From: Bill Hawkins <billlee123456@gmail.com>
Sent: Wednesday, January 06, 2016 11:24 AM
Subject: [External_Sender] AVP/DAB invites comments from NEI, UCS, SCE, MHI, Global Consultants and NRC on the Root Cause of San Onofre's Unit 3 Replacement Steam Generators Tube Leak to enhance nuclear safety understanding of SG Tube Ruptures due to in-plane f...

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Flag Status: Flagged

Root Cause of San Onofre's Unit 3 Replacement Steam Generators Tube Leak

Introduction: SONGS was a two-reactor Pressurized Water Reactor (PWR) nuclear power plant (NPP) located in California, USA. SONGS consisted of two twin units (unit 2 and unit 3) each rated at 3358 MWt (1180 MWe). SONGS was majority owned and operated by Southern California Edison Company (Edison). SONGS unit 2 began commercial operation in 1983 and unit 3 in 1984.

Each of the SONGS units were originally equipped with two CE Model 3340 recirculating steam generators. The OSGs were designed for a 40-year service life. Over the years of operation of the PWR plants, it became evident that the steam generator tubes, made predominantly of Alloy 600, were susceptible to inter-granular attack (IGA) and primary water stress corrosion cracking (PWSCC). These corrosion mechanisms were resulting in tube degradation necessitating plugging large numbers of tubes. In addition, the SONGS OSG design has shown to be susceptible to tube through-wall wear and severe corrosion of the tube supports. It became evident that the OSGs would have to be replaced much sooner than stipulated by their design service life.

In November 2001, SCE formed a team to study the viability of replacing the Unit 2 and Unit 3 original steam generators. SCE performed an assessment of six steam generator vendors, which included vendor benchmarking, development of the replacement steam generator design specifications, a steam generator request for proposal, and a steam generator bid evaluation. In September 2004, SCE selected Mitsubishi Heavy Industries (Mitsubishi) as the manufacturer of the replacement steam generator.

After fabrication of the Unit 2 steam generators was complete in April 2008, Mitsubishi performed hydrostatic pressure tests of the primary and secondary sides of the Unit 2 steam generators. In July 2008, after completion of the hydrostatic pressure tests, AREVA performed the baseline eddy current pre-service examinations of the Unit 2 steam generators at the Mitsubishi facilities in Japan. The final inspections for the Unit 2 steam generators were completed in September and October 2008, followed by filling the primary and secondary sides of the Unit 2 steam generators with nitrogen. The Unit 2 steam generators were shipped from the Mitsubishi facilities in December 2008 and received on site at SONGS in February 2009. In July 2009, AREVA performed the final eddy current pre-service examination on the Unit 2 steam generators at SONGS. The baseline and final eddy current pre-service examinations were performed in Japan and at SONGS, respectively, to assess whether any changes to the steam generator tubing had resulted from shipping, and no changes were identified. The Unit 2 steam generators were installed during a refueling outage, between September 2009 and April 2010. On April 13, 2010, Unit 2 returned to power operations.

The fabrication of the Unit 3 steam generators 3E0-89 and 3E0-88 was completed between September 2004 and April 2010. The design specifications of the Unit 2 and Unit 3 steam generators were the same when the contract between the licensee and Mitsubishi was signed; however, due to a fabrication issue, there was a modification to the divider plate-to-channel head weld requirements for the Unit 3 steam generators, and to the classification of the Unit 3 tubesheet material. In March 2009, after initial fabrication of the Unit 3 steam generators was complete, Mitsubishi performed hydrostatic pressure tests of the primary and secondary sides of the Unit 3 steam generators. After completion of the hydrostatic pressure tests, a visual inspection of the steam generator reactor coolant side revealed cracks in the welds that join the divider plate to the channel head of both steam generators.

The repairs to the Unit 3 steam generators were completed in late March (3E0-89) and early April (3E0-88) of 2010. In late April 2010, the Unit 3 steam generators passed the primary hydrostatic pressure re-tests. In June 2010, AREVA performed a final eddy current pre-service examination of the steam generators at the Mitsubishi facilities in Japan, and this was used as the baseline pre-service examination for the Unit 3 steam generators. The steam generators were shipped from the Mitsubishi facilities in Japan in early August 2010, and arrived at SONGS in early October 2010. The Unit 3 steam generators were installed during a refueling outage, between October 2010 and February 2011. On February 18, 2011, Unit 3 returned to power operations.

Unit 2 was shut down for a scheduled refueling outage on January 10, 2012. Steam generator tubing inspections in steam generator 2E0-89 found unexpected wear caused by retainer bars on two tubes that required plugging in accordance with the technical specifications. Steam generator tubing inspections in steam generator 2E0-88 found wear on four tubes that required plugging in accordance with the technical specifications. Anti-vibration bars caused the wear on two of the tubes and retainer bars caused the wear on the other two tubes. Because of the unexpected wear, the licensee preventatively plugged 94 tubes in steam generator 2E0-89 and 98 tubes in steam generator 2E0-88. Fifteen of the tubes in steam generator 2E0-89 were stabilized prior to plugging, and 18 of the tubes in steam generator 2E0-88 were stabilized prior to plugging.

On January 31, 2012, Unit 3 control room operators received an alarm that indicated a primary-to-secondary reactor coolant leak from steam generator 3E0-88. The alarm received was from the main condenser air ejector radiation monitors, which continuously samples from a vent line for the purpose of rapidly identifying steam generator tube leaks. Although the leak rate was small, it increased enough in a short period of time for the SCE to perform a rapid shutdown. The plant never came back on line.

Fluid Elastic Instability: In a tube array, a momentary displacement of one tube from its equilibrium position will alter the flow field and change the forces to which the neighboring tubes are subjected, causing them to change their positions in a vibratory manner. When the energy extracted from the flow by some of the tubes due to higher velocity and higher void fraction dry steam exceeds the energy dissipated by damping by the tubes and its supports, it produces Fluid Elastic Instability.

You boil nearly the same amount of water in two kettles. One kettle is supplied with more heat than the other one. The water in the kettle with more heat will boil first with higher velocity and higher void fraction dry steam than the other kettle. Due to sever operational conditions, the San Onofre Unit 3 RSGs had more primary heat, higher velocity and higher void fraction dry steam compared to the Unit 2 RSGs milder operational conditions, which had less primary heat, lower velocity and lower void fraction wet steam. Higher velocity and higher void fraction dry steam caused in-plane fluid elastic instability and tube-to-tube wear in Unit 3 RSGs. Lower velocity and lower void fraction wet steam did not cause in-plane fluid elastic instability and tube-to-tube wear in Unit 2 RSGs.

