did not result from in-plane vibration of the tubes since all available eddy current data clearly support the analytical results that in-plane vibration could not have occurred in these tubes, and (b) Operational data – ATHOS Model shows no differences in Units 2 & 3

A.4.3.5 – AREVA OA Conclusions – Based on the extremely comprehensive evaluation of both Units, supplemented by thermal hydraulic and FIV analysis, assuming, a priori, that TTW via in-plane fluid-elastic instability cannot develop in Unit 2 would be inappropriate.

A.4.3.6 – SONGS Insider Investigator Unit 2 FEI Conclusions – Due to higher SG pressure (Range 863 – 942 psi) and lower thermal megawatts compared to Unit 3, FEI did not occur in Unit 2. This is consistent with the position of RCE Team Anonymous Member. NRC AIT Report, SCE, Westinghouse and AREVA conclusions on Unit 2 FEI are inconsistent, confusing and inconclusive.

A.4.4 - Possible RSG Degradation Causes:

1. MHI did not benchmark the computer codes for CE steam generators or used 100% mock up for SONGS High Steam Flows and SCE did not check their work.

2. SONGS Certified Design Specification did not specify the value of FEI or SR and MHI did not design the RSGs for in-plane vibrations.

3. SONGS Certified Design Specification implicitly implied MHI to avoid the NRC License Amendment Process and make the tube bundle as tall as possible to achieve the maximum heat transfer area.

4. SCE or MHI did not review NUREG-1841 to see how Westinghouse and BWI were designing CE Replacement Generators AVBs to avoid excessive tube vibrations and areas with high dry steam.

5. The World's Foremost Renowned Professeur Titulaire, Michel J. Pettigrew, Ecole Polytechnique de Montreal, on the subject of fluid elastic instability and turbulence-induced vibration in 1970's states, "It is concluded that, although there are still areas of uncertainty, most flow-induced vibration problems can be avoided provided that nuclear components are properly analysed at the design stage and that the analyses are supported by adequate testing and development work when required. There has been no case yet where vibration considerations have seriously constrained the designer."

6. SCE/MHI did not review the research papers Published by Pakistani Researchers in 2003 and by Dr.Pettigrew and Dr. Mureithi published in 2006, which states "In nuclear power plant steam generators, U-tubes are very susceptible to undergo fluid elastic instability because of the high velocity of the two-phase mixture flow in the U-tube region and also because of their low natural frequencies in their out of plane modes. In nuclear power plant steam generator design, flat bar supports have been introduced in order to restrain vibrations of the U-tubes in the out of plane direction. Since those supports are not as effective in restraining the in-plane vibrations of the tubes, there is a clear need to verify if fluid elastic instability can occur for a cluster of cylinders preferentially flexible in the flow direction. Almost all the available data about fluid elastic instability of heat exchanger tube bundles concerns tubes that are axisymmetrically flexible. In those cases, the instability is found to be mostly in the direction transverse to the flow. Thus, the direction parallel to the flow has raised less concern in terms of bundle stability."

7. Westinghouse OA ATHOS Analysis shows Unit 2 had 99.6% vapor fraction (FEI) and fluid velocities of 28 feet/sec, but based on results of ECT inspection, Westinghouse concludes that unit 2 did not experience FEI. Westinghouse also states, "Test data shows that the onset of in-plane (IP) vibration requires much higher velocities than the onset of out-of-plane (OP) fluid-elastic excitation. Hence, a tube that may vibrate in-plane (IP) would definitely be unstable OP. A small AVB gap (3 Mil) that would be considered active in the OP mode would also be active in the IP mode because the small gap will prevent significant in-plane motion due to lack of clearance (gap) for the combined OP and IP motions. Thus, a contact force is not required to prevent significant IP motion. Manufacturing Considerations: None were extensively treated in the SCE root cause evaluation."

8. AREVA states, "At 100% power, the thermal-hydraulic conditions in the U-bend region of the SONGS replacement steam generators exceeded the past successful operational envelope for U-bend nuclear steam generators based on presently available data. The primary source of tube-to-AVB contact forces is the restraint provided by the retaining bars and bridges, reacting against the component dimensional dispersion of the tubes and AVBs. Contact forces are available for both cold and hot conditions. Contact forces significantly increase at normal operating temperature and pressure due to diametric expansion of the tubes and thermal growth of the AVBs. After fluid elastic instability develops, the amplitude of in-plane motion continuously increases and the forces needed to prevent in-plane motion at any given AVB location become relatively large. Hence shortly after instability occurs, U-bends begin to swing in Mode 1 and overcome hindrance at any AVB location."

9. Average heated length of the tubes is too much (730 inches in RSGs versus 680 inches in OSGs). Unit 3 has historically produced more power than Unit 2 (1186 MWe vs. 1183 MWe, 1178 MWe vs. 1172). Westinghouse states, "In the U-bend region, the gap velocities are a strong function of power level. The steam flow in the bundle is cumulative and increases as a function of the power level and the bundle height which causes high fluid quality, void fraction, and secondary fluid velocities in the upper bundle."

10. RSGs were operating at a circulation ratio of 3.3. Most of The CE RSGs are running at a circulation ratio of 4.0 or more.

A.4.5 - Defects or Deviations:

The design of San Onofre Replacement Steam generators (RSGs) are identical (Neglecting the impact of Units 3 and Unit 2, Tube-to-AVB contact forces due to manufacturing errors – See Item A.4.6 below). As shown below, SONGS Unit 2 potentially did not suffer in-plane fluid elastic instability due to operation at higher steam pressures and lower RCS flows. SONGS Unit 3 suffered in-plane fluid elastic instability due to operation at lower steam pressures and higher RCS flows. This conclusion is consistent with Westinghouse Operational Assessment, but challenges the SCE, NRC AIT, AREVA and MHI conclusions. NRC AIT Report, SCE, MHI and AREVA conclusions on Unit 3 and Unit 2 FEI are incomplete, inconsistent, confusing and inconclusive and based on faulty computer simulations and hideous testing data (Shielded under the false pretense of Proprietary information). The analysis in these reports does not meet the intent of NRC CAL ACTION 1, which states "Southern California Edison Company (SCE) will determine the causes of the tube-to-tube interactions that resulted in steam generator tube wear in Unit 3, and will implement actions to prevent loss of integrity due to these causes in the Unit 2 steam generator tubes. SCE will establish a protocol of inspections and/or operational limits for Unit 2, including plans for a mid-cycle shutdown for further inspections."

Repeated requests to NRC AIT Leader, NRC SONGS Special Panel and NRC Region IV Allegation Coordinator to examine carefully the operational difference between Units 2 & 3 and determine its impact on the tube-to-tube interactions that resulted in steam generator tube wear in Unit 3, and actions to prevent loss of integrity due to these causes in the Unit 2 steam generator tubes have not been addressed to date. NRR has not asked SCE in its RAI(s) the impact of operational differences between Units 2 and 3 on Unit 2 and Unit 3 tube-to-tube wear. Honorable NRC Commissioner Mr. Apostolakis was totally confused on Unit 2 FEI inconsistent statements by SCE, Westinghouse and AREVA. The Author tried to tell this information to SCE and MHI Management in June 2012, but of no avail (See copy of attached Emails and SG Nuclear Notifications).

A.4.6 - Contact Force Differences between SONGS Units 2 and 3:

NRC AIT, SCE and MHI state that supports were better in Unit 2, so no tube-to-tube wear occurred in Unit 2. Fabrication differences during manufacture of SONGS RSGs caused difference of contact forces in supports between Units 2 & 3. Let us now examine that whether insufficient contact tube-to AVB forces in the Unit 3 upper tube bundle caused "fluid-elastic instability" which was a significant contributor to the tube-to-tube wear resulting in the tube leak.

