Large and ASSOCIATES

DESIGNATED ORIGINAL BY PM

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE PETITION REVIEW BOARD

Docket Nos. 50-361 and 50-362

DECLARATION OF JOHN LARGE
IN SUPPORT OF 2.206 PETITION
BY FRIENDS OF THE EARTH

I, John Large, hereby declare as follows:

1. My name is John Large. I am the Chief Executive at Large & Associates, located at The Gatehouse, 1 Repository Road, Ha Ha Road, London, United Kingdom SE18 4BQ.

2. I prepared the attached report:

   Review of Tube Wear Identified in the San Onofre Replacement Steam Generators - Mitsubishi Reports UES-20120254 Rev.0 (3/64) and L5-04GA588(0) together with Other Relevant Information, R3218-A2, March 27, 2013

I declare under penalty of perjury under the laws of the United States of America that the foregoing is true and correct.

Dated: March 27, 2013

BY: ______________________

John Large

REVIEW

MITSUBISHI HEAVY INDUSTRIES LTD

i) **ROOT CAUSE ANALYSIS REPORT FOR TUBE WEAR IDENTIFIED IN THE UNIT 2 AND UNIT 3 STEAM GENERATORS OF SAN ONOFRE NUCLEAR GENERATING STATION - UES-20120254 REV.0 (3/64), *C* OCTOBER 2013**

ii) **SUPPLEMENTAL TECHNICAL EVALUATION REPORT, L5-04GA588(0), *C* JANUARY 2013**

iii) **E BAUMGARTNER, MHI LETTER TO A T HOWELL, NRC OF FEBRUARY 25, 2013**

iv) **RESPONSE OF SOUTHERN CALIFORNIA EDISON TO MOTION OF FRIENDS OF THE EARTH AND WORLD BUSINESS ACADEMY FOR EXPEDITED CONSIDERATION OF CERTAIN PHASE III ISSUES, BEFORE PUBLIC UTILITIES COMMISSION OF CALIFORNIA, 12-10-013 MARCH 22, 2013**

L&A REF: R3218-A2 03 27 13

CLIENT: FRIENDS OF EARTH

LARGE & ASSOCIATES

CONSULTING ENGINEERS
LONDON

This is a review of the redacted MHI Proprietary *Root Cause Analysis* released by the NRC on March 8, 2013. This Review also considers a second MHI *Supplemental Technical Evaluation Report* released by the NRC on March 8, 2013, the MHI letter of February 25, 2013 introducing the redacted version of the RCA to the NRC, and the response of SC Edison to the FoE Motion made before the Public Utilities Commission of the State of California.

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TUBE WEAR IDENTIFIED IN THE SAN ONOFRE REPLACEMENT STEAM GENERATORS MITSUBISHI REPORTS UES-20120254 REV.0 (3/64) AND L5-04GA588(0) TOGETHER WITH OTHER RELEVANT INFORMATION

SUMMARY AND FINDINGS

This is a review of the redacted Mitsubishi Heavy Industries (MHI) Ltd proprietary *Root Cause Analysis* (RCA)[1] report released into the public domain by the Nuclear Regulatory Commission (NRC) on March 8 2013, following earlier calls for its release into the public domain by Senator Boxer and Representative Markey.

The copy of the MHI RCA report is complete comprising 68 pages of text and tables. The released non-proprietary version has been redacted at a number of locations, namely partially at pages 3, 10, 18, 19, 21, 25 and 26 with the majority of the text redactions removing component dimensions and other details of a proprietary nature – singly and overall, the redactions do not taken as a whole detract from the RCA narrative and findings.

This review also refers to a second MHI report released by the Nuclear Regulatory Commission (NRC) on March 8, 2013. This second report, the *Supplemental Technical Evaluation Report* (STE),[2] adds to the earlier MHI tube wear report included in the Southern California Edison (SCE) *Confirmatory Action Letter* (CAL) submission. Similarly, the STE report is redacted at locations throughout the 68 pages of text and figures, although *Section 4*, considering the joint SCE and MHI involvement during the design stages from early 2005, includes only a few isolated instances of redaction of what are obviously component dimensions and inter-component clearances.

Three further relevant documents became publicly available during the final stages of the preparations of this review, these are: the Intertek updated Operational Assessment,[3] the SCE response[4] of March 22, 2013 to the Public Utilities Commission motion made by Friends of the Earth, and the MHI cover letter[5] introducing the RCA to the NRC which provides further MHI commentary on the RCA and STE reports.

For a background of the steam generator tube degradation at the San Onofre Nuclear Generating Station (SONGS) there are a number of chronological narratives of the events leading up to the withdrawal of all four RSGs at SONGS, for example United States Nuclear Regulatory Commission Region IV, *San Onofre Nuclear Generating Station – NRC Augmented Inspection Team Report 05000362/2012007*, July 18, 2012, SCE, Enclosure 2, *Songs Return to Service Report*, October 3, 2012 and the Large & Associates Affidavit *Response to Atomic Safety and Licensing Board's Factual Issues*, January 22, 2013.

GENERAL FINDINGS

In the absence of a detailed rebuttal from SCE, the MHI RCA and STE reports claim that:

1 *San Onofre Nuclear Generating Station, Unit 2 & 3 Replacement Steam Generators, Root Cause Analysis Report for Tube Wear Identified in the Unit 2 and Unit 3 Steam Generators*, MHI UE-2012025 Rev 0, Non-Proprietary c October 2011, redacted form released by the NRC on March 8 2013 - hereafter the MHI RCA team undertaking the root cause analysis report will be referred to as RCA to avoid confusion with MHI when it was involved in the earlier design and manufacturing activities and, similarly, the MHI Supplemental Technical Report will be referred to as STE – although note that, essentially, RCA and MHI are the same entity and share the same commercial interests - the location of the text referred to in the RCA report is shown page number and paragraph thus [p53,¶6] .

2 Attachment 4, *Supplemental Technical Evaluation Report*, MHI Document L5-04GA564 Tube Wear of Unit-3 RSG – the location of the text referred to in the STE report is shown page number and paragraph thus {p53,¶6}.

3 SCE, Enclosure 1, Amendment I *Operational Assessment for SONGS Unit 2 Steam Generators for Tube-to-Tube Wear Degradation 100% Power Operation Case*, Intertek AES 13018304-2Q-1 March 2013, March 14 2013 - similarly, text location is shown <p2, ¶3>.

4 *Response of Southern California Edison to Motion of Friends of the Earth and World Business Academy for Expedited Consideration of Certain Phase III Issues*, Before Public Utilities Commission of California, 12-10-013 March 22, 2013 – similarly, text location is shown ¦p2, ¶3¦.

5 E Baumgartner, MHI Letter to A T Howell, NRC of February 25, 2013 – similarly, text location is shown |p2, ¶3| – text cited from other sources and reports are shown thus |p35, ¶5|.

1) SCE was involved in the overall and detailed design of the replacement steam generators (RSG) for the San Onofre Nuclear Generating Station (SONGS) at the specification and from the early stages of the design process, namely

   i) in September 2004 when SCE issued the *Certified Design Specification* (CDS) spelling out the design strategy of the anti-vibration bar (AVB) support systems that were to prove crucial in the tube degradation performance of the RSGs, and

   ii) from about May-June 2005 when SCE, along with MHI, formed the *AVB Design Team* charged with investigating, amongst other things, the high local void fraction[6] in the two-phase flow regime predicted by the MHI computer analysis of the then developing RSG detailed design;

2) In both of these roles SCE was involved in the thermal-hydraulic modeling of the two-phase flow regimes acting within the RSG, crucial for successful design, to the extent that

   i) in September 2004 SCE specified in the CDS that it required to *'approve'* the modeling software codes, and *'all thermal-hydraulic aspects of the RSG design'*; and thereafter

   ii) through to the end of 2006, as a joint member of the AVB Design Team, SCE was involved in identifying practical means of curtailing the high void fraction, some of which involved evaluation of very substantial design changes to the RSGs, although SCE, jointly with MHI, agreed not to implement any of these because *'unacceptable consequences'* would arise;

3) one difficulty and possible *'unacceptable consequence'* identified by MHI was SCE's CDS constraint clauses, including *Cl 3.6.1* stipulating the intended use of the provisions of 10 CFR §50.59 to minimize the impact of the RSGs on the existing plant licensing basis, and the *Cl 3.9.1* prerequisite to closely match the RSGs to the original Combustion Engineering SGs in '*in form, fit, and function*'; so that

4) any such modifications and/or departures from the original SG design should not impede the ability to justify the final RSG design under the provisions of 10 CFR §50.59, that is whereby SCE provided the NRC with assurance that the RSG design would not give rise to any detriment to the established SONGS nuclear safety case and, in doing so, there would be no need to apply for a *License Amendment.*

Overall, the unsatisfactory and clearly defective design of the installed RSGs at San Onofre introduced performance uncertainties and, some would claim, added risk of radiological incident at these two nuclear plants. It might be argued that with prior knowledge of a number of these uncertainties before SONGS Units 2 and 3 were returned to service operation, SCE should have taken the opportunity to revisit the 10 CFR §50.59 screening process - if it had done so then, surely, it would have foreseen the need for the RSG design and function to be subject to a License Amendment.

However, other than some dimensional modifications between the production Unit 2 and 3 that, incidentally, quite inadvertently served to delay (but not eliminate) the onset of tube-to-tube wear (TTW) in Unit 2, the SCE-MHI RSG design remained fundamentally flawed. Indeed, when installed and in operation at San Onofre, the RSG steamside flow regimes deviated from that modeled, so much so, that the feedwater recirculation ratio was under par, the tube pitch velocity was high and steam voidage extensive with boiling most probably developing in lower zones of the hot leg and, particularly in the vibration prone tube bundle U-bend, in-plane fluid excitation dominated. Moreover, the SCE specified AVB design functionality provided little or no-restraint in the *in-plane* direction or, where individual AVBs inadvertently provided such in-plane restraint, the flow mechanisms present were able to wear this down, leading to a situation where TTW was active and extensive (Unit 3) or where TTW was committed to but had yet to fully develop (Unit 2).

In effect, the outcome of the joint design effort of SCE and MHI requires the introduction of a unique regulatory regime tailored to the operating aberrations of the San Onofre Unit 2 plant – this will entail a re-interpretation of the license *Technical Specification* (TS) if the plant is resume power in compliance with its

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6 *Void Fraction* – is an index of the volume of steam to water in a two-phase fluid, expressed as a fraction of unity, so a Void Fraction of 0.75 comprises 75% vapor phase (ie steam) and 25% liquid (water) by volume. Local Void Fraction is an important contributor to flow induced tube rattling since increasing void is accompanied by decreasing viscosity and the ability of the two-phase fluid to dissipate or damp out any energy input to immersed structures (ie tubes subject to cross flows) from the *pitch* or *dynamic velocity* – the balance of dynamic velocity *vs* damping, with account of the stiffness (ie excitation frequency) of the immersed structure is described by as *fluid elastic instability* or FEI.

Operating License – in this way the RCA and the more recent STE report have bearing on the future operation of the San Onofre Unit 2 plant.

5) SCE's latest revision (March 18) to its response to the CAL, is the Intertek amended *Operational Assessment* (OA),[36] particularly (for the sake of argument here accepting Intertek's projections of the TTW wear rates *vs* probability of tube burst):

   i) Intertek's projected trends <p37, Figure 1.5-1> show that even at 70% reduced power output, the restarted Unit 2 RSGs would have a remaining operational life of about 1.35 years before the tube burst criterion is exceeded;

   ii) at which time (skipping the SCE nominated inter-inspection period of five months), the RSGs would have to be taken out of service and those tubes (TTW, TSP- and AVB-to-tube wear) failing the Operating License TS would have to be plugged and isolated from pressurized service operation, along with other zones of tubes requiring preventative plugging to suit the changing thermal-hydraulic flow regimes in the RSGs.

   In this way and depending on the number of tubes requiring isolation and preventative plugging, it might be possible for SCE to eke out the operating life of the Unit RSGs for two or three more fuel cycles before the percentage limit on the number of plugged tubes was reached, that is taking the plant to a state of unserviceability with four to five years compared to the intended design life of forty or so years.

6) However, the Intertek OA is presented with a number of shortcomings, particularly arising from its lack of transparency, including:

   i) there remains some ambiguity about Intertek's derivation of the tube-to-tube wear (TTW) coefficient from the Unit 3 tube wear experience, because this seems to assume that TTW commenced from the start-up of Unit 3 and was not delayed, thereby commencing at some time into Fuel Cycle 16 – this would result in a lower wear (rate) coefficient;

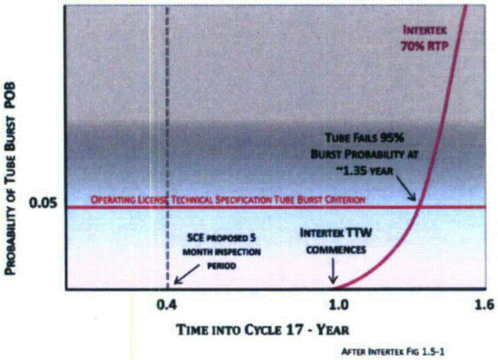
   ii) whereas the STE report shows the wear rates of both TTW and AVB-to-tube wear in Unit 3 were dominated by the impact wear coefficient, it is not clear if and how this much greater impact wear mode (x~20) is taken into account in the Intertek predictions;

   iii) the Intertek analysis is confined to TTW, only considering TSP- (tube support plate) and AVB-to-tube wear in respect to their respective contributions to the onset of TTW – this neglects the AVB-to-tube wear as potential source of a steam generator tube rupture in its own right;

   iv) Intertek's projection for the 70% RTP, Unit 2 Cycle 17 TTW comprises two phases (see graphed **line** – right):

      a) first, the AVB-to-tube restraint is worn away so that the AVB has little or no preload force in the in-plane direction – this first or prerequisite stage preceding the onset of TTW takes, according to Intertek, about 1 year, and

      b) second, the TTW wear rate is such that the tube passes the 95% tube burst criterion (—) in about 0.35 years thereafter – this means that the tube is considered to have failed at about 1.35 years into Cycle 17;



After Intertek Fig 1.5-1

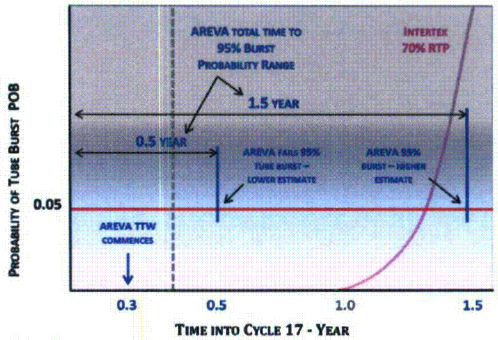
   v) Intertek's projections do not compare at all favorably with the same periods evaluated by AREVA OA.[7] For AREVA (show **THUS**):

7 Attachment 6 – Appendix B: SONGS U2C17 - ***Steam Generator Operational Assessment for Tube-to-Tube Wear***, AREVA – the data presented here relates to *Figure 8-3* but this has been declared proprietary information and thus cannot be reproduced here – instead the set points of the AREVA AVB and TTW wear phases have been taken from the same but non-proprietary information available in the text of Appendix B – see Large & Associates Affidavit *Response to Atomic Safety and Licensing Board's Factual Issues*, January 22, 2013.

a) the period whilst the AVBs are *'slackening off'* before TTW commences is ~0.3 year compared to 1 year by Intertek; and

b) similarly, the equivalent total time to tube burst being AREVA in the range of 0.5 to 1.5 year compared to ~1.3 year Intertek

Such uncertainties and differences with the other CAL OAs – here just comparison has been made to AREVA at the neglect of the other OAs - render the reliability of the Intertek wear rate and tube burst projections open to challenge.



