CHAIRMAN Resource

From: Bill Hawkins <billlee123456@gmail.com>
Sent: Monday, January 04, 2016 11:23 AM
Subject: [External_Sender] Fwd: "Penny wise and pound foolish", SONGS Unit 3 RSGs in-plane FEI, Dings & Dents & Tube-to-Tube Wear - SCE has no choice but to release the SONGS Units 2 & 3 Operational Data to gain credibilty - SCE/MHI Lawsuit appears to be on Fak...

Low steam pressures are severe for tube vibrations. Somebody within SCE Organization (With MHI's concurrence or not) made a decision to increase to steam pressures and RCS Flow rate in Unit 3 RSGs to minimize ding/dent indications, maintain mechanical damping and thus minimize tube vibrations. AVP Experts across the "Seven Seas" state, "The Primary side flow is higher in Unit 3 RSGs (79E6 lb./hr.) than Unit 2 RSGs (75E6 lb./hr.), which equates to more primary energy transfer to the Unit 3 RSGs secondary side. The OSGs had primary side flow of 72E6 lb./hr. **The importance of this undiscovered/hidden fact during SCE, MHI and NRC investigations is that Unit 3 RSGs had more heat energy transfer from the hottest channels (region of tube-to-tube wear) than the OSGs and Unit 2 RSGs hottest channels to the Unit 3 RSGs secondary side. These adverse phenomena caused elevated steam and void fractions much higher (than the SG average) in Unit 3 RSGs secondary side than the OSGs and Unit 2 RSGs secondary side.** These AVP observations are consistent with a new proprietary nuclear industry study (Impact of High Moisture Carryover on Turbines) published in November 2015, which states, "Core exit steam quality (Note 1 & 2) in 'hot' channels are much higher than the core average, resulting in elevated steam velocity exiting the core."

This experiment fired back, produced dry steam with elevated steam velocities and zero damping, which caused in-plane FEI in Unit 3, in-plane motion of some of the tubes with large amplitudes in the zone of highest void fractions or highest heat flux. These tubes moved in the middle of the AVBs without being restrained by any of the AVBs, made tube-to-tube contact and produced tube-to-tube wear leading to retirement and decommissioning of San Onofre Units 2 & 3.

In-plane FEI did not occur in Unit 2 RSGs because of lower steam pressure and lower RCS Flows. The double tube-to-AVB contact force, more number of dent/ding signals, higher AVB Twist and crooked AVBs unverified theories/analysis/testing results developed by SCE/MHI (ignoring the operational differences between Units 2 & 3) for preventing in-plane FEI in Unit 2 are completely repudiated/rejected based on the reports and interviews listed below.

Some body within the SCE AVB Team, Design Team, Root Cause Team, Shift Managers, Senior Leadership and Operations made that decision to exceed the Unit 3 RSGs Functional acceptance criteria. SCE, MHI, NRC, CPUC and state AG needs to conduct investigation to find out who authorized or made that decision. All the investigations conducted to date have been meaningless or geared in the wrong direction. Truth can be suppressed but not hidden forever like the Hotel Bristol notes and SCE/MHI decision not to inform NRC about their failed efforts to reduce void fractions and improve circulation ratios. This is not a matter of money but a question of nuclear safety.

---------- Forwarded message ----------
From: **CHAIRMAN Resource** <CHAIRMAN.Resource@nrc.gov>
Date: Mon, Jan 4, 2016 at 6:19 AM
Subject: RE: "Penny wise and pound foolish", SONGS Unit 3 RSGs in-plane FEI, Dings & Dents & Tube-to-Tube Wear - SCE has no choice but to release the SONGS Units 2 & 3 Operational Data to gain credibilty - SCE/MHI Lawsuit appears to be on Fake, Fl...
To: Bill Hawkins <billlee123456@gmail.com>

This is to acknowledge receipt of your communication to the Chairman of the U.S. Nuclear Regulatory Commission. Thank you.

Office of the Secretary

U.S. Nuclear Regulatory Commission

From: Bill Hawkins [mailto:billlee123456@gmail.com]
Sent: Monday, January 04, 2016 12:47 AM
Subject: [External_Sender] "Penny wise and pound foolish", SONGS Unit 3 RSGs in-plane FEI, Dings & Dents & Tube-to-Tube Wear - SCE has no choice but to release the SONGS Units 2 & 3 Operational Data to gain credibilty - SCE/MHI Lawsuit appears to be on Fake, Fl...

The nuclear safety question is, "Why is SCE hiding the Units 2 & 3 operational data. Ratepayers have been charged $3.7 Billion by SCE for Units 2 & 3 retirement & operational data. Therefore, Units 2 & 3 operational data are public records and why does not SCE want to release it AVP or post it on the www.songscommunity.com?"

SCE needs to explain why there are alarming differences between San Onofre Units 2 & 3 cycle 16 operational data records found in the San Onofre operator logs and the data found in their plant computer system shown in the NRC AIT Report (per NRC AIT Charter) and the data used by SCE and its Global Consultants in their documents (SCE CDS, SCE Unit 3 Root Cause Evaluation, Westinghouse Operational Assessment, SCE 50.59, MHI Root Cause Evaluation and 2 Papers by Edison Engineers). SCE needs to explain why they refuse to release this data on flimsy grounds, which make no sense, because this data now has no proprietary or monetary value except to hide SCE wrongdoing. SCE needs to explain why it is claiming that the NRC AIT used analytical assumptions to derive the results, and not the actual Units 2 & 3 Operational Data to evaluate the difference between Units 2 & 3 as mandated by NRC AIT Directive that was written by Elmo Collins NRC Region IV Administrator. The NRC is an Official US Government Agency, which does not use fake data and does not lie in its reports. Therefore, SCE statements must be construed to be misleading, erroneous and a futile attempt to hide SCE's design and operational mistakes that were responsible for the failure of the San Onofre Replacement Steam Generators.