A.4.6.1 - MHI states, "By design, U-bend support in the in-plane direction was not provided for the SONGS SG's". In the design stage, MHI considered that the tube U-bend support in the out-of-plane direction designed for "zero" tube-to-AVB gap in hot condition was sufficient to prevent the tube from becoming fluid-elastic unstable during operation based on the MHI experiences and contemporary practice. MHI postulated that a "zero" gap in the hot condition does not necessarily ensure that the support is active and that contact force between the tube and the AVB is required for the support to be considered active. The most likely cause of the observed tube-to-tube wear is multiple consecutive AVB supports becoming inactive during operation. This is attributed to redistribution of the tube-to-AVB-gaps under the fluid hydrodynamic pressure exerted on the tubes during operation. This phenomenon is called by MHI, "tube bundle flowering" and is postulated to result in a spreading of the tube U-bends in the out-of-plane direction to varying degrees based on their location in the tube bundle (the hydrodynamic pressure varies within the U bend). This tube U-bend spreading causes an increase of the tube-to-AVB gap sizes and decrease of tube-to-AVB contact forces rendering the AVB supports inactive and potentially significantly contributing to tube FEI. Observations Common to BOTH Unit-2 and Unit-3: The AVBs, end caps, and retainer bars were manufactured according to the design. It was confirmed that there were no significant gaps between the AVBs and tubes, which might have contributed to excessive tube vibration because the AVBs appear to be virtually in contact with tubes. MHI states, "The higher than typical void fraction is a result of a very large and tightly packed tube bundle, particularly in the U-bend, with high heat flux in the hot leg side. Because this high void fraction is a potentially major cause of the

tube FEI, and consequently unexpected tube wear (as it affects both the flow velocity and the damping factors)."

A.4.6.2 – AREVA states – "The primary source of tube-to-AVB contact forces is the restraint provided by the retaining bars and bridges, reacting against the component dimensional dispersion of the tubes and AVBs. Contact forces are available for both cold and hot conditions. Contact forces significantly increase at normal operating temperature and pressure due to diametric expansion of the tubes and thermal growth of the AVBs. After fluid elastic instability develops, the amplitude of in-plane motion continuously increases and the forces needed to prevent in-plane motion at any given AVB location become relatively large. Hence shortly after instability occurs, U-bends begin to swing in Mode 1 and overcome hindrance at any AVB location."

A.4.6.3 – Westinghouse states, "Test data shows that the onset of in-plane (IP) vibration requires much higher velocities than the onset of out-of-plane (OP) fluid-elastic excitation. Hence, a tube that may vibrate in-plane (IP) would definitely be unstable OP. A small AVB gap that would be considered active in the OP mode would also be active in the IP mode because the small gap will prevent significant in-plane motion due to lack of clearance (gap) for the combined OP and IP motions. Thus, a contact force is not required to prevent significant IP motion. Manufacturing Considerations: There are several potential manufacturing considerations associated with review of the design drawings based on Westinghouse experience. The first two are related to increased proximity potential that is likely associated with the ECT evidence for proximity. Two others are associated with the AVB configuration and the additional orthogonal support structure that can interact with the first two during manufacturing. Another relates to AVB fabrication tolerances. These potential issues include: (1) The smaller nominal in-plane spacing between large radius U-bend tubes than comparable Westinghouse experience, (2) The much larger relative shrinkage of two sides (cold leg and hot leg) of each tube that can occur within the tubesheet drilling tolerances. Differences in axial shrinkage of tube legs can change the shape of the U-bends and reduce in-plane clearances between tubes from what was installed prior to hydraulic expansion, (3) The potential for the ends of the lateral sets of AVBs (designated as side narrow and side wide on the Design Anti-Vibration Bar Assembly Drawing that are attached to the AVB support structure on the sides of the tube bundle to become displaced from their intended positions during lower shell assembly rotation, (4) The potential for the 13 orthogonal bridge structure segments that are welded to the ends of AVB end cap extensions to produce reactions inside the bundle due to weld shrinkage and added weight during bundle rotation, and (5) Control of AVB fabrication tolerances sufficient to avoid undesirable interactions within the bundle. If AVBs are not flat with no twist in the unrestrained state they can tend to spread tube columns and introduce unexpected gaps greater than nominal inside the bundle away from the fixed weld spacing. The weight of the additional support structure after installation could accentuate any of the above potential issues. There is insufficient evidence to conclude that any of the listed potential issues are directly responsible for the unexpected tube wear, but these issues could all lead to unexpected tube/AVB fit-up conditions that would support the amplitude limited fluid-elastic vibration mechanism. None were extensively treated in the SCE root cause evaluation."

A.4.6.4 – John Large States, "Causes of Tube and Restraint Component Motion and Wear: My study of the various OAs leads me to the following findings and opinion that; (i) degradation of the tube restraint localities (RBs, AVBs and TSPs) occurs in the absence of fluid elastic instability (FEI) activity; (ii) TTW, acknowledged to arise from in-plane FEI activity, generally occurs where the AVB restraint has deteriorated at one or more localities along the length of individual tubes; (iii) the number of tube wear sites or incidences for AVB/TSP locations outstrips the TTW wear site incidences in the tube free-span locations. I find that the 'zero-gap' AVB assembly, which features strongly in the onset of TTW, is clearly designed to cope only with out-of-plane tube motion since there is little designed-in resistance to movement in the in-plane direction – because of this, it is just chance (a combination of manufacturing variations, expansion and pressurization, etc) that determines the in-plane effectiveness of the AVB; (iv) Uniquely, the SONGS RSG fluid regimes are characterized by in-plane

activity, which is quite contrary to experience of other SGs used in similar nuclear power plants in which out-of-plane fluid phenomena dominate. Moreover, from the remote probe inspections when the replacement steam generator (RSG) is cold and unpressurized, I consider it impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state., and (5) v) The combination of the omission of the in-plane AVB restraints, the unique in-plane activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed. John Large continues, "Phasing of AVB-TSP Wear -v- TTW: I reason that, overall, the tube wear process comprises two distinct phases: First, the AVB (and TSP) -to-tube contact points wear with the result that whatever level of effectiveness is in play declines. Then, with the U-bend free-span sections increased by loss of intermediate AVB restraint(s), the individual tubes in the U-bend region are rendered very susceptible to FEI induced motion and TTW. Whereas the OAs commissioned by SCE broadly agree that the wear mechanics comprises two phases, there are strong differences over the cause of the first phase comprising in-plane AVB wear: AREVA claim this is caused by in-plane FEI whereas, the contrary, Mitsubishi (and Westinghouse) favor random perturbations in the fluid flow regime to be the tube motion excitation cause. Put simply: (i) if AREVA is correct then reducing the reactor power to 70% will eliminate FEI, AVB effectiveness will cease to decline further and TTW will be arrested; however, to the contrary, (ii) if Mitsubishi is right then, even at the 70% power level, the AVB restraint effectiveness will continue to decline thereby freeing up longer free-span tube sections that are more susceptible to TTW; or that (iii) the assertion of neither party is wholly or partly correct. As I have previously stated, I consider that AVB-to-tube wear is not wholly dependent upon FEI activity.