In Summary: Essentially, SCE's Intertek (and other OAs) approach in justifying the restart of Unit 2 to nuclear power operation assumes that it is permissible, under regulatory safeguard, to operate a plant in which a key nuclear safety related component, the RSG tubing,[8] is permitted to progressively degrade, not necessary linearly, at an increasing risk of failure. Whereas, the structural integrity performance criterion (*Cl 5.5.2.11*) of the operating license TS requires that a ***margin*** against tube burst be maintained during normal operation, etc., it does not provide specifically for a situation where the ***margin*** against tube failure is reducing in a stepped and non-linear fashion – see forgoing diagrams and <p37, Figure 1.5-1>.

Put simply, the SCE approach in justifying the restart of Unit 2 calls for an interpretation of the TS along the lines that a tube burst is permissible after a certain period of projected operation (here for the AREVA lower band case of 6 months) if it is planned to remove the plant from operation before that time period expires - in other words, is it acceptable to have a higher probability of a tube burst developing as the length of time in operation nears the designated tube burst time. Put another way, can the TS be satisfied for a certain period and not satisfied for an additional time beyond that period - for example the AREVA 6 months to tube burst - or must the TS be satisfied, save for reasonably unforeseen anomalies, for the entire projected life of the RSG, say 30 to 40 years?

An important point here is that a ***margin*** is established and maintained to accommodate uncertainties, unforeseen variations, etc., being an intrinsic part of the system defense in depth that should not be forsaken to accommodate a design shortfall that could or should have been corrected during the design process. It follows that a sound interpretation of the Operating License Technical Specification is that a set margin against tube burst must be maintained for the entire, projected design life of the RSGs – that is for a period of 30 to 40 years.

SC EDISON'S KNOWLEDGE OF AND INVOLVEMENT IN THE DESIGN PROBLEMS

SCE's latest response to the FoE 10 CFR 2.206 Petition states that until recently (2012)[9] it had no knowledge of the design inadequacies that resulted in fluid elastic instability (FEI) (and, hence, accelerated tube wear) in the replacement steam generators (RSG) |p11, 2 ¶4|:[10]

"... *As a result of its **recent evaluations**, SCE has determined that MHI's thermal-hydraulic analysis code did not predict the fluid elastic instability that occurred in the RSGs. That concern, however, **was not known** during the design and manufacturing of the RSGs. Therefore, those concerns could not have been a basis for a license amendment and do not provide any basis for an allegation that SCE violated 50.59 in 2009-2011."*

my ***emphasis***

8 The RSG tubing is a key nuclear safety SSC (structures, systems and components) that forms the fission product boundary of the reactor coolant circuit

9 SCE is reported to have vigorously denied that it was aware of any design flaws in the steam generators during the design stages – LA Times, February 26 2013 *'Cost for troubled San Onofre plant? $400 million and growing'*

10 Docket N° 50-361 and 50-362 Response to Friends of the Earth 10 CFR 2.206 Petition, January 9, 2013

More recently, following public release of the RCA and STE reports, SCE stated:[11]

"... At no time was SCE informed that the maximum void fraction or flow velocities estimated by MHI could contribute to the failure of steam generator tubes ... At the time, the design was considered sound."

However, the RCA report suggests otherwise, giving account of when and the extent of SCE's involvement in and knowledge of the uncertainties and inadequacies of the RSG design, for example:

[p48, (3) ¶2] the [link] gives the location of the text extract in the RCA and directs to the where the point is considered in greater detail in this Review – my **emphasis** throughout this and following text extracts	*Also **MHI and SCE recognized** that the SONGS RSG steam quality (void fraction) was high and MHI performed feasibility studies of different methods **to decrease it**. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and **MHI and SCE agreed** not to implement them.*

Since void fraction[12] is a direct contributory factor of FEI, knowledge of a high void fraction will forewarn of the potential for FEI (and hence the risk of accelerated tube wear), thus MHI's computer modeling prediction of high void fraction directly related the possible presence of damaging levels of FEI in the RSG tube bundles.

Also, SCE's response analysis to FoE's Allegation |p18, Appendix 1|[10] states:

*"... At the time the RSGs were designed, MHI performed analysis that demonstrated that the **steam** {void fraction} in any area of the tube bundles **would be low enough to provide the required damping**, and that the quality of the steam in the vast majority of the secondary side of the steam generators would be even less. Furthermore, MHI analyzed the potential for fluid elastic vibration, and determined that conditions were stable SCE's root cause evaluation has determined that FEI did occur. However, **SCE had no evidence of that beforehand**."*

my **emphasis** and *{clarification}*

According to the RCA report, not so because SCE was part of the *AVB Design Team* charged with investigating the cause(s) of high void fraction (and the associated FEI) and how it could be eliminated from the RSG design:

[p17, ¶2]	*Early in the project, **MHI and SCE formed an AVB Design Team** with the goal of minimizing U-bend vibration and wear.*
[p22, ¶2]	*However, the AVB Design Team recognized that the design for the SONGS RSGs resulted in **higher steam quality (void fraction)** than previous designs and had considered making changes to the design **to reduce** the void fraction (e.g. using a larger downcomer, using large flow slot design for the tube support plates and even removing a TSP). But each of the considered changes had unacceptable consequences and the AVB Design Team agreed not to implement them. Among the difficulties associated with the potential changes was the possibility that making them could **impede the ability to justify the RSG design under the provisions of 10 C.F.R §50.59.***

11 Power Engineering *'Southern California Edison Comments on MHI Evaluation of San Onofre Nuclear Plant Steam Generators'*, Business Wire, March 8 2013

12 In gas-liquid two-phase flow, the void fraction is defined as the fraction of the flow volume that is occupied by the gas phase. The void fraction will vary from location to location in the recirculating flow around the SG tube bundle (depending on the two-phase flow pattern), it will fluctuate with time and its value is usually time averaged. In local FEI flow situations the fluid provides the damping or dissipation of the energy entering the situation, so its efficiency in the damping role reduces as the steam content increases (high void fraction) and, with this, is an increase in the two-phase fluid volume and a corresponding raising of the dynamic pressure ($v2$), to the extent that the energy input increases and the situation becomes unstable.

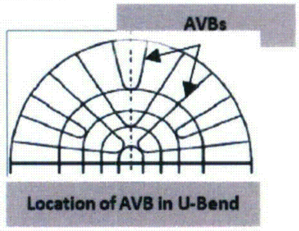
[p48, (3) ¶2]	*Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and **MHI and SCE agreed** not to implement them.*

The STE report provides further details about the SCE-MHI AVB Design Team {p51, ¶6}:

> *In **mid-2005** a **joint SCE /MHI AVB Design Team** was formed for the purpose of minimizing the potential for tube vibration and wear in the SONGS RSGs. For the first six months, video meetings were scheduled every two weeks and technical or design review meetings were held on a two month cycle.*

The RCA covering letter from MHI to the NRC also refers to SCE's involvement in the AVB Design Team Ip2, ¶2I:[5]

> *As part of the design process, Southern California Edison (SCE) and MHI formed a special AVB Design Team to develop an effective anti-vibration design, focusing on the design of the anti-vibration bars (AVBs) that provide support for the tubes in the U-bend region of the replacement steam generators.*



The primary role of the AVB assembly is to inhibit the onset of tube vibration in the U-bend region of the RSG tube bundle, so the *AVB Design Team* would have been acutely aware of the high void fraction being predicted at the time and, from that, of the need to control and suppress FEI in the U-bend region of the tube bundle - the importance of accounting for and managing FEI in recirculatory steam generators has been established for several decades.[13] Since SCE was a member of the *AVB Design Team* it would have been privy to all of this information, particularly since it had agreed with MHI not to implement certain changes to the RSG design to reduce the void fraction.

Similarly, when questioned by the NRC at the November 30, 2012 Public Meeting, SCE responded specifically on the issue of void fraction,[14] here alluding that it had not known of the high void fraction at the early stages of the design process (ie the *'2005 timeframe'*):[15]

> *"... Werner {NRC} - "Just so we are clear the underprediction of the velocity by FIT III was not recognised - the problem of the model when it was changed from square pitch to triangular pitch a number of years ago - but the void fraction even under FIT-III while not predicting 99.6% was predicting 95% which was still high and was a matter of concern back in the **2005 timeframe** – I know that still being looked that was **a matter of concern a number of feasibility studies were conducted to try to lower the void fraction before the steam generators were fabricated** but apparently it was not - so - we will need to understand that better as we go forward"*
>
> *... Palmisano {SCE} - "We have as well – we have asked MHI for a better explanation of that it and we are looking at it ourselves because as you say the void fraction was high it was not predicted as high 99.5% it was high **it was questioned ultimately** the calculations and the operating experience showed even with that void fraction the system should have been effective it was not – clearly thats a failure several reasons for that failure that have to be dealt with."*

my *{clarification}* and ***emphasis***

13 The seminal work in this area is *Fluidelastic Vibration of Heat Exchanger Tube Arrays*, Journal of Mechanical Design – Volume 100 – April 1978, H J Connors although there are earlier references to this topic which is rooted in the *Navier-Stokes Equations*, particularly *Stoke's Law* of the 1850s.

14 At a guess, the design intent for the SONGS RSG steam quality (~void fraction) would have been around a maximum of 90% in the U-bend area of the tube bundle, although value is redacted in the STE {13, Figure 2.2-1}.

15 NRC-Edison exchange at the *SONGS CAL Response Public Meeting*, November 30 2012 - 0 1hr 52 minutes into session.

Again, according to the RCA report:

[p48, (3)]	*A **special AVB team** was formed and included industry experts to conduct an extensive design review process in 2005 / 2006 . . . Also **MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high** and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and **MHI and SCE agreed** not to implement them.*

The STE report reiterates this with some additional detail {p56, ¶4.1.3}:

> *In the **May 2005** Design Review meeting, MHI presented an RSG performance calculation showing **high projected void fraction**. It was decided that MHI would perform a parametric analysis to determine **how the void fraction could be reduced** while maintaining the other design requirements.*

The MHI covering letter provides further topic-specific information on the particular interest of the AVB Design Team Ip2, ¶3I:[5]

> ***Specifically**, the AVB Design Team focused on preventing out-of-plane fluid elastic instability (FEI). design and fabricate the SONGS replacement steam generators to prevent tube wear.*

So it seems that the May 2005 Design Review Meeting, attended by both SCE and MHI, recognized undesirable implications (ie tube rattling and wear) of the *'high projected void fraction'* in the then developing RSG design. This most probably motivated the decision *'in mid-2005'* to form *'a joint SCE / MHI AVB Design Team'* and, if so, there is some ambiguity over Palmisano's response to the NRC that the high void fraction *'was questioned ultimately'* because, according to the RCA and STE reports, the AVB Design Team was constituted immediately following the MHI's reporting of the high void fraction result.

These comparisons between the highly confidential RCA report and SCE's published and public recollections provide quite different accounts of how much was known and by whom.

Put simply, according to the RCA, then with SCE being involved in the design process as part of the *AVB Design Team*, SCE would have known of the design uncertainties (eg unacceptably high void fraction and the direct link to FEI), and it would have had knowledge about this in or about May 2005. Thereafter, with its close involvement in the AVB Design Team, SCE would have been aware of and involved in attempts to modify the design of the RSG internals to reduce FEI contributory factors, including lowering of the high void fraction.

In fact, MHI state in both RCA and STE {p56, ¶4.1.3} reports that the AVB Design Team (comprising both SCE and MHI, advised by an {p55 ¶2} *'independent U-bend tube vibration expert'*) recognized a potential requirement to substantially modify the design:

[p22, ¶2]	*However, the AVB Design Team **recognized** that the design for the SONGS RSGs resulted in **higher steam quality (void fraction)** than **previous designs** and had considered making changes to the design to reduce the void fraction (e.g. using a larger downcomer, using large flow slot design for the tube support plates and even removing a TSP). But each of the considered changes had unacceptable consequences and the AVB Design Team agreed not to implement them.*

> *Over the next five months, MHI evaluated alternative design modifications to increase the RSG*

> *circulation ratio (and thereby reduce the maximum void fraction). The design alternatives included a larger downcomer, larger TSP flow area, and removing one TSP. None of these alternatives had a large enough effect on the maximum void fraction to justify such a significant change.*

SC EDISON'S INVOLVEMENT IN THE AVB DESIGN TEAM

a) **Inclusion of SC Edison in the Design Process:** The usual arrangement between the Client (here SCE or its nominee) and the supplier (MHI) was established via the *Certified Design Specification*. Typically this would have involved SCE at arms-length in decisions relating to details of the design, such as the AVBs.

Unusually, however, as the purchasing client, SCE built-in a requirement in the *Certified Design Specification* (reasonably assumed to be part of the contract) whereby it would specify the principal functionality mode of the AVBs. In this respect:

i) SCE required MHI to detail design and incorporate into the RSGs a type of AVB that resulted in a *'zero tube-to-flat bar gap'* - it is important to understand that under this arrangement alone SCE would **not have** itself designed the AVB, but it did specify basic and underlying geometry of the AVB that was to be developed and detail designed by MHI.

ii) SCE also specified that the AVB should function specifically in the *out-of-plane* (OOP) direction, but it was tacit on the functioning of the AVB in the *in-plane* (IP) direction.

iii) MHI was also required to submit the a) final design and b) method of manufacturer of the AVB to SCE for its approval.