The Nuclear Regulatory Commission (NRC) is now involved in a 3-year new investigative experiment costing an undisclosed amount of money funded by the nuclear industry and US ratepayers to determine why in-plane fluid elastic instability (IPFEI) occurred in San Onofre Unit 3's Replacement Steam Generators (RSGs). NRC currently does not know what corrective and regulatory actions are needed for preventing the occurrence of IPFEI in existing/new steam generators to ensure the safety of all American Reactors and their Steam Generators.

Only SCE can tell why SCE decided to exceed the redline of Unit 3 RSGs Functional Testing, Screening & Design Limits to reduce the number of dents/ding within 7% per steam generator in the zone of high void fractions? Was it a safety, design or a financial issue, or all? The San Onofre Units 2 & 3 Operational conditions cannot be reproduced in any laboratory steam generator experimental studies. Since this data is nuclear safety-related and SCE has been paid $3.7 Billion dollars by the public for their mistakes in Units 2 & 3, this data becomes a People's and Public Record. SCE has no legal and moral authority to withhold this data. Since CPUC controls the funds allocated to SCE, CPUC can pursue SCE that It will be in the best public interest of the SCE to post the entire data on www.songscommunity.com website within 30 days of this email as requested by AVP Attorney's Letter to SCE's Managing Attorney, Mr. Douglas Porter.

AVP's Expert Panel's benchmarking of the NRC Licensed and in-plane qualified Palo Verde and ANO-2 RSGs Alloy 690 tubes heat transfer area/thermal megawatt ratio calculations show that as-designed/operated and unlicensed San Onofre's RSGs designed by SCE were capable of only producing 1600 MWT (operational flexibility of ± 2%, 1568 -1632 MWt) of safe thermal power instead of the 1729 MWt thermal power specified by SCE to maximize their profits. AVP Expert's cannot find any documents showing how SCE justified the power of 1729 MWTs in the new RSGs since there was no comparative design/operational, thermal-hydraulic, tube wear/vibrational, stress, fatigue and impact on the balance of plants systems/components analysis done between the OSGs and the RSGs.

After the Unit 3 tube leak and retirement of San Onofre Units 2 & 3, SCE stated, "Mitsubishi Heavy Industries (Mitsubishi) failed to offer any viable, implementable and licensable plan that would safely and reliably restore the RSGs to 100-percent power(1729 MWt) for their promised 40- year operational life." It was impossible for any manufacturer to re-build the RSGs to deliver 100-percent power of 1729 MWt for their promised 40- year operational life designed by SCE due to factors stated above. Since SCE & MHI did not figure out why in-plane FEI occurred in Unit 3, both were unable to come up with a viable, implementable and licensable plan that would safely and reliably restore the RSGs to 100-percent power(1600 MWt) for their promised 40- year operational life. For the same reason, SCE after spending 100's of million dollars utilizing Global Consultants and performing 170,000 tube inspections and plugging hundreds of tubes was unable to convince NRC Atomic Safety Licensing Board, John Large, Dr. Joram Hopenfeld, AVP Experts and Public that Unit 2 RSGs at 70% power (Velocity -12 feet/second, void fractions - 92%, circulation ratio - 4.9) was not safe.

A small change in velocity and steam quality can cause some of the tubes to exceed their critical velocity and cause them to travel them in the in-plane direction with large amplitudes, hit other tubes with violent forces resulting in tube-to-tube wear and high cycle thermal fatigue cracks in Alloy 690 Tubes. Based on the AVP Technical Analysis, NRC AIT Report and Westinghouse Operational Assessment, the void fractions, velocities, steam flow, primary flow, circulation ratio and thermal power in Unit 3 RSGs were 99.6%, 28.3 feet/second, 7.62 Million Lbs./hour, 79.79 Million Lbs./hour, 3.2 and 1729 MWt respectively. Based on the AVP Technical Analysis, NRC AIT Report and MHI Repair Plan, the void fractions, velocities, steam flow, primary flow, circulation ratio and thermal power in Unit 2 RSGs were < 99.3%, < 25 feet/second, 7.59 Million Lbs./hour, 75.76 Million Lbs./hour, 3.5 and 1729 MWt respectively. The increased primary flow rate in Unit 3 RSGs, higher steam pressure, lower circulation ratios and higher feedwater flow rate produced elevated steam velocities and dry steam in a localized area of the hot-leg side with the highest heat flux and a tightly packed tall tube bundle. The elevated steam velocities and dry steam caused in-plane FEI, tube-to-tube wear, a tube leak, 320 failed tubes and uncontrolled out-of-plane vibrations in Unit 3 RSGs. The lower primary flow rate in Unit 2 RSGs, lower steam pressure, higher circulation ratio and lesser feedwater flow rate produced steam velocities lesser than Unit 3 RSG and wet steam (void fractions < 99.0%) instead of dry steam (void fractions > 99.6%). These factors resulted only in uncontrolled out-of-plane vibrations in Unit 2 RSGs and no in-plane FEI/tube-to-tube wear. The double tube-to-AVB contact force, more number of dent/ding signals, higher AVB Twist and crooked AVBs unverified theories/analysis/testing results developed by SCE/MHI (ignoring the operational differences between Units 2 & 3) for preventing in-plane FEI in Unit 2 are completely repudiated/rejected based on the following reports and interviews:

A. **Interviews with Unit 3 Root Cause Team and San Onofre Insiders**

B. **Interviews with members of the NRC AIT Team**

C. **Analyses by AVP's Nuclear Safety Systems Experts across Seven Seas**

D. Analyses by Union of Concerned Scientists

E. **Analyses by Independent Consultants to NRC**

F. **NRC Atomic Safety Licensing Boarding San Onofre Ruling**

G. **Westinghouse Operational Assessment**

H. **Westinghouse and B&WI, Canada studies on egg-crates with lattice grids**

I. **MHI Advanced AVB Testing & Repair Plan**

J. SCE's White Paper: San Onofre Nuclear Plant Replacement Steam Generators

K. **SCE/MHI Meeting Minutes**

L. **Design of NRC Licensed in-plane Palo Verde and ANO-2 RSGs**

M. **Improving like for-like RSGs paper by Bob Olech, P.E, former SCE Heat Transfer Expert**

N. NRC Steam Generator Manual

O. SCE 2001 NRC Approved Original Steam Generators Power Uprate Application

P. AREVA Operational Assessment

Q. **Dr. Pettigrew Research Papers and comments to NRC Commissioners**

Part 1 - MHI Root Cause Analysis and Supplemental Technical Evaluation Report- RSGs Design Basis - March 2013

Early in the project, SCE and MHI formed an AVB Design Team with the goal of minimizing U-bend tube vibration and wear. The AVB Design Team conducted numerous technical and design review meetings. The agreed-upon tube bundle U-bend support design and fabrication were as follows:

- Six (6) V-shaped AVBs (three sets of two) were to be provided between each tube column (12 AVB intersections total around the U-bend).
- Tube and AVB dimensional control, including increasing the AVB thickness was to achieve an effective "zero" tube-to-AVB gap under operating (hot) conditions with gap uniformity and parallelism being maintained throughout the tube bundle. Effective "zero" gap was desirable as an industry practice in order to maximize the effectiveness of the supports. The tube and AVB tolerances were to be tighter than that of any prior MHI SG.
- Excessive preload contact force was to be avoided in order to minimize ding/dent indications, and to maintain mechanical damping and thus minimize tube vibration.

Part 2 - SCE/MHI Review and Technical Meetings, August 17-20, 2005

RCS Flow Rate Design Value: SCE does not want the RCS flow rate to exceed ( ) % and has set this as a warranty penalty threshold (section 1.16.5.6). The CDS (section 3.2.0.2) states that the RCS flow rate should not exceed ( ) %. MHI has selected a ( ) % as the design value (target) necessary to meet the SCE objectives. SCE wants to optimize the design point to provide more margin for steam pressure, so wants MHI to shift the ( ) % target to ( ) %. There was a lengthy discussion of how to deal with this situation. The procedures that will be used to evaluate the RSG performance (i.e. RCS flow rate, RSG steam pressure, etc.) are needed, including a description of how measurement error will be factored into the interpretation of the results. It is MHI's action to propose the procedures that will be developed cooperatively will be factored into the interpretation of the results.

Part 3 - SCE/MHI Technical Meeting Report for the meetings during the week of March 28, 2005

The Tube-to-AVB Gap Influence on. Tube Support ( ) stated that ideal tube support is achieved when the tube is in contact at each of its tube supports. Where the structure must be designed with clearances between tubes and supports, this ideal support condition can still be achieved by one-sided tube support (i.e. the tube is located so the tube-to-support clearance is all on in one side of the tube). Furthermore, when a tube is placed between two AVBs, it is the smaller gap that determines the effectiveness of the tube support. This means the worst-case (i.e. largest possible gap) is when the tube is equidistant from its supports. So, for a U-tube delta-G value of ( )" the maximum gap used to determine the effectiveness of the support and used in the tube wear calculation is half of the total gap (( )).

Part 4 - Improving like for-like RSGs by Bob Olech, P.e., Former SCE Heat Transfer Expert and MHI

A paper published by Boguslaw Olech, P.E., former heat transfer expert, Southern California Edison Company and Tomoyuki Inoue, Mitsubishi Heavy Industries in International Engineering Magazine in January 2012 reprinted from a paper published at ICAPP 2011, 2-5 May 2011, Nice, France (paper 11330) stated, "Even though all design and fabrication challenges were addressed during manufacturing, it was not known if the as designed and fabricated RSGs would eventually perform as specified. To verify this, the RSGs were functionally tested after installation in the plant after unit re-start from the replacement outage. The following essential operating parameters were verified through functional tests. **Heat transfer (steam pressure):** As-designed, the RSGs operating at full (100%) reactor rated power with the reactor coolant temperature at the design point were expected to generate steam whose pressure was to be no less than 816 psia (and no greater than 900 psia) at the steam outlet nozzle. As-tested, one RSG generated steam at approximately 831 psia (5.73 MPa) and the other one at approximately 837 psia. The authors wish to acknowledge all Edison and MHI personnel involved in the SONGS steam generator replacement project for their efforts to make this project a success." All the SCE documents list 833 psi as the operating pressure for both San Onofre RSG Units 2 & 3.

Part 5 – SCE Root Cause Evaluation: Unit 3 Steam Generator Tube Leak/TTW

SONGS Unit 3 started commercial operation in April 1984 with new Combustion Engineering (CE) steam generators. The Unit was safely operated at 100% nominal power for 15 refueling cycles. In October 2010, Unit 3 was taken offline for the Cycle 16 Refueling and Steam Generator (SG) Replacement Outage. On January 31, 2012, Unit 3 was operating at nominal 100% full power and approximately half way through the first operating cycle after SG Replacements. SONGS Operators identified a primary to secondary system leak in Unit 3 SG E088.