A.4.6.5 - Violette R., Pettigrew M. J. & Mureithi N. W. state (Ref. 1 – See below), "In nuclear power plant steam generators, U-tubes are very susceptible to undergo fluid elastic instability because of the high velocity of the two-phase mixture flow in the U-tube region and also because of their low natural frequencies in their out of plane modes. In nuclear power plant steam generator design, flat bar supports have been introduced in order to restrain vibrations of the U-tubes in the out of plane direction. Since those supports are not as effective in restraining the in-plane vibrations of the tubes, there is a clear need to verify if fluid elastic instability can occur for a cluster of cylinders preferentially flexible in the flow direction. Almost all the available data about fluid elastic instability of heat exchanger tube bundles concerns tubes that are axisymmetrically flexible. In those cases, the instability is found to be mostly in the direction transverse to the flow. Thus, the direction parallel to the flow has raised less concern in terms of bundle stability." Reference 1: Fluid-elastic instability of an array of tubes preferentially flexible in the flow direction subjected to two-phase cross flow, Violette R., Pettigrew M. J. & Mureithi N. W., 2006, http://yakari.polytechnique.fr/people/revio/masters_research_subject.html

A.4.6.6 - Dr. Pettigrew (Presentation to NRC Commission, February 2013): So, you notice the U-bend — the plane of the U-bend is being installed, and on top of the U-bends are bars. They are anti-vibration bars. And so you can see here that from the point of view of out-of-plane motion, the tubes are really very well supported because you have a large number of bars all around; but from the point of view of in-plane motion, there's really no positive restraint here to prevent the tube to move in the in-plane direction. Essentially, it relies on friction forces to limit the vibration.

A.4.6.7 – Contact Force Definition: Contact force is the force in which an object comes in contact with another object. Some everyday examples where contact forces are at work are pushing a car up a hill, kicking a ball, or pushing a desk across a room. In the first and third cases the force is continuously applied, while in the second case the force is delivered in a short impulse. The most common instances of contact force include friction, normal force, and tension. Contact force may also be described as the push experienced when two objects are pressed together. The MHI-designed AVBs had zero contact forces in Unit 3 to prevent in-plane fluid elastic

instability and subsequently, wear occurred under localized thermal-hydraulic conditions of high steam quality (void fraction) and high flow velocity. Large u-bends were moving with large amplitudes in the in-plane direction without any contact forces imposed by the out-of-plane restraints. The in-plane vibration associated with the wear observed in the Unit 3 RSGs occurred because all of the out-of-plane AVB supports were inactive by design in the in-plane direction. The Unit 3 tube-to-AVB contact force for the tubes with tube-to-tube wear (TTW) was zero. That is why they did not restrain the tubes in the in-plane direction (like a sports car moving with very high speed in freeway express lanes passing by a stalled police car with empty guns and disabled communication systems).

A.4.6.8 – Contact Force Conclusions: SONGS Unit 3 RSG's were operating outside SONGS Technical Specification Limits for Reactor Thermal Power and Current Licensing Basis for Design Basis Accident Conditions. I agree with MHI that high steam flows and cross-flow velocities combined with narrow tube pitch-to-diameter ratio caused elastic deformation of the U-tube bundle from the beginning of the Unit 3 cycle, which initiated the process of tube-to-AVB wear and insufficient contact forces between tubes and AVBs. Tube bundle distortion is considered a major contributing cause to the mechanism of tube-to-tube/AVB/TSP wear seen in the Unit 3 SG's. After 11 months of wear, contact forces were virtually eliminated between the tube and AVBs in the areas of highest area of Unit 3 wear as confirmed by ECT and visual inspections. I conclude that FEI and MHI Flowering effect redistributed the tube-to-AVB gaps in Unit 3 RSG's. FEI did not occur in Unit 2, because of the absence of high steam dryness and NOT the better supports and/or differences in fabrication, which resulted in substantially increased contact forces (reduced looseness) between tubes and AVBs for Unit 2 and prevented FEI from occurring. My findings on Unit 2 FEI are consistent with the findings of AREVA, Westinghouse, John Large, SONGS RCE Anonymous Root Cause Team Member and latest research performed by Eminent Professor Michel Pettigrew and others in 2006. In-plane fluid elastic instability did not happen in Unit 2 because of operational differences, so therefore double contact forces and better supports is just conjecture in Unit 2 to justify the restart of an Unsafe Unit 2.

A.4.7 – Dings and Dents Conclusions: A analysis performed by AREVA shows that there are more dents and dings in SG 2E-089 (Unit 2) compared to SG 3E-089 (Unit 3) by a factor of about 13. Overall, analysis found that nearly 12,000 contact indications were found in both Unit 2 steam generators as opposed to just under 4,100 contact indications in both Unit 3 steam generators. Even more alarming is that fact that these indications in Unit 2 were primarily found distributed very distinctly across entire rows of steam generator tubes, much more so than Unit 3. This testing is performed by measuring signals between supports and tubes inside of the steam generators. When they are in contact together a signal will be registered and based on the strength of the contact one can correlate the size and impact of the indications on the tubes. What these results infer is that there is a large discrepancy between the amount of tubes out of place and touching the supports in the Unit 2 and Unit 3 steam generators. Considering the fact that Southern California Edison has repeatedly stated that steam generators are of like design and that no evidence or data has been provided which showed any design deviation in this regard between the two units, it is likely that this accelerated wear seen in Unit 2 occurred within the last cycle of operation. Simply this means that for every one indication found in Unit 3 steam generators, three indications were found in Unit 2 steam generators. Though the Unit 3 steam generators failed more catastrophically, it appears from this analysis that there is a much larger pool of tubes out of alignment and in direct contact with support plates in Unit 2. During any operation, it is presumed that there will be some vibration and movement of all of the tubes in steam generators, but this is offset by supports and spacing between tubes. However in this case, nearly 12,000 tubes in Unit 2 are already in contact with supports, meaning that with any vibration or movement more contact and ergo contact indications will occur in the tubes regardless of operational power rates.

A.4.8 – RSG 10CFR50.59 Conclusions: The values of the RSG major design parameters are different than the values of the corresponding OSG parameters. The RSG steam flow is slightly higher, the outlet steam pressure

is lower and the moisture content is considerably lower than the values for the OSGs. These changes are in a non-conservative direction (increased void fractions) and constitute a significant reduction in margin of safety and increase in probability of cascading tube ruptures over the OSGs.

The RSG heat transfer area is larger than the OSG area (116,100 ft2 vs. ~105,000 ft2) and the RSG tube bundle is taller than the OSG bundle. The larger and taller RSG tube bundle along with unauthorized and untested design changes provided the mechanism for increased void fractions, fluid velocities, fluid elastic instability, flow-induced random vibrations, high cycle thermal fatigue and Mitsubishi Flowering Effect. These factors indicate that the RSGs performed worse than, the OSGs during the events that credit natural circulation. The RSG primary side volume is larger than the OSG volume (2003 ft3 vs. 1895 ft3). Due to this increase, more radioactivity will be released to the environment during multiple tube ruptures caused by anticipated operational occurrences and main steam line break. The RCS volume increase will also result in a slight increase of the containment flooding level, following a LOCA.

The RSG tube wall is thinner than the OSG tube wall (0.0429 in. vs. 0.048 in.). The analysis concluded that a tube would have to be plugged if it contained a flaw to a depth lesser than that for the OSGs (35% vs. 44%). This reduction of the tube plugging limit is non-conservative because hundreds of SONGS Unit 2 & 3 RSGs exceeded this limit and were operating beyond their license.

Based on the above, it is concluded that, the proposed activity significantly and adversely affects the steam generator ability to:

(1)Function as a part of the RCPB
(2)Transfer heat between RCS and main steam system and

(3)Remove heat from the RCS to achieve and maintain safe shutdown following postulated accidents (other than the large break LOCA).

Therefore, it is concluded, that the replacement adversely affected the ability of performing or controlling these design functions. Based on the above, changing the OSGs to RSGs changed an SSC in a manner that adversely affected UFSAR-described design functions or that had an adverse effect on the method of performing or controlling UFSAR-described design functions.