MHI Root Cause states, "The evaluations found that the average contact force in the Unit 3 RSGs was smaller than the average contact force in the Unit 2 RSGs. The difference in the contact forces between the Unit 2 and Unit 3 RSGs is caused by the reduction in dimensional variations during the manufacture of the Unit 3 RSGs, mainly due to improvement of the control over tube and AVB dimensions in the manufacture of the Unit 3 RSGs. The reduced contact forces resulted in far more tubes in the Unit 3 RSGs experiencing tube-to-tube wear than those in the Unit 2 RSGs." Based on actual advanced testing of the AVBs in MHI Repair Plan and review of Palo

Verde RSGs tube supports, AVP/DAB rejects the speculative and assumed assumptions of MHI Root Cause, SCE and NRC Reports. AVP/DAB Safety Team's Global Experts findings are based on universal laws of energy, physics and worldwide steam generator operational experience and cannot be refuted by any SCE, MHI or NRC Technical Panel and CPUC/Court of Law.

This analysis pertains to the Root Cause of the damage caused by in-plane fluid elastic instability (IPFEI), which occurred in San Onofre's Unit 3 Replacement Steam Generators (RSGs). It has taken AVP/DAB Safety Team (AVP/DAB) almost 4 years to complete, using "critical questioning and an investigative attitude" and includes the help of many independent experts from around the globe. These experts have exceptional academic credentials in nuclear, chemical, mechanical, heat transfer, fluid and steam generator engineering. Parts of these analyses have been sent to the Nuclear Regulatory Commission (NRC), Southern California Edison (SCE), Mitsubishi Heavy Industries (MHI) and Administrative Law Judge (ALJ) Melanie Darling's appointed California Public Utilities Commission (CPUC) Safety Engineer. The NRC, SCE, MHI and the CPUC *have to date been unable, unwilling and/or are for whatever reason not in a position to refute the findings* of the AVP/DAB Safety Team.

Summary: AVP/DAB's global experts completely disagree with:

1. The inconclusive and speculative studies prepared (at the cost of hundreds of millions of dollars to ratepayers) by SCE's SG Recovery and Root Cause Evaluation (RCE) Teams (which included the services of [friendly] industry experts in SG design, manufacturing, operation, and repair to [supposedly] ensure a complete understanding and verification of the condition, extent, cause, and corrective actions required).

2. The questionable evaluations prepared by MHI and the NRC Augmented Inspection Team (AIT) on the root cause of the failure of San Onofre's new RSGs. These reports were primarily meant to obscure SCE's design flaws while also covering up SCE's operational errors, thereby helping SCE and SDG&E collect billions of dollars from ratepayers, thanks to the CPUC's *back room* settlement and its own flawed RSG investigation.

AVP/DAB will show, quoting from SCE's own documents (and others), that the new Unit 2 & 3 RSGs were not designed to prevent in-plane fluid elastic instability in regions of dry steam during their operation, a condition which SCE then subjected them to, which caused them to fail so soon after installation, instead of lasting 40 or more years as promised by SCE.

The Root Cause was SCE's failure to properly design and operate San Onofre's RSGs

The legality of the CPUC-SCE San Onofre *Settlement* is now in question, since it must be based upon factual information about how SCE designed the new San Onofre Units 2 & 3 Replacement Steam Generators (RSGs), how they operated them, and what exactly caused them to fail so soon after being installed, since SCE designed them to last 40 or more years.

Specifically, SCE needs to explain why the cycle 16 operational data found in the San Onofre operator logs and the plant computer system is significantly different for Units 2 & 3 RSGs from what SCE and its global consultants used in their justification documents ([assumed the same data for Units 2 & 3 RSGs] - SCE CDS, SCE Unit 3 Root Cause Evaluation, Westinghouse Operational Assessment, SCE 50.59, MHI Root Cause Evaluation and two papers by Edison engineers). AVP/DAB believes that the data proves that SCE is trying to use the same unverified argument for the difference in data between Units 2 and 3 being "like-for-like" as it used previously when they said that the RSGs were "like-for-like" to the original steam generators (OSGs), a comparison that was prima facie proved wrong (at the cost of billions of dollars) when Unit 3's RSG sprung a radioactive leak. This technical data is now legal evidence because it proves that SCE committed multiple design and operational errors which violated the federal 10 CFR 50.59 and 50.90 regulations, Appendix A to 10CFR 50, General Design Criterion 14 and RSGs functional testing acceptance criteria. This data (identified by the NRC) and the MHI Repair Plan were kept secret by SCE. The operational data, MHI Repair Plan along with the analysis by AVP/DAB Experts explains what caused the San Onofre Unit 3 tube leak of radioactive reactor core coolant on January 31, 2012. The data also identifies the Root Cause of the thousands of damaged tubes in both San Onofre Units 3 and Unit 2 RSG's which was termed by NRC as a "very serious" safety issue. As a result of SCE's design and operational errors, <u>the almost-new San Onofre RSG's had more damaged and/or plugged tubes than the rest</u>

of the US power plants combined, which is unprecedented in the history of the U.S. Operating Nuclear Fleet.* *Credit: SanOnofreSafety.org

It is important to note that all this ongoing tube damage was unknown to SCE because SCE did not pay attention to the extra vibration monitoring system alarms in Unit 3 RSGs while operating them with higher primary temperatures, both of which were contributing indicators to in-plane fluid elastic instability. This RSG damage was only discovered during the Unit 3 Leak Root Cause investigation. SCE Root Cause states, "There were four factors associated with the VLPMS that indicate it may have been capable of detecting the tube-to-tube vibration that was occurring during the cycle: 1. Multiple alarms on various VLPMS channels after new SGs were installed in R3C16. 2. Primary side of the SG were inspected in U3C16, and no indication of loose parts were found on 3E088 (NMO 800842826) or 3E089 (NMO 800842830). This indicates that the alarms could have been initiated by secondary side noise. 3. Westinghouse Impact Analysis of Unit 3 determined impacts to be metal to metal (Attachment 3 Westinghouse, "SONGS Unit 3 Impact Analysis" ITS3206 Rev. 0)." 4. Valid alarms were seen in Unit 3 and not in Unit 2 during the C16 operating cycle. The valid alarms could have been caused by the tube to tube contact. September 8 – Reactor trip on both Units due to system disturbance. September 11 – Unit 3 return to full power."