SCE's Unit 3 Tube Leak Root Cause Analysis prepared by a team of experts including Bob Olech states, "The facts identified in this analysis indicate that even though the Unit 3 tube bundle components (tubes and [Anti-Vibration Bars (AVBs)]) might have been fabricated and assembled better, the tube-to-AVB as-built gaps might have been in fact larger in the Unit 3 RSGs as suggested by the ECT results. Based on this, it cannot be ruled out that the tube-to-AVB gaps are larger and more uniform in the Unit 3 RSGs than the Unit 2 RSGs. This might have resulted in reduction of the tube-to-AVB contact force and consequently in multiple consecutive AVB supports being inactive. Inactive tube supports might have resulted in tube-to-tube wear."

Part 6 - Unit 3 Root Cause Team and San Onofre Insiders

One Key member of the Root Cause Team disagreed with the results of SCE Unit 3 Root Cause Evaluation and said in June 2012 that the Unit 3 tube leak and tube-to-tube wear was caused in Unit 3 RSGs because Unit 2 & 3 RSGs were operating at different steam pressures and not because the tube-to-AVB contact forces/gaps were different. SCE ignored the warnings of SCE insiders in June 2012 that SCE Unit 3 Root Cause Evaluation was defective and SCE will never get NRC's permission to restart Unit 2. SCE and their Global Consultants reports based on SCE supplied erroneous Units 2 & 3 operational data, showed that Unit 2 & 3 RSGs were operating at the same exact pressure of 833 psi and RCS flow rate of 209,880 gpm. In a meeting with NRC in 2013, NRC AIT Leader Greg Werner confirmed by showing San Onofre Control Room Charts that Unit 3 RSGs were operating at 942 psi and Unit 2 RSGs were operating at 833 psi. NRC AIT Report noted that Unit 3 ran with slightly higher primary temperatures, about 4°F higher than Unit 2. Based on NRC AIT Report, it is assumed that the primary loop volumetric flow rate in Unit 3 RSGs was 104,000 gpm and primary loop volumetric flow rate in Unit 2 RSGs was 102,000 gpm The procedures developed to by MHI to evaluate the RSG performance (i.e. RCS flow rate, RSG steam pressure, etc.) were not used when SCE changed the RCS flow rate and steam pressure of Unit 3 RSGs, which caused in-plane FEI and tube-to-tube wear. AVP has unable to confirm with MHI that the operational changes in Unit 3 RSGs were made with MHI's concurrence. MHI Root Cause states, "This analysis focused on mechanical differences because T/H conditions were expected to be similar." NRC Alt Report states, "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.The above analyses apply equally to Units 2 and 3, so it does not explain why the accelerated fluid-elastic instability wear damage was significantly greater in Unit 3 steam generators. The ATHOS thermal-hydraulic model predicts bulk fluid behavior based on first principals and empirical correlations and as a result it is not able to evaluate mechanical, fabrication, or structural material differences or other phenomena that may be unique to each steam generator. Therefore this analysis cannot account for these mechanical factors and differences which could very likely also be contributing to the tube degradation."The Primary side flow was higher in Unit 3 RSGs (79E6 lb./hr.) than Unit 2 RSGs (75E6 lb./hr.), which equates to more primary energy transfer to the Unit 3 RSGs secondary side. The OSGs had primary side flow of 72E6 lb./hr. (**AVP Note:** Interpolated from the NRC AIT Report - Primary loop volumetric flow rate from U2 -204,400 gpm, U-3, 208,000 gpm, Westinghouse Operational Assessment – 79.78E6 lb./hr., U3 SCE RCE -OSG -198,000 gpm, RSGs- 209,880 gpm, SCE CDS – OSG MWt -1705, RSG MWt -1729, SONGS Steam Generator Manual, Coolant flow rate, each: (U2) 75.76E6 lb./hr.; (U3) 79.79E6 lb./hr. The importance of this undiscovered fact during SCE, MHI and NRC investigations is that Unit 3 RSGs had more heat energy transfer from the hottest channels (region of tube-to-tube wear) than the OSGs and Unit 2 RSGs hottest channels to the Unit 3 RSGs secondary side. These adverse phenomena caused higher velocities and void fractions in Unit 3 RSGs secondary side than the OSGs and Unit 2 RSGs secondary side. These AVP observations are consistent with a new proprietary nuclear industry study (Impact of High Moisture Carryover on Turbines) published in November 2015, which states, "Core exit steam quality in 'hot' channels are much higher than the core average, resulting in elevated steam velocity exiting the core."

Part 7 - NRC AIT Report, July 2012

Per Region IV written instructions, NRC AIT Report stated, "The team reviewed Unit 2 and 3 Cycle 16 operational data records found in operator logs and the plant computer system. The team focused on differences in configuration and operation between Units 2 and 3. The team evaluated full power operational data between Unit 2 and Unit 3 steam generators after each were replaced. From this data the team compared key plant parameters and other indications such as temperature, flow, power, pressure, and vibration and loose parts monitoring alarms. The team reviewed operational differences between Units 2 and 3 in order to gain information and to assess if these differences could have had an impact on the observed differences in the steam generator tube wear between the units. The team performed a number of different thermal-hydraulic analysis of Units 2 and 3 steam generators. The output of the various analyses runs where then compared and reviewed to determine if those differences could have contributed to the significant change in steam generator tube wear. It was noted that Unit 3 ran with slightly higher primary temperatures, about 4°F higher than Unit 2. Other differences were noted in steam and feedwater flow but none of the differences were considered sufficient to significantly affect thermal hydraulic characteristics inside the steam generators. The different analyses included:

- Lower bounding thermal hydraulic analysis using the steam generator base design condition, where primary inlet temperature was 598°F, and an upper bound case where primary inlet temperature was 611°F as identified in Mitsubishi Document L5-04GA021, Revision 3

- Varying steam generator pressures from 833 to 942 psia

- Steam mass flow rates from 7.59 to 7.62 Mlbm/hr

- Primary loop volumetric flow rate from 102,000 to 104,000 gpm, and

- Recirculation ratio from 3.2 to 3.5.