However, the isolation of SCE from the AVB detailed design process changed with the formation of the AVB Design Team early following the May 2005 Design Review meeting - from that date, as a joint member of the AVB Design Team, SCE would have been confronted not only with the detailed design of the AVB assembly but also with the modeling results of the thermal-hydraulic two-phase flow regimes, including the all important void fraction and pitch velocity being critical contributory components to FEI.

b) **SCE Involvement in the AVB Design Team:** At some date early in the project, STE states this to be following the May 2005 Design Review meeting, SCE and MHI formed the *AVB Design Team*. This seems to have involved SCE in a much more hands-on role with both the design and manufacturing processes for the AVBs, including attendance at *'numerous technical and review meetings'* {p54, ¶6}:

> *The AVB Design Team **generated many action items** and **answered many questions**, several of which dealt with **high void fraction** and **how to minimize it**. This process **continued through the end of 2006**. The AVB team investigated instances of U-bend tube degradation using the INPO, NPE (Nuclear Power Experience), and NRC databases and studied whatever could be found describing the design of other similarly large SGs.*

Again according to RCA, via the AVB Design Team, SCE would have been instrumental in design decisions on crucial aspects of the AVB design strategy and application, including determining the number of AVBs, the tube and AVB dimensional control to achieve the *'zero-gap'* functionality and, along with this, formulating the strategy to minimize AVB-to-tube preload in order to minimize ding and dent indications.

As described by both RCA and STE reports, SCE's involvement in the overall design strategy, the detailed design of the AVBs and other aspects of the tube bundle, seems to have been within the terms of the *Certified Design Specification* (CDS).[26] In fact, the CDS is quite specific in regard to SCE's specification of the AVB design, thereby setting the AVB design strategy of *'zero tube-to-AVB gap-zero force'* - ie no preload in the *in-plane* direction [p19, figure]:

[p8, ¶3]	*3.10.3.5 . . . The Supplier shall **develop** and submit for Edison's **approval** an **Engineering** and **Fabrication** Gap Control Methodology describing control*

	*of an effective "zero" tube-to-flat bar gap, gap uniformity and parallelism of the tube bundle in the **out-of-plane direction** prior to tube fabrication.*

The choice of this *'zero gap'* AVB strategy, then recognized to be a fundamental design parameter, was a strong contributory factor that resulted in the undesirable vibration performance, AVB-to-tube wear and TTW of the RSGs when placed in service – this was specified by SCE in or about September 2004 and must have been adhered to through to October 2005 when the final 2V x 3 AVB detailed design was finalized and adopted. SCE as a member of the AVB Design Team, active between May-June 2005 through to at least October 2005, must have been fully aware of the AVB design and would have, most probably, participated in the development of the AVB detailed design.

However, SCE disputes the close and participatory role in the detailed design process via its joint membership of the AVB Design Team as referred to by the RCA and STE reports and, from the onset of the contract of September 2004, as specified by the CDS. In opposing FoE's Motion to the California Public Utilities Commission, SCE state that although it ¦p7, ¶1¦:[4]

> *. . . actively oversaw and challenged MHI's design of the RSGs. SCE, however, is not expert in steam generator design, and as provided in the parties.' contract, MHI was ultimately **responsible** for design decisions.*

This contrasts with MHI view ¦p2, ¶4¦:[5]

> *The design through an extensive design review process which included the participation of third-party experts. MHI and SCE recognized that the thermal-hydraulic conditions were high compared to previous steam generator designs. In particular, the higher steam quality was seen as increasing the potential for steam generator tube "dry out" (potentially increasing concentration of impurities on the tubes and increasing the risk of corrosion) and decreasing damping conditions for out-of-plane vibration.*

CONSEQUENCES OF SC EDISON'S INVOLVEMENT IN THE DESIGN PROCESS

a) **Cause of Tube Wear:** It is now established that the unacceptable incidence, rates and severity of tube wall thinning (tube wear) extant in the two Unit 3 RSGs arose because of a combination of:

   i) the lack of contact or preload force and friction force between the AV bar and individual tubes acting in the IP direction, directly because of the *zero tube-to-flat bar gap* design of the AVB specified and approved by SCE; and

   ii) the loss of tube motion restraint at the AVB-to-tube 'contact' points released the free-span sections of the tubes to vibrate, resulting in tube-to-tube wear (TTW); and also

   iii) where a degree of pre-load force (inadvertently) existed at some of the AVB-to-tube locations, adjacent tubes that had no pre-load[16] impacted on the restrained tubes, eventually wearing away the AVB points of restraint.

   The consequences of ii) and iii) foregoing are the direct result on the inappropriateness of the SCE-MHI jointly specified AVB function and, similarly, the final AVB detailed design.

   Similar levels of AVB-to-tube wear have developed in Unit 2 and further TTW is expected to occur if Unit 2 re-enters service.

b) **Other Tube Wear Incidence:** Two other forms of tube wear have been found in both Units 2 and 3 RSGs, these are:

16 No pre-load was the intended design condition – this could also result for tubes where the restraint had been worn away by tube motion in the unrestrained *in-plane* direction,

i) the degraded (worn and slackened) AVB-to-tube wear incidences are also most likely to have contributed to the TSP-to-tube wear; whereas

ii) the tube wear found in tubes adjacent to the retainer bar (RB) is not connected to the design and functioning of the AVBs.

MHI identifies the mechanistic causes of the various types of tube wear found in the SONGS Units 2 and 3 RSGs:

[p12, ¶5.3] my added *{explanation}*	*... fluid elastic instability {FEI} as the mechanistic cause of the tube to tube {TTW} wear, turbulence induced vibration (often referred to as "random vibration" because the excitation modes over time are unpredictable) as the mechanistic cause of the tube to AVB wear, and turbulence induced vibration of the retainer bar as the mechanistic cause of the retainer bar to tube wear.*

ROLE OF MHI IN THE DESIGN PROCESS

a) **RB-to-Tube Wear:** RCA admits that the design of the small diameter retainer bars was never checked by MHI at the design stage for susceptibility to resonate when subject to random fluid (turbulent flow) excitation.

b) **Thermal Hydraulic Modeling and FIT-III:** Similarly, it was solely MHI who conducted the thermal-hydraulic flow analysis using its FIT-III software, although as a member of the *AVB Design Team* SCE would have been aware of the thermal-hydraulic modeling results. As previously noted, the first results of the FIT-III seemed to have been reported to SCE at the May 2005 Design Review Meeting.

It transpires that MHI not only failed to adequately validate the FIT-III software but, also for the specific SONGS application, MHI incorrectly set up crucial flow modeling parameters[17] as these related to the triangular pitched tube sets of the SONGS RSGs. Even with the triangular pitch error corrected, generally, the FIT-III code under-predicted the tube gap (pitch) velocity by about a factor of x2 when compared to other established methods [TABLE 3].

The involvement of SCE in the thermal-hydraulic modeling of the RSGs, a crucial element in the early stage design process, is specified in the CDS [p21], the NRC[20] opines that SCE accepted and, it must be assumed, approved the use of the MHI FIT-III thermal-hydraulic code |p35, ¶5|:[18]

> *Mitsubishi's FIT-III thermal-hydraulic code was **accepted by SCE** for the design of the replacement steam generators.*

Again, because of its involvement in the SCE-MHI AVB Design Team from about May-June 2005, SCE learnt of the high void fraction prediction for the two-phase flow and, according to the RCA, SCE was involved in evaluation of *'making changes to the design to reduce the void fraction'* over the five months following.

In addition, a particular problem with the FIT-III thermal-hydraulic code that '*predicted nonconservative low velocity and low void fraction results*' has been identified by the NRC.[19] In this respect the NRC Augmented Inspection Team reviewed[20,21] the SCE-MHI cause evaluation for organizational and programmatic factors

17 This appears to a have been a simple arithmetical mistake.

18 San Onofre Nuclear Generating Station – *NRC Augmented Inspection Team Report 05000361/2012007 and 05000362/2012007*, July 18 2012.

19 NRC Inspection Reports 05000361/2012007 and 05000362/2012007 – Augmented Inspection Team, NRC Incident Management Program.

20 San Onofre Nuclear Generating Station – *NRC Augmented Inspection Team Follow-Up Report* 05000361/2012010 and 05000362/2012010, November 2012.

that might have resulted in the nonconservative low pitch velocity and void fraction results of the thermal-hydraulic model, although at the time of the AIT reporting (November 2012) the evaluation was still being finalized and had not then be approved by SCE.

c) **In- and Out-of-Plane FEI:** The FIT-III software could not account for IP FEI that, in the SONGS RSG flow geometry, was subsequently found to dominate the flow regime in the TTW zone of the tube bundle hot-leg to U-bend region.

Jointly in their roles on the AVB Design Team, both SCE and MHI failed to recognize the dominance of FEI and other fluid flow driven phenomena in the IP direction and, accordingly, it did not modify the AVB design to provide effective (design function) restraint to IP tube motion and wear.

On its part, MHI considers this [p22, ¶4] *'because contemporary knowledge and industry U-tube SG operation experience did not indicate a need to consider in-plane FEI'*. However, it might be construed to be somewhat disingenuous to suggest that *in-plane* FEI is a novel and unresearched phenomenon - it is not, there being a wealth of research papers, guidance notes and standards many of which predate the design of the SONGS RSGs.[13, 22] In other words, knowledge of the fluid flow regimes and structural response in both OOP and IP directions is very established and well understood. In this respect, there is no reason why the joint SCE-MHI AVB Design Team should have had any reason to overlook analyzing the potential instability in the IP direction in the tube bundles of the SONGS replacement steam generators.

LICENSE AMENDMENT PROCESS – 10 CFR §50.59

The information provided in the RCA Report strongly hints that because the then FIT-III predicted void ratio was high, the RSG designs were comprehensively reviewed *'in 2005/2006'* – see (DIAGRAM 1).[23] However, by that date in the design and manufacturing processes, the design should have been largely settled with the purchase of bought-out items, particularly with the vast quantity of customized tubing ordered, if not then already in manufacture.

Although RCA admits that SCE and MHI as the *AVB Design Team*, had then jointly considered making some quite substantial changes to the RSG design in order to reduce the higher steam quality (void fraction) - *'using a larger downcomer, using large flow slot design for the tube support plates and even removing a TSP'* – SCE seems to have held sway with its commitment to minimize the *'change'* of the RSG design, over the original Combustion Engineering SGs, in order not to provoke a license amendment under the 10 CFR §50.59 process.

The RCA report specifically identifies SCE's requirement to comply with the 10 CFR §50.59 screening process and therefore circumvent the need for a License Amendment:

[p22, ¶2] my truncation . . .	*However, the AVB Design Team recognized that the design for the SONGS RSGs resulted in higher steam quality (void fraction) than previous designs and had considered making changes to the design to reduce the void fraction . . .* ***Among the difficulties associated with the potential changes was the possibility than making them could impede the ability to justify the RSG design under the provisions of 10 C.F.R §50.59.***

21 By all accounts, the NRC is still reviewing and/or has yet to publish the AIT findings (mid-March 2013).

22 Much of the steam generator tubing specific applications work was completed by a Westinghouse employee H J Connors, whose standard work was published as *Flow-Induced Vibration and Wear of Steam Generator Tubes*, Nuclear Technology, V55, November 1981, although interest in this area with SG manufacturers having their own design guidelines, as well as the American Society of Mechanical Engineers (ASME), there are many publications and guides on this topic which predate the SONGS design phase, for example Au-Yang M K, *Flow Induced Vibration of Power and Process Plant Components*, 2001 ASME. Pettigrew and the others cited in the MHI RCA and STER reports are latecomers to this long established and well understood phenomenon.

23 DIAGRAM 1 is a *'rough-and-ready'* compilation of the revision dates given in the RCA Time Line diagrams [p55 to 59] but which are also interesting in another important respect. This is because many of the revisions referred to would have been raised at the various Design Review and AVB Design Team meetings so, it follows, there must be minuted action lists, or similar, for these meetings – such action lists would provide a greater and more reliable insight into the roles and responsibilities assumed by each of the parties in the overall and detailed design processes.

And, similarly, the RCA *'Change Analysis'* concludes:

[p48, ¶(3)] my truncation . . .	*Also MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and MHI and SCE agreed not to implement them. It was concluded that the final design was optimal based on the overall RSG design requirements and constraints.* ***These included physical and other constraints on the RSG design in order to assure compliance with the provisions of 10 C.F.R. §50.59.***

And, similarly, the STE report sets out the constraint on the design imposed by SCE {p51, ¶4.1.1}:

> *The general design requirements, performance requirements, and design criteria for the SONGS RSGs were set forth in SCE's "Certified Design Specification (CDS), SO23-617-01 (Ref. 8)". Significant features of the CDS were* ***the intended use of the provisions of 10 C.F.R. §50.59 to minimize the impact of the RSGs on the existing plant licensing basis*** *(CDS 3.6.1) and the requirement to closely match the dimensions and function of the OSGs (CDS 3.9.1). These features meant that the RSGs needed to* ***"be as close as possible to the existing steam generators in form, fit, and function"*** *(CDS 3.6.1.1).*

As does the MHI covering letter to NRC Ip3, ¶1I:[5]

> *Because the physical size and number of tubes of the replacement steam generators had already been established in accordance with the requirements of the SCE design specification, including the requirement that the replacement steam generators be designed to criteria consistent with the application of 10 C.F.R. 50.59, the evaluated design changes, while significant in scope, had minimal impact on the circulation ratio.*

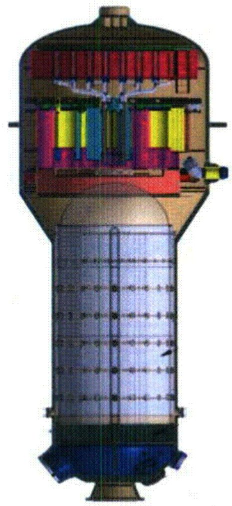
In effect, this 10 CFR §50.59 constraint, that seems to have been imposed solely in order to avoid the need for a license amendment, meant that {p51, ¶3} *'overall RSG had to fit within the size, weight, and volume limits related to those of the OSG, the tube bundle heat transfer area was to be maximized'* – an engineering challenge that SCE and MHI failed to meet, with this being the fundamental root cause of the severe and intolerable levels of tube degradation experienced in the San Onofre Units 2 and 3 replacement steam generators.

