CHAIRMAN Resource

From: Bill Hawkins [billlee123456@gmail.com]
Sent: Sunday, May 12, 2013 1:45 AM
Subject: San Onofre Sad Saga Continued - NRC/SCE/MHI/SCE Experts and Public Awareness Series - SCE Designed and MHI Fabricated 21st Century Safest & Innovative Replacement Steam Generators

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]
Kendra Ulrich states, "Edison's restart scheme has major holes in it: the root cause of the problems hasn't been found, Edison's own experts disagree on even the secondary cause of the issues, and disagree on the length of time left before another accident could occur -- which could be somewhere in the range of a few months to a little over a year. In short, Edison is rushing to restart the reactor in time for summer, when they can maximize their profits, which -- unfortunately for the people of Southern California -- means putting safety at the bottom of the list. But the Nuclear Regulatory Commission has the power to stop the restart, by rejecting Edison's license amendment and allowing the public to get involved in the process." **Dr. Joram Hopenfeld** and **SONGS Insiders** say, "Operating at 70% power may reduce risk, but it is no guarantee of safety. "Maybe the vibrations wouldn't be as severe, but it doesn't mean they are going away," Hopenfeld said. SONGS Insiders say, "If an accident like Main Steam Line Break happens, (an) emergency plan is not geared to handle such a public safety devastation. Those things have never been practiced or demonstrated in a drill scenario." According to a highly placed retired SCE manager, "[redacted]."

Conclusions: Edison says, "It had no knowledge of the SONGS SGs issues until January 2012" is completely misleading and erroneous. Based on an internal San Onofre investigation conducted last year and rejection of SCE & MHI Root Cause Evaluation and NRC AIT Report, it is concluded that Edison Engineers most likely knew of SG issues in 2006, but did not speak up loud enough for fear of retaliation or firing. Therefore, truth can only be proven, if Edison team of California Professional Engineers (Mechanical, Pipe Supports, Civil and Steam Generators) are made to testify under oath to prove that they had no PRIOR knowledge of the steam generator issues until January 2012. In addition, all the meeting notes and videos of SCE/MHI Joint AVB Team need to be examined in detail, which hopefully are in possession of SCE and MHI and have not been tampered (Billion Dollar Question).

Issues: MHI Root Cause states, "Also MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction(e.g., using a larger downcomer, using larger flow slot design for the tube support plates, and even removing a TSP)) by a greater amount were considered, but these changes had unacceptable consequences and MHI and SCE agreed not to implement them. Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG design under the provisions of 10 C.F.R. §50.59. Thus, one cannot say that use of a different code than FIT-III would have prevented the occurrence of the in-plane FEI observed in the SONGs RSGs or that any feasible design changes arising from the use of a different code would have reduced the void fraction sufficiently to avoid tube-to-tube wear. For the same reason, an analysis of the cumulative effects of the design changes including the departures from the OSG's design and MHI's previously successful designs would not have resulted in a design change that directly addressed in-plane FEI."

A review of Edison Procurement documents for replacement generator indicates that these documents were prepared, checked and certified by an Edison Design Team of California Professional Engineers (Mechanical, Pipe Supports, Heat Transfer, Civil and Steam Generators.) These documents state, "The tables in the 'Original Design Data' column specify the values for the existing steam generators and in the "Replacement Design Data" column specify the values for the RSG, either imposed by Edison or proposed by the Supplier. Edison prefers that the imposed values be maintained to the greatest extent possible, but understands that some trade-offs may be necessary, and is willing to work with the Supplier to accommodate these trade-offs. The Supplier may address with Edison any deviations from the values specified. The Supplier shall prepare and submit for Edison's review a Design Review Item List, which shall identify the design issues and fabrication/assembly activities, which if not correctly handled/performed may result in inadequate RSG performance, or degradation of performance, or reduction of the RSG service life. The list shall provide a detailed description of the issues, explain how they are addressed in the RSG design and fabrication/assembly plan or procedures, and identify the areas where trade-offs may be required."