Operational data reported by SCE and its Global Consultants for San Onofre Units2 & 3 RSGs

- Units 2/3 RSGs, steam generator pressure = 833 psia
- Units 2/3 RSGs, steam generator mass flow rate = 7.59 Mlbm/hr
- Units 2/3 RSGs, primary loop volumetric flow rate = 209,880 gpm,
- Units 2/3 RSGs, primary operating temperature (Thot) = 98 degrees F

Part 8 - Why there are differences between SCE Unit 3 RCE and NRC AIT Report

SCE needs to explain to its southern California ratepayers, and the AVP Technical Panel, why there are alarming differences between San Onofre Units 2 & 3 cycle 16 operational data records found in the San Onofre operator logs and the data found in their plant computer system shown in the NRC AIT Report (per NRC AIT Charter) and the data used by SCE and its Global Consultants in their documents (SCE CDS, SCE Unit 3 Root Cause Evaluation, Westinghouse Operational Assessment, SCE 50.59, MHI Root Cause Evaluation and 2 Papers by Edison engineers).

SCE needs to explain why they refuse to release this data on flimsy grounds, which make no sense, because this data now has no proprietary or monetary value except to hide SCE wrongdoing. SCE needs to explain why it is claiming that the NRC AIT used analytical assumptions to derive the results, and not the actual Units 2 & 3 Operational Data to evaluate the difference between Units 2& 3 as mandated by NRC AIT Directive that was written by Elmo Collins NRC Region IV Administrator. The NRC is an Official US Government Agency, which does not use fake data. Therefore, SCE statements must be construed to be misleading, erroneous and a futile attempt to hide SCE's design and operational mistakes that were responsible for the failure of the San Onofre Replacement Steam Generators.

The Nuclear Regulatory Commission (NRC) is now involved in a 3 year new investigative experiment costing an undisclosed amount of money funded by the nuclear industry and US ratepayers to determine why in-plane fluid elastic instability (IPFEI) occurred in San Onofre Unit 3's Replacement Steam Generators (RSGs). NRC currently does not know what corrective and regulatory actions are needed for preventing the occurrence of IPFEI in existing/new steam generators to ensure the safety of all American Reactors and their Steam Generators.

Only SCE can tell why SCE decided to exceed the redline of Unit 3 RSGs Functional Testing, Screening & Design Limits to reduce the number of dents/ding within 7% per steam generator in the zone of high void fractions? Was it a safety, design or a financial issue, or all? The San Onofre Units 2 & 3 Operational conditions cannot be reproduced in any laboratory steam generator experimental studies. Since this data is nuclear safety-related and SCE has been paid $3.7 Billion dollars by the public for their mistakes in Units 2 & 3, this data becomes a People's and Public Record. SCE has no legal and moral authority to withhold this data. It will be in the best public interest of the SCE to post the entire data on www.songscommunity.com website within 30 days of this email as requested by AVP Attorney's Letter to SCE's Managing Attorney, Mr. Douglas Porter. No false excuses are acceptable.

Part 9 – Steam Generator Tubes, Dings & Dents

There is a significant difference between an indication of wear, which could be anything from a scratch to a rub mark, and the potential for failure. The signal can be indicative of a ding, bulge, loss of wall thickness, buff mark, permeability variation and other kinds of tube anomaly? Having a dent in your car door does not make the vehicle unsafe. Similarly, having a dent in steam generator tube does not make the steam generator tube unsafe, unless SCE/MHI had prior knowledge to indicate that dings and dents in zones of high void fractions has the potential to challenge loss of tube wall thickness, mechanical damping or a tube rupture.

Part 10 - San Onofre Unit 3 Replacement Steam Generators, Dings & Dents & TTW

Experience, benchmarking and literature has shown that the steam generators at San Onofre were not designed to prevent in-plane fluid elastic instability in a dry steam environment. SONGS Unit 3 RSGs TTW confirms that void fractions more than 99.3% (dry steam) cause elevated steam velocities initiating in-plane tube motion with large amplitudes, tube-to-tube contact and TTW. Worldwide benchmarking and latest research of operating steam generators including SONGS Unit 2 RSGs confirms that void fractions less than 99.3% (wet steam) do not cause in-plane FEI and tube-to-tube wear. SONGS Unit 2 RSGs void fractions were less than 99%. That is why SONGS Unit 2 RSGs did not experience in-plane tube motion and tube-to-tube wear.