B. SONGS Replacement Steam Generators 10CFR50.92 Evaluation

B.1 - Condition of Unit 2 steam Generators:

SONGS Unit 2 & 3 RSGs are of the same design. Therefore, the description of unit 3 provided below is also applicable to Unit 2. SONGS Unit 3 RSGs' unprecedented tube failure and massive tube and AVB/TSP degradation occurred due to fluid elastic instability, flow-induced random vibrations, Mitsubishi Flowering Effect and high cyclic fatigue under the following unique circumstances:

(1) U-tube bundle areas with high dry steam will experienced double in-plane velocities (> 50 feet/sec, based on review of MHI Root Cause, Dr. Pettigrew and other research papers published between 2006-2011) compared with out-of plane velocities assumed (25 feet/sec) to have been predicted by Outdated Out-of-Plane Westinghouse /NRC /MHI /AREVA ATHOS Computer Models,
(2) Lack of positive in-plane restraints and zero damping,

(3) Large number of SONGS Units 2/3 RSG U-bends with tube clearances of only 0.05 inches (Design 0.25 inches, Industry Norm > 0.25 inches),

(4) Excessive number of tubes with narrow tube pitch to tube diameter,

(5) Low in-plane frequency tubes and retainer bars compared with MHI SGs' higher in-plane frequency tubes and retainer bars,

(6) SONGS' tubes being much longer than Westinghouse Model 51 steam generators (Average length of heated tube = 730 inches) and other MHI SGs,

(7) MHI RSGs' unique floating tube bundle with degraded Retainer Bars can collapse due to 100% tube uncovery for 10 minutes under MSLB SG Depressurization, Multiple SGTR SG over-pressurization and lifting of SG Relief Valves, Combination of MSLB and SGTR Conditions, Release of 100% RCS Iodine to Environment,

(8) Large amount of uncertainties and unverified assumptions in MHI, AREVA, Westinghouse and Intertek's contact force (zero for in-plane vibrations), wear rate and tube stress calculations (4.6 ksi versus 16-17 ksi) and computer modeling, and,

(9) Incomplete tube inspections in SONGS Unit 2. Incubating macroscopic circumferential cracks caused by fluid elastic instability, flow-induced random vibrations and high cycle thermal fatigue are extremely difficult to detect and be accurately sized by nondestructive evaluation techniques including X-ray, ultrasonic, and eddy current based bobbin coil probes, mechanically rotating pancake coil (RPC), etc., which have been used in 170,000 SONGS Tube inspections. State-of-the-art systems: Zetec MIZ-80 iD system, Tecnatom TEDDY+, Circular TE and TM, transmit-receive eddy current array probe C-3 and other specialized radiographic probes capable of detecting sub-surface cracks caused by high cycle thermal fatigue have not been used in the 170,000 SONGS Tube Partial and Limited Inspections as shown below for Unit 2 due to access problems in the most problematic innermost sections of the U-Tube Bundle, the high cost, lack of availability of highly specialized tools and contractors, radiation doses, and time considerations in a rush to start Unit 2. The inspection scope defectively designed and degraded SONGS Unit 2 RSGs should have covered 100% hot leg and cold leg tube inspections, 100% of dents or dings, 100% of tube inspections in the tight radius U-bends, 100% area of the Top of the Tube Sheet and Tube Support Plates.

B.2 - SONGS and Offsite Emergency Plans

Current SONGS Updated FSAR, Emergency Plans, San Diego County Multi-hazard Regional Emergency Operations Plans, IPC/Orange County & Other Offsite/State of CA Plans and NRC Emergency Rules/Guidance, SONGS Drills and Exercises are based on a slow occurring Steam Generator Tube Leakage/Rupture caused by anticipated operational transients, which are significantly flawed based on the SONGS Unit 2 realistic scenario described below.

B.3 - Main Steam Line Break In Unit 2:

A potential main steam line break occurs outside Containment in SONGS Unit 2 operating at 70% power. This event causes a simultaneous reactor, turbine, feedwater and reactor coolant trips and MSIVs Close (Conservative assumption for the benefit of SCE). Due to feedwater pump trip and SG U-tube bundle depressurization, the RSG U-bundle secondary water level will shrink and tubes will be uncovered for a period of at least 10 minutes and experience a sharp drop/increase in secondary side pressure. The entire sub-cooled feedwater inventory contained in the faulted RSG will instantaneously flash to high dry steam and over-pressurize the steam generators. Loss of Turbine load will also over-pressurize the steam generator. Main steam safety valves located outside the containment will progressively open to prevent over-pressurizing the steam generators and connect the faulty generators to the environment via open steam safety valves. Now for the next 10-15 minutes, the Control Room is busy in trying to trouble shoot and diagnose the changing plant conditions and flipping through 1000 pages of Emergency and Abnormal operating procedures to determine the correct course of mitigation actions.

Meanwhile, during the 10 to 15 minutes, the combination of resonant, out-of-plane, in-plane vibrations, jet impingement forces, and RSG debris will cause large axial, bending, dynamic and cyclic loads on all the tubes, tube support plates, retainer bars and anti-vibration structure. The strength of the welded and mechanical connections of these low frequency retainer bars, retaining bars and bridges have not been analyzed for the effects of these cumulative loads to prevent AVB structure displacement, deformation or collapse during loss of offsite power. The displacement, deformation or collapse of AVB structure introduce new and significant axial, bending, dynamic and cyclic loads, which can potentially cause thousands of worn, cracked, plugged and stabilized tubes to exceed their high cycle fatigue stress levels several times than the allowed tube ASME Endurance Limit of 13.6 ksi. If this happens, multiple circumferential tube ruptures will occur at tube-support plates, mid-spans, free spans and tube-to-anti-vibration bar notched interfaces due to macroscopic circumferential cracks caused by tube-to-tube wear and high cycle thermal fatigue. Since all the steam from the RSG would escape to the environment, the iodine-131 from un-partitioned reactor coolant leaking out the rupture tubes will also escape to the environment in less than 10 minutes with 60 tons of radioactive coolant and steam. Consistent with Fukushima Task Force Lessons Learnt and NRC Commissioner Meeting Transcripts, this event will be considered as a beyond design basis event, and SONGS Operators will be unable to take any timely mitigation actions in a radiation/steam environment to stop a severe nuclear accident in progress and notify the Offsite Agencies.

If the prevailing winds are towards San Clemente, consistent with NRC Inspector General Reports, NRC and Government Studies and observations of SONGS Emergency Plan Drills for the last six years, SCE and Offsite agencies would not have time to respond, notify, evacuate, shelter or give Potassium Iodide to the affected residents within the 10-mile affected emergency planning zone. ODAC, Offsite field monitoring teams, Emergency Vehicles, Helicopters, Orange County Hospitals capabilities will be severely limited or non-functional in a high radiation environment to operate and rescue/transport/shelter disabled, sick, elderly, children, transients and other affected citizens. The casualties, and short, long-term cancer affects to the affected population and ingestion pathway will depend upon the iodine spiking factor and the duration of blowdown, but the offsite releases will significantly exceed the NRC approved SONGS Control Room limit of 5 Rem Total Effective Dose Equivalent (TEDE), and the Exclusion Area Boundary and Low Population Zone limit of 2.5 Rem TEDE.

NOTE: While this event is occurring, San Diego County, Orange County and Marine Corps Base Camp Pendleton won't be able to send radiation monitoring teams into areas around the plant due to high radiation levels to locate the plume and take soil and air samples to determine the extent of the release off plant grounds. That offsite field monitoring data, along with the data from the plant wound not be able to sent to the Offsite Dose Assessment Center (ODAC) located in MESA Emergency Operations Facility for making Protective Action Determinations. The offsite plans are recommended to be revised and feasibility demonstrated via an Emergency Plan Drill using Alternate and Parallel Emergency Operation Facilities located in Irvine and San Diego. The Three Mile Island nuclear accident was not as serious as Chernobyl, but was very confusing and chaotic. 40,000 gallons of radioactive waste was released in the Susquehanna River. 140,000 pregnant women and small children were evacuated as a precautionary measure, but cancer risk was not a serious threat.