AVP/DAB Comment: San Onofre's RSG failures illustrates that the intense high surface temperatures in the RSGs steam generator tubes were caused by localized film boiling (the surface of the SG tubes were covered with a blanket of dry steam instead of wet steam which aids in heat transfer and tube stability). This was caused by inadequate circulation ratios, higher primary flows and low feedwater flow regions in the areas with the highest primary heat flux inside a tightly packed tube bundle because the RSG's were designed for "additional" power generation instead of RSG longevity. This faulty RSG design, along with SCE "experimental" operation led to the radioactive leak in Unit 3 as well as all the tube damage in both Unit 2 and Unit 3. This finding is consistent with

NRC Steam Generator Manual and Boiling Heat Transfer Text Books, which state, "If circulation is inadequate, the heat transfer surfaces tend to become blanketed with steam rather than continuously wetted by a steam/water mixture. This significantly reduces the heat transfer of the steam generators." A European Research Paper on film boiling states, "Normally the tube surfaces are effectively cooled by feedwater due to nucleate boiling (surface of tubes covered with a thin film of liquid feedwater). However, when the heat flux exceeds a critical value the heat transfer from the tube surface into the feedwater deteriorates, with the result that a drastically increased tube surface temperature occurs. The mechanisms of critical heat flux are: (a) Formation of hot spots under a growing bubble. Here when a bubble grows at the heated wall a dry patch forms underneath the bubble as the micro-layer of liquid under the bubble evaporates. In this dry zone, the wall temperature rises due to the deterioration in heat transfer, and (b) Near-wall bubble crowding and inhibition of vapor release. Here a "bubble boundary layer" builds up on the surface and vapor generated by boiling on the surface must escape through this boundary layer. When the boundary layer becomes too crowded with bubbles, vapor escape is impossible and liquid cannot penetrate to the heated wall and cool it, the surface becomes dry and overheat gives rise to burnout." NRC AIT Report states, "It was noted that Unit 3 ran with slightly higher primary temperatures, about 4°F higher than Unit 2. SCE Engineering personnel also compared hot leg temperature changes linked to Unit 3 operations from February 18, 2011, to January 31, 2012, and confirmed about 30 valid alarms during this period were not associated with thermal transients." The higher primary temperature in Unit 3 was another contributing indicator that should have alerted the operator that some abnormal transient was occurring in Unit 3 RSGs, which was absent in Unit 2.

EMAIL TO SCE MANAGEMENT

From: [Redacted]

Date: June 10, 2012 2:01:46 PM PDT

To: [Redacted]

Cc: [Redacted]

Subject: SONGS Steam Generators Tube leaks - CONFIDENTIAL EMAIL-

Please do not discuss without [Redacted] Permission - Thank You

Dear [Redacted]

I decided to Nominate SONGS SVP/CNO and SLT because of their open, warm and facilitative leadership, which is unprecedented in the History of SONGS. Their continuous commitment to the Nuclear Safety Culture, SONGS Excellence Guidebook and concern for Operational Excellence and Public Health & Safety reinforces and strengthens that belief every day. My candid discussions with [Redacted] on various plant issues have been helpful in generation of several Nuclear Notifications. I have been researching for weeks quietly at home on my time on the Unit 3 Steam Generator E088 Tube Leak. This includes review of SG RCE, reports by Environmental Groups, SG Problems and Solutions on US, European and Russian Websites (including MHI, CE, Westinghouse, Babcock & Wilcox, Framatome, Tihanga, AREVA, and others) extensive discussions with my friends, who specialize in Thermodynamics, Heat Transfer and Fluid Flow and some plant personnel. I may have the reasonable cause and potential solution for these leaks. Just trying to help MHI & SONGS. I would like to meet with you and [Redacted], sometimes next week in MESA to discuss confidentially my ideas on the Unit 3 Steam Generator E088 Tube Leak.

Thanks and Sincerely [Redacted]

SONGS Anonymous Nuclear Notification - 202016239 Lack of candor related to SONGS restart - 06/08/2012 07:49:26 (GBU_NN_USER)

Problem description:

The Senior Leadership Team and the managers of the SG Recovery Project have failed SONGS and the employees of SONGS. For weeks, rumors were rampant that SONGS would not restart until late in the summer but all schedules and information stated the close breaker date for Unit 2 was July 1 with Unit 3 following a few weeks later. Employees were asked to keep working the same accelerated pace to meet schedule dates, even when asked to address the rumors. Then, after leaving work yesterday evening and arriving home, I find out that the local news reported SONGS would remain shut down until after the summer months. Yet no information was provided by the SLT at SONGS nor was any communication provided by Rosemead. Clearly the SLT and the SG Recovery managers do not trust their employees and do not respect or value the work performed to date. Additionally, the work performed to date for the recovery shows that Unit 2 could be safely restarted and run for a period of time, and then shut down for inspections. Yet the management at SONGS is reluctant to advocate the restart of the unit. The perception by the public and the employees of SONGS is that the problem of tube vibration is so difficult that is cannot be solved, perhaps ever.

Conclusions: Leadership has failed to provide clear, open, and honest communications with SONGS employees. This leads to low morale and cynicism. If SONGS ever restarts, the low morale could challenge nuclear safety. Leadership has failed to have a clear vision of the tube vibration issue and how to move the station forward. Too many consultants are obscuring the goal and offering too many opinions that leadership feels need to be addressed prior to restart. We will never restart if leadership doesn't make specific choices based on evidence available, not opinions.

Recommended Actions

Replace the SLT with people who understand the need for clear communications, who value SONGS employees, and can make decisions based on facts and evidence, not opinions offered by outside consultants.

San Onofre RSGs, February 2013, Peter T. Dietrich, SONGS CNO Presentation to NRC

"• All three conditions (High velocity, high dryness and loose supports* (AVP/DAB Note: contact force < 1 newtons) required for FEI existed concurrently for many tubes in Unit 3 RSGs

• Two conditions (High velocity & high dryness) existed in Unit 2 but better supports (AVP/DAB Note: contact force > 2 newtons*) prevented in-plane FEI in Unit 2 RSGs

• Detailed tube bundle model shows that differences in fabrication result in substantially increased contact forces (AVP/DAB Note: better supports) between tubes and AVBs for Unit 2

• Significant independent expert participation has been instrumental in developing conclusions**

• Operating experience is being shared***

• Unit 3 AVB twist lower due to greater pressing force. Confirmed by nearly 3 times more dent/ding signals in Unit 2 (AVP/DAB Note: See below for detailed explanation)

• Unit 3 tube roundness is more controlled. Confirmed by pre-service measurements."

<u>AVP/DAB Notes:</u>

* SCE CDS specified AVBs to prevent only out-of-plane vibration and SCE/MHI RCEs, Dr. Pettigrew and John Large confirmed it. SCE specified zero gap and low tube-to-contact force (<1 Newton) in the zone of high void fractions to in order to minimize ding/dent indications, and to maintain mechanical damping (by sliding of the tube along the AVB) and thus minimize tube vibrations based on MHI Root Cause and SCE/MHI Meeting Notes. According to MHI Repair Plan (Obscure MHI document found in CPUC Library & Published in December 2012), thicker AVBs with tube-to-contact force > 30 Newtons are required to prevent in-plane FEI, tube-to-tube contact. Differences in fabrication resulted in insignificant increases contact forces (reduced looseness) between tubes and AVBs (thin & crooked AVBs) for Unit 2 means that Peter Dietrich was providing misleading and materially false information to NRC.