The question SCE should answer to the public, "Why were the void fractions, velocities and operating conditions in Unit 2 and 3 replacement steam generators different?" The void fractions in San Onofre Unit replacement steam generators were > 99.6% because SCE exceeded the redline of Unit 3 RSGs functional acceptance testing criteria to reduce secondary side vibrations to target the number of ding signals within 7% per steam generator due to concerns with zero tube-to-AVB gap and zero tube-to-AVB contact force. As St. Lucie SVP Joseph Jensen says, "Dent/Dings in steam generators do not make the tubes unsafe." To reduce the number of dents/ding within 7% per steam generator and maximize the heat generation to meet the SG Technical specification limits of 3458 MWt, SCE raised the Unit 3 RSGs steam pressure, primary flow rate and feedwater flow rate, which caused dry steam with elevated steam velocities and hydro-dynamic pressures. These adverse operating conditions caused the tubes to move in the in-plane direction with large amplitudes due to zero damping, making tube-to-tube contact, one tube leak, 2 non-leaking tubes and TTW in 160 tubes. This experiment by SCE of exceeding the Unit 3 RSGs functional acceptance testing criteria to reduce vibrations to target the number of ding signals within 7% per steam generator was done without any T/H, FIV, tube wear and vibrational analysis. SCE did not identify any need to exceed the Unit 2 RSGs functional acceptance testing criteria to reduce vibrations to target the number of ding signals within 7% per steam generator due to larger tube-to-AVB gaps and higher tube-to-AVB contact force. Small increases in tube-to-AVB gaps and the contact force has the potential to decrease the tube-to-AVB wear (number of dings & dents) and increase the number of out-of-plane active supports but not susceptibility to in-plane FEI by increasing the number of active in-plane supports and in-plane compressive/friction forces required to pin down and clamp in-plane of the tubes. The small differences in the steam quality between Unit 3 and Unit 2 RSGs (causing elevated steam velocities, dry steam and zero damping in Unit 3 RSGs) explains the differences in significant tube-to-tube wear in Unit 3 RSGs and zero tube-to-tube wear in Unit 2 RSGs. That is why SONGS Unit 2 RSGs experienced 3 times more dings/dents than Unit 3 RSGs, but did not experience in-plane tube motion and tube-to-tube wear. This statement is consistent with Westinghouse Operational Assessment, SCE/MHI Meeting Notes & MHI Repair Plan and completely rejects/refutes the highly speculative and bogus dent/ding signals, twist, loose supports and manufacturing improvements theories promoted by SCE, MHI and AREVA based on unverified computer, statistical, testing and length of time observation comparison models.

The numerous technical papers reports prepared by Nuclear Safety Systems Experts across Seven Seas and others also evaluated the unexpected tube-to-AVB wear observed in the Unit 2 and Unit 3 RSGs. The evaluation has led to the conclusion that the thousands of tube-to-tube wear indications in Unit 3 RSGs were caused by the presence of thousands of small tube-to-AVB gaps full with dry steam caused by higher primary side flow and lower circulation ratios in Unit 3 RSGs. The number of Unit 2 tube-to-AVB dings/dent indications are three times more than that of Unit 3 due to Unit 2 RSGs small tube-to-AVB gaps full with lower void fraction wet steam and lower steam velocities. This is consistent with the fact that the Unit 2 RSGs have lower void fractions and velocities than the Unit 3 RSGs due to lower primary flows and higher circulation ratios.

Part 11 – Analyses by AVP's Nuclear Safety Systems Experts across Seven Seas

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1. There are a lot of contradictions on everyone's part, SCE, MHI and NRC.

2. The Primary side flow is higher in Unit 3 RSGs (79E6 lb./hr.) than Unit 2 RSGs (75E6 lb./hr.), which equates to more primary energy transfer to the Unit 3 RSGs secondary side. The OSGs had primary side flow of 72E6 lb./hr. (**AVP Note:** Interpolated from the NRC AIT Report - Primary loop volumetric flow rate from U2 -204,400 gpm, U-3, 208,000 gpm, Westinghouse Operational Assessment – 79.78E6 lb./hr., U3 SCE RCE -OSG -198,000 gpm, RSGs- 209,880 gpm, SCE CDS – OSG MWt -1705, RSG MWt -1729, SONGS Steam Generator Manual, Coolant flow rate, each: (U2) 75.76E6 lb./hr.; (U3) 79.79E6 lb./hr. **The importance of this undiscovered fact during SCE, MHI and NRC investigations is that Unit 3 RSGs had more heat energy transfer from the hottest channels (region of tube-to-tube wear) than the OSGs and Unit 2 RSGs hottest channels to the Unit 3 RSGs secondary side. These adverse phenomena caused higher velocities and void fractions in Unit 3 RSGs secondary side than the OSGs and Unit 2 RSGs secondary side.** These AVP observations are consistent with a new proprietary nuclear industry study (Impact of High Moisture Carryover on Turbines) published in November 2015, which states, "Core exit steam quality (Note 1 & 2) in 'hot' channels are much higher than the core average, resulting in elevated steam velocity exiting the core."

AVP Explanation 1: Steam Quality - Enclosure 2, SONGS Unit Return to Service Report, www.songscommunity.com, states, "Steam quality, defined as mass fraction of vapor in a two-phase mixture, is an important factor used in determining SRs [AVP NOTE: SRs, or Stability Ratios > 1 indicates in-plane fluid elastic instability, Unit 3 RSGs SR > 1 (in-plane fluid elastic instability), Unit 2 RSGs SR < 1 (No in-plane fluid elastic instability)]. Steam quality is directly related to void fraction for a specified saturation state. This description is important when considering effects on damping. Damping is the result of energy dissipation and delays the onset of FEI. Damping is greater for a tube surrounded by liquid compared to a tube surrounded by gas. Since quality describes the mass fraction of vapor in a two-phase mixture, it provides insight into the fluid condition surrounding the tube. A higher steam quality correlates with dryer conditions and provides less damping. Conversely, lower steam quality correlates with wetter conditions resulting in more damping, which decreases the potential for FEI. Steam quality also directly affects the fluid density outside the tube, affecting the level of hydrodynamic pressure that provides the motive force for tube vibration. When the energy imparted to the tube from hydrodynamic pressure (density times velocity squared or ρv2) is greater than the energy dissipated through damping, FEI will occur. When steam quality decreases, the density of the two-phase mixture increases, decreasing velocity. Since the hydrodynamic pressure is a function of velocity squared, the velocity term decreases faster than the density increases. Small decreases in steam quality significantly decrease hydrodynamic pressure and the potential for FEI." (AVP Note: The small difference in the steam quality between Unit 3 and Unit 2 RSGs (causing elevated steam velocities, dry steam and zero damping in Unit 3 RSGs) explains the differences in significant tube-to-tube wear in Unit 3 RSGs and zero tube-to-tube wear in Unit 2 RSGs due to consistent with the new industry study and analysis by Anonymous AVP Experts across the "Seven Seas". This statement is consistent with Westinghouse Operational Assessment, MHI Repair Plan and design of world's largest steam flows and NRC licensed, in-plane qualified RSGs with 15% higher power than San Onofre (15% shared owned by SCE).