If the prevailing winds are towards the Pacific Ocean and San Diego, the Public and SONGS worker casualties will be minimum, and short, long-term cancer affects to the affected human and marine population will depend upon the iodine spiking factor and the duration of blowdown, exceed the NRC approved SONGS Control Room limit of 5 Rem Total Effective Dose Equivalent (TEDE) and the Exclusion Area Boundary and Low Population Zone limit of 2.5 Rem TEDE. The impact on Marine Life and 50 Mile Ingestion Pathway is undetermined.

B.4 – SCE 50.92 License Amendment

SCE has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment", as discussed below:

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

Response: Yes

As shown above, the proposed changes affects the probability of multiple SG Tube Ruptures due to a potential main steam line break design basis accident. These changes are in a non-conservative direction (increased void fractions) and constitute a significant reduction in margin of safety and significant increase in probability of cascading tube ruptures over the OSGs. Operation at reduced power is not acceptable under the current licensing basis and operation of the plant will not remain bounded by the assumptions of the analyses of accidents previously evaluated in the UFSAR.

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

Response: Yes, see above

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

Response: Yes, see above

Please free to write an email, if you require more information. Thanks for your time.

Joosten, Sandy

From: Vinod Arora [vinnie48in@gmail.com]
Sent: Tuesday, May 14, 2013 6:03 PM
Subject: Thought for the Day - San Onofre Sad Saga Continued

SONGS has a poor work planning, work control, fire safety and maintenance history. Quality of NRC Inspections & Reports needs to be cross checked with SONGS Nuclear Oversight Findings, SONGS Division Self Assessment Reports and SONGS Open NMOs to make an intelligent determination and develop aggressive line of defensible questioning for reasonableness of 2012 O&M and Capital Project Expenditure Costs.

Joosten, Sandy

From: Vinod Arora [vinnie48in@gmail.com]
Sent: Wednesday, May 15, 2013 12:08 AM
Subject: VASSSM - Edison Nuclear Industry & Potential Cabinet Connections Put Pressures On Soft and Gentle NRC Commission Chairman

Edison's Adamant and Powerful (Anti-Safety, Anti-Southern Californian Nuclear Industry Lobby & Potential Cabinet) Connections Exert Pressures On NRC Commission Chairman - Chairman throws doubts and stones on decision of its own Independent and Intelligent ASLB for Public Hearings - NRC chief: No hearing required at ailing Cal nuke – How does Honorable Dr. Macfarlane know, what is the right decision to make - No background in FEI, SG Design or EP - Profits Potentially Prevail Over Safety - Clashes with Senator Barbara Boxer, who provides oversight over NRC

Top US nuclear regulator says no hearing required on San Onofre, despite board ruling

Associated Press – 4 hrs ago

LOS ANGELES (AP) -- The Nuclear Regulatory Commission has not decided whether it will hold a public hearing on a plan to restart the troubled San Onofre nuclear power plant in California, the nation's top nuclear regulator said Tuesday.

NRC Chair Allison Macfarlane told reporters in Washington, D.C., that she is aware of strong public interest in California and among some members of Congress for a public hearing, but added that a ruling this week by an NRC licensing panel does not require such a hearing be held.

"There are potential opportunities for public hearings," Macfarlane told reporters after a speech to the nuclear industry. She called the situation at San Onofre complex with "multiple moving parts right now."

The Atomic Safety and Licensing Board, an independent arm of the agency, sided Monday with environmentalists who have called for detailed public hearings on Southern California Edison's restart proposal.

Macfarlane's statement appears to put her at odds with U.S. Sen. Barbara Boxer, D-Calif., who said in a statement Monday that the board's decision established "a legal framework for a full public hearing before any final decision on the restart of the San Onofre nuclear power plant is made."

The plant between San Diego and Los Angeles hasn't produced electricity since January 2012, after a small radiation leak led to the discovery of unusual damage to hundreds of tubes that carry radioactive water.

Edison wants to run the Unit 2 reactor at no more than 70 percent power for five months, which it projects will stop damage to its steam generator tubing that sidelined San Onofre more than a year ago.

In its ruling, the licensing board called Edison's restart plan an "experiment."

Macfarlane says with or without a public hearing, a decision on the restart plan will not be made until at least late June.

In a brief statement, Southern California Edison said it's evaluating the board's ruling.

The company noted it had separately submitted paperwork to change the seaside plant's operating rules to permit the single reactor to run at reduced power, down from the now-required 100 percent. Parent company Edison International raised the possibility last month of retiring the plant if it can't get one reactor running later this year.

Joosten, Sandy

From: Vinod Arora [vinnie48in@gmail.com]
Sent: Thursday, May 16, 2013 12:41 AM
To: CHAIRMAN Resource
Subject: San Onofre Sad Saga - An example and appeal to the NRC Chairman for the Safety of 8.4 Million Southern Californians
Attachments: Senator Boxer LetterC 09-27-12 - D Final as Mailed-1.docx

Follow Up Flag: Follow up
Flag Status: Flagged

Note: Office of the Secretary, U.S. Nuclear Regulatory Commission

Would you be please kind enough to deliver this message and the attached letter to Dr. Macfarlane

Honorable Dr. Macfarlane,

Bluntly declaring "I am angry" about the IRS scandal, President Barack Obama said late Wednesday that Treasury Secretary Jack Lew had forced out Steven Miller, the acting commissioner of the Internal Revenue Service. Speaking in the East Room of the White House, Obama called Miller's ouster "the first step" to prevent similar misconduct in the future and vowed to "do everything in my power" to make sure it never happens again. Obama said he had reviewed the Treasury Department Inspector General's report that details how the IRS targeted conservative groups for special scrutiny when they applied for tax-exempt status.

According to San Onofre Insiders and 8.4 Million Southern Californians, there are too many allegations of violations of federal safety regulations, examples of serious Management misconduct, discrimination, retaliation, poor maintenance and work practices, and financial irregularities by SCE in the design, operation and investigation of San Onofre Replacement Steam Generators. Before, NRC even thinks of granting permission to restart Unit 2, all these allegations have to be thoroughly investigated and presented in plain English and Spanish by NRC in lengthy public hearings to 8.4 Million Southern Californians in accordance with the ASLB Ruling, President's Open Government Initiative and U.S. Sen. Barbara Boxer, who said the ASLB's decision established "a legal framework for a full public hearing before any final decision on the restart of the San Onofre nuclear power plant is made."

If 8.4 Million Southern Californians are not satisfied, then the NRC and SCE have only the following choices:

1. SCE contracts Westinghouse to repair or replace the San Onofre Generators in the shortest time using multiple tube manufacturers and fabricators to duplicate san Onofre Steam Generators like Palo Verde Steam Generators at 100% capacity, or

2. SCE contracts MHI to repair or replace the San Onofre Generators in the shortest time using multiple tube manufacturers at 85% capacity using third party validated and independently verified conservative design, full-scale mock-up testing and precision manufacturing (It has been proven beyond the shadow of doubt that NEI Qualified, "US Nuclear Plant Designer" MHI does not have the organization, tools, skills, technology and capability to design and fabricate these generators to operate at 100% capacity as we saw with the destruction of 21st Century Safest and Innovative defectively-designed and degraded generators.)

In the end, NRC commission cannot or should not be pressured to put "Profits over Safety" by EIX/SCE Management, Attorneys, Politicians, Lobbyists and Nuclear Industry. EIX/SCE gets a handsome 10.45% yearly return on its $32 Billion Transmission and Distribution Investment in addition to all the power purchases and generous operating expenses paid by the helpless Rate Payers. Therefore, SCE cannot lose money in any situation. Meeting Peak Summer Demand and closing San Onofre is a myth and illusion created by EIX/SCE to pressure NRC for granting permission to restart defectively designed and degraded Unit 2, so they do not have to refund Billions of Dollars to Ratepayers for overcharging and pay for their mistakes in the San Onofre Generators Replacement Project. America is a Democratic Society and following the example of His Excellency, The President of United States, you should protect the safety of the people rather than succumbing to the pressures of EIX/SCE Management, Attorneys, Politicians, Lobbyists and Nuclear Industry.