Some of the Edison specified parameters responsible for higher steam quality (higher void fraction, fluid elastic instability) are as follows:
OSG - San Onofre Original Combustion Engineering Steam Generators
RSG - San Onofre SCE Designed and MHI Fabricated 21st Century Safest & Innovative Replacement Steam Generators

Steam Generaor Thermal Rating (@100% Reactor Power – MWt - OSG – 1,705, RSG – 1,729
Steam Pressure (@100%power) – Beginning of Life (Guaranteed Value) – psia - OSG - 900, RSG - 817
RCS Design Flow (0% Tube Plugged) – gpm – OSG – 198,000, RSG – 209,880 (16%)
Circulation Ratio (@100%power) – OSG – 3.2, RSG – 3.3 817
Steam Moisture Content, % -OSG < 0.20 –RSG - < 0.10
Increasing the heat transfer area by 11%,
Addition of 377 new tubes (4% heat transfer area)
Increasing the average length of heated tubes by 50 inches (Equivalent addition of 650 tubes or 7% heat transfer area)

There are hundreds of operating steam generators in the world, which have avoided 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 psia, Steam Moisture Content < 0.20 % and 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 states, "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."

The closure of SONGS' units 2 and 3 has cost $470 million, which has been absorbed by the region's customers and the company's shareholders. Southern California Edison, is trying to figure out its next move: That is, it is asking the Nuclear Regulatory Commission to allow it to restart Unit 2 and to gradually rev it up to 70 percent of its capacity. If it is unable to do so, the utility says that it may retire both units at the end of this year.
"Without a restart of Unit 2, a decision to retire one or both units would likely be made before year-end 2013," says Theodore Craver, chief executive of Edison International that is the holding company for SoCalEd, in a conference call. "There are many potential decision scenarios involving Unit 2 and Unit 3. They all have different implications for grid reliability, customer costs, attainment of greenhouse gas and air quality objectives."
Other than permanently shutting down the units, SoCalEd could choose to work with the vendor that installed the steam generators to fix the problem. That would take five years, meaning that costs would continue to add up while the facility would not be bringing in any revenues.

All this has created a mess between SoCalEd and Mitsubishi Corp., which installed the steam generators that have defective parts. The vendor says that its liabilities are limited to $139 million, which is considerably less than the overall maintenance costs, not to mention to the loss of business. It also says that it had informed the utility in 2009 of the malfunctions, which led to the radiation leaks in 2012.

Mitsubishi maintains that a strategic business decision was made in 2009 to fix the problem, as opposed to re-install different parts. SoCalEd, however, denies that supposition, adding that it had no knowledge of the issue until January 2012. That's when the tubes began leaking.

Now both companies are being accused by citizen groups and policymakers of improprieties and caring more about profits than they do safety. Specifically, Rep. Ed Markey, D-Mass. wants the U.S. Securities and Exchange Commission to probe into whether investors were intentionally misled.

"Investors presumably would want to know whether a company is choosing not to implement additional safety protocols because such actions might require a nuclear reactor to go through a more strenuous licensing process," says Rep. Markey, who is top Democrat on the House Natural Resources Committee. "Such choices could be evidence of poor management or even possible civil liability."

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From: Bill Hawkins [billlee123456@gmail.com]
Sent: Monday, May 13, 2013 10:24 PM
Subject: San Onofre Sad Saga Continued - NRC/SCE/MHI/SCE Experts/CPUC and Public Awareness Series

SCE is on the road to being Unpopular and Bankrupt without Public Support - Ted, Ron, Michael and Pete will find themselves alone holding the Bag Full of RW - Holding of Useless Proprietary Information is hurting SCE, its Vendors and NRC - It makes Public more suspicious of wrongdoing by SCE, its Vendors and NRC

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

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]

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 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 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.

