UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

SOUTHERN CALIFORNIA EDISON COMPANY

(San Onofre Nuclear Generating Station, Units 2 and 3)

Docket Nos. 50-361-CAL & 50-362-CAL

ASLBP No. 13-924-01-CAL-BD01

22 January 2013

RESPONSE TO ATOMIC SAFETY AND LICENSING BOARD'S FACTUAL ISSUES

1ST AFFIDAVIT OF JOHN H LARGE

ADDED NOTE: In this version of the Affidavit a few sections of text taken from SCE's proprietary documentation has been redacted thus < ***redacted proprietary information*** **. . .**>.

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SUMMARY OF THE 1ST AFFIDAVIT OF JOHN LARGE

S0 **SUMMARY:** Here I provide a summary of my 1st Affidavit that follows.

S1-4 **PREAMBLES**

I am John Large, a UK citizen and a Chartered, Consulting Engineer with considerable experience in nuclear engineering matters.

On December 8, 2102 I was instructed by Friends of the Earth to prepare a response to certain of the Factual Issues raised by the ASLB in its order of December 7, 2012. For this affidavit I have referred to the proprietary versions of the documents submitted by Southern California Edison (SCE) in its response to the Nuclear Regulatory Commission's *Confirmatory Action Letter* (CAL) of March 27 2012 and, more generally, to other relevant documents and data available in the public domain.

S5 **REVIEW OF THE FLUID ELASTIC INSTABILITY AND TUBE WEAR**

Section 5 of my affidavit assesses the results of the tube inspections, the analyses undertaken by SCE and its Operational Assessment (OA) consultants, and how their findings might be practicably implemented. I have expressed my findings in the following sections:

S5.4 **Tube Inspections:** I present a summary of the tube wear as related to the anti-vibration bars (AVB), the tube sheet plates (TSP) and tube-to-tube (TTW) in TABLE 1.

My principal finding is that tube wear occurs not just at

i) the tube-to-tube or TTW free-span locations; but also at

ii) various tube restraint components, such as the retainer bars (RBs), AVBs and TSPs, all of which serve to restrain and/or provide points of fixity to the individual tubes.

So, it follows, to comprehend and predict the risk of TTW, a thorough understanding of performance and wear of the replacement steam generator (RSG) restraint components is required.

S5.5 **Causes of Tube and Restraint Component Motion and Wear:** My study of the various OAs leads me to the following findings and opinion that

i) degradation of the tube restraint localities (RBs, AVBs and TSPs) occurs in the absence of *fluid elastic instability* (FEI) activity;

ii) TTW, acknowledged to arise from *in-plane* FEI activity, generally occurs where the AVB restraint has deteriorated at one or more localities along the length of individual tubes; and

iii) the number of tube wear sites or incidences for AVB/TSP locations outstrips the TTW wear site incidences in the tube free-span locations.

I find that the *'zero-gap'* AVB assembly, which features strongly in the onset of TTW, is clearly designed to cope only with *out-of-plane* tube motion since there is little *designed-in* resistance to movement in the *in-plane* direction - because of this, it is just chance (a combination of manufacturing variations, expansion and pressurization, etc) that determines the *in-plane* effectiveness of the AVB.

iv) Uniquely, the SONGS RSG fluid regimes are characterized by *in-plane* activity, which is quite contrary to experience of other SGs used in similar nuclear power plants in which *out-of-plane* fluid phenomena dominate.

Moreover, from the remote probe inspections when the replacement steam generator (RSG) is cold and unpressurized, I consider it impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state.

v) The combination of the omission of the *in-plane* AVB restraints, the unique *in-plane* activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed.

Realistically, because of these difficulties and uncertainties I consider the outcome of the OA assessments not to have the reliability and confidence that I would expect for the safe operation of a nuclear power plant.

S5.6 **Retainer Bar Vibration and Tube Wear:** I find that the wear at the tube row immediately adjacent to the RB, although serious in itself, plays no significant part in the potential for TTW.

S5.7 **Phasing of AVB-TSP Wear -*v*- TTW:** I reason that, overall, the tube wear process comprises two distinct phases: First, the AVB (and TSP) -to-tube contact points wear with the result that whatever level of effectiveness is in play declines. Then, with the U-bend free-span sections increased by loss of intermediate AVB restraint(s), the individual tubes in the U-bend region are rendered very susceptible to FEI induced motion and TTW.

Whereas the OAs commissioned by SCE broadly agree that the wear mechanics comprises two phases, there are strong differences over the cause of the first phase comprising *in-plane* AVB wear: AREVA claim this is caused by *in-plane* FEI whereas, the contrary, Mitsubishi (and Westinghouse) favor random perturbations in the fluid flow regime to be the tube motion excitation cause.

Put simply:

i) if AREVA is correct then reducing the reactor power to 70% will eliminate FEI, AVB effectiveness will cease to decline further and TTW will be arrested; however, to the contrary

ii) if Mitsubishi is right then, even at the 70% power level, the AVB restraint effectiveness will continue to decline thereby freeing up longer free-span tube sections that are more susceptible to TTW; or that

iii) the assertion of neither party is wholly or partly