A V P ARORA INTERNATIONAL INC.

N U C L E A R   S A F E T Y

STATEMENT BY VINOD K. ARORA

NUCLEAR SAFETY ENGINEER

TO THE NUCLEAR REGULATORY COMMISSION

Submitted June 18, 2016

Vinod Arora is the Chief Executive Officer of the not for profit AVP International. He is the former nuclear safety engineer for Southern California Edison's San Onofre Nuclear Generating Station. The following pages contain Mr. Arora's analysis of confidential Nuclear Regulatory Commission documents that were withheld from the public until they were secured by AVP International under the Freedom of Information Act. For more information visit http://www.avparorainternational.org/ or email vinnie48in@gmail.com.

For background or to arrange an interview, contact Charles Langley at (858) 752-4600.

1. SCE was operating San Unit 3 Replacement Steam Generators (RSGs) with more primary energy than Unit 2 Replacement Steam Generators (RSGs).

2. Therefore, Unit 3 RSGs velocities were high and steam was nearly dry compared to Unit 2 RSGs, where velocities were lower and steam was wet.

3. The Unit 3 high velocities and dry steam created large amplitude in-plane tube vibrations, which caused tubes to slide over the AVBs with no in-plane friction force to restrain tube movement. The moving tubes hit other tubes in the same column with large kinetic energy, created a tube leak, 2 practically leaking tubes, tube-to-tube wear in 326 tubes, 8 failed tubes, 1767 tubes-to-AVB wear and 463 tubes-to-TSP wear in 11 months or weeks. The lack of in-plane support was the result of SCE RSGs being designed for out-of-plane U-bend support with "zero" gaps in the hot condition and no consideration for U-bend support in the in-plane direction.

4. The Unit 2 low velocities and wet steam created low amplitude in-plane tube vibrations, which caused tubes to hit AVBs in the out-of-plane direction with evidence of orbital tube movement relative to the anti-vibration bar and tube. The low amplitude in-plane tube vibrations only created tube-to-AVB damage but prevented tube-to-tube wear in 22 months in out-of-plane FEI qualified RSGs.

5. SCE says that primary energy was the same in both Units and damage in Unit 3 was considerably more than Unit 2 because of manufacturing differences and errors in *Mitsubishi Code.* Units 2 & 3 replacement steam generators had more primary energy than the original Units 2 & 3 steam generators.

6. NRC Internal, NRC Atomic Safety Licensing Board, NRC Independent Consultants, Union of Concerned Scientists and AVP Reports overwhelming reject SCE's unverified analytical assumptions and false material statements designed to: (1) Dupe southern California Ratepayers of $3.3 Billion Dollars, and (2) lay all the blame on MHI Errors and NRC Regulatory Delays to hide its negligence in the design and operation of RSGs and avoidance of a NRC License Amendment,

Summary: SCE is misleading and making false material representations about NRC AIT, when it states that operational data for San Onofre Units 2 & 3 was exactly the same like the "Improving Like-for-Like RSGs." Based on a review of Actual Range of San Onofre Units 2 & 3 RSGs operational data published in the NRC AIT Report, NRC Internal Files, witnessing of the actual data in a AVP/NRC AIT Meeting documented in early 2013, NRC ASLB Ruling and MHI Repair Plan, AVP concludes that San Onofre Units 2 & 3 RSGs operational data is significantly different.. SCE statements also strongly imply without any legal and scientific basis that NRC AIT used "Analytical assumptions and results, not operational data" in an official NRC AIT Report in violation of the directive issued by Retired NRC Region IV Administrator, Elmo Collins (shown in the attached NRC AIT Team Charter ML12075A258). NRC is a US Government Nuclear Safety and Regulatory Agency with International Reputation and US Government Charter responsible to the US President/Congress, which does not publish fake reports. The only way, SCE can prove Albert Einstein, NRC Internal Files and AVP Chemical, INPO 1 Reactor and NSSS Engineers are wrong is by publishing the San Onofre Units 2 & 3 RSGs operational data (with legitimate evidence that withstands scrutiny) and is in a form, which can be certified by AVP and another third party without any conflict of interest with SCE (Like Dr. Budnitz).

1. Email From: Tom Palmisano

9:19 AM (June 27, 2016 ago)

To: me (Vinod Arora), John Kotek (DOE), David Victor (CEP Chairman)

Thank you for attending the June 22 CEP meeting.