** Significant independent expert participation was not successful in developing correct conclusions for Unit 3 in-plane FEI because SCE provided incorrect information for differences in Units 2 & 3 operational data to these experts.

*** SCE is not releasing Units 2 & 3 operational data, so this is a false claim.

SCE/AVB Team benchmarking of ANO-2 RSGs for design of San Onofre RSGs

"MHI Root Cause States, "MHI considered the changes in the SONGS design from previous steam generator designs and compared the basic design parameters of the SONGs RSGs (e.g., heat transfer area, circulation ratio, steam pressure, etc.) with other steam generator designs. Further, as part of the development of the SONGS RSG design, MHI conducted a detailed comparison between design of a plant whose SGs were similar to the proposed RSGs (the "comparison" or"(ANO-2) reference" plant). The AVB Design Team included consultants with knowledge and experience in the design and construction large U-bend SGs. One consultant had experience with the design of a plant whose SGs were similar to the proposed RSGs (the "comparison" or" reference" plant). Together, the AVB Design Team concluded that the SONGS RSGs had more tube vibration margin than the comparison plant, which had experienced only a small number of tube wear occurrences. This conclusion was due to the following considerations:(i) SONGS RSG tubes are larger, have thicker walls, and are stiffer than those of the comparison plant; (ii) the SONGS distances between AVB tube supports are shorter than those at the comparison plant; (iii) SONGS has 12 AVB tube supports where the comparison plant only has 10; (iv) SONGS's tube-to-AVB gap requirement was more stringent than that of the comparison plant."

SONGS Unit 3 Tube Leak Root Cause Evaluation and ANO-2 RSGs

"**Question:** What is different or has changed when comparing SONGS Replacement SG to ANO-2

SONGS operates at 13.6% higher thermal power:

SONGS Pwr 3458, ANO-2 MWt 3044 MWt (Note 1)

Different size tube diameter (d) and pitch (P)

Tube Index= P/d, SONGS Tube Index 1.33-1.433, ANO-2 1.518-1.672 (Note 2)

SONGs has less flow area: - Smaller wrapper area and less TSP flow area

- Tighter U-bend Area, AVB Configuration is different"

AVP/DAB Notes:

- Note1 – AVP/DAB's Safety Team Expert'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 shows that as-designed/operated by SCE, and unlicensed San Onofre's RSGs with Alloy 690 tubes 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/DAB 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. The increased primary flow in Unit 3 RSGs, smaller heat transfer and less flow area produced elevated steam velocities and dry steam caused in-plane FEI and uncontrolled vibrations. The smaller heat transfer and less flow area produced elevated steam velocities, wet steam and uncontrolled vibrations in Unit 2 RSGs.

- Note 2 - SONGS P/d was too small and feedwater/primary flow too high compared with ANO-2 resulting in much higher velocities and void fractions. The tighter U-bend Area and different AVB configuration in

SONGs compared with ANO-2 increased friction losses, elevated steam velocities and uncontrolled vibrations in Unit 2 RSGs.

- Note 3 - SONGS velocities, void fractions, feedwater flows, steam pressure, circulation ratio and primary flow for Unit 3 RSGs: 28 feet/second, 099.6%, 7.62 Mlbs/hr., 942 psi, 3.2, 79 Mlbs./hr – In-plane FEI & Out of Controlled Random Vibrations. High steam pressure reduced the tube vibrations but it also reduced the circulation ratios, which produced dry steam.

- Note 4 - SONGS velocities, void fractions, feedwater flows, steam pressure, ciirculation ratio and primary flow for Unit 2 RSGs: 25 feet/second, 98.9%, 7.59 Mlbs/hr., 833 psi, 3.5, 75 Mlbs./hr – No In-plane FEI, Only Out of Controlled Random Vibrations

- Note 5 - ANO-2 SONGS velocities & void fractions: 18 feet/second & 98.5 % - No In-plane FEI & No Out of Controlled Random Vibrations – Westinghouse Operational Assessment

What causes the difference between in plane and out of plane damage to RSG tubes?

When tubes move in the out-of-plane direction, they hit against their AVBs resulting in tube-to-AVB wear, which causes dings/dents to the tubes.

When tubes moves in the in plane direction, they hit or rub against other tubes in the same row of tubes at the next higher elevation, which causes tube-to-tube wear (extrados and intrados incidences).

AVP/DAB's bases its interpretation of the RSG damage assessment of Unit 3 verses Unit 2 upon the analysis performed by AREVA and confirmed by Peter Dietrich in presentation to NRC on February 7, 2013. It showed that 4,100 dings/dent indications were found in both of Unit 3 steam generators as opposed to nearly 12,000 dings/dent indications were found in both of Unit 2 steam generators. (Source: http://enformable.com/2013/02/areva-analysis-of-san-onofre-unit-2-steam-generators-reveals-new-twist-in-restart-debate/) This indicates that:

(1) Unit 3 RSG tubes were moving predominantly in the in-plane direction due to high velocities, dry steam, no tube damping, insufficient or zero tube-to-AVB in-plane friction forces, hitting other tubes causing significant tube-to-tube wear, which is consistent with MHI Root Cause Analysis,

(2) Unit 3 RSG tubes were barely touching/sliding over the AVB supports due to insignificant in-plane friction and pinning force causing an insignificant amount of tube-to-AVB wear, therefore they had 4,100 dings/dent indications,

(3) Unit 2 RSGs had only 2 tubes that experienced slight amount of tube-to-tube wear,

(4) If the AVB is in contact with the tube but there is insufficient contact force to lock the two together, there will be relative motion between the two and wear will occur. Unit 2 RSG tubes were moving predominantly in the out-of-plane direction causing significant tube-to-AVB damage of nearly 12,000 dings/dent indications because

the contact force was only 2 Newtons, whereas as a contact forces > 15 Newtons with thicker AVBs was required to lock the tubes and AVBs to prevent dings & dents shown in AREVA Operational Assessment. The in-plane fluid elastic instability and tube-to-tube wear did not occur because Unit 2 RSGs velocities were 25 feet/second, void fractions were less than 99% due to lower primary flow of 77.5 Million Pounds/Hour in Unit 2 RSGs compared to higher primary flow of 79 Million Pounds/Hour in Unit 3 RSGs

The direction of tube movement, the number of dings/dents along with the steam conditions in Units 3 & 2 can be explained by the differences in operational conditions between Unit 3 RSGs (e.g., Primary flow of 79 Million Pounds/Hour Steam void fractions – 99.6% & steam velocities -28 feet/second, steam pressures - 942 psi, circulation ratio-3.2, etc.) and Unit 2 RSGs (e.g., Primary flow of 77.5 Million Pounds/Hour Steam void fractions – 98.9% & steam velocities -25 feet/second, steam pressures - 833 psi, circulation ratio-3.5, etc). Unit 3 RSG tubes were barely touching/sliding over the AVB supports due to insignificant tube-to-AVB, in-plane friction and pinning force, and no liquid causing an insignificant amount of tube-to-AVB wear, therefore they had 4,100 dings/dent indications,

This explains why Unit 3 steam generators had three times lesser the number of dings/dents compared with Unit 2 steam generators.