AVP Explanation 2: Void Fractions - MHI Root Cause Evaluation, page 23 states, "The highest void fraction is located in the (Added by AVP: Unit 3 RSGs) U-bend region, where the maximum value is estimated by ATHOS to be 99.6% (0.4% of the volume is occupied by saturated liquid water). The highest void fraction calculated using ATHOS for prior MHI-designed SGs was 98.5%. The higher void fraction is a result of a large and tightly packed tube bundle and the relatively high heat flux in the upper hot leg side of the tube bundle. The Unit 2 and Unit 3 RSGs have identical operating conditions and the displayed thermal hydraulic results are applicable for all four SONGS RSGs." (/AVP Note: 0.4% of the volume occupied by saturated liquid water is entrapped or entrained in dry steam. This 0.4% water is not available, resulting in no liquid film on the surface of the hot primary reactor coolant tubes for cooling of the tubes to transfer the primary energy to the secondary side (defeats the function of the steam generator) and prevent film boiling. MHI Statement and Root Cause Analysis regarding Unit 2 and Unit 3 RSGs having identical operating conditions is incorrect because SCE supplied wrong Units 2 & 3 operational data to MHI as discussed previously on page 1 and NRC AIT Report. MNES President Yoshinobu Shibata like SCE and NRC has failed to respond to AVP and San Diego Channel 10 News Reporters inquiries regarding the validity of SCE/MHI (including their global consultants) Printed Reports and Press Statements, "The Unit 2 and Unit 3 RSGs have identical operating conditions." The Unit 2 and Unit 3 RSGs different operating conditions as recorded in the San Onofre Operator Logs and Plant Computer System as reported in the NRC AIT Report per AIT Charter, witnessed by AVP Member in meeting with the NRC AIT Team and reported by San Onofre Root Cause Team Guru only explains the difference in tube damage and behavior between Units 2 & 3 and nothing else as imagined/rationalized by SCE and its global consultants, NRC and MHI).

3. "The change from the lattice structure (**AVP Note 1:** To trefoiled drilled plate tube support plates, along with the removal of stay cylinder and the addition of more additional longer tubes in order to increase the thermal performance of RSGs to 1729 MWt [OSGs = 1705 MWT] done without a through NRC 50.90 License Amendment) added additional frictional resistance and in turn inhibited adequate circulation and recirculation flow."

AVP Note 2: Along with ignoring the impact of this significant design change, SCE exceeded the operational redline of Unit 3 RSGs functional testing limitations by boosting the power of primary coolant by 5%, reducing the wall thickness of the tubes by 10%, increasing the RCS Temperature by 10%, changing the steam pressure from 833 psi to 942 psi and increasing the feedwater flows which resulted in an additional 15,000 cubic feet/hour of steam. These changes explain the occurrence of regions of steam dry out and high steam velocities experienced in the middle elevations of the Unit 3 RSGs tube bundle which caused in-plane FEI and tube-to-tube wear. Unit 3/Unit 2 RSGs calculated secondary pressure loss is 58-54 psi (based on an under-prediction of velocity by Mitsubishi a factor of 3 due to calculation of incorrect pressure loss coefficients and flow area per Mitsubishi and NRC Reports). SCE OSG specified secondary pressure loss in CDS was 36 psi, SCE RSG specified secondary pressure loss in CDS was 19 psi. SCE's California Registered Engineers should have caught this mistake at the design stage in 2005, which could have averted the whole San Onofre RSGs Debacle. SCE Engineers did not catch this mistake even in 2012 during the preparation of Unit 3 Tube Leak Root Cause Evaluation (AVP feels because it would have identified SCE design and operational failures). NRC ASLB pointed to these adverse design changes between OSGs and RSGs performed without a NRC License Amendment in May 2013. After spending hundreds of millions of dollars on global consultants and performing 170, 000 tube inspections, SCE should have realized their design/operational mistakes. Instead of admitting their mistakes, SCE first secured their "behind closed doors" $3.3 Billion Dollar San Onofre Settlement profit with the CPUC. AVP Note: The CPUC later fined SCE $16.7 million (for unreported communications about the talks that utility representatives had with regulators over the closed San Onofre nuclear plant) which was only about 1/2 of one percent of SCE's total settlement profit of $3.3 billion. SCE then announced the preplanned retirement of both San Onofre Units 2 & 3 on June 7, 2013, blaming NRC for regulatory delays, MHI for RSGs manufacturing failures and economic uncertainty for their RSGs repairs/operation. This adverse and significant AVP finding is consistent with Westinghouse, Babcock & Wilcox and AVP Technical Reports. SCE response to NRC finding was that since none of the other

Mitsubishi Customers did check the velocities, SCE was justified in not checking the Mitsubishi calculated velocities and void fractions. Based on a review of SCE/MHI Meeting Notes, AVP concludes that SCE was more interested in checking Mitsubishi's cost and schedule for maximizing the profits from new RSGs.

4. "An interesting point noted was the middle of the tube bundles is now occupied by more tube bundles which inhibits the recirculation flow that the old SG would have had. In the old OSGs, the void space above the stay cylinder would help not only in redistribution of the flow on the inner and outer portions of the tube bundles but also provided the adequate cooling of the bundles and provided the necessary damping that was missing in the upper portions of the RSGs."