I am responding to your recent email related to the SONGS Units 2 & 3 RSG operation data. In regard to your request, I note that the SCE Law Department previously responded to your counsel in May 2015 and March 2016 that SCE would not agree to produce RSG operational information. The NRC has also completed its inspections and reviews regarding the RSGs, and considers the matter closed. We will not provide the RSG operational information you requested. Respectfully, Tom Palmisano

2. SCE 10 CFR 50.59 Screen Order: NECP 800175663

Description: Steam Generator Replacement Mstr ECP U2

Operation No: 0510 Sub Op: Op Wrkctr: ED_SG SGRP

The saturated steam pressure of 833 psia is the best estimate pressure at the steam generator outlet nozzle with the reactor coolant inlet (Thot) temperature of 598 oF (corresponding to the Tcold of 541.3oF), reactor coolant best estimate flow rate (79.79E6 lb/hr), no (0%) tubes plugged and an assumed fouling factor (0.9E-4 ft2hroF/Btu). The actual pressure will vary depending on the actual values of these parameters during plant operation. The RSGs are qualified to operate in the Thot range from 598 to 611oF, which corresponds to the Tcold range from 541.3 to 555.4oF. However, NECP 800071702 and this 50.59 screen only allows the plant to operate up to Thot <= 598oF, as additional analyses are required to be performed to evaluate the rest of the RCS and support systems' ability to operate above Thot > 598oF.

3. Improving Like-for-Like RSGs

Boguslaw Olech, P.E., Southern California Edison Company, 14300 Mesa Rd., SanClemente, CA 92674, USA, Email: bob.olech@sce.com.Tomoyuki Inoue, Mitsubishi Heavy Industries Ltd. (MHI), 1-1 Wadasaki-cho 1-Chome, HyogoKu, Kobe, Japan 652 8585, Email: tomoyuki_inoue@mhi.co.jp.

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. This article was based on a paper published at ICAPP 2011, 2-5 May 2011, Nice, France, paper 11330.

Results: 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.

Reactor coolant flow rate: As-designed, with the RSGs operating at full (100%) reactor rated power with reactor coolant temperature at a design point, the ‘as-measured’ reactor coolant flow rate was expected not to exceed 106.5% of the original volumetric design flow rate. As-tested, the reactor coolant flow rate was 104.35% (206,613 gpm) of the original design flow rate (198,000 gpm - OSGs).

Primary-to-secondary leakage: As-designed, the RSGs were not supposed to exhibit a detectable primary-to-secondary leakage with the primary side at 2250 psia, and the secondary side at the normal operating pressure and temperature. As-tested, a primary-to-secondary leakage of less than 1 gallon per day (3.87 litres/day) was reported when the plant stabilized at full (100%) reactor rated power.

4. SCE Root Cause Evaluation: Unit 3 Steam Generator Tube Leak and Tube-to-Tube Wear Condition Report: 01836127,
Revision 0, 5/7/2012 San Onofre Nuclear Generating Station (SONGS)

General Parameter	OSG	RSG
Thermal rating, MWt	1729	1729
Number of Tubes	9350	9727
Heat Transfer Area, ft2	105,000	116,100
UA, Btu/hr o F	1.5E8	1.49E8
Tubes Outside Diameter, in.	0.750	0.750
Tube Wall Thickness, in.	0.048	0.0429
Tube Pitch, in.	1.0 triangular	1.0 triangular
Tube Plugging Margin, %	8	8
Primary Side		
Design Pressure, psia	2500	2500
Design Temperature, F	650	650
Operating Pressure, psia	2250	2250
Operating Temperature (Thot), F	611.2	598.0
Operating Temperature (Tcold), F	553.0	541.3
Reactor Coolant Flow (at cold leg temperature), gpm	198,000	209,880
Reactor Coolant Volume, ft3	1895	2003
Secondary Side		
Design Pressure, psia	1100	1100
Design Temperature, F	560	560
Operating Pressure (@100% power), psia	900	833
Operating Temperature (@100% power), F	532	523
Steam Flow, lb/hr	7,414,000	7,588,000
Steam Moisture Content, %	<0.20	<0.10
Feedwater Temperature, F	445	442
Blowdown Flow, lb/hr	151,000	154,860
Dimensions		
Top of the Tube Bundle, in.	381.0	388.2
Overall Height (including support skirt), in.	786	785.6
Upper Shell OD, in.	264.125	264.125
Lower Shell OD, in.	172.375	174.65
Dry Weight, lbm	1,242,366	1,286,200
Flooded Weight, lbm	1,971,840	2,041,300
Operating Weight, lbm	1,505,437	1,548,700

5. NRC AIT Report July 2012, Page 22

(2) Operational Differences: 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

(AVP Added -Per RSG 204,000 – 208,000 GPM) (10872 lbm/second -11081.8 lbm/second)

• Recirculation ratio from 3.2 to 3.5.