TUBE WEAR IDENTIFIED IN THE SAN ONOFRE REPLACEMENT STEAM GENERATORS

MITSUBISHI UES-20120254 REV.0 (3/64)

SAN ONOFRE REPLACEMENT STEAM GENERATORS

After 25 years of operation, the Southern California Edison (SCE) operator of the San Onofre Nuclear Generating Station (SONGS) replaced the original steam generators of Units 2 and 3 reactor power plants. The four Mitsubishi Heavy Industries (MHI) replacement steam generators (RSGs) were installed and commissioned into service in April 2010 and February 2011 in the Unit 2 and 3 plants respectively.



Essentially, each RSG comprises a cylindrical jacket inside which is a bundle of 9,700 or so tubes receiving hot, pressurized water directly from the nuclear reactor (primary) circuit (RPC). The primary water flows up, around and down inside the individual tubes of the U-bend tube bundle entering and exiting from the divided calandria lower section of the RSG. To prevent boiling in the primary circuit, the operating pressure inside the tubes is maintained at about twice that of the surrounding steam raising (secondary) circuit. Each RSG increased the number of 0.75 inch diameter by 0.043 inch wall thickness tubes from 9,400 to about 9,700, with the design differing substantially in a number of respects to the original SGs, particularly in the restraint means of the individual tubes and tube bundle in the top or U-bend region of the RSG.

Feedwater, from the separate steam raising (secondary) circuit, enters the top annular section of the jacket to flow down, around and up through the tube bundle, extracting heat from the outer surfaces of the tubes and forming a two-phase fluid of water and steam. This two-phase fluid develops as it flows over the tube banks, progressively diminishing in liquid (water) whilst increasing in vapor (steam) content, accompanied by an increasing volume and, hence, rate of flow (velocity) across the tube arrays making up the tube bundle. The steam component is collected and dried in the uppermost and larger diameter domed section of the RSG jacket, where it exits to the main steamline to drive the turbine sets located outside the reactor island containment

WITHDRAWAL FROM SERVICE OF THE RSGS

On January 31 2012, while the Unit 2 was in refueling outage, the virtually identical Unit 3 was forcibly shut down when an alarm alerted SCE operators that a breach had occurred with reactor primary circuit (RPC) water leaking across the RSG tube interface to the secondary steam circuit. This leak emanated from a single tube, although subsequent post-shutdown and reactor cool-down, non-destructive inspection of all of the tubes in the Unit 3 RSGs revealed very significant rates of tube wear in both RSGs. For example, each Unit 3 RSG exhibited approximately 5,000+ indications of wear localities, with many tubes having wear indications at more than one locality and of differing degrees of wear severity, with a total of about 900 individual tubes affected in each. Because of the depth and length of certain of the tube wear scars, a number of tubes were subjected to in situ hydrostatic pressure testing in March 2012, which resulted in 8 tube failures, all located in one of the Unit 3 RSGs.

As a result of the tube failure in Unit 3, the tube bundles of the Unit 2 RSGs were subject to further and more detailed inspections. These inspections revealed unacceptably high incidences of tube wear mirroring certain aspects of the Unit 3 and, as a result, Unit 2 was held shut down pending further investigation and approval from the Nuclear Regulatory Commission (NRC) to restart.

NUCLEAR SAFETY FUNCTION OF THE RSGS

Coupled to and receiving high pressure water directly from the reactor pressure vessel, via the primary circuit, each RSG fulfils three functions:

- the RSG tubes collectively provide the large heat transfer surface necessary to raise steam in the separate, secondary steam raising circuit operating at a lower pressure than the RPC;
- in the event of an untoward event in the RPC, for example loss of pumping power, the SGs continue to dissipate the post-trip decay heat in the reactor fuel core by natural circulation; and

- the SG tubing forms (the by far largest) part of the reactor coolant pressure boundary (RCPB), thereby confining radioactive fission products in the RPC from the secondary steam circuit and, ultimately, uncontrolled release to the environment.

Rupture of a single RSG tube is itself a design basis accident (DBA), with the plant systems expected to cope with such an event without incurring any significant radiological consequences off-site. However, multiple RSG tube failures/ruptures (primary-to-secondary leakage) present a number of very serious safety implications and radiological consequences that are beyond the DBA, including opportunity for sufficient volume of the radioactive reactor coolant water to bypass the primary containment (ie the domed structure) of the nuclear island.

CONFIRMATORY ACTION LETTER OF MARCH 2012

Following its own further investigation, the Nuclear Regulatory Commission (NRC) issued a *Confirmatory Action Letter* (CAL) in March 2012 specifying detailed prerequisite actions that had to be completed by SCE before restarting either or both Unit 2 and 3 nuclear power plants. SCE undertook further investigations and inspections of the defective RSGs, it consulted with MHI who reported its reasoning for the accelerated rates of tube wear and failures, and it instructed external consultants (AREVA, Intertek APTECH and Westinghouse - WEC) each to prepare independent *Operational Assessments* (OAs) based on returning the Unit 2 plant to service at a 70% thermal power rating specified by SCE.

On the basis of this OA preparatory work and with its previous excessive wear and preventative tube plugging, in October 2012 SCE submitted its response to the CAL and its application to restart Unit 2 to nuclear operation to the NRC. Essentially, SCE claims to have fulfilled the prerequisites of the CAL, particularly that the cause of the excessive tube wear was fully understood, that further tube wear could be managed, and that it was safe to return the Unit 2 plant to operation at 70% power at no additional radiological risk to members of the public.

UNDERSTANDING THE ROOT CAUSE OF THE TUBE DEGRADATION

SCE nominates the root cause of the tube degradation to be [p4, ¶6][24]

> *"... The mechanistic cause of the tube-to-tube wear was identified as FEI {fluid elastic instability}, involving the combination of localized high steam/water velocity (tube vibration excitation forces), high steam void fraction (loss of ability to dampen vibration), and insufficient tube to AVB contact forces to overcome the excitation forces. The corrective actions to prevent recurrence of FEI include lowering power operations to reduce tube excitation forces and improve the ability to dampen vibration..."*
>
> my clarification {...}

thereby confining itself to the *'mechanistic cause'* rather than identifying the underlying circumstances and decisions that resulted in the failure of the RSG design.

In fact, the most probable underlying cause of the failure is rooted in SCE's reasoning that the RSGs were to replicate the original Combustion Engineering SGs as closely as practicable. This, it argued, would enable the RSG replacement program to proceed without having to seek an amendment to the Operating License via the 10 CFR §50.59 screening process.[25] There are a number of sources of this resolve of SCE to almost doggedly comply with 10 CFR §50.59 and therefore avoid entering the route of License Amendment that would have been required if the RSGs differed significantly in design and function to the OSGs.

For example, the recently disclosed MHI STE[2] report identifies specific clauses of *Certified Design Specification*[26] (CDS) that constrained the RSG design [p51, ¶4.1.1]:[2]

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24 SCE, *Root Cause Evaluation*, Ref 201836127 Rev 0, REC May 7 2012

25 NRC Part 9900 10 CFR Guidance *10 CFR 50.59 Changes, Tests and Experiments*, March 13 2001

26 Certified Design Specification SO23-617-01 Rev 3, 2004 – bound to the 2004 contract between SCE and MHI for the supply of the RSGs,

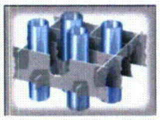
> "... *Significant features of the CDS were the intended use of the provisions of 10 C.F.R. §50.59 to minimize the impact of the RSGs on the existing plant licensing basis (CDS 3.6.1) and the requirement to closely match the dimensions and function of the OSGs (CDS 3.9.1). These features meant that the RSGs needed to "be as close as possible to the existing steam generators in form, fit, and function" (CDS 3.6.1.1).*"

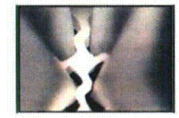
The first publicly available example of this requirement to comply with §50.59 seems to have been stated by SCE when presenting |Slide 8|[27] to the NRC at a public meeting in June 2006.

In effect, compliance with §50.59 required the RSG designer/manufacturer MHI to conform to the overall dimensions of Combustion Engineering OSG design, facilitate the same reactor coolant circuit temperature, pressure and flow conditions and satisfy the Technical Specification[28] of the Operating License. MHI hint at the strictures placed upon the design in its technical evaluation report of the tube wear [p10, Summary]:[29]

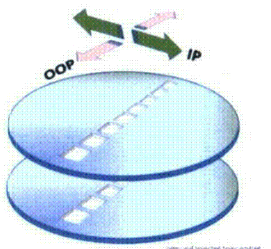
> "... *The SONGS RSGs were specified, designed and fabricated as replacements on a **like-for-like basis** for the original steam generators in terms of fit, form and function with limited exceptions, and were replaced under the **10CFR50.59** rule. The CDS{Certified Design Specification} for the design and fabrication of the RSGs (SO23-617-01, Revision 3) specified the limiting design parameters and materials. Thus, replacement steam generator design with 3/4" tube diameter arranged in 1" triangular pitch, which was the same as in the original steam generators, and the larger heat transfer area than in the original steam generators, was optimal. The other parameters/materials not specified by CDS were established/ selected in the design process. The SONGS RSGs were designed and fabricated to achieve an "effective zero gap" as required by CDS Rev. 3 in order to minimize its potential for tube wear...*"
>
> my ***emphasis*** and *{added explanation}*

However, 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 (ie 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 (right).



Although SCE has not (publicly at least) explored how these and other smaller departures from the OSG and, generally, conventional steam generator design, contributed the changes in the thermal-hydraulic environment of the secondary steamside circuit, very certainly the fluid flow within the RSGs was quite unique over the original Combustion Engineering OSGs in at least two important respects:

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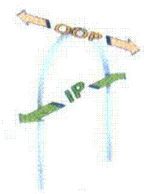
27 ML121350603 - Meeting Handouts from June 7, 2006 Public Meeting with Southern California Edison to Discuss Steam Generator Replacement Project Overview. (22 page(s), 6/7/2006)

28 NRC *Attachment 1, Volume 7, San Onofre Nuclear Generating Station, Improved Technical Specifications Conversion*, ITS Section 34 Reactor Coolant System (RCS), June 2010.

29 Attachment 4: MHI Document L5-04GA564 - Tube Wear of Unit-3 RSG Technical Evaluation Report

1) **Fluid Elastic Instability Activity:** First, the flow conditions in the hot-leg side of the U-bend comprised high flow velocities and increasing void fraction (increased fraction of steam making up the two-phase fluid), to the effect that the energy input to the tube structures, the dynamic pressure ($\sim\rho v^2$), exceeded the means of energy dissipation or output, essentially the damping provided by the fluid which is strongly and inversely related to the steam content. Increase of steam content (eg higher void fraction) reduces the damping and, correspondingly, the larger specific volume of the two-phase fluid results in a higher impinging velocity. In other words, if the velocity is sufficiently high and the void fraction large, the energy balance can only be maintained with the tubes themselves dissipating the energy by being induced into mechanical motion – the point at and beyond which this energy system becomes unstable this is referred to as fluid elastic instability (FEI). Since the FEI spreads over rows and columns of tubes, many tubes are likely to be induced into relatively large amplitude oscillatory (vibratory) motion to the extent that physical clashing will occur.

2) **In- vs Out-of-Plane FEI:** The second aspect unique to the SONGS RSG was the dominant direction of the FEI being in the *in-plane* (IP) direction, that is with the flow disturbance in the line of the principal axis of the tube bundle - along the *columns* of tubes - rather than, as expected, in the *out-of-plane* (OOP - side-to-side) direction across the tube *rows* that is typically found on conventional steam generator design. In the *in-plane* axis the individual tube wrap over the U-bend is significantly stiffer, so less likely to vibrate in a low frequency but greater amplitude motion, than the tube's less stiff *out-of-plane* axis.



If and how the flow area and flow path changes to the steamside of the RSGs determined the *in-plane* dominance and the high void fraction has not been explained by SCE, although the high flow resistance of the broached TSPs may have resulted in a much lower circulation of the steamside fluid body, so low as to produce very high void fraction (>90%) in the hot-leg U-bend zone, a fact that did not escape the NRC[30] when questioning SCE at the November 30, 2012 Public Meeting:[31]

> "... *Werner {NRC} - "Just so we are clear the underprediction of the velocity by FIT III was not recognised - the problem of the model when it was changed from square pitch to triangular pitch a number of years ago - but the void fraction even under FIT-III while not predicting 99.6% was predicting 95% which was still high and was a matter of concern back in the 2005 timeframe – I know that still being looked that was a matter of concern a number of feasibility studies were conducted to try to lower the void fraction before the steam generators were fabricated but apparently it was not - so - we will need to understand that better as we go forward"*
>
> ... *Palmisano {SCE} - "We have as well – we have asked MHI for a better explanation of that it and we are looking at it ourselves because as you say the void fraction was high it was not predicted as high 99.5% it was high it was questioned ultimately the calculations and the operating experience showed even with that void fraction the system should have been effective it was not – clearly thats a failure several reasons for that failure that have to be dealt with."*

SCE's recollection of its knowledge of the design situation in 2005/2006 seems contrary to the MHI RCA account:

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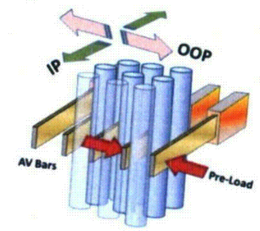
30 Of course by that date (November 2012) NRC had the benefit of hindsight and, particularly, the ATHOS thermal-hydraulic results as described by the redacted STE report {p22, ¶2} '*Fig. 2.3.4-1 shows the results of the three-dimensional thermal hydraulic analysis of SONGS Unit 2 and 3 SGs. This analysis was performed after the discovery of the tube wear, using the ATHOS computer code developed by EPRI. The highest void fraction is located in the 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 is | |. 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*' – my unredacting with data from the NRC statement.

31 NRC-Edison Public Meeting, November 30 2012 0 1hr 52 minutes into session.

> "... *A **special AVB team** was formed and included industry experts to conduct an extensive design review process in 2005 / 2006... Also **MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high** and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and **MHI and SCE agreed** not to implement them.*
>
> my ***emphasis***

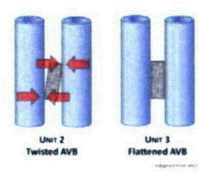
The TSPs and AVBs provide points of restraint that arrest tube motion. The TSPs acting at seven locations along each of the hot- and cold-leg sections of the tube bundle capture the individual tubes, via a broached piercing which minimizes lateral and radial tube motions whilst allowing some flow continuity at that particular tube location. The twelve AVBs act to restrain the tubes in the *out-of-plane* (OOP) direction by the tubes reacting against the AV bar which, itself, reacts against the next and successive *rows* of tubes. In this way the system of sandwiched AVBs obtains stiffness and restraint via the collective inertia of the tube bundle. Normally, because the dominant direction of motion experienced in SG tubing is in the *out-of-plane* direction (that is the least stiff axis of the individual U-bend section of a tube), the restraint acting against tube motion in the *in-plane* (IP – along the *columns* of tubes) direction is considered of secondary importance.