NRC News, May 13 (Reuters) - ASLB: San Onofre Confirmatory Action Letter Process Offers Opportunity for Adjudicatory Hearing: The Atomic Safety and Licensing Board (ASLB) has decided partially in favor of a public interest group that petitioned for a hearing on the NRC's Confirmatory Action Letter process regarding steam generator issues at the San Onofre nuclear

power plant in California. The ASLB is a three-member board of administrative judges independent of the NRC staff that conducts adjudicatory hearings on major agency licensing actions.

The board's decision concludes that this particular Confirmatory Action Letter process, in which San Onofre seeks to restart Unit 2, is effectively a license amendment proceeding. Therefore the Atomic Energy Act and NRC rules give the public the opportunity for an adjudicatory hearing. The Board's decision provided the public interest group, Friends of the Earth, with the relief it requested – namely, the opportunity for a hearing on the license amendment. Accordingly, the Board's decision terminates the proceeding at the Board level. The Board also offered reasons why this decision applies only to the unusual facts in the San Onofre process and not to the whole category of Confirmatory Action Letters.

Reaction to ASLB Ruling: Damon Moglen of Friends of the Earth called the ruling "a complete rejection of Edison's plan to restart its damaged nuclear reactors without public review or input." An SCE spokeswoman said the utility was still reviewing the ruling and declined to comment. Edison's Chief Executive Ted Craver has said the utility may decide by year end to retire one or both San Onofre reactors if its restart request is denied, citing uncertainty over NRC timing and SCE's ability to recover costs related to the extended outage. The reactor can only restart if the NRC concludes it can operate safely. Pressure has been growing on the NRC and the utility to agree to a full review of safety issues at San Onofre from elected officials and anti-nuclear groups. The board concluded that SCE's restart plan, known as the Confirmatory Action Letter process, is effectively a license amendment proceeding that gives the public the right to a hearing with testimony and cross-examination of witnesses.

CPUC News: Two California Public Utility Commission Judges have banned the media and the public from videotaping the hearings on the broken San Onofre nuclear plant run by SCE. The chair of the California Public Utility Commission is the former CEO of SCE and has taken favors from non-profit corporations funded by the SCE. Governor Brown who represents the utility industry has kept this questionable chair in his position of regulating the utilities in California. One of the judges Melanie Darling literally went out of control at the last hearing and tore down a banner after the hearing was adjourned. Maybe she does not like seeing herself in action so shutdown the cameras. Public Groups are requesting that the Commission provide a good-quality webcast of the entire week of evidentiary hearings currently scheduled for May 13-17, 2013. California Public Utilities Commission is strongly advised to allow citizens to videotape the hearings pursuant to the Bagley-Keene Act, in order to maximize transparency in this case and provide public access, especially for affected people, who live near the San Onofre nuclear plant, 450 miles away from the Commission's courtrooms.

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.

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

SCE states, "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 lb/hr (vs. 7.414E6 lb/hr for OSGs) of 833 psia (vs. 900 psia for OSGs) saturated steam with void fraction of 99.6% (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.).
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 the 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:

1. 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 fluid elastic instability and flow-induced 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 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 causes tube leakage, tube failures at MSLB test conditions and or unprecedented tube-tube wear, Tube-to-AVB/Tube Support Plates wear, as we saw in SONGS Unit 3. 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 drops. This lowering of steam temperature combined with high heat flux in the hot leg side of the U-tube bundle causes steam dry-outs to form (Vapor fraction >99%), 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 as seen in SONGS Unit 3 RSG's. In summary, FEI is a phenomenon where due to SONGS RSGs design intended for high steam flows causes the tubes to vibrate with increasingly larger amplitudes due to the fluid effective flow velocity exceeding its specific limit (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 found to have a 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
DAB Safety Team SG Secondary U2 Pressure Range 863 -942 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. SCE/MHI did not review the research papers 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."

6. 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."

7. 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."

8. 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."