6. Table 1 - NRC Internal files

Parameter	Plant 1 RSG	SONGS Low Thot	SONGS High Thot	Plant 2 OSG
Heat Transfer Area (ft**2)	-	115500	-	-
Design Power (MWt)	~1350	1729.00	-	~1700
No. Tubes (0% TP)	9000	9727	-	9350.00
Layout	U-Bend, Triangular	U-Bend, Triangular	-	Sq-Bend, Triangular
Dome Pressure (psia)	**909.5**	**840.0**	**942.0**	**900.0**
**Dome Temp. (Deg F)	533.2	523.8	537.4	531.9
Density Of Sat. Liq. (lbm/ft**3)	47.0	47.6	46.8	47.1
Density Of Sat. Vapor (lbm/ft**3)	2.02	1.85	2.10	2.00
*Feedwater Flow Rate (lbm/sec)	820.8	1053.9	1058.3	1050.3
*Prim. Flow Rate (lbm/sec)	10528.8	11081.8	10872.0	10277.5
*Bundle Flow Rate (lbm/sec)	2797.3	3523.7	3483.5	3584.5
*Steam Flow Rate (lbm/sec)	820.4	1058.7	1062.6	1050.3
Circulation Ratio	**3.42**	**3.33**	**3.28**	**3.42**
Downcomer Liquid Level (in)	411.3	451.6	-	438.6
Mass in Shroud (lbm)	25272.9	32371.8	33959.35	42629.5
Exit Steam Quality (%)	99.8	99.8	-	99.8
Calc. Power (MWt)	1352.4	1724.0	-	1708.0
**Feedwater Temperature (Deg F)	435.0	442.0	442.0	445.0
****Primary Inlet Temp. (Deg F)**	**598.9**	**599.8**	**611.6**	**611.0**
****Primary Outlet Temp. (Deg F)**	**552.2**	**542.1**	**554.9**	**552.2**
Max Void	**0.980**	**0.998**	**0.998**	**0.970**
Max Quality	**0.670**	**0.945** 941	**0.940**	**0.610**
****Max Field Velocity (ft/s)**	**12.139**	**21.325**	**19.685**	**14.764**
- (m/s)	**3.700**	**6.500**	**6.000**	**4.500**
* Note for 1/2 Generator				
** in Effective Area				

Primary Flow Rate Low Hot (lbm/sec) = 11081 = 208,000 gpm = 79 Mlb/hr

Primary Flow Rate High Hot (lbm/sec) = 10528.8 = 204,000 gpm = 77.5 Mlb/hr

7. Table 2 - AVP Analysis of NRC AIT & SCE Units 2 & 3 Operational Data

Operating Parameter	NRC AIT Case 1 - NRC Internal Files SONGS Low Hot Unit 2	NRC AIT Case 2 - NRC Internal Files SONGS High Hot Unit 3	Unit 3 Tube Leak Root Cause Evaluation San Onofre Units 2 & 3 Design Data	SCE RSGs Units 2 & 3 SCE RSGs Functional Acceptance/10 CFR 50.59 Screen Criteria	San Onofre Original Steam Generators Units 2 & 3 Data	Notes
Reactor Coolant Flow (gpm)	208,000	204,000	209,880	206,613	198,000	**1**
Reactor Coolant Flow (Mlb/hr)	79	77.5	79.76	78.5	75	**1**
Steam Pressure (psi)	840	942	833	831-837	900	**2**
Steam Flow, lb/hr	7,590,000	7,620,000	7,588,000	7,588,000	7,414,000	**3**
RCS Operating Temperature (Thot), degrees F	599.8	611.6	598.0	598.0	**598.0 (11)**	**4**
RCS Enthalpy (Thot) Btu/lb	612.52	634.73	611.25	611.25	**611.25 (11)**	
RCS Temperature (Tcold), degrees F	**542.1**	**554.9**	**541.3**	**541.3**	**541.3 (11)**	
RCS Enthalpy (Tcold) Btu/lb	539.37	559.1	538.39	538.39	**538.39(11)**	
Circulation Ratio	3.33	3.28	N/A	N/A	Varying/ N/A	
Velocity (feet/second)	25	> 31	N/A	N/A	23	**5**
Peak Void Fractions	98.3 - 99.2	> 99.6	N/A	N/A	96.1	**5**
Bulk Void Fractions	98.3	99.3	99	97.4	Varying/ N/A	**7**
RCS Energy MBtu/Hr	5778.85	5861.3	5,811.32	5719.51	**5464.5(11)**	**5, 10**
Stability Ratio	0.9	1.15	N/A	> 0.75	0.675	**6**
Contact Force	2 Newton	1 Newton	0	0	Varying/ N/A	
In-Plane FEI	NO (8)	YES (9)	NO	NO	**NO**	