AVP Note: Based on the Westinghouse Operational Assessment (Enclosure 2, San Onofre Nuclear Generating Station Unit 2 Return to Service Report, Tables 8-2 & 8-3,www.songscommunity.com), "the void fractions, velocities and thermal power in OSGs were 96.1%, 22.8 feet/second and 1709 MWt respectively." Based on the AVP Technical Analysis, NRC AIT Report and Westinghouse Operational Assessment, the void fractions, velocities, steam flow, primary flow, circulation ratio and thermal power in Unit 3 RSGs were 99.6%, 28.3 feet/second, 7.62 Million Lbs./hour, 79.79 Million Lbs./hour, 3.2 and 1729 MWt respectively. Based on the AVP Technical Analysis, NRC AIT Report and MHI

Repair Plan, the void fractions, velocities, steam flow, primary flow, circulation ratio and thermal power in Unit 2 RSGs were < 99.3%, < 25 feet/second, 7.59 Million Lbs./hour, 75.76 Million Lbs./hour, 3.5 and 1729 MWt respectively. Therefore, by removing the stay cylinder and adding more tubes in the RSGs, SCE not only reduced the recirculation ratios but also the cooling and damping in the upper portions of the RSGs. These adverse SCE design and operational parameters resulted in IPFEI and uncontrolled out-of-plane random vibrations which together destroyed the almost new RSG's.

5. "MHI proposes to increase the thickness of the AVB bars as well as increase the size of the retainer bar. The influence of the slender bars would influence the natural frequency and the additional mass would most certainly help prevent movement. I believe this was a major contributor of their tube failure because they did not adequately restrain the tubes in the upper portion. Movement in the upper portion most likely translated to wear in the lower portions of the tube bundles."

AVP Note: These comments are consistent with AVP Technical Panel Research, Dr. Pettigrew's studies, design of in-plane qualified PVNGS RSGs and the MHI repair Plan to eliminate in-plane fluid elastic instability and control out-of-plane random vibrations.

6. "We are trying to reconcile the difference in the operational parameters between SG2 and SG3. The primary side and secondary side have differences...do you know why if the RSG were identical that they would have such large differences in operational parameters?"

AVP NOTE: The large differences in operational parameters between Unit 2 and 3 Unit 3 RSGs were caused by SCE's increasing the steam pressure to reduce the differential pressure between the primary and secondary side to minimize the tube vibrations and the number of dings/dents (tube wear) in Unit 3 RSGs due to zero gaps and low tube-to-AVB contact force in the zone of high void fractions. Another SCE benefit of these changes was to increase the turbine efficiency due to higher steam saturation temperature and more amount of steam for higher electrical power output to the grid. Left unsaid was that this higher output also generated about $6 million per year in additional profits for SCE (based on turbine textbooks and conversations with San Onofre shift managers). The zero gaps and low tube-to-AVB contact forces as designed in Unit 3 RSGs was implemented because of SCE's AVB Design Team's inability to reduce high void fractions and improve circulation ratios due to the priority to increase output power due to these of significant design changes and SCE's decision not to inform NRC. MHI implemented these untested and unverified changes with SCE's approval without any thermal-hydraulic, tube wear and vibration analysis in order to focus on better control of the AVB and tube fabrication dimensions in the Unit 3 RSGs in order to maintain SCE's manufacturing timeline. SCE and

Mitsubishi engineers wrote in a joint paper published in France in May 2011, "Even though all design and fabrication challenges were addressed during manufacturing, it was not known if the as designed and fabricated RSGs would eventually perform as specified. To verify this, the RSGs were functionally tested after installation in the plant after unit re-start from the replacement outage. The following essential operating parameters were verified through functional tests. Heat transfer (steam pressure). As-designed, the RSGs operating at full (100%) reactor rated power with the reactor coolant temperature at the design point were expected to generate steam whose pressure was to be no less than 816 psia (and no greater than 900 psia) at the steam outlet nozzle. As-tested, one RSG generated steam at approximately 831 psia (5.73 MPa) and the other one at approximately 837 psia. The authors wish to acknowledge all Edison and MHI personnel involved in the SONGS steam generator replacement project for their efforts to make this project a success."

7. We are curious on the level in the SG. From the SG procedure, the steam separator was partially submerged. There is a rotating vane that is supposed to separate the liquid vapor mixture our experience is that it should see only two phase flow and not saturated fluid hence it loses its effectiveness. Was there any issues found on the turbine blades when they were inspected i.e. more noise during operation, lower power output, etc."

AVP Note: Only SCE can tell the people of Southern California, AVP Technical Panel and CPUC if there any issues found on the Unit 3 turbine blades when they were inspected (i.e. more noise/vibration during operation, damage or lower power output, etc., which would have indicated abnormal operational conditions that should have been immediately investigated by SCE.

Part 12 - Nuclear Industry Expresses Doubts About SCE RCE/MHI RCA & STER, NRC AIT Report & NRC SONGS Lessons Learned

In 2015, "While discussing their response to the degradation observed at San Onofre, the industry indicated that it could not do a formal lessons learned evaluation since much of the information is not publicly available (i.e., it is proprietary); however, they did indicate that testing regarding the major cause of degradation (i.e., in-plane fluid-elastic instability) was warranted and was being pursued. The industry will be working with various vendors to determine an appropriate test matrix with a targeted completion date for this matrix in March 2015. The testing will be done in Canada at facilities where the Canadians are doing some of their own testing on this phenomenon. The series of tests proposed by Canadian Nuclear Labs is a phased project and will be finished in approximately 3 years. Prediction of the final solution is difficult (new Connor's constant, more effective supports, etc.). Goal is to understand what leads to the onset of in-plane fluid elastic instability. Utilities may be able to use results to avoid operating in these regimes. SG designers may be able to understand their margins in operating SGs and avoid it with new designs (Reference: NRC ADAMS ML15043A610 & ML1506400)."