Please feel free to call or write to me on any technical or performance issue relating to San Onofre.

Sincerely,

Vinod Arora, P.E. (CA - Mechanical Engineering)

M.S. Engineering (Environmental/Civil Engineering), University of West Virginia

Graduate Level Courses in Chemical Engineering (University of Maryland & Newark College of Engineering)

B.S. Chemical Engineering with minor in Civil, Electrical, Mechanical and Materials Engineering

Ex San Onofre Systems, Hazards, HELB, FP and EP Engineer/Auditor

Former Certified Asbestos Consultant to State of California

CEO/President AVP Arora International Inc. [A Non-Profit Engineering Corporation]

Cell: (714) 305-1903, Home: (714) 281-8600

FYI see below and attached letter

vinod arora <vinnie48in@gmail.com> 11/1/12

to info

November 1, 2012

Rufus Gifford
National Finance Director
Obama for America

Dear Mr. Gifford,

Would you be kind enough to pass the following message

Thanks and God Bless you and Your Family for supporting His Excellency Honorable President Barack Obama and Gracious First Lady Mrs. Obama. Your are truly helping All Americans.

Sincerely
Vinod Arora, PE
8840 East Wiley Way
Anaheim Hills, CA 92808
(714) 305-1903
****

Attention: To His Excellency Honorable President Barack Obama and Gracious First Lady Mrs. Obama

Respected Sir/Madam,

Your track record and spectacular performance on All fronts for ALL Americans have assured your 100% win in the coming elections and my hearty congratulations for being the "One of the Greatest and the Next President of the United States of America." Election Night will hold victory for you and your family and celebrations for ALL Americans. I have never being wrong in my predictions. Truth and care for ALL prevails over the power of Money, False Advertising, Confusing Claims, Erroneous Poll Projections and False Promises for Jobs for 12 Million Americans without Federal Participation/Assistance to Small and Large Businesses.

GOD Bless You and Your Family

Sincerely

Vinod Arora, PE

8840 East Wiley Way

Anaheim Hills, CA 92808

(714) 305-1903

September 26, 2012

To: U.S. Senator Barbara Boxer
70 Washington Street, Suite 203
Oakland, CA 94607
Oakland Office Phone: (510) 286-8537

From: Vinod K. Arora, PE (Mechanical, CA)
8840 East Wiley Way
Anahiem Hills, CA 92808
SS#: Included in the Original Letter
Cell: (714) 305-1903
Home: (714) 281-8600

Subject: **The Number 1 US Nuclear Safety Concern – San Onofre Steam Generators**

Honorable and Respected Senator,

Thank you very much for promptly responding to my email enquiry. We appreciate very much your hard and dedicated work for improving the life of **ALL Americans** as the Chairman of the U.S. Senate Committee on Environment & Public Works.

1. Background

SONGS Unit 3 had been operating for approximately 1 year following the replacement of the new steam generators, when the control room operators received alarms on January 31, 2012, indicating that reactor coolant was leaking into one of the steam generators (3E088). The leak was unexpected, although small, had increased to approximately to 75 gallons per day and increased enough in a short period of time to warrant the precautionary shutdown of the Unit 3 reactor. The first indication of the leak was that the main condenser air ejector radiation monitors reached their alarm set-points. This event caused a direct release of radioactivity to the atmosphere. The radiological release resulted in an estimated 4.52 E-5 mrem dose to the public.

Southern Californians were very lucky *this time*, because a potentially serious nuclear accident in progress was stopped because of a high radiation reading on the main condenser air ejector radiation monitors, which happened to be maintained and in working condition! In March 2012, eight steam generator tubes in San Onofre Unit 3 Steam Generator E-088 had failed their pressure testing and therefore were plugged. According to NUREG-1841 and Fairewinds Report, San Onofre's eight plugged tubes are an anomaly for the ***entire*** operating history of US nuclear industry." NRC website just makes a casual note of the Unit 3 tube leak by stating, "The leak, although small, had increased enough in a short period of time to warrant the precautionary shutdown." If these non-safety related radiation monitors were ***Inoperable*** (e.g., out of service for repairs, needed spare parts, etc.) or the Operators had not performed the precautionary shutdown of the Unit 3 reactor, this unanticipated leak could have developed into one full tube rupture, eight full tube ruptures, or undetermined amount of tube ruptures, and potentially released significant amounts of radiation. This unanticipated radiological release from the newest, safest, most efficient 21st century Steam Generator in the USA [The New 700 Million Dollar Steam Generator (SG) Replacements costs paid for by the Southern Californian Ratepayers] could have potentially and adversely affected the health and safety of all Southern Californian residents plus the transient population within the 10-mile Emergency Planning Zone.

Data obtained from public sources for nuclear plants that have experienced major steam generator tube leakage events (shown below) even with one tube rupture shows the significance of what could have happened to Southern California, if the eight steam generator tubes in San Onofre Unit 3 Steam Generator E-088 had failed concurrently while Unit 3 was operating. The data shown below contradicts NRC Website statement about SONGS Tube Leak and Staff correspondence dated May 20, 2009, which states, "By means of plant technical specifications, licensees are required to assure with high confidence that steam generator tubes have sufficient integrity to survive normal operations as well as possible design basis accidents, such as the rupture of a main steam line outside of the containment boundary.' 'Current TS requirements relating to SG tube integrity provide reasonable assurance that all tubes will exhibit acceptable structural margins against burst or rupture under normal operating conditions and DBAs, including MSLB, and that leakage from one or multiple tubes under DBAs will be limited to very small amounts, consistent with the applicable regulations for offsite and control room dose (ref. ADAMS Accession Number ML091320055)."

Year	Plant	Location	Flow Rate	Rupture or Leak
1975	Point Beach 1	Wisconsin	125 gal/min	Rupture
1976	Surry 2	Virginia	330 gal/min	Rupture
1979	Prairie Island 1	Minnesota	390 gal/min	Rupture
1982	Ginna	New York	630 gal/min	Rupture
1987	North Anna 1	Virginia	600 gal/min	Rupture
1989	McGuire 1	North Carolina	500 gal/min	Rupture
1993	Palo Verde 2	Arizona	240 gal/min	Rupture
2000	Indian Point 2	New York	90 gal/min	Rupture

At around 13: 50, on February 9th, 1991, leakage of about 55 tons of primary coolant occurred due to the failure of one SG tube in a steam generator built by Mitsubishi in the No. 2 pressurized water reactor at the Mihama nuclear power station in Japan. This accident was the first level 3 nuclear accident in Japan. This accident ignited social concerns in Japan because it shattered the industry myth of 100% safe nuclear reactors!

The International Nuclear Events Scale (INES) - A level one is a minor event, and the highest level is a level seven, which is a major nuclear accident, so the level number increases with the scale of the nuclear accident). For example, the loss of coolant that occurred in Three Mile Island was ranked a level five, while both Fukushima Daiichi and Chernobyl Nuclear Disasters were ranked level seven or major accidents.