SCE was operating Unit 3 differently (exceeding the "redline" of RSGs Pressure/Temperature Functional Acceptance, Design & Screen Limits) than Unit 2 (Operating within/below the "redline" of RSGs Functional Acceptance, Design & Screen Limits).

AVP/DAB Safety Team concludes that SCE was operating Unit 3 differently than Unit 2 in order to:

(1) Reduce the dings/dents occurring in their defective design (that specified zero tube-to-AVB gap and zero AVB contact force, *both done without NRC approval, since they were made without the required thermal-hydraulic and tube wear/vibration analysis*) by increasing the steam pressure to reduce tube vibrations and tube-to-AVB wear, and

(2) To generate more power (and operational profits) by increasing the primary temperature.

AVP/DAB Note: SCE failed to investigate the cause of extra alarms and/or lower the higher primary temperature because SCE was in hurry to restart Unit 3 in order to maximize their profits. SCE has now refused to release their operational data and chosen to decommission San Onofre, in the hopes of stoping all investigations into its RSG design and operational wrongdoing. AVP/DAB Safety Team has determined that SCE -- not southern California ratepayers -- is solely responsible for the multi-billion-dollar failure of San Onofre and the data proves it.

Part I – Expert Rebuttal of the NRC, SCE and MHI's Root Cause Conclusions (Patial)

Part I A - March 2013 - John Large (and AVP/DAB) on differences between RSGs and OSGs.

NRC Atomic Safety Licensing Board in its "Landmark San Onofre Ruling" on May 13, 2013 stated, "The replacement steam generators for Units 2 and 3, which were manufactured by Mitsubishi Heavy Industries

(MHI), differ in design from the original steam generators. For example, each replacement steam generator (1) has 9,727 tubes, which is 377 more tubes than are in the original; (2) does not have a stay cylinder supporting the tube sheet; and (3) has a broached tube design rather than an "egg crate" tube support. As discussed infra Part II.B.2, a licensee must obtain a license amendment from the NRC if a change to its facility triggers the safety standards described in 10 C.F.R. § 50.59. Despite the design differences mentioned above between the replacement and original steam generators, SCE concluded that the replacements were a like-for-like change that did not require a license amendment." The SCE Unit 3 Tube Leak Root Cause Evaluation states, "Changing design from the original SG to the Replacement SG, is not causal factor in itself for tube to tube wear with regards to: (1) Departure from the OSG design in terms of tube U-bend configuration and U-bend support configuration, (2) Departure from the OSG design in terms of replacing the stay cylinder with the divider plate and separator configuration, and (3) Departure from the OSG design in terms of tube straight leg support configuration." The NRC AIT Report repeats SCE and MHI evaluations and states, "With regard to the major design changes between the original and replacement steam generators, the updated final safety analysis report did not specify how the original steam generators relied on special design features such as the stay cylinder, tubesheet, tube support plates, or the shape of the tubes to perform the intended safety functions."

NOTE: NRC, SCE, MHI and NRC ASLB never listed the impact of the differences (weaknesses) between RSGs and OSGs. Rather, NRC, SCE and MHI described at great length the advantages of RSGs over OSGs in order to justify SCE's bogus use of a 10 CFR 50.59 Screening and Evaluation instead of a through NRC 50.90 License Amendment.

John Large, the International Chartered Engineer from London, England states, "Adhering to the original OSG design involved the compromise of increasing the total tube heat transfer surface area, and hence a higher number of tubes, as a result of changing the tube material from Inconel 600 to Inconel 690 with its reduction of heat transfer coefficient of about 11%. This required a commensurate increase in the number of tubes from about 9,400 in each OSG to about 9,700 in each RSG. Also, the physical restrictions on the overall dimensions of the RSG and, directly from this, maintaining the same secondary side flow area, required removing the tubesheet supporting pedestal or stay cylinder so that additional tubes could be accommodated in the central zone of the tube bundle. This meant that the cylindrical void above the tube support sheet could no longer function as in the traditional design of steam generator that provides for a column of relatively water-rich (i.e. liquid phase) feedwater to rise up through the centre of tube bundle to reach the U-bend and reduce the steam quality. Another quite radical design departure from the OSG involved the replacement of the more conventional 'eggcrate' or lattice horizontal tube supports (near right), with seven tube support plates (TSP) comprising solid steel plates broached with trefoil apertures through which the individual tubes passed (far right). This change has two significant outcomes in that compared to the eggcrate supports, the greater blockage to ascending flow presented by trefoil broached tube support plate, together with the loss of the flow column above the stay cylinder location, required slots to be cut across the centre out-of-plane direction of the tube bundle to facilitate the upward and circulation flow of feedwater." AVP/DAB'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 shows that as-designed/operated by SCE, and unlicensed San Onofre's RSGs with Alloy 690 tubes 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/DAB 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.*

<u>Part I</u> B – July, 2012 – NRC Independent Consult. Report Preceded The NRC AIT Report

Note: *In March, 2012, after the formation of the NRC AIT, the new NRC Chairman appointed a group of NRC Independent Consultants (aka Beckman & Associates who were SG experts not affiliated with the NRC), who were* *chartered with performing an independent evaluation of San Onofre in order to find any perceived gaps in the response actions taken by the NRC Region IV AIT Team or SCE due to the Unit 3 RSG tube leak. Their primary focus was to assess the differences in damages between Units 2 & 3 due to tube-to-AVB gaps, contact forces and thermal/hydraulic related aspects of operation. After an intense & in-depth review, the Independent Consultants identified significant, conflicting and adverse "gaps" in the NRC AIT Team and SCE's Unit 3 Tube Leak Root Cause Evaluation Reports. The Independent Consultants issued significant and adverse findings in their report that did not agree with the San Onofre Unit 3 RSGs Tube Leak SCE, MHI and NRC AIT Reports. The NRC Independent Consultants report was submitted to the NRC on July 13, 2012 prior to issuance of NRC AIT Report on July 18, 2012, but was it was withheld from the public for over a year to hide SCE's errors until it was attached to the NRC San Onofre Final Inspection Report issued on September 20, 2013. Highlights from their critical report are listed (verbatim) in bullets below:*