8. Table 1 Notes

1. RCS Flow - Unit 2 Operating Outside Functional Testing Criteria.

2. High Steam Pressure - Unit 3 Outside Functional Testing/Design/Screen Criteria. Pressure beneficial for reducing tube vibrations due to defective AVB design and increasing the turbine efficiency to deliver more power to the grid for higher profits.

3. Steam Flow - Units 2 & 3 Operating Outside Design Criteria

4. RCS Operating Temperature (Thot), - Units 2 & 3 Outside Functional Testing/10CFR 50.59/Design/SCE Specification Criteria of

5. Outside Industry Norm. In-plane Fluid Elastic Instability was observed when void fractions were > 99.3% in the Unit 3 hot-leg side with the highest heat flux, velocities > 31 feet/second and stability ratios > 1.15.

6. Tube-to-AVB contact forces > 30 Newtons are required to prevent in-plane fluid elastic instability per Mitsubishi Repair Plan with thicker AVBs, when void fractions > 99.3%. Higher the void fractions are, higher the steam velocities are. The thicker AVBs basically impose compressive forces on the tubes and restrain them from moving like a mechanical spring loaded clamp physically locks the tubes. SCE/MHI AVB team were relying on U-bends contacting one side the AVBs basically remaining elastically stable & staying on the AVBs due to their dampening strength. With dry steam in Unit 3 RSGs, high steam velocities exceeded the critical velocities of the tubes and dampening strength and tubes slided along the AVBs and contacted other tubes. However, wear patterns in Unit 2, which had operated the longest, did not have the wear pattern seen in Unit 3 steam generators, which showed evidence of extended tube wear scaring attributed to in-plane motion or vibration, with evidence of orbital tube movement relative to the anti-vibration bar and tube. These results demonstrate that high amplitude in-plane tube vibration did not occur in Unit 2 because the dampening strength of the tubes was not exceeded due to lower velocities. Lower void fraction wet steam.

7. ATHOS or Manual Calculations can predict only Bulk Void Fractions and not localized Peak Void Fractions due to differing tube-to-tube clearances and tube-to-AVB gaps, varying feedwater flows and RCS hot-side heat flux.

8. NRC Internal Case 1 Represents Unit 2 RSGs with no in-plane fluid elastic instability.

9. NRC Internal Case 2 Represents Unit 3 RSGS with in-plane fluid elastic instability.

10. Both Units 2 & 3 were being operated outside the RSGs Functional Acceptance Criteria. Unit 3 had more primary energy than Unit 2 resulting in higher velocities and higher void fractions compared to Unit 2.

11. New Information discovered from NRC Internal Notes on 7-18-2016 – "From the commissioning days in early 80s until late 90s both SONGS units operated with Tcold = 553°F. RCS temperature was reduced from 553 to 540...541°F in the late 90s to mitigate aging issues that developed in the original SONGS SGs with Alloy 600 tubing. Use of this reduced temperature program continued as aging issues arose with Alloy 600 use in the Reactor Vessel Head. With the replacement of the SG's and the Reactor Vessel Head and the use of stainless steel support materials and Alloy 690 tubing, SONGS planned to restore the RCS temperature to 550°F in the 2012 timeframe." It means that Units 2 & 3 replacement steam generators had more primary energy than the original Units 2 & 3 steam generators.