9. RSGs were operating at a circulation ratio of 3.3. Most of The CE RSGs are running at a circulation ratio of 5.0 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**

CHAIRMAN Resource

From: Bill Hawkins [billlee123456@gmail.com]
Sent: Friday, May 17, 2013 1:26 PM
Subject: San Onofre Sad Saga Continued - CPUC SONGS First Phase Hearings, ASLB Ruling - SCE/MHI/NRC/8.4 Million Southern Californians Awareness Series

Updates

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]

Summary: Based on the information provided below, SCE has not wisely spent the money in 2012 in resolving the safety problems with Unit 2 and all the money spent on preparing Confusing, Conflicting and Incomplete Unit 2 Return to Service Reports and 170,000 Tube Inspections is considered a complete waste of Ratepayers/EIX Shareholders and SONGS Workers Money. Therefore, the SCE expenses spent on Unit 2 in 2012 are highly questionable and unreasonable and do not ensure that Unit 2 can be operated safely at 70% power. The ASLB ruling is crystal clear that Unit 2 operation at 70% power constitutes a test or experiment that meets the criteria in 10 C.F.R.
§ 50.59(c)(2)(viii) for SCE to require a full 10CFR 50.90 license amendment with lengthy trial-like public hearings, where 8.4 Million Southern Californians and Independent safety Experts can question SCE, its contractors and NRC Experts. Congress has commanded that licensees may not, under penalty of law, deviate from the terms of their reactor operating licenses. If a licensee is unable to operate a reactor in strict accordance with its license, it must seek authorization from the NRC for a license amendment (10 C.F.R. §§ 50.59, 50.90 to 50.92), which is a process that triggers a right to request an adjudicatory hearing by 8.4 Million Southern Californians, whose financial and safety interests have been compromised by SCE, NRC and CPUC unlawful activities since 2004 in the San Onofre Steam Generator Replacement Project.

NOTE: Atomic Safety Licensing Board AND NRC Office of the Inspector General are not empowered "to supervise or direct NRC Staff regulatory reviews." Therefore, NRC Commission under pressure from Anonymous Adamant Cabinet Secretaries, EIX/SCE Financial Backers, Politicians, Lobbyists, Attorneys, Nuclear Energy Institute, Nuclear Industry and Utility Groups can reverse ASLB ruling or any potential adverse findings from the NRC Office of the Inspector General.

San Onofre Units 3/2 FEI: Fluid elastic instability] occurs when the amount of energy imparted on the tube by the fluid is greater than the amount of energy that the tube can dissipate back to the fluid and to the supports. During in-plane [fluid elastic instability], tubes within the same column are excited by the fluid and move with the plane of the column, resulting in tube-to-tube contact and wear of the tubes. Based on a review of review of all the available to-date research papers on fluid elastic instability and flow-induced random vibrations, NUREG-1841, Power-uprate applications, SONGS Procedures, AREVA, Westinghouse, John Large, Arnie Gundersen, MHI and NRC AIT Reports, it is concluded that FEI (high dry steam and high fluid
velocities) in San Onofre Unit 3 Replacement Steam Generators (RSGs) was caused by lack of in-plane anti-vibration bars, narrow tube pitch to tube diameter, extremely tall tubes, high RCS flows, high steam flows and low steam pressures (conducive to film boiling). Even though Unit 2 and Unit 3 RSGs are of the same design, FEI in Unit 2 was averted by operation at lower RCS flows, lower steam flows and higher steam pressures (Conducive to nucleate boiling).

Dangers of Unit 2 Restart: Unit 2 at reduced power operation can experience multiple tube ruptures due to tube-to-tube wear and instant burst of existing incubating tube cracks (especially at geometric
discontinuities) caused by fluid elastic instability, flow-induced vibration and Mitsubishi Flowering Effects and flashing feedwater jet impingement during anticipated operational transients (SG Pressurization and opening of main steam line relief valves located outside containment) and main steam line breaks outside containment (SG Depressurization with failure of MSIV to close). During these events, SG will be connected to the environment and 60 tons of radioactive coolant with Iodine-131 and other radioactive gases will escape to the environment with steam. These events are not analyzed in the SONGS FSAR and the impact of radiological releases on SONGS Control Room, Workers, 10-mile EPZ Residents and 50-mile Ingestion Pathway is undetermined.