From 4, Probable Cause Evaluation

- "It was concluded from this review that the fluid condition alone did not explain the abnormal tube-to-tube wear based on the standard FEI criteria of ASME Boiler and Pressure Vessel Code, Section III, Division1, Appendix N, Section N-1330, 1998. As noted in both the AIT draft report and the SCE RCE, no tube wear of the type detected in the Unit 3 replacement steam generators has been previously observed in other recirculating SGs in the domestic fleet. The observed severe tube-to-tube wear was also restricted to a small region of the tube bundle cross section. Based on the wear indications, the U-bend tube support by the anti-vibration bars was ineffective in a small region of the tube bundle for in-plane tube vibrations. Various assumptions were made to rationalize the mechanism, which caused the observed tube-to-tube wear including: (a) insufficient tube support at multiple anti-vibration bar locations, (b) vibration in an in-plane mode, (c) gaps between tubes and anti-vibration bars, and (d) spreading of the upper U-bend structure due to fluid dynamic forces and thermal effects."

AVP/DAB Comment: This rejects the conclusions of NRC AIT Report, SCE and its global consultant's published reports.

- "No evidence was found in the reviewed documentation that was pertinent to the following: (a) the cause for the abnormal and localized high void fraction in the replacement steam generator design, which historical degradation information would indicate was absent in the original steam generator design; and (b) the existence of the localized high void fraction and flow velocities, as calculated by ATHOS, and originally by FIT-III with lower flow velocities, that has apparently been accepted without question as a replacement steam generator design feature. No evidence was found that the flow conditions, void fractions, flow velocities and temperatures of the tube bundle were compared against the replacement steam generator design basis. Rather, the thermal-hydraulic analysis results were accepted, initially by design, and again during the extensive review process.'

'The conditions leading to the wear were calculated with the EPRI ATHOS code, a standard steam generator design tool. The calculated high void fraction region included the smaller region of tubes with the observed wear. It was not obvious that a small region with conditions which caused tube wear can be extracted from the ATHOS results."

AVP/DAB Comment: This rejects the conclusions of NRC AIT Report, SCE and its global consultant's published reports.

From 5. Design and Manufacturing Differences

- "The average gap between outermost tubes and anti-vibration bars for the Unit 2 and Unit 3 SGs was reported by Mitsubishi Report L5-04GA564, Rev. 2 to be: steam generator 2E0-88, 0.59 mils; steam generator 2E0-89, 0.76 mils; steam generator 3E0-88, 0.15 mils; steam generator 3E0-89, 0.21 mils. This report additionally stated that the average of the gaps between the outermost tubes and the central columns is essentially the same between the Unit 2 and Unit 3 SGs. This data obviously does not support the premise that more uniform manufacturing practices for Unit 3 steam generator tube bundles resulted in less contact force between anti-vibration bars and tubes. In the absence of more dimensional information for the steam generator tube bundles, it is not believed possible to explicitly define the number of active supports in the Unit 2 and Unit 3 SGs."

AVP/DAB Comment: This rejects the conclusions of NRC AIT Report, SCE and its global consultant's published reports.

From 5. Design and Manufacturing Differences

- "Data was not specifically searched for during this review to allow formal assessment of the technical credibility of eddy current testing inspection for measurement of gaps between tubes and anti-vibration bars; i.e., it is currently unknown whether a qualified Examination Technique Specification Sheet (ETSS) exists for this measurement. Difficulties in use of the CERTREC system for information retrieval negatively affected conduct of this review. Paragraph 4.1.2 of Mitsubishi Report L5-04GA564, Rev. 2 states, in part, with respect to comparison of bobbin probe signals in the Unit 2 and Unit 3 SGs for estimating tube-to-anti vibration bar gap sizes "...This data did not reveal significant differences and indicates that the gaps in the affected region of the tube bundle are below 20 mils (0.5 mm). However, the average voltage signal in the Unit 3 SGs is slightly lower than the average signal in the Unit 2 SGs, indicating that the average gap size in the Unit 3 SGs is slightly larger than in the Unit 2 SGs, and indicating that the average contact force between the tubes and anti-vibration bars during operation may be lower in the Unit 3 SGs." These comments are believed to be speculative, and rely on a global number of unknown technical credibility for predicting values in a bundle sub-population. Review of Figure 4.1.2-1 in Report L5- 04GA564, Rev. 2 indicates the potential fallacy in making these projections. Specifically, Figure 4.1.2-1shows virtually identical average absolute signal amplitude signals at anti-vibration bar locations for SGs 2E0-88 and 3E0-89, SGs that have shown significant differences in operational tube wear behavior.

Accordingly, this review concluded that eddy current testing inspection measurements of tube-to-anti-vibration bar gap were of questionable value in assessment of likely tube wear behavior."

AVP/DAB Comment: This rejects the conclusions of NRC AIT Report, SCE and its global consultant's published reports.

From 7. Operational Impacts

- "The AIT identified an unresolved item requiring further review pertaining to whether SCE appropriately reviewed and dispositioned numerous steam generator loose parts alarms during Unit 3 operation. Similar steam generator loose parts alarms did not occur during Unit 2 operations in Cycle 16, raising the question of whether the Unit 3 alarms were potentially indicating steam generator tube-to-tube contact during power operations. It was noted from review of the licensee RCE report that Westinghouse had performed an analysis of the various alarms for the licensee. Westinghouse concluded that the vibration and loose parts monitoring system events for both SGs were the result of true metallic impacts and not false indications from electrical noise or fluctuations in background noise. The alarm events were noted to be similar to events that occur when SGs shift during reactor coolant system temperature transients, but it could not be conclusively stated without additional data that the events were from the same source. During this independent review, it was ascertained that the accelerometer skirt location did not appear to comply with the requirements of the Design Specification SO23-617-01, "Specification for Design and Fabrication of Replacement Steam Generators for Unit 2 and Unit 3," Revision 4. Specifically, Section 3.9.3.19, Loose Parts Monitoring Provisions, required mounting pads for sensors to be installed on the external surface of the inlet side of the channel head and on the lower shell. One pair of mounting pads (one active and one reserve) was required to be located with a vertical alignment above the tube-sheet, and one pair with a vertical alignment below the tube-sheet. Revision 4 of Design Specification SO23-617-01 was approved on July 28, 2010, which postdates the Unit 2 return to power on April 13, 2010 after replacement steam generator installation. The circumstances pertaining to relocation of sensors to a lower sensitivity measurement location, approval of this change, and the continuing conflict with current design specification requirements were not available for review."