In the SONGs RSGs the AVB design strategy intended to achieve a *'zero bar-to-tube gap'* functionality when in the hot, pressurized condition. This meant that at *zero-gap* although the AV bar and tube would be in contact, there would be no clamping or preload force present between the successive AVBs and tubes. This *zero-gap-NO-contact force* functionality was to minimize point contact with the tubes and the undesirable formation of dings and dents in the tube wall but, to disadvantage, no contact force meant that the friction force restraining tube motion in the *in-plane* direction (both in-and-out and up-and-down) was also zero or minimal.

In effect, any significant *in-plane* fluid excitation force had the potential to excite individual tubes into motion in the *in-plane* direction because of the lack of *in-plane* restraint wore down the surfaces of the ABV and tube, leading to greater freedom to move in both *in-* and *out-of-plane* directions. Individual tubes so released in this way were free to vibrate and impact against neighboring tubes, with their motion similarly wearing down the AVB-to-tube restraint, thus releasing that tube to repeatedly *'bump'* against its neighbor, eventually leading to the progressive wearing down of whatever level of unintended preload force existed at each AVB. This 'bumping' mechanism provides a means by which loss of AVB-to-tube effectiveness can advance through the area of the tube bundle where *in-plane* fluid forces were active – it is also characterized with a time dimension as each AVB-to-tube contact wears away and permits the individual tube motion to be transferred to its neighbor.

The second role of the AVB system is to curtail the free-span tube length between successive AVBs. Pinning down the individual tubes in this way, effectively chops the free-span U-bend into (here 12) short sections between the hot- to cold-leg top TSPs. This raises the OOP fundamental frequency of vibration of the tube free-span sections between each successive AVB restraint location with, in the optimum design, the resonant frequency being taken above any excitation frequency active in the fluid (turbulence, vortex shearing, etc). Even in situations where the OOP FEI is vigorous, the lower amplitude motion of the pinned short sections of free-span tube will tend to govern and inhibit tube-to-tube clashing and TTW. The effectiveness of this second role of *out-of-plane* restraint is strongly influenced by the *in-plane* degradation of the AVB when subject to the *'bumping'* phenomena. Progressive bumping loss of AVB effectiveness along a single tube results in longer free-span tube length, a lower fundamental resonance frequency and a higher amplitude vibration for that tube, leading to tube-to-tube clashing and TTW.



Incidentally, tube *'bumping'* explains the different time periods over which AVB-to-tube wear and TTW occurred in Units 2 and 3. The incidence of TTW was delayed in Unit 2 because the tube bundle in each to the two RSG had been assembled with distorted or warped AVBs, whereas the same AV bars of Unit 3 were corrected[32] to achieve the design intent of *zero-gap-no-contact force*. The warped AV bars effectively pre-loaded ➨⬅ the AVB-to-tube contact with an *in-plane* friction force, thereby unintentionally providing the AVB with effective *in-plane* restraint that has delayed, but not halted, the advance to TTW.

MHI *Root Cause* Evaluation and Reporting

In or about late March 2012, MHI formed a specialist team of engineers to determine the causes of the tube failures and excessive in-service wear of the RSGs. The MHI *Root Cause Analysis* (RCA) was undertaken by MHI under its own *Corrective Action program* comprising, according to RCA, *"after-the-fact hindsight-based analysis"*. During about the same period of time SCE launched its investigation involving three external consultants for what it describes as *'Operational Assessments'* in support of its *Return to Service* proposal.

There seems to have been no coordination and/exchange of information, etc., between the RCA evaluation and SCE commissioned investigations with, indeed, RCA proclaims:

a) *"... These results and much of the information considered in this evaluation were not available to the organizations, management, or individuals during the period that relevant actions were taken and decisions were made."*

The RCA team produced a highly confidential report in or about October 2012 – this RCA internal report has now been released by the NRC in a slightly redacted form.

Structure of the RCA Investigation

The RCA team split the investigation in to a) physical causes and affect, and b) human factors.

A) Physical Causation and Affect

In scope, RCA considers the various modes of tube wear and the different fluid interactions that give rise to the individual tube and component motions.

Tube Wear: RCA considered the i) individual tubes fretting against each other to produce tube-to-tube wear (TTW), and three specific components, the ii) retainer bars (RB), iii) anti-vibration bars (AVB) and iv) tube support plates (TSP) in contact with tubes with relative motion between the two giving rise to tube wear.

Fluid Excitation: The fluid interaction considered were a) fluid elastic instability (FEI) in the *in-plane* (IP) direction and, separately, b) fluid turbulence also referred to as random vibration.

The RCA conclusions can be summarized as follows:

Table 1 RCA's Analyses of Various Modes of Tube Wear[33]

Tube Wear Mode	RCA Wear Type	FEI	Tube Random Excited Vibration	AVB Assembly	Comments
TTW	1	■ in plane		□ inactive AVB	FEI positively identified in U-bend region
AVB	2		■		FEI not positively identified
TSP	3		■		FEI not positively identified
RB	4			■ retainer bar	RB vibrates - no tube motion active

B) Human Factors

A combination of investigative tools was deployed with the following overall conclusions on the root causes:

32 This distortion occurs when the length of bar is wrapped back on itself to form the 'V' or hairclip shape, with the bar warping particularly towards the nose section. For Unit 2 the AV bars were subject to a pressing flat force of ~3 tons, but this did not completely flatten the bar so for Unit 3 the pressing force was increased to ~10 tons.

33 Attachment 4: MHI Document L5-04GA564 - Tube Wear of Unit-3 RSG Technical Evaluation Report, Mitsubishi Heavy Industries SO23-617-1-M1538 Rev 0.

1) **AVB Contact Force:** During the design stages, there was insufficient definition of an objective specifying the need for a prescribed level of contact (pre-load) force at each of the AVB-to-tube contact points.

2) **Retainer Bar Vibration:** The design overseeing process failed to ensure that the retainer bars was subject to an analytical check for vulnerability to flow induced excitation and large amplitude vibration.

As a result of 1) and 2) three in-house design and control procedures were revised.

CONTRACTUAL OBLIGATIONS BETWEEN THE PARTIES

The RCA report provides little insight into the contractual obligations between the parties, SCE and MHI, and nothing on any overseeing involvement of the Nuclear Regulatory Commission (NRC).

Thermal-Hydraulic Modeling: Modeling the two-phase flow regimes (thermal-hydraulic) within the steamside of the RSG is a crucial, early stage element in the early stage design process. The NRC[20] opines that SCE accepted and, it must be assumed, approved the use of the MHI FIT-III thermal-hydraulic code |p35, ¶5|.[34] The requirement for SCE to approve the thermal-hydraulic code to be used by MHI seems to have been is specified in terms of a *Performance Analysis Report* referred to in Cl 3.8.2 of the CDS:[35]

"... *3.8.2 **Performance Analysis Report** (Thermal Hydraulics Report)*

> *The Supplier shall prepare and submit for **Edison's approval** a Performance Analysis Report documenting **all thermal-hydraulic aspects of the RSG design** ... The Performance Analysis Report shall include all computer codes and modeling for the thermal-hydraulic performance of the RSGs...*
>
> *The **thermal-hydraulic design parameters** for the RSGs are specified in Table 3A-1. The numerical values of these parameters are either **imposed** or shall be proposed by the Supplier. Based on the values of these parameters, the Supplier shall calculate the expected thermal-hydraulic performance of the RSGs. Where applicable, the calculations shall be performed for a power level range from 0 to 100% power. The report shall also include the mathematical model, analytical methods and data used to calculate the RSG performance...*
>
> *The Performance Analysis Report shall contain, as a minimum, the following information:*
>
> ...
>
> > *Detailed calculations and graphs showing circulation ratio vs. power level. Include graphs showing the expected circulation ratio as a function of tube bundle fouling and tube plugging level (up to the allotted maximum)...*
> >
> > ...
> >
> > *Detailed calculations and data showing secondary side cross-sectional (horizontal) flow velocities at selected elevations within the RSG. These should specifically address velocities on top of the tubesheet and where boiling and sludge accumulation may occur...*"

my ***emphasis*** and *truncation...*

This CDS clause requires SCE to approve *'all thermal-hydraulic aspects'* of the RSG design which certainly would have included the FIT-III modeling of the two-phase fluid flow regimes throughout the RSG. What cannot be determined, however, is the extent to which SCE specified the thermal-hydraulic design parameters (void fraction, recirculation ratio, pitch velocity, etc), because of the unavailability of Table 3A-1.

SCE also specified in the CDS what is expected of the supplier (MHI):

"... *9.3.7 Tube Supports*

---

34 San Onofre Nuclear Generating Station – *NRC Augmented Inspection Team Report 05000361/2012007 and 05000362/2012007*, July 18 2012.

35 The CDS extract here is taken from a list of excerpts from SO23-617-1, although the revision number and authenticity cannot be verified.

The RSG shall be equipped with tube supports that adequately support the tube bundle and ***facilitate internal circulation****. The tube supports shall be of a broached plate type and . . . the tube support design shall:*

Preclude tube damage due to wear caused by ***flow induced vibration*** *(FIV) . . . Minimize* ***secondary side pressure loss*** *. . Provide the tube-to-tube support contact length such as to* ***minimize tube wear****. . .Ensure that the relative tube/tube support motions during* ***normal*** *and accident transients shall not result in* ***tube lockup****.*

. . .

*The Supplier shall address analytically support design as related to RSG thermal-hydraulic performance (****flow rates, pressure drops, circulation ratios, vibrations****, etc.) and . . . Tube supports shall be designed to ensure that the* ***potential for "dryout"*** *(presence of high quality fluid) is minimized at the tube-to-tube support intersections.*

The Supplier shall specify the parameters of ***" minimum dryout"*** *(****maximum fluid quality****) . . . The Supplier shall* ***provide experimental justification*** *demonstrating that "dryout" does not occur with the optimized design selected for the tube supports and tube bend supports.*

The Supplier shall address ***flow-induced*** *and* ***turbulence-induced vibration of the tube supports*** *to demonstrate that fatigue failures, and excessive fretting and wear of the tubes will not occur. The tube arrangement and support design shall ensure that the* ***effective cross-flow velocity*** *at design conditions for any span will be such that a sufficient margin exists to prevent tube high cycle fatigue . . .* ***Specifically****, the Supplier shall demonstrate that its design will* ***minimize vibration-induced tube wear*** *or fatigue in the* ***tube bend area*** *of the tube bundle. The Supplier shall perform a stability analysis of the tubes both in the tube bend region and over the straight length.* ***All thermal-hydraulic aspects*** *of the tube support design listed above shall be documented in the* ***Performance Analysis Report*** *. . . "*

my ***emphasis*** and *truncation . . .*

So, clearly, from the onset (September 2004) SCE was aware of the importance of settling the crucial thermal-hydraulic parameters early in the design process. The first reporting of the presence of a high void fraction was at the Design Review Meeting of May 2005 {p56, ¶4.1.3} which provides a set date at which SCE would have been aware that some design revision or change was necessary to avert the potential for *'dryout'* and the implications that this had for RSG performance and accelerated tube degradation.

AVB Design and Manufacture: RCA reproduces a short text extract [p8, ¶2] of the SCE *Certified Design Specification SO23-617-01 Rev 3*:

b) *". . . 3.10.3.5 . . . The Supplier shall* ***develop*** *and submit for Edison's* ***approval*** *an* ***Engineering*** *and* ***Fabrication*** *Gap Control Methodology describing control of an effective* ***"zero" tube-to-flat bar gap****, gap uniformity and parallelism of the tube bundle in the* ***out-of-plane direction*** *prior to tube fabrication."*

my ***emphasis***

This specification is of interest in that it clearly states (although note that it is presented in isolation by RCA) that there should be no pre-load force acting on the tubes, and hence no friction force between the AV bar and tube to inhibit movement in the IP direction. The specification also requires the *'gap statistical size . . . shall not exceed 0.003 inch'* which effectively restrains *out-of-plane* (OOP) motion of the tube.

Also, according to this extract, SCE not only specified this crucial design parameter but was also required to approve the manufacturing process (*Fabrication Gap Control Methodology*).

This clear demarcation between the parties is schematically shown as follows:

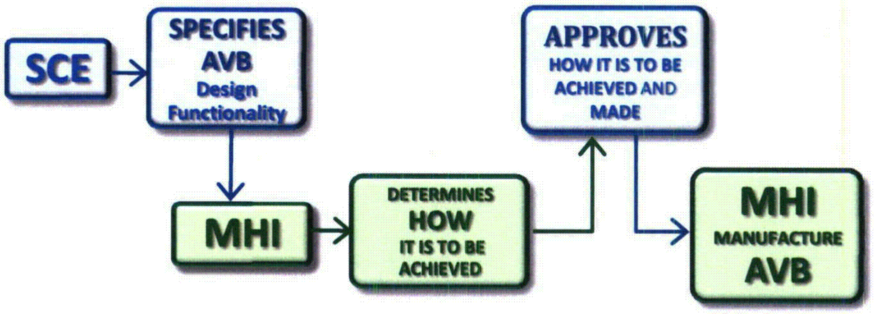


FIGURE 1 AVB DESIGN ARRANGEMENTS ACCORDING TO THE CERTIFIED CONTRACT SPECIFICATION

The *Certified Design Specification* (CDS - see *3.10.3.5 extract* above) is quite clear with, at the pre-design contract stage, SCE specified both:

a) the underlying type of AVB restraint (*zero tube-to-flat bar gap*) which, because the gap is specified (*0.003"*), can only mean that there was to be no contact pre-load force and, hence, no friction force restraining tube motion in the IP direction; and

b) that this design objective should apply in the OOP (*out-of-plane*) direction, at the disregard of tube motion in the IP direction.

In other words, according to this contract clause SCE specified, but did not itself design, the function of the AVB effectiveness – that was for MHI to design (*develop*) – and once that it had been designed, SCE was to certify (*approve*) the design fit for purpose in both design (*Engineering*) and manufacturing (*Fabrication*) senses.