SCE Unit 2 Return to Service Reports: AREVA, Westinghouse, SCE and NRC AIT Report have not addressed the synergic affects of tube-to-tube wear and high cycle metal fatigue. MHI Fatigue stress calculations are significantly flawed.

Anti-Vibration Bars: MHI designed anti-vibration bars are not capable of stopping the large amplitudes of tubes, violent tube-to-tube impact/sliding/denting, fretting, tube-to-tube wear and high cycle metal fatigue cracks caused by very high void fractions (100%) and high steam velocities (> 50 feet/sec). MHI inadequate and double contact forces and loose/tight supports theory as the difference between Units 2 & 3 damage based on fabrication or manufacturing difference is just a conjecture and based on faulty computer simulations, unproven statistical analysis, hideous testing data (Redacted under the false shield of proprietary information), incomplete and questionable ECT, Visual and Dings/Dents Analysis. A review of literature indicates that there are anti-vibration bar designs available, which are capable of stopping the in-plane vibrations of tubes due to high steam velocities and void fractions, but these are all in the experimental, testing or research stages, and not installed in operating steam generators. The best thing is to avoid these adverse flow regimes by keeping velocities below 20 feet/sec, operate with steam pressures > 950 psi, recirculation ratios
> 4.5 and void fractions around 96.5%.

Tube Inspections: 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 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. Bobbin Coil ECT probes have several limitations: (1) A general inability to permit characterization of identified degradation (e.g., axial, circumferential, or volumetric; single or multiple axial indications; etc.), (2) Relative insensitivity to detecting circumferentially oriented tube degradation, and (3) Limited capability to detect degradation in regions with geometric discontinuities (e.g., expansion transitions, U-bends, and dents) and deposits. Rotating probes generally contain one to three specialized test coils. The coils used in the rotating probe head at a specific unit depend on many factors, including optimizing the coils for detecting the forms of degradation to which a tube may potentially be susceptible. The coils used on a rotating probe include (1) a pancake coil (which is

sensitive to both axially and circumferentially oriented degradation), (2) an axially wound coil (which is sensitive to circumferentially oriented degradation), (3) a circumferentially wound coil (which is sensitive to axially oriented degradation), or
(4) a plus-point coil (which reduces the effects of geometry variations in the tube and is sensitive to both axially and circumferentially oriented degradation). The major disadvantage of the rotating probes is their slow inspection speed (typically less than 1 inch per second). Because of this slow inspection speed, rotating probes are only used at specific locations (e.g., U-bends, sleeves, expansion transitions, dents, locations where there is a bobbin coil probe indication, locations where a more sensitive inspection is needed, and locations susceptible to circumferential cracking).

ASLB News: ASLB concluded in its May 13, 2013 ruling that SCE's Unit 2 Restart Plan, if implemented, would (1) grant SCE authority to operate without the ability to comply with all technical specifications; (2) grant SCE authority to operate beyond the scope of its existing license; and (3) grant SCE authority to operate its replacement steam generators in a manner that constitutes a test or experiment that meets the criteria in 10 C.F.R. § 50.59(c)(2)(viii) for seeking a license amendment. For these three independent reasons, this CAL process constitutes a de facto license amendment proceeding that is subject to a hearing opportunity under section 189a of the AEA. For the foregoing reasons, we resolve the first issue referred by the Commission in the affirmative, concluding that the CAL process for SONGS Units 2 and 3 constitutes a de facto license amendment proceeding that is subject to a hearing opportunity under section 189a of the AEA. Our resolution of the first issue grants Petitioner the relief it seeks in its contention; namely, the opportunity for an adjudicatory hearing incident to the license amendment proceedings for the restart of Units 2 and 3. Petitioner's contention is thus moot, which renders moot the second issue referred by the Commission. The proceeding before this Board is therefore terminated.