After the SONGS accident, International Nuclear Expert Arnie Gundersen has stated in several reports, "By claiming that the SONGS steam generator replacements were a *like-for-like* design and fabrication, Southern California Edison avoided the more rigorous NRC license amendment process. 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. Fairewinds believes that vibration within the tubes in both Unit 2 and Unit 3 were due to the simultaneous implementation of untested manufacturing and design changes made by the Edison/MHI to the replacement steam generators. Edison

has plugged 3.7 times more tubes in its San Onofre Units 2 and 3 replacement steam generators than the nationwide total of all plugs inserted in all the remaining replacement steam generators in all US nuclear reactors. Moreover, the NRC and Edison have not openly informed the public about the magnitude of tube generator plugging at San Onofre compared to the remainder of U-tube damage at other US reactors. Edison maintains that San Onofre Unit 2 should be allowed to restart because it has a steam generator that is better than the one in San Onofre Unit 3. Fairewinds notes, however, that the Unit 2 steam generators would be the worst in the nation except that San Onofre Unit 3 has earned that dubious distinction. Moreover, the replacement steam generators at San Onofre Unit 2 and Unit 3 were designed to the same specifications, were modeled using the same flawed and inadequate computer codes, and both units have the same high velocity fluid elastic instability (FEI) that is one of the causes of the significant tube degradation. Each Unit's replacement steam generators have the same failure modes. Both Unit 2 and Unit 3 have experienced tubes colliding with each other (tube to tube damage) as well as tubes colliding with structures inside the steam generator that were designed to prevent tube damage (anti-vibration bars, tube support plates, and retainer bars). Both Units experienced wear due to FEI (fluid elastic instability) and turbulence induced vibration. In addition, Unit 2 exhibited wear due to the collision of a foreign object with the tubes while Unit 3 did not."

These unprecedented design changes made without NRC approval, under the pretense of *like-for-like* replacement in terms of "form, fit and function" by SCE Engineers resulted in (data based on www.nrc) 1595 tube wears (4,721 total wear indications) in Unit 2 and 1806 tube wears (10,284 total wear indications) in Unit 3. The NRC AIT Report states that the damage was caused due to high steam flow and velocities, high steam voids and multiple areas of no contact between the anti-vibration bars and the tubes. Out of the total 1595 tubes damaged tubes in Unit 2, only 510 tubes have been plugged/staked. Plugging 510 tubes in Unit 2 will not eliminate the fluid elastic instability or the flow-induced vibrations in the remaining 1,085 damaged tubes by allowing operation of Unit 2 at reduced power.

Professor Daniel Hirsch in his September 12, 2012 report stated, "San Onofre Unit 2 and Unit 3 are both very ill nuclear plants. Unit 3's fever is slightly higher, but both are in serious trouble. What they are experiencing is not just normal wear due to "settling in" purportedly experienced with similar replacement steam generators. They are far, far outside the norm of national experience. And Unit 2 cannot be said to be acceptable for restart, any more than Unit 3. Unit 2 has hundreds of times more bad tubes and a thousand times more indications of wear on those tubes than the typical reactor in the country with a new steam generator, and nearly five times as many plugged tubes as the rest of the replacement steam generators, over a comparable operating period, in the country combined. Restarting either San Onofre reactor with crippled steam generators that have not been repaired or replaced would be a questionable undertaking at best." Dale Bridgenbaugh, Internationally Known Nuclear Engineer stated in the above report, "It is my opinion that measures necessary for the future safe operation of either of these unit have not been adequately put forth at this time, and that operation with or without reduced power of Unit 2 should not be authorized."

2. The Number 1 US Nuclear Safety Concern – San Onofre Steam Generators

If SONGS Unit 2 is allowed to operate at reduced power, an un-isolable main steam line break accident can occur at any time, due to a postulated design basis earthquake and/or any other associated failure. Due to this event, the depressurization of the steam generator caused by the steam line break coupled with the excessive vibrations due high differential pressure (> 2250 psi), high reactor coolant water temperature inside the tubes, the compact space between the SONGS U-Tube Bundle and the moisture separators (compared with other Mitsubishi Steam Generators) and the steam over-pressurization would cause the elastic deformation (NRC AIT Report, Mitsubishi Preliminary Cause Evaluation) of the taller U-tube bundle due to increased U-tube bundle height, high localized steam-voids or dry-outs (two-phase mist region, almost devoid of water in undefined central portion of the U-tube bundle above the 7th support plate) and narrow-pitch/tube diameter ratio. This unanalyzed and rare phenomena not experienced in the Steam

Generators operating history, in turn, *would cause the onset of fluid elastic instability conditions* due to the 100% localized steam voids in the central portion of U-Tube bundle above the 7th Support Plate. The *fluid elastic instability conditions* would result in further lowering the already low in-plane tube clearances (Attributed to unanalyzed effects because of addition of more tubes to achieve more thermal MWt out of the SGs). The combination of these factors along with a poorly designed anti-vibration support structure [low damping capability of the support structure (i.e., the tube support plates, the tube-sheet, and the anti-vibration bars)] would result in excessive and violent vibrations, cause tubes to hit each other in the in-plane direction, result in leaking tubes, which would cause high-pressure primary sub-cooled water jets. These high-pressure jets would cut holes into other already worn tubes and create undetermined number of cascading tube ruptures.

The cumulative effects of the above conditions along with the unanalyzed effects of plugged and staked tubes would rupture other damaged, plugged, staked and worn tubes. The amount of leaking reactor coolant through these ruptured tube cuts is beyond the analyzed limits of a SONGS UFSAR Analysis [Three combined independent events loads (DBE + MSLB + LOCA)] that would be released via the blowing radioactive steam carrying un-partitioned reactor coolant from the un-isolated steam generator into the environment. This uncontrolled radiological accident would release significant amounts of radiation, which could adversely affect the health and safety of all Southern Californian residents plus the transient population within the 10-mile Emergency Planning Zone. We believe that this scenario can also progress to a nuclear meltdown of the reactor due to potential errors by plant operators unable to diagnose and control rapidly changing plant conditions due to the confusion caused by the non-user friendly and complex, abnormal, emergency operating and emergency plan implementing procedures.

This scenario is a departure from a method of evaluation described in the UFSAR used in establishing the SONGS design bases or in the safety analyses and requires a NRC 50.90 License Amendment before SONGS Unit 2 or 3 can be allowed to restart. A permission by NRC for SONGS restart of either Unit 2 or 3 without the 50.90 License Amendment would be construed as: (1) Repeat violation of NRC 50.90 License Amendment Process by SCE, (2) Violation of SCE's Overriding Obligation to protect the health and safety of Southern Californians from radiological accidents, and (3) Inconsistent with NRC long history of commitment, transparency, participation, and collaboration with public in oversight of Nuclear Reactor regulatory activities.

3. Insights into the San Onofre Steam Generators Debacle

According to SONGS anonymous and concerned current managers, former employees and Press Reports, SCE did not fully disclose to the NRC of the numerous untested and unanalyzed design changes, which were made in a rush to generate more power in order to make Billions of Dollars in profit by replacing the SONGS Original CE Steam Generators with the SCE designed and MHI fabricated *safest, most efficient 21st century machinery* (Source: SONGS Chief Nuclear Officer Pete Dietrich, Jan 10 2012, Marketwatch). "The installation is 'a major milestone in the station's history, said Ross Ridenoure, Southern California Edison senior vice president and chief nuclear officer. We're committed to making sure it's done right." [Source: January 28th, 2009, LA Times]." "Edison President John R. Fielder said new steam generators are cheaper for ratepayers than building new power plants or buying power on the open market. [Source: December 16th, 2005, LA Times]." "The new steam generators are designed to last longer, said Mike Wharton, manager of the steam-generator replacement project. 'They are designed for 40 years,' he said. 'We expect we'll actually be able to get 60 years out of them … better materials, better design. You learn over the course of years what works well and what doesn't, and you try to build it into the next generation.' [Source: December 24th, 2009, OC Register]."

Avoiding a thorough required NRC review, combined with the lack of Solid Teamwork & Alignment between SCE & MHI, and the lack of *benchmarking* on the design and fabrication details of Palo Verde and other CE Replacement Generators is probably either the Root Cause or a significant contributor to the 1.3 Billion Dollar San Onofre Replacement Steam Generator debacles. A review of NUREG-1841 and Palo

Verde Uprate Application indicates that Westinghouse and Babcox & Wilcox International have designed

more than 100 CE and other Replacement Steam Generators Anti-vibration bar and tube support plates to mitigate the adverse effects of high steam velocities and flows, localized steam voids on the tube wear from the effects of turbulence-induced vibrations and fluid elastic instability both in the in-plane and out-of plane directions.