However, RCA makes a second reference to SCE's role [p17, ¶2]:

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

- *Six (6) V-shaped AVBs ... between each tube column (12 ... around the U-bend).*
- *Tube and AVB dimensional control ... effective "zero" tube-to-AVB gap ... to maximize the effectiveness of the supports ...*
- *Excessive preload contact force was to be avoided in order to minimize ding/dent indications, and to maintain mechanical damping and thus minimize tube vibration."*

my ***emphasis*** and *truncation* ...

This undated modification to the contractual arrangements, as claimed by RCA, considerably changes the role and responsibility of SCE in the AVB design and manufacturing processes:

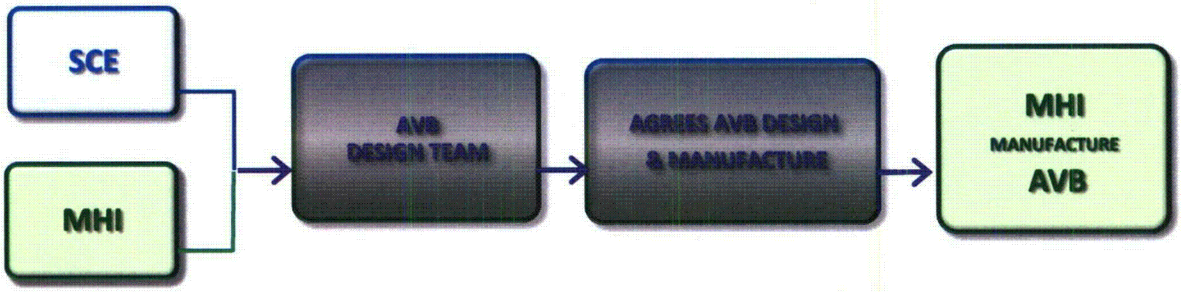


FIGURE 2 ARRANGEMENTS ACCORDING RCA SCE/MHI COMBINED AVB DESIGN TEAM SPECIFICATION

Interpretation of these arrangements suggests that SCE had much hands-on involvement with decisions relating to the detailed design and manufacture of the AVB. In this joint role SCE was involved at all stages of the design that arrived and approved the incorrect design option for the AVB restraint and, moreover, it was also involved in the flawed method of manufacture which resulted in warped AVB bars that were subsequently corrected for the later manufactured Unit 3 RSGs (incidentally, reducing the quite fortuitous pre-loading of some of the AVB-to-tube contact points that were, as it transpired to be, the most obvious reason for the delay the onset of TTW in Unit 2).

Pinpointing the Date of the RSG U-Bend Design Difficulties: Elsewhere in the RCA report the date of formation of the AVB Design Team is given [p48, (3)]:

d) "... *A* ***special AVB team*** *was formed and included industry experts to conduct an extensive design review process in 2005 / 2006 to optimize the U-bend design and address* ***the technical issues****. The team concluded that the SONGS design was significantly more conservative than previous designs in addressing U-bend tube vibration and wear.* "

my emphasis

and the *'technical issues'* alluded to are identified in the immediately following paragraph of the RCA text [p48, (3) ¶2]:

e) "... *Also MHI and SCE recognized that the SONGS RSG steam quality (void fraction) was high and MHI performed feasibility studies of different methods to decrease it. Several design adjustments were made to reduce the steam quality (void fraction) but the effects were small. Design measures to reduce the steam quality (void fraction) by a greater amount were considered, but these changes had unacceptable consequences and* ***MHI and SCE agreed*** *not to implement them. . .*"

my emphasis

In other words, at a relatively early stage of the design process, in or about 2005 to 2006, both SCE and MHI were aware of the unacceptably high void fraction, they jointly (via the AVB Design Team) evaluated methods to reduce this, but both agreed not to implement any significant modifications because [p48, (3) ¶2] because:

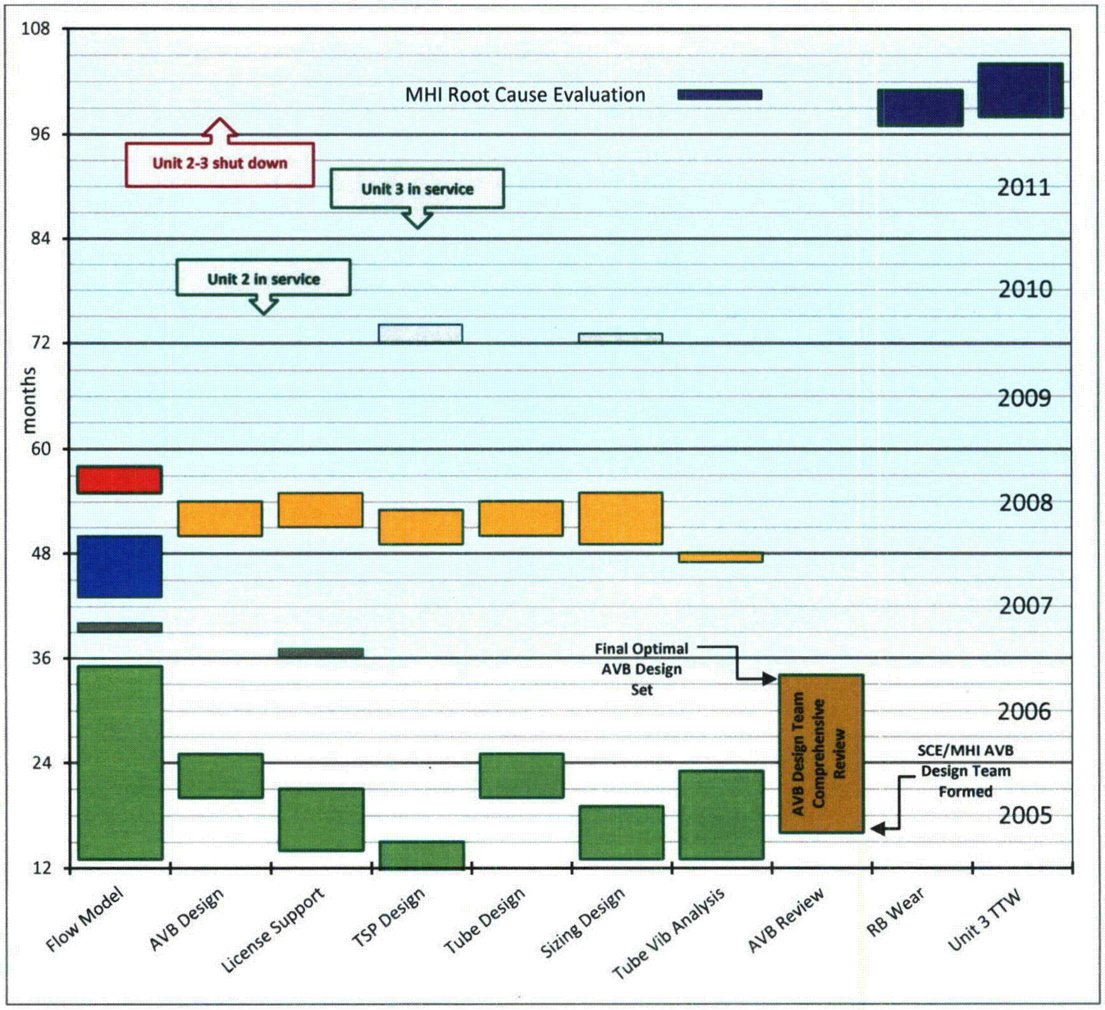
f) "... *These included physical and other constraints on the RSG design in order to assure compliance with the provisions of 10 C.F.R. §50.59.* "

Design Development and Changes – 10 CFR §50.59: Not unexpectedly, during the development of any custom engineered product issues arise requiring changes and/or modification as the design evolves.

The RSG design process would have involved a number of substantive issues (fluid flow, AVBs, TSPs, etc), each comprising a wide range of topics (eg AVB position, numbers, contact force, vibration, etc) – the iterative solution of the individual topics complete, as a whole, the issues each of which marks a progressive step along the path of the completion of the design overall. A means of tracking the design progress is to follow the revisions to the design over the period from order to final manufacture and commissioning. Essentially, this assumes that each revision represents the close-out of a particular design topic, thus yielding a sense of progress of and activity levels in the topics of interest.

For the SONGS RSGs the overall period stretched from the initial order, in or about September 2004, through the six year period to 2010 and 2011 when the Units 2 and 3 RSG were installed and commissioned into service at San Onofre.

DIAGRAM 1 TIMELINES FOR SELECTED DESIGN ACTIVITY TOPICS



As evident from **DIAGRAM 1,** most of the design activity ■ seems to have been settled by 2005 to late 2006. Note that the critical topics *AVB Design* and *Review*, and *Tube Vibration Analysis* seem to have been closed out early in 2005-06, although the flow modeling continues through with revisions being raised through to late 2008.

Of interest here is the flurry of activity ■ across the topics in the first half of 2008 following activity in the thermal-hydraulic flow modeling ■ (FIT-III and/or ATHOS) in late 2007, and which is topped off with final bout of flow modeling ■ in the 3rd to 4th quarter of 2008.

Of course, such deduction that the *AVB Design Team* (SCE and MHI) found something amiss with the SONGS RSG design around May 2005 is based on much speculation around the recorded design, etc., revisions. Nevertheless, if the design was proceeding smoothly and without hitch, it is difficult to reason why i) the FIT-III thermal-hydraulic modeling continued through to early 2008 and why this is followed by ii) the second flurry of design activity ■ occurring over the first half of 2008.

RCA states [p8, ¶1] that

g) *". . . The Certified Design Specification ,. . . states that SCE intended to use the provisions of 10 CFR §50.59 as the justification for the RSG design, which imposed physical and other constraints on the characteristics of the RSG design in order to assure compliance with that regulation."*

In other words and if the RCA is correct, by minimizing the departure from the original Combustion Engineering SG the *'change'* specified in the 10 CFR §50.59 process would be acceptably small so as not to require a license amendment. In taking this line SCE must have been cognizant to the risk that significant change to the design and/or performance (both for normal and abnormal operation) of the RSGs could have resulted in a knock-on or crosscutting demand on other components of the RPC.

In fact, at some time during the design process, although it is not stated when, RCA claim that the *AVB Design Team* (SCE and MHI) recognized a potential requirement to substantially modify the design [p22, ¶2]:

h) "... *However, the AVB Design Team recognized that the design for the SONGS RSGs resulted in higher steam quality (void fraction) than previous designs and had considered making changes to the design to reduce the void fraction (e.g. using a larger downcomer, using large flow slot design for the tube support plates and even removing a TSP). But each of the considered changes had unacceptable consequences and the AVB Design Team agreed not to implement them. Among the difficulties associated with the potential changes was the possibility than making them could impede the ability to justify the RSG design under the provisions of 10 C.F.R §50.59...*"

Clearly, additional design effort was given to evaluating major changes in the RSG design, for example *'larger downcomer, larger flow slots, etc'*. However, RCA does not elaborate further on the *'unacceptable consequences'* which seem, on the face of it, to be confined to the sphere of activity of the *'AVB Design Team'*. That said, such major component changes (*larger downcomer*) would have, surely, been beyond the purview of the *AVB Design Team* alone.

An interesting aside is given by Item CA 6 of the *Corrective Action Matrix* [p27, S6.0] requiring MHI to review and correct other areas of the primary circuit pressure boundary (of which the RSGs form just one part):

i) "... *other SG design procedures and primary pressure boundary components (Reactor vessel, Core internals, Pressurizer...) using senior engineers to determine if other design features have assumptions that are not programmatically captured and evaluated.*"

And, also by RCA's concern about the interrelationship between the RSG and the other parts of the primary pressure boundary design [p32, S7]:

j) "... *Root Cause 1 is associated with the design program and procedures not capturing necessary design elements affecting the primary pressure boundary... the extent of cause applied to the SG design program and areas of design outside the SG program that could impact the primary pressure boundary.*"

my truncation ....

This could suggest that the proposed revisions to the design to *'reduce the void fraction'* of the involved measures that affected the performance of other components in the RPC. There is a strong hint of how possible modifications to the RSG design might crossover to other RPC components in another document (*CDS Confirmation Items related to SG Re-design – Nov 09 2012*).

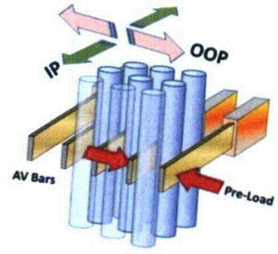
The RCA team concludes that [p47, (2)]:

k) "... *The primary conditions related to the thermal hydraulic and tube supporting condition that are now understood to be related to in-plane FEI are as follows:*

• *Increase in tube bundle heat transfer surface area (11%)*	Result of changing from Inconel 600 to Inconel 690 because of lower heat transfer coefficient.
• *Increase in number of tubes (5%)*	Arises from Inconel 600/690 switch – means an increase of the average steamside flow velocity.
• *Removal of Stay Cylinder*	Frees up space to accommodate tubes but added tubes on topside of tube sheet fill rise column and inhibit recirculatory flow around tube bundle.
• *Change from lattice bars to trefoil broached tube support plates*	Introduces greater flow resistance to recirculatory flow and necessitates added flow slots in tube plate in out-of-plane direction
• *Change from CE to MHI Moisture separators*	
• *Power level / operating temperature / tube plugging margin...*"	

my added comment

General Review of the RCA Report UES-20120254 Rev.0 (3/64)



It is now accepted that the principal cause to the excessive rates and severity of tube wear sustained by the SONGS RSG derives from omission in the AVB design to specify a pre-load acting on the tubes between the bar components of the AVB. Simply, the pre-load force clamps the tubes between successive lines of AVBs thereby inhibiting tube motion in both the *in-plane* IP and *out-of-plane* OOP directions. At an early stage SCE specified a *'zero gap'* AVB strategy which resulted in no contact and no preload force and, hence, no or marginal friction force restraint in the *in-plane* direction.

Comparisons with Other Steam Generators

RCA identify the replacement steam generators at Fort Calhoun nuclear power plant to be the only other MHI SGs operating in the United States [p10, ¶4.0], noting that all MHI SGs worldwide are of a different design, having a variety of tubes sizes and pitches, and operating conditions. Although not specifically stated, it is implied that the Fort Calhoun RSGs deployed a similar AVB design.