9. AVP Conclusions - Since Edison/Mitsubishi Engineers were not sure if the as-installed San Onofre Unit 3 Replacement Steam Generators (RSGs) will perform as-designed for out-of-plane U-bend support with "zero" gaps in the hot condition and no consideration for U-bend support in the in-plane direction., RSGs were functionally tested by Edison and Mitsubishi. The Unit 3 RSGs were tested at primary reactor coolant energy of 5,720 Million Btu/hour, steam pressure not to exceed 900 psi and a primary reactor coolant temperature not to exceed 598 degrees F. The RSGs passed functional testing and were accepted by Edison from Mitsubishi. When Unit 3 Tube Leak occurred on January 31, 2012, the Unit 3 RSGs were being operated by SCE outside the tested configuration at primary reactor coolant energy of 5,860 Million Btu/hour, steam pressure of 942 psi and a primary reactor coolant temperature between 602 and 611 degrees F. This additional primary energy of 140 Million Btu/hour (5,860-5,720) in Unit 3 RSGs above the tested configuration and additional primary energy of 50 Million Btu/hour (5,860-5,810) above the design configuration increased the steam quality, produced more quantity of higher velocities drier steam and created zero damping. These high amplitude in-plane tube vibrations caused tubes to slide over the AVBs with no in-plane friction force to restrain tube movement, make tube-to-tube contact, a tube leak in Unit 3, 2 practically leaking tubes, tube-to-tube wear in 326 tubes, 8 failed tubes, 1767 tubes-to-AVB wear and 463 tubes-to-TSP wear in 11 months or weeks (SONGS Unit 3 Root Cause Evaluation, NRC SONGS Lessons Learned).

NRC States, "Dry saturated steam at about 850 psia has a density of about 2.0 lbm/ft3, and the NRC, Mitsubishi and Westinghouse results predict that there are significant areas in the U-bend, in the affected region, where velocities are high and steam is nearly dry. Peak velocities and void fraction in the vicinity of angles of 30 degrees support the tube-to-tube wear patterns found in the Unit 3 steam generators."

AVP Note: NRC, Mitsubishi and Westinghouse did not make a similar observation for Unit 2 because velocities were lower and there was no dry steam in Unit 2. Only dry steam causes in-plane fluid elastic instability (Westinghouse & B&W RSGs Design Basis).

All the SCE and MHI Cause Evaluation are Speculative, Incorrect, Fake and Invalid. These conclusions were concurred by INPO Plant 1 NSSS Supervisor on 7/15/2016.

10. NRC ASLB May 2013 Ruling – "Section 10CFR 50.59 permits changes with respect to components (i.e., steam generators) without a license amendment under prescribed conditions that assure the replacement components are sufficiently similar to the original so that safety requirements are maintained or improved. 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. ' '(1) The restart of Unit 2 would grant SCE authority to operate without the ability to comply with all applicable technical specifications; (2) The restart of Unit 2 would allow SCE to operate beyond the scope of its existing license; and (3) SCE's Unit 2 Return to Service Plan includes a test or experiment that meets the criteria in 10 C.F.R. § 50.59 that require a license amendment.' 'SCE and its contractors have evaluated the in-plane tube-to-tube wear due to fluid elastic instability and have developed a theory to explain its occurrence and to predict how it can be avoided. As a result of comparing the thermal hydraulic conditions in the SONGS replacement steam generators with those of other steam generators, SCE concluded that the likelihood of fluid elastic instability will decrease if the steam quality in the steam generators is reduced (i.e., if the moisture content of the steam is increased). SCE determined that a reduced steam quality results in greater "damping" within the steam generators, which decreases the potential for fluid elastic instability. 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 [fluid elastic instability]."

11.. NRC Inspection Report September 2013 - Dry saturated steam at about 850 psia has a density of about 2.0 lbm/ft3, and the NRC, Mitsubishi, and Westinghouse results predict that there are significant areas in the U-bend, in the affected region, where velocities are high and steam is nearly dry. Peak velocities and void fraction support the tube-to- tube wear patterns found in the Unit 3 steam generators.

12. Notes from NRC AIT Internal Files Dated April 2013 Received by AVP on July 10, 2016

AVP - The quality of the NRC AIT Report improved with time, but not good enough to prevent retirement of Units 2 & 3 and all the radioactive, criminal, economic, nuclear safety and financial mess created by SCE's negligence and greed.

NRC Review of Arnie Gundersen's paper: More specifically, Fairewinds believes that the NRC would have identified the inadequacy of the Mitsubishi Heavy Industry computer-code-applied to validate the tube design and vibration prior to fabrication. Mitsubishi computer code was not simply capable of analyzing Combustion Engineering (CE) designs like San Onofre and was only qualified for Westinghouse design that are not similar to the original Combustion Engineering (CE) design. In NRC licensing jargon, the Mitsubishi design codes were not benchmarked for CE design.

AVP - NRC AIT Reviewer writes True meaning if SCE had applied for a NRC license amendment, the entire SCE created San Onofre Mess could have been avoided.