From 6.2.4 Tube Wear at tube support plates above the Tube Sheet Hot-Spot

- "Mitsubishi Report L5-04GA564, "Tube Wear of Unit-3 RSG – Technical Evaluation Report," Revision 2, includes figures showing the distribution of tube wear at tube support plate 1 to 7 levels and in the U-bend sections. The region at the tube support plate 1 level with TWD tube wear is immediately above the location where nucleate boiling at the tube sheet level started. A reasonable supposition is that flow caused this local wear. The correlation between the tube wear increasing upward from tube support plate to tube support plate and the location of the hot spot appeared to be systematic in the replacement steam generators. The coolant flow between tubes in the straight tube sections is predominantly axial, upward with low cross-flow velocities. Mitsubishi postulated in L5-04GA564, Rev. 2, that turbulent excitation was the potential cause for wear at tube support plates. The evaluation did not specifically address the small region of tube wear shown in Figure 2-6 that was observed at tube support plates 1 through 7."

- "The correlation between boiling in a small region at the bottom of the tube bundle, based on Mitsubishi Document calculations, and the observed region of tube wear increasing from tube support plate 1 levels upward into the U-bend region has apparently not been addressed. Some mechanism is moving the tubes and causing the tube-to-tube support plate wear in a small region. Standard steam generator evaluation procedures may not model in sufficient detail the unusual flow pattern evolving from the tube-sheet."

From 6.2.3.2 ATHOS and AREVA CAFCA Analyses

- "One observation from examining these calculation results was that the nucleate boiling started in a region outside the outer row of stay rods. Figure 8.3-3 of Document L5-04GA521, Rev. 3 showed local void fractions, and Figures 8.3-1 and 8.3-2 showed the flow velocities above the tube-sheet. Nucleate boiling at the tube-sheet surface is an anomaly not intended to occur in the intended design as described in Mitsubishi Document L5-04GA510, Rev.5. The consequences of this inconsistency between the replacement steam generator system design and calculated thermal-hydraulic performance has not been addressed. It was also noted that Westinghouse in their independent ATHOS analysis indicated that their design approach precluded boiling at the tube-sheet surface. The graphs of flow pattern above the tube-sheet indicate a region of low flow velocities where higher void fractions than in the surrounding fluid are indicated. A potential cause for these low flow velocities, a region of almost stagnant flow, is the higher flow resistance for the cross-flow from the wrapper inlet ports into the tube bundle due to the smaller pitch-to-diameter ratio of the replacement steam generators than in the original steam generators. A comparison of the replacement steam generator thermal-hydraulics with that of the original steam generator was not found in either the AIT or the SCE RCE reports, which could aid in the determination of the cause for the flow abnormalities in the replacement steam generator."

- "The standard deviation for tube O.D. in the Unit 2 and Unit 3 replacement steam generators was calculated by Mitsubishi to be: steam generator 2E0-88, 0.71 mils; steam generator 2E0-89, 0.71 mils; steam generator 3E0-88, 0.63 mils; steam generator 3E0-89, 0.55 mils. Mitsubishi postulated in Report L5-04GA564, Rev. 2 that improved dimensional controls for Unit 3 replacement steam generators such as anti-vibration bar thickness, tube roundness, and gaps between tubes and anti-vibration bars probably resulted in less contact force between the tubes and the anti-vibration bars. This difference in standard deviation values for the O.D. of the tube populations in the individual SGs is considered by this review to have minimal effect, particularly if one takes into consideration that the low radius U-bends are the biggest contributor to tube ovality and higher G-values. The localized region of tube wear, however, is located in high row number tubes where variations in G-values would not be expected in the large radius U-bends during ongoing production. It is believed that any conclusions to be drawn from contribution of tube ovality to wear differences between Unit and Unit 3 SGs should specifically consider the Unit 2 and Unit 3 G-Values of tubes in the Unit 3 sub-population wear region."

- "Mitsubishi Report L5-04GA564, Revision 2, noted that the number of adjustments to tube bending radius was smaller for the Unit 3 SGs than for the Unit 2 SGs. Specifically, the reported values were: steam generator 2E0-88, 265; steam generator 2E0-89, 390; steam generator 3E0-88, 132; steam generator 3E0-89, 149. The inference drawn was in the context of promoting greater uniformity in tube to anti-vibration bar gaps. The required profile for U-bends is established on an inspection layout table and needed local, minor radius adjustments are made to assure conformance to the required profile. Absent the existence of additional information, there is no apparent basis to believe that the number of local adjustments to U-bends has any relevance to observed steam generator tube degradation."

From 6.2.3.1 Mitsubishi FIT III Code Analysis

- "Document L5-04GA521, Three Dimensional Thermal and Hydraulic Analysis (FIT III Code Analysis) calculated the void fraction and flow velocities of the tube bundle in some detail. The void fraction, flow velocities and temperatures throughout the tube bundle are presented and are the input to the vibration calculations. Issues with this code were addressed in the draft AIT inspection report. The AIT inspection noted that the Mitsubishi reported flow velocities and void fractions appeared to be low and, as a result, performed independent calculations using the EPRI ATHOS code. These calculations found much higher velocities in the tube bundle. The cause for the FIT-III discrepancy has currently not been resolved. The AIT inspection also noted that the validation and verification of the FIT-III code did not provide sufficient evidence that the code had been adequately benchmarked. The AIT inspection concluded, without performing vibration analysis, that the higher flow velocities and void fractions were the cause of the observed FEI and tube wear."

From 6.3 Summary

"In summary, the simplified, postulated scenario leading to the damaging tube vibrations of the

U-bend tubes is as follows:"

- "The secondary flow from the wrapper inlet port to the tube bundle does not penetrate the bundle because the flow resistance of the replacement steam generator bundle with a smaller pitch-to-diameter ratios higher than in the original steam generator. The smaller pitch-to-diameter ratio of the replacement steam generator tube bundle increases the cross-flow resistance in the tube bundle. As a result, the penetration of the flow from the wrapper ports into the tube bundle and the flow distribution in the bundle changes when compared with the original steam generator. The inlet flow stagnated in a region outside the outer row of stay rods, visible on the flow velocity plots of ATHOS and FIT-III. No evidence was found that this change was considered in design."

- "Nucleate boiling occurs at the tube sheet level with low cross flow, a hot spot location, which is inconsistent with the replacement steam generator system design parameters."

- "Above the hot spot, undefined flow conditions cause a small group of tubes to vibrate starting at the tube support plate1 level. Tube wear progressively increases to the upper tube support plate 7."

- "The localized high velocity flow with high void fraction causes the U-bend tube bundle to vibrate violently in a small region above the TWD wear at the tube support plate levels."

- "More evaluations would be required to substantiate the postulated scenario as the source of the high void fraction and velocity in a specific U-bend region."