Comparison of the significant design parameters for the SONGS RSG compared to the Fort Calhoun RSGs are as follows:

Table 2 Comparison of Design Parameters SONGS -v- Fort Calhoun

Design Parameter	SONGS	F Calhoun
Tube Diameter and Wall Thickness	=	=
Tube Pitch and Geometry	=	=
Pitch to Diameter Ratio	=	=
AVB to Tube Gap	Zero Gap/Contact†	Larger Gap/No Contact
Overall RSG Size		Smaller
No of AVBs		Fewer
No if Tubes		Fewer
U-bend Radius		Smaller
Maximum Steam Void		Lower
No of Fuel Cycles	1	3
TTW, AVB, RB and TSP Wear	TTW, AVB, RB and TSP	None Reported

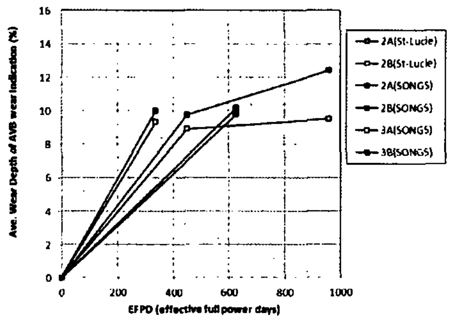
† aimed for but not achieved with any consistency

However, there is reference [p17, ¶8] to a *'similar'* plant whose SGs which had experienced *'only a small number of wear occurrences'*, although this comparative plant is not identified. RCA also provides other comparisons between the SONGS RSGs and MHI triangular tube pitch designed SGs [p46, Change Analysis].

The MHI STE report {p34, ¶3.2}[2] gives a second but unidentified example as *'Plant-A'*

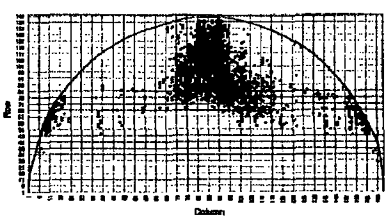
> "... *Tube wear patterns similar to those observed at SONGS were reported at the Plant-A large U-bend steam generators that were replacements for CE manufactured OSGs (See NRC ADAMS ML11270A015 and ML093230226). The Plant-A steam generators were designed by another vendor. They are slightly smaller than the SONGS steam generators but have U-bend tubes, flat bar AVBs, and BEC type TSPs, that are similar to the SONGS RSGs, except SONGS features a 12 AVB design and Plant-A has an 8 AVB design.*"

In fact *'Plant-A'* is readily identified from the NRC ADAMS ML references to be Unit 2 of the Combustion Engineering St Lucie nuclear power plant located at Hutchinson Island, Florida. The SGs at St Lucie were replacement units designed by AREVA that were brought into commission in 2009. The St Lucie RSGs, although slightly smaller than the SONGS steam generators, have U-bend tubes, broached TSPs, and the original stay cylinders replaced, like the MHI SONGS design, with a plain division plate, although the tube occupancy in the bundle space above the freed-up central tube sheet area is not known.



Like SONGS, the anti-vibration supports are flat bar AVBs, although St Lucie deploys 8 AVBS compared to 12 AVBs at SONGS. It is not known if the AREVA St Lucie AVB strategy is either preloaded or, as at SONGS, nominally a *zero-gap-zero-contact* design.

Not unlike SONGS Unit 2, the tube degradation at St Lucie is dominated by AVB-to-tube wear The distribution and numbers of AVB-to-tube wear incidences within the tube bundle is not dissimilar to the SONGS Unit 2, although the wear depth at St Lucie seems to have reached a plateau (right) {p37, Figures 3.2.1-4 and 5}.[2] The reason for this apparent plateau is unclear - it may be indicative of the type of tube vibration mechanism or an effect of the support condition, but it is clear that the number of tubes with tube-to-AVB wear at St Lucie is increasing.



The high incidence and dominance of AVB-to-tube wear at St Lucie, (right) which is not believed to have zones of unacceptably high void fraction and pitch velocity in the tube bundle, generally endorses the MHI finding that the SONGS AVB-to-tube incidences arise because of random fluid mechanisms (ie turbulence) and not FEI. If this is a correct diagnosis then SCE's plan to reduce the Unit 2 operation to 70% rated thermal power, thereby eliminating FEI, may be to no avail because there is no guarantee that the same or, indeed, new random fluid mechanisms will persist at such a reduced power level.

DETAILED TECHNICAL EVALUATION UNDERTAKEN BY RCA

The RCA investigation team referred to and it is assumed relied upon previous technical evaluations, including specifically:

1) Tube Wear of Unit-3 RSG Technical Evaluation Report, L5-04GA564, Rev 9
2) Retainer Bar Tube Wear Report, L5-04GA561, Rev 4
3) Validity of Use of the FIT-III Results During Design, L5-04GA591, Rev 3
4) Supplement Technical Evaluation Report L5-04GA588 draft

These internal MHI reports are not publicly available.

Tube Wear Types: The RCA investigation concentrates its effort on what it considers to be three significant wear categories:

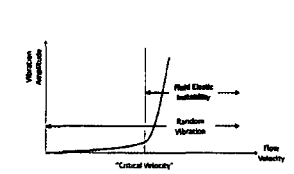
1) **TTW due to IP FEI** - found in the U-bend region, located between AVB points of contact in the free-span sections, with many of the TTW tubes also exhibiting AVB- and TSP-to-tube wear – for tubes with TSP-to-tube wear at the top TS plate, the investigation concluded [p13, ¶5.3.1] that *'the entire tube, including its straight region, is vibrating'*.

   This wear category attracts an interesting footnote [p13, f1] suggesting that i) certain tubes were impacting on stationary tubes and ii) that the two TTW tubes in Unit 2 had distinctive wear scars to those more established TTW tubes in Unit 3.

2) **Retainer Bar-to-Tube Wear** - the worn tubes were free of other wear scars along their lengths so, RCA concludes, that the source of the wear is a vibrating RB excited by random (turbulent) fluid flow locally.

3) **AVB-to-Tube Wear but No Free-Span Wear** - tubes with wear at the AVB-to-tube contact points, where there is no free-span TTW and where the AVB scar is short and localized.

Main Wear Mechanisms: RCA provides a somewhat textbook explanation of fluid elastic instability (FEI) accompanied by the rather generalized diagram (left), although it suggests that a fuller and more SONGS-specific analysis is presented in the unavailable *Supplemental Technical Evaluation Report* {see previously item 4), p28}.



However, RCA makes two incisive statements [p15, ¶2], these being:

l) *"... due to ineffective support for the tubes in the in-plane direction resulting from the very small and uniform tube-to-AVB gaps, some of the tubes exceeded the fluid elastic critical velocity resulting in in-plane FEI, which in turn produced large amplitude tube-to-tube wear."*

m) *"... tubes with low or no contact force in the region of highest void fraction are most susceptible to this mechanism .. "*

In other words, here RCA is acknowledging the cause of TTW to be a combination (or coupling) of IP FEI and lack of a pre-loaded clamping force acting on the tubes at the AVB locations. There are similarly pointed statements relating tube wear to random vibration [p16, ¶1]:

n) *"... tube wear at the AVB intersections with no wear in the tube free span sections is due to turbulence induced vibration caused by insufficient contact force between tube and the AVBs due to very small, uniform tube-to-AVB gaps"*

Meaning that the AVB wear and loss of IP effectiveness is caused by random fluid turbulence and not FEI (so AVB deterioration could continue even if Unit 2 was returned to service at 70% power).

On TTW the finding [p16, ¶2] that

o) *"... the wear scars indicate that tubes were generally vibrating in their first fundamental in-plane mode, which implies that none of the twelve (12) AVB supports were restraining tube motion."*

p) *"... Yet, it also indicates that the tube-to-AVB gaps are very small and uniform, because none of the tubes exhibited out-of-plane FEI, which is the tube's preferential fluid elastic vibration mode."*

RCA follows these two conclusions that *'the tube-to-AVB contact forces were negligible'* and that *'the tube-to-AVB gaps... were very small'* by which it acknowledges that the original design intent was achieved (ie *"zero" tube-to-flat bar gap*). In other words, here RCA admits that the intended design feature of no pre-load at the AVB-to-tube contact localities was a major contributing factor in the AVB- and TSP-to-tube wear, and TTW experienced in Unit 3.

It follows, that the similar AVB- and TSP-to-tube wear pattern established in Unit 2 is clearly a portent of developing TTW should Unit 2 be restarted.

Also, MHI's design analysis assumed just one of the AVBs to be ineffective against OOP [p18, ¶4], concluding that this showed an adequate margin against FEI excitation

RCA then argues that the original design, which did not take account of IP FEI, arose because the industry practice was then that IP FEI did not have to be considered if OOP FEI was controlled [p16, ¶3]. However, the weakness here is the omission of consideration of random turbulence that RCA earlier admits was a strong factor in the deterioration of the AVB effectiveness [p16, ¶1]. This should be considered alongside RCA's caution that [p17, 3rd bullet]:

q) *"... Excessive preload contact force was to be avoided in order to minimize ding/dent indications, and to maintain mechanical damping and thus minimize tube vibration."*

RCA also makes much of the role that the inter-surface (AVB-tube) liquid film plays in damping (ie dissipating-out the tube motion), emphasizing that in the higher U-bend region the higher steam quality (void fraction) reduces the damping [p20, ¶2]. This is a confused, if not incorrect, interpretation of the hydrodynamic forces in play at the AVB-tube interface – MHI or SCE's other consultants do not quantify this effect upon which certain of their arguments for restarting Unit 2 rely.

Modeling the IP and OOP FEI: The other significant failing of the MHI design process relates to the errors produced by its own in-house FIT-III software routines [p21, ¶1-4].

Interestingly, RCA does not provide a convincing argument why its FIT-III software did not identify any level of IP FEI activity, other than the countenance that the other industry-standard flow modeling software (ATHOS) also would not have identified an inadequate IP FEI margin.

These errors, which resulted in a gross mistake in underestimating the IP FEI active in the tube bundle can only be at the sole failing of MHI itself.

It is also known that the *Certified Design Specification* required [p10, ¶3(1)] that the RSGs remained primary-to-secondary leak tight for the duration of the *Warranty Period* (although the actual period is not cited in the RCA Report).

In-Bundle Flow Prediction: RCA concedes that the FIT-III flow analysis modeling software provided misleading results underestimating the crucial two-phase fluid velocity and voidage particularly in the U-bend region. As well as inappropriate values for pressure loss coefficients and two-phase mixture density applied for the tube bundle flows, RCA admit [p21, ¶2-3] that the geometric definition of the tube array gaps was incorrect and insufficiently trialed by bench testing (only one of five possible verification tests was undertaken), an error that may have arisen when the FIT-III program was adapted from treating square to triangular pitched tube arrays.

Interestingly, there occurs a flurry of revisions in the thermal and hydraulic parametric calculations (FIT-III etc) from August 2007 through to November 2009, even though all other revisions had ceased and the RSG design had been effectively closed-out by October 2006 (other than some later for changes to the RB in March 2008– see [p57-58, Attachment 5]). This activity might coincide with MHI first becoming aware of the possible errors in the FIT-III predictions of IP and OOP velocity and voidage distributions in the tube bundle.

MHI claims to have built in two conservatisms to provide a margin against FEI induced tube vibration, one of which was a weighting factor or multiplier of x1.5 applied to the averaged FIT-III tube gap velocities although it transpired that this was not an added conservatism but a requirement |p50, ¶2|[19]– indeed, a gauge of the FIT-III prediction underestimate for the peak gap velocity of a comparative tube can be drawn from the following |p51, ¶1|:[19]

TABLE 3 COMPARISON OF EFFECTIVE PEAK GAP VELOCITIES (TUBE R142C88)

THERMAL-HYDRAULIC MODELING CODE	EFFECTIVE PEAK VELOCITY m/sec
NRC ATHOS	5.2
MHI ATHOS	5.6
MHI FIT-III	2.5 (NOT WEIGHTED)

However, so far as safeguarding against IP FEI, RCA concludes that the use of the alternative ATHOS software would not have initiated a change in design [p22, ¶1]:

r) "*. . . If the steam quality (void fraction) predicted by FIT-III had been the same as the ATHOS calculated value, and if the appropriate tube to tube gap had been utilized to compute the flow velocity, MHI would have identified a decreased margin against out-of-plane FEI. In that case, MHI might have incorporated an additional AVB to increase the design margin against out-of-plane FEI, but would not have taken measures to protect against in-plane FEI . . .*"

In an astonishing turn-around of logic, RCA claim [p22, ¶2] that in:

s) "*. . . not using ATHOS, which predicts higher void fractions than FIT-III at the time of design represented, at most, a missed opportunity to take further design steps, not directed at in-plane FEI, that might have resulted in a different design that might have avoided in-plane FEI.*"

Interestingly, the RCA team sum up the section of the report dealing with *Thermo-Hydraulic Conditions* when considering what it refers to as the *'organizational and programmatic cause'* for the *in-plane* FEI to be [p22, ¶4]

t) "... *The underlying reason for this insufficiency is that the MHI SONGS RSG design did not consider the phenomenon of in-plane FEI because contemporary knowledge and industry U-tube SG operation experience did not indicate a need to consider in-plane FEI.*"

This somewhat misleading statement might be construed to suggest that in-plane FEI is a novel and unresearched phenomenon - it is not there being a wealth of research papers, guidance notes and standards many of which predate the design of the SONGS RSGs. Moreover, there is little distinction to be had between *in-plane* (IP) and *out-of-plane* (OOP) FEI since it simply refers to the two principal axes of the tube bundle – typically the design focuses on OOP because the tube is less stiff in this direction, the assumption being that if the FEI inhibition in the OOP direction is effective then the FEI induced vibration in the stiffer IP direction will also be inhibited.

COMPARISON OF DESIGN AND HINDSIGHT ANALYSES OF THE SONGS RSGS

Table A3-1 of the STE report {p67, APPENDIX 3}, although redacted in part, highlights the differences between the analyses undertaken at the design stage (ie that overseen by the AVB Design Team) in about mid-2005 to late-2006 with the post-shutdown evaluation of the tube wear of both Units 2 and 3 in January 2011. The salient differences being:

AVB Support Condition: At the time of the design the maximum number of consecutive AVB supports that were considered to be inactive was 2 (out of 12) per tube, whereas the post-shutdown evaluation found that the dispersion and extent of tube wear required 8 consecutive AVBs to be inactive. Moreover, the STE analysts gave some thought to the differing requirements for IP and OOP support at the AVB-to-tube interface, this being {p64,¶4}:

> "... *Active condition against out-of-plane FEI: Narrow gap that is small enough to produce tube-to-AVB contact and mechanical damping (contact force is not necessary)*
>
> *Active condition against in-plane FEI: Tube-to-AVB contact force sufficient to produce friction that inhibits in-plane tube displacement is required...*"

and in doing so concluded the very opposite to the September 2004 SCE CDS specification Clause 3.10.3.5 requirement that 'preloading' at the AVB-to-tube should be avoided with the *'zero tube-to-AVB gap-zero force'* design functionality.