1. The inspectors identified a licensee-identified non-cited violation of 1 O CFR Part 50, Appendix B, Criterion 111, "Design Control" for the licensee's failure to verify the adequacy of

the replacement steam generator design for Units 2 and 3 with respect to the susceptibility of the retainer bars in the anti-vibration bar assembly to flow-induced vibration. 1 O CFR Part 50, Appendix B, Criterion Ill, Design Control, requires in part, that design control measures shall be established to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, SONGS failed to ensure that there was sufficient analytical effort in the design methodology of the anti-vibration bar assembly to support the conclusion that tube wear would not occur as a result of contact with the retainer bars due to flow-induced vibration. Consequently, the inadequate design of the retainer bar allowed the smaller diameter retainer bars to vibrate during normal operation to the extent of causing wear on the adjacent tubes, which challenged the integrity of the Reactor Coolant System boundary. The inspectors determined that SONGS failed to verify the adequacy of the retainer bar design as required by procedure S0123-XXIV-37.8.26

2. SONGS and MHI both had the opportunity and had numerous discussions about issues with FIT-III and the apparent disconnect between results that FIT-Ill provided as compared to other similar replacement S/Gs. In addition, many of the items that attributed to both the FEI and AVB issues appear to have been discussed and discounted during early design meetings (2005 time frame).

3. The lack of in-plane support was the result of SCE RSGs being designed for out-of-plane U-bend support with "zero" gaps in the hot condition and no consideration for U-bend support in the in-plane direction. For example, the visual observations from the Unit 2 and 3 did not show large gaps between the anti-vibration bars and tube. The anti-vibration bars appeared to be straight with no detectable abnormalities with weld caps or upper structure orientation. However, wear patterns in Unit 2, which had operated the longest, did not have the wear pattern seen in Unit 3 steam generators, which showed evidence of extended tube wear scaring attributed to in-plane motion or vibration, with evidence of orbital tube movement relative to the anti-vibration bar and tube. This result show that high amplitude in-plane tube vibration did not occur in Unit 2.

4. Modeling two-phase flows is very complex since it exhibits various flow regimes, or flow patterns, depending on the void fraction of the two-phase fluid and the flow rate. Additionally, flow patterns can be irregular or chaotic. However, averaged behavior based on conservation equations can be used to model one-dimensional steady state and transient two-phase flow in reactors and steam generators with control volume approach. Obviously, some simple

transients are relatively easy to model and can be validated with data while others are more difficult and data to validate the results are scarce. Three-dimensional analysis with axial, radial and tangential control volumes adds additional complexities of momentum equations for the added directions. The 30 code uses a porous media approach to represent local geometries and needs customized pre-processors to model modern designs of anti-vibration bars and tube support plates. The above analyses apply equally to Units 2 and 3 and so do not explain the accelerated FEI wear damage in Unit 3. This thermal-hydraulic model predicts bulk fluid behavior based on first principals and empirical correlations so it is not able to evaluate mechanical, fabrication, or structural material differences that may be unique to each steam generator.

5. Dry saturated steam at about 850 psia has a density of about 2.0 lbm/ft3, and the NRC, Mitsubishi and Westinghouse results predict that there are significant areas in the U-bend, in the affected region, where velocities are high and steam is nearly dry. Peak velocities and void fraction in the vicinity of angles of 30 degrees support the tube-to-tube wear patterns found in the Unit 3 steam generators.

13. The Parties and Documents, which directly, indirectly and partially support AVP Conclusions that Unit 3 RSGs tube-to-tube wear (In-plane Fluid Elastic Instability) was caused due to differences in San Onofre Units 2 & 3 Operational Data are as follows:

1. San Onofre Insiders
2. Nuclear Reactor & NSSS Engineers
3. International Industry PhD Chemical and Heat Transfer Engineers
4. Nuclear Industry Proprietary Studies
5. Retired Ph.D. NRC/Westinghouse Engineer
6. NRC Atomic Licensing Safety Board
7. NRC Independent Consultants
8. Industry Independent Consultants
9. Union of Concerned Scientists
10. Design of NRC Licensed IPFEI Palo Verde & ANO-2 RSGs

11. Portions of NRC Final Inspection Report

12. Mitsubishi Repair Plan

13. Portions of NRC AIT Report

14. Dr. Pettigrew's Research Papers

15. Design of South Korean RSGs

16. SCE's Design Engineers Published Industry Papers

17. Industry's Concerns Expressed in NRC Documents in response to SONGS Lessons Learned &. Westinghouse Operational Assessment