The SCE Team did not follow the NRC's Section Chief advice on "Critical Questioning & Investigative Attitude" and read the World's Foremost Expert research papers on preventing the adverse and expensive effects of Turbulence-induced vibrations and Fluid Elastic Instability. That means that SCE Engineers also potentially violated the SONGS Human Performance Tools and Procedures, by not using the above described critical attributes.

4. Recommendations For Uncovering the Truth About San Onofre's Steam Generator Debacle

Senator, you have my written consent to open a confidential formal joint enquiry by the US Departments of Justice, Labor and the Committee on Environment and Public Works into the dark inner secrets of the SONGS Steam Generator Degradation and Safety Concerns. Please ensure that this investigation does not breach the confidentiality of my family and my children. This investigation is of vital importance to ALL Americans and should also lead to improvements of the safety records at all the other US Nuclear Power Plants. Only an Expert Independent Steam Generator Panel aided by investigators from the Joint Commission can determine the Real Root Cause and find the Truth by examining the SONGS RSGs MHI Procurement/Technical Bid Evaluation, SONGS Performance and Design Specifications and all the QA, Design, Root Cause, Test Reports and interviewing under oath all the Responsible parties both from MHI and SCE. This investigation will help CPCU, NRC, NEI, INPO and other US Agencies responsible for the protection of health and safety of Americans from potential radiological accidents, which can be caused by natural forces, human errors, equipment failures poor training, poor operating procedures and production/profit oriented management, etc.

Mitsubishi has been qualified as a "US Nuclear Manufacturer/Designer" by NEI in 2009 and is trying to dominate the future US Nuclear Energy Market by very aggressively marketing their Advanced Pressurized Water Reactor (APWR) 1700 MWe design by claiming, " US APWR is more efficient with greater output than any previous power plant. This design has been slightly modified to satisfy U.S. and international utility requirements as the US-APWR, and it will become the design that we sell around the globe. The US-APWR satisfies our customers' requirements with the best performance for safety, economy, operation and maintenance."

In a Mitsubishi Heavy Industries, Ltd. Technical Review Vol. 43 No. 4 (Dec. 2006), The company claimed, "On the basis of the manufacturing technology, quality control and process management cultivated for decades in PWR plants in Japan, MHI has organized a system for shipping and delivering high-quality reactor products to overseas customers within a short period, as requested. In the export business, not only in terms of equipment quality but also through a quality assurance system for export, quality records conforming to local laws, regulations, and codes can be compiled and shipped together with products. Quality record documents including design drawings, manufacturing procedures, and inspection records are highly evaluated by customers. The know-how of exports of PWR plant components obtained so far will be fully fed back to the replacement work of equipment in overseas plants and the construction work of new plants, and MHI is determined to fulfill its duties as a leading PWR plant equipment company to contribute to the stable supply of electric power all around the world."

Note: A joint venture company has been established by Luminant and Mitsubishi Heavy Industries (MHI) to support the construction of US-APWR at Comanche Peak in Texas. □ The companies' move is in line with statements made in December 2008 when they submitted a combined construction and operating license application for the two new reactors they want to build at Comanche Peak. The 1700 MWe APWRs

should begin operation before 2020. MHI has submitted application for Standard Design Certification for the U.S. Advanced Pressurized-Water Reactor (US-APWR), a 4,451-MWt pressurized-water reactor (PWR) on

December 31, 2007. The staff of the U.S. Nuclear Regulatory Commission (NRC) is currently performing a detailed review of that application. The Final NRC Safety Evaluation Report (SER) is projected to be complete in March 2015.

As NRC Section Chief, of MIT Intelligence, said to his dedicated staff, "Before approving any Licensing Documents/Inspection Reports, Please, "Read and reread in-between the lines', use a 'Critical Questioning & Investigative Attitude' and 'Solid Teamwork & Alignment' between NRC Staff, Licensee, Manufacturers and Vendors." This critical advice is consistent with the teaching of the World's Foremost Expert on preventing the adverse and expensive effects of Turbulence-induced vibrations and Fluid Elastic Instability in Nuclear Power Plant Components and their Heat Exchangers.

The DAB Safety Team says, "*After the San Onofre debacle, NRC, NEI, INPO, Utilities and other Agencies need to double their efforts to ensure that the MHI technology is safe, reliable and affordable for the US Public.*"

My Team and I are willing to meet with you, NRC Chairwoman Allison Macfarlane and other Federal Officials in Washington, D.C., to provide further details in this matter which we strongly believe is of the utmost public safety and to also discuss the other San Onofre Safety and Management retaliation techniques being used against workers, who express nuclear safety concerns. These acts of SCE Management retaliation are in blatant violation of Federal Regulations.

Sincerely yours

Actual Signatures on File

Vinod K. Arora, P.E.

The DAB Safety Team

DISCLAIMER

The opinions expressed in this document are solely based on Author's personal experience, research, observations and recommendations and need to be verified and validated before implementation.

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About the Author:

Vinod K. Arora, PE spent the last 15 years at SONGS as a hazard barrier engineer (Fire Protection Engineering Group), fire protection and emergency preparedness (Nuclear Oversight Engineering Group) functional area owner and auditor.

The author does not claim to be a Steam Generator Expert. The views expressed in this report are based on research conducted by him based on the advice of an anonymous NRC Branch Chief, who said "Read and reread in-between the lines', use a 'Critical Questioning & Investigative Attitude' and 'Solid Teamwork & Alignment'." The author read approximately 25 research papers on "Preventing the adverse and expensive effects of Turbulence-induced vibrations and Fluid Elastic Instability in Nuclear Power Plant Components" published by the World's leading experts and researchers from Canada, US, Korea and Taiwan. He read NUREG-1841, Power Uprate applications and literature published by Westinghouse, BWI International and MHI on design and fabrication details for "Preventing the adverse and expensive effects of Turbulence-induced vibrations and Fluid Elastic Instability in Nuclear Steam Generators."

The author has performed extensive independent research and held discussions with SONGS current concerned key personnel associated with the Replacement Steam Generator Project Investigations, Shift Managers and industry experts on this subject. In addition, he has reviewed the articles published by Fairewinds Associates, Professor Daniel Hirsch, News Media Reports and NRC Preliminary Augmented Inspection Team Report (ML 2012007). He has also reviewed other SONGS plant documents (2001- to date) and production data sheets, which are not available to public.

The author is a Professional Mechanical Engineer, a former Senior Shift Chemical/Process Engineer and a 30-year veteran of the nuclear power industry. He is an extremely hard working multi-disciplined engineer with chemical, mechanical, environmental, fire protection, hazard barrier, and emergency preparedness experience as well working as an auditor.

The author earned a BS in Chemical Engineering from the University Institute of Chemical Engineering & Technology, UICTE (earlier known as Department of Chemical Engineering & Technology, DCET). Since its inception in 1958, the Institute has established itself as one of India's premier institutions engaged in imparting quality technological education and providing support to research and development activities. The author earned a MS in Engineering from the Department of Civil and Environmental Engineering at West Virginia University, at Morgantown. The institution is committed to meeting the economic and environmental challenges of our times by educating professionals with cutting-edge skills and conducting innovative, cross-disciplinary research to solve our nation's infrastructure challenges.

The author was previously an asbestos/lead consultant to the California Office of the State Architect and an environmental consultant to many cities and industrial/commercial clients in California, South and North Carolina, Illinois, Indiana and Virginia. He has won a number of awards for both his service and ability to perform critical engineering evaluations. He is currently the CEO/President of a Non-Profit Public Service Engineering Corporation.