The outcome of this was that when Unit 3 was first installed all or most of the AVBs were OOP active but none were IP active. Similarly, for Unit 2 where there was some AVB-to-tube preload (as a result of the unplanned for AV bar distortion) all or most of the AVBs were active in both OOP and IP directions. By comparison of the post January shutdowns, it is clear that the wear dispersion and extent at the AVB supports for Unit 2 has followed the same pattern as Unit 3 where the progression went on to TTW. However, because of the unintentional preload, the Unit 2 AVBs started off active but have, because of the tube-to-tube bumping phenomenon, many have degraded to inactive, so much so, that the Unit 2 tube bundle is at some way on the path towards or is immediately vulnerable to the onset of TTW.

From its observations the STE report concludes:

> "... *In U-bend SGs, if the number of active supports against out-of-plane FEI is identical to the number of active supports against in-plane FEI, the critical velocity for out-of-plane FEI is always lower than what is required to produce in-plane FEI because the natural frequency of out-of-plane FEI is lower than that of in-plane FEI. Therefore out-of-plane FEI will occur before in-plane FEI...*"

In other words, as the ABV effectiveness in the IP direction decreases, the IP FEI begins to dominate triggering at a <u>lower</u> critical velocity than that in the OOP direction. The correlation between the FEI respective IP and OOP critical velocities is referred in the STE report {p65, Figure A2-1} that, although completely redacted is summarized {p64, ¶9}:

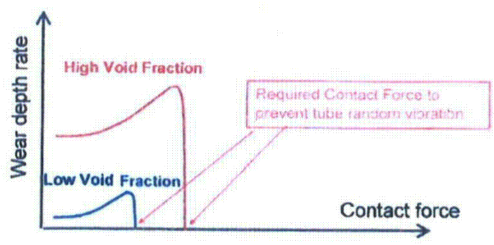
> ". . . *If the number of active supports that prevent in-plane FEI becomes sufficiently less than the number of supports that prevent out-of-plane FEI, the critical velocity of in-plane FEI becomes lower than that of out-of-plane FEI. . .*"

The implication of this is twofold: first, in specifying the *'zero tube-to-AVB gap-zero force'* design functionality of the AVBs, SCE may not have understood or misjudged the importance of the IP *vs* OOP juxtapositioning being determined by the AVB support effectiveness and, second, it is not clear that SCE, via its latest submission[36] in response to the CAL for the 100% rated thermal power (RTP) case (and also for the previously submitted 70% RTP case), has taken this into account.

Random Excitation Force Basis: Another significant difference between the AVB Design Team approach and the post January 2012 shut down evaluation relates to the availability of random fluid mechanisms (turbulence, etc) to induce tube motion. The AVB Design Team relied upon a data from a single phase (water) flow tests, whereas the post shutdown evaluation involved data from a series of two-phase tests.

Justification for relying solely upon single phase test data is on the premise that (random) flow turbulence by itself is insufficient to produce displacements for the tubes to engage in tube-to-tube wear (TTW - in both OOP and IP directions). However, this ignores to potential loss of effectiveness of the AVB-to-tube support by random turbulent flow mechanisms.

The incidence of the MHI Type 2 AVB-to-tube wear, where tube motion at the AV bar contact interface is not impeded by a contact or preload force (ie as specified by SCE's *zero tube-to-flat bar gap'* CDS requirement) is probably represented by a single-phase fluid test – ie random flow induced AVB-to-tube motion is minimal. However, where the operational RSG flow regime includes a significant amount of voidage then a contact or preload force is necessary to prevent the tube lifting off the AV bar and to commence sliding (and wear) – as the void fraction increases then a greater degree of contact force in required to prevent tube lift-off – see {p40, Figure 3.3-2} right.



So, it follows the single phase test data relied upon by the AVB Design Team during 2005-2006 would have provided misrepresentative results: First, the AVBs modeled for the tests would most likely have been of near perfect geometry and unlike the AVBs of the in-service RSGs, particularly Unit 2, these would not have include a scatter of then unintentional preload forces. Second and importantly, the single phase liquid in the test would not have included any vapor/gas phase (ie the void fraction) so the relationship between increasing void fraction and contact force necessary to prevent tube lift-off and sliding (above right) would not have been calibrated.

As reported in both RCA and STE reports SCE, via the May 2005 Design Review meeting and subsequent meetings of the AVB Design Team, knew of the FIT-III and ATHOS projections of high void fraction flow regimes in the U-bend region but, seemingly, both MHI and SCE failed to act on this by not reviewing the reliance on the single phase test data. A result of this inaction was, it might be reasoned {p38, ¶5 . . . p41, ¶3}

> ". . . *the turbulence induced (random) tube vibration associated with the small gaps and small contact forces combined with the lower tube damping in the high void fraction regions is sufficient to produce the observed wear {at the AVB-to-tube interfaces} . . . When the contact force is sufficiently high to prevent random tube vibration, the tube-to-AVB wear becomes negligible. The magnitude of the contact force that prevents random tube vibration is a function of the void fraction, with a higher contact force being needed in the regions of higher void fraction (steam quality).*"
>
> my *{clarification}* and truncation . . .

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36 SCE, Enclosure 1, Amendment I *Operational Assessment for SONGS Unit 2 Steam Generators for Tube-to-Tube Wear Degradation 100% Power Operation Case*, Intertek AES 13018304-2Q-1 March 2013, March 14 2013

Sliding vs Impact Wear Rates: Table A3-1 of the STE report {p67, APPENDIX 3} reveals the types of wear coefficient applied at the AVB Design Team involvement and subsequently in the post shutdown evaluations. Although the values of the individual wear coefficients have been redacted, it is quite clear that the AVB Design Team relied solely upon a *fretting* wear coefficient whereas the post shut down evaluations included *impact* wear coefficients.

To demonstrate the AVB-to-tube wear and TTW that occurred in the operational Units 2 and 3, it is necessary that the analysis:

i) assumes that the degraded tube frets and impacts rather than solely rubs surface-to-surface:

- for the ABV-to-tube interface this requires consecutive AVBs with small clearances and/or small contact forces that permit ABV-to-tube impacting to occur within the gap – in the AVB case the magnitude of the impacting force may increase as the AVB-to-tube gap wears and increases
- for TTW the adjacent tubes have to be free space separated and periodically clash together to produce an impact force;

ii) with impacting conditions established, the wear coefficient has to include components in account of fretting and impacting, with this combination coefficient being significantly larger;[37] and

iii) to model the excitation function:

- for the AVB-to-tube interface, the random excitation forcing function derived from two-phase fluid testing (see above) has to be incorporated; and
- for TTW the FEI mode (IP and/or OOP) with, for IP, particular regard given to the in-plane distortion or *'flowering'* of the tube bundle.

In other words, the analytical approach and test data acquisition adopted at the design stage (2005-2006) would seem to have fallen far short of the level of technical and intellectual resources subsequently found necessary to understand the tube wear once that it had occurred. Even without the benefit of hindsight, this does call into question the thoroughness of SCE's design specification and its direct involvement in the detailed design and development of the San Onofre replacement steam generators.

IN CONCLUSION

The RCA and STE reports claim, that during the period 2005 to 2006, as members of the *AVB Design Team*, both SCE and MHI were aware of the high void fraction and the potential for vigorous FEI activity and that, moreover, they both explored means (various design changes) to lower the void activity and suppress FEI. It is surprising therefore, that no practicable design changes were implemented during this period to eliminate the uncertainties about the thermal-hydraulic conditions within the San Onofre replacement steam generators.

LARGE & ASSOCIATES
CONSULTING ENGINEERS, LONDON

37 The wear coefficient data in the STE report is redacted but, not untypically, the difference between fretting and impact coefficients would be of the order x20 to 30.

February 12, 2013

Mr. Sher Bahadur
Chairman, Petition Review Board
U.S. Nuclear Regulatory Commission
Sixteenth Floor
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Re: Request for Disclosure of MHI Report in the §2.206 Petition Review Process Regarding the 10 CFR § 50.59 Review for the Replacement Steam Generators at San Onofre Units 2 and 3.

Dear Chairman Bahadur:

On February 6, 2013, Senator Barbara Boxer and Congressman Edward Markey sent a letter to NRC Chairwoman Allison Macfarlane referring to a 2012 Mitsubishi Heavy Industries (MHI) document entitled, "Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators at San Onofre Generating Station" (Report). The Report's contents as described by their letter have direct bearing on the matters raised by the June 18, 2012 petition filed by Friends of the Earth (FoE), which the NRC is considering under 10 C.F.R. § 2.206. In the interest of a complete and accurate record in NRC's review of FoE's petition, FoE requests that the 2012 MHI Report and all other documents in the possession of the staff or Commission regarding the void fraction and potential for fluid elastic instability (FEI) in San Onofre Units 2 or 3 be included in the record of this proceeding and publicly released.

According to the Boxer/Markey letter, the MHI Report shows that (1) not only were Southern California Edison (SCE) and MHI aware that there were design problems with the replacement steam generators (RSGs) at San Onofre, but also that (2) SCE and MHI rejected modifications that could have mitigated these safety risks because the changes would have triggered a license amendment process under 10 C.F.R. § 50.59. According to the Members of Congress, the Report states, "Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG design" without triggering the need for a license amendment, a process which, to SCE, was as an "unacceptable consequence."

1707 L St, N.W. • Suite 850 • Washington, DC 20036 • Phone: (202) 452-9200
Web: www.ayreslawgroup.com

In its response to the SCE submission filed February 6, 2013, FoE stated that SCE was aware of the prospect that the design of the RSGs would result in FEI that could damage the tubes in the RSG. As the basis for this contention, FoE quoted a colloquy between the NRC's inspection lead at San Onofre and SCE's Vice President for Engineering during the November 30, 2012, NRC Meeting in Laguna Hills, California. In that conversation, the SCE official admitted that SCE understood that in their design the void fraction would be too high, leaving the tubes open to the effects of FEI.

According to the Boxer/Markey letter, the information in the MHI Report confirms FoE's statement that SCE had knowledge of the safety risks presented by the design of the RSGs, notwithstanding SCE's claims to the contrary in its January 9, 2013 "Response to Friends of the Earth 10 CFR 2.206 Petition." In the context of an argument that § 50.59 did not require a license amendment for the RSGs, SCE conceded that the conditions resulting in the tube leak in Unit 2 were "adverse" to a design function but then stated categorically that no license amendment was needed under § 50.59 because SCE did not know about those conditions.[1] It would appear that the MHI document referred to by Senator Boxer and Representative Markey provides proof that the SCE statement is untrue.

The Boxer/Markey letter raises serious questions about SCE's representations to the NRC about the state of its knowledge at the time it performed the 50.59 analysis – questions which could confirm Mr. Gundersen's contention that SCE was more intent on avoiding a regulatory program than in assuring the health and safety of the millions of people who live in proximity to the San Onofre plant.

In the interest of assuring that the 2.206 panel has a complete and accurate record before it, FoE calls upon the Chairman to direct that the MHI Report and other documents regarding the void fraction and potential for FEI in San Onofre Units 2 or 3 be placed in the record of this proceeding and made available to the members of this panel and all the parties. Given the Commission's stated commitment to an open and transparent process in this important matter of public safety, these documents should also be publicly disclosed in full.

Sincerely,

/s/ Richard Ayres

Richard Ayres
Counsel for Friends of the Earth
(202) 452-9300
ayresr@ayreslawgroup.com

[1] Southern California Edison Company, Response to Friends of the Earth 10 CFR 2.206 Petition (Jan. 9, 2013) at 9. *See also* page 11 ("That concern [FEI], however, was not known during the design and manufacturing of the RSGs.") and page 12 ("If the RSGs had been designed and manufactured in accordance with the procurement specification, the leak and tube wear would not have occurred.").

Cc: Brian Benney, Petition Manager
Lee Banic, Petition Coordinator
Molly Barkman March, Office of General Counsel
David Beaulieu, Office of Nuclear Reactor Regulation
Art Howell, Region IV
Greg Werner, Region IV

Enclosure: Letter to Chairwoman Macfarlane from Senator Boxer and Representative Markey

Congress of the United States
Washington, DC 20515

February 6, 2013

The Honorable Allison M. Macfarlane
Chairman
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Dear Chairman Macfarlane:

We have become aware of new information contained in a 2012 Mitsubishi Heavy Industries (MHI) document entitled "Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators of San Onofre Generating Station" (Report).

We strongly urge the Nuclear Regulatory Commission (NRC) to promptly initiate an investigation concerning the troubling information contained in this Report.

The Report indicates that Southern California Edison (SCE) and MHI were aware of serious problems with the design of San Onofre nuclear power plant's replacement steam generators before they were installed. Further, the Report asserts that SCE and MHI rejected enhanced safety modifications and avoided triggering a more rigorous license amendment and safety review process.

For example, the Report states that although SCE and MHI accepted some adjustments to the replacement steam generators, further safety modifications were found to have "unacceptable consequences" and were rejected: "Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG [replacement steam generator] design" without the requirement for a license amendment. The Report also indicates that SCE's and MHI's decision to reject additional safety modifications contributed to the faulty steam generators and the shutdown of reactor Units 2 and 3.

This newly-obtained information concerns us greatly, and we urge the NRC to immediately conduct a thorough investigation into whether SCE and MHI did in fact fail to make needed safety enhancements to avoid the license amendment process.

All people in our nation, including the 8.7 million people who live within 50 miles of the San Onofre plant, must have confidence in the NRC's commitment to put safety before any other concern.

We believe this alarming Report raises serious concerns about SCE's and MHI's past actions. Safety, not regulatory short cuts, must be the driving factor in the design of nuclear facilities, as well as NRC's determination on whether Units 2 and 3 can be restarted.

We look forward to your prompt response detailing how public safety will be assured in light of this information. If you have any questions, please have your staff contact Dr. Michal Freedhoff of Rep. Markey's staff at 202-225-2836 or Grant Cope of Chairman Boxer's staff at 202-224-8832.

Sincerely,

Barbara Boxer
Chairman
Senate Committee on Environment and Public Works

Edward J. Markey
Member of Congress