AVP/DAB Comment: The AVP/DAB Safety Team Technical Analysis supports the analysis of the NRC Independent Consultants, which basically means that more energy was being transferred in the Unit 3 RSGs as compared to Unit 2 RSGs, which resulted in nucleate boiling in the Unit 3 RSGs starting at the tube sheet level. Also, it is crystal clear that the secondary side higher pressure (942 psi) steam in Unit 3 RSGs required less energy (661 Btu/lb.) to saturate compared with the lower pressure (833 psi, 683 Btu/lb.) steam in Unit 2 RSGs above the tubesheet. Therefore, with more primary energy available in Unit 3, Unit 3 produced a greater amount of steam (15,000 cubic feet/hour more than Unit 2) and had a higher vapor fraction (> 99.6%) of dry steam as compared to Unit 2's lower vapor fraction (< 98.9%) wet steam. This data, which is consistent with the Mitsubishi Repair Plan, explains why Unit 3 RSGs were susceptible to in-plane FEI and experienced so much tube-to-tube wear and why Unit 2 RSGs were NOT susceptible to in-plane FEI and did not experience tube-to-tube wear like Unit 3. (Note: Unit 2 did experience large amounts of tube to AVB wear, which was only discovered later during the AIT investigation after the Unit 3 radioactive leak, but it was caused by SCE's RSG design, not by in-plane FEI.) Therefore, AVP/DAB fully supports the NRC Independent Consultants analysis and completely rejects the conclusions of NRC AIT Report, SCE and its global consultant's published reports.

<u>Part I</u> C – November, 2015 – Seven *Quotes* From A Nuclear Safety Systems Expert:

<u>Part I</u> C 1. ***"There are a lot of contradictions on everyone's part, SCE, MHI and NRC".***

<u>Part I</u> C 2. ***"The Primary side flow is higher in both SG3 (79E6 lb./hr.) and SG2 (77.5E6 lb./hr.) which equates to more energy transfer to the SG. The old SG had about 75E6 lb./hr. The importance of this is that from SG3 had more heat energy transfer than the old SG."***

AVP/DAB Note 1: Interpolated from the NRC AIT Report - Primary loop volumetric flow rate from U2 -204,400 gym (77.5E6lbs/hr), U-3, 208,000 gpm (79E6lbs/hr), U2- Feedwater Flowrate – 7.588E6 lbs./hr., U3 - Feedwater Flowrate – 7.62E6 lbs./hr., U2- Steam Pressure – 833 psi, U3 - Steam Pressure – 942 psi, 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 (5% more than OSGs & 2% more than Unit 2 RSGs) in the hottest channels (region of tube-to-tube wear), higher velocities and void fractions than the OSGs and Unit 2 RSGs hottest channels.** These AVP/DAB 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 is much higher than the core average, resulting in elevated steam velocity exiting the core.”

“**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.”

AVP/DAB Comment 1: As you can see by reviewing the data in Note 1, to increase the steam pressure in Unit 3 RSGs from 833 to 942 psi, an increase in feedwater flow by 0.5% was required. So the energy in primary flow in Unit 3 RSGs should have been increased by 0.5% also (It is a 1:1 ratio of transfer of energy from the primary to the secondary side assuming no heat loss from the steam generator). But the ratio of actual transfer of energy from the primary to the secondary side in the Unit 3 RSGs was was much more than actually needed (2.0: 0.5) resulting in elevated steam velocities, higher quality and drier steam and higher primary temperatures in Unit 3 RSGs compared with lower steam velocities, lower quality wet steam and lower primary temperatures in Unit 2 RSGs. The NRC AIT Report indicates that Unit 3 primary temperature was higher by 4 degrees than Unit 2 indicating confirmation of elevated steam velocities and higher quality and drier steam and film boiling in Unit 3 RSGs meaning presence of extra primary energy in Unit 3 RSGs. This indicates that the RSG performance (i.e. RCS flow rate, RSG steam pressure, etc.) procedures were never developed by MHI or developed incorrectly. It is also possible that SCE never used these procedures or performed any thermal-hydraulic, flow-induction vibration, or tube vibration and wear analysis before exceeding the redline of RSGs functional acceptance criteria.

AVP/DAB Explanation 1 RE: 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/DAB 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/DAB Note: The small difference in the steam quality between Unit 3 and Unit 2 RSGs explains the difference in significant tube-to-tube wear in Unit 3 RSGs and Zero tube-to-tube wear Unit 2 RSGs due to elevated steam velocities and zero damping consistent with the new industry study and analysis of AVP/DAB Experts).

AVP/DAB Explanation 2 RE: Void Fractions - MHI Root Cause Evaluation, page 23 states, "The highest void fraction is located in the (Added by AVP/DAB: 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/DAB 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/DAB 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/DAB 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).

Part I_C 3. "***The change from the lattice structure*** (**AVP/DAB 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/DAB 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/DAB 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/DAB 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/DAB finding is consistent with Westinghouse, Babcock & Wilcox and AVP/DAB 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/DAB concludes that SCE was more interested in checking Mitsubishi's cost and schedule for maximizing the profits from new RSGs.

<u>Part I</u> C 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/DAB 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/DAB 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/DAB 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.

Part I C 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/DAB Note: These comments are consistent with AVP/DAB Safety 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.

<u>Part I</u> C 6. ***"I am 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/DAB 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 increae 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." Only SCE can tell the people of Southern California, AVP/DAB Safety Team's Technical Panel and the CPUC why SCE decided to exceed the redline of Unit 3 RSGs Functional Testing, Screening & Design Limits and why they are refusing to release the Units 2 & 3 Operational Data, unless it is to hide their own wrongdoing .

Part I C 7. "***We are curious on the level in the SG. From the SG procedure you sent, the steam separator was partially submerged. There is a rotating vane that is supposed to separate the liquid vapor mixture but my experience in BWRs 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/DAB Note: Only SCE can tell the people of Southern California, AVP/DAB Safety Team 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 I D – 2012 - Union of Concerned Scientists

Dave Lochbaum, Director of the Union of Concerned Scientists Nuclear Safety Project, one of the nation's top independent experts on nuclear power told NRC in October 2012 and April 2014, "Manufacturing process improvements between the fabrication of the Unit 2 replacement steam generators and the Unit 3 replacement steam generators resulted in the latter having "smaller average tube-to-AVB contact force" making them "more susceptible to in-plane vibration. However, this explanation is not well documented and therefore appears to be more convenient than factual. Actual differences in operating conditions (e.g., RCS temperature and pressure) seem to offer more likely causal factors."

<u>Part I</u> E – 2015 - Nuclear Industry Express Doubts About NRC SONGS Lessons Learned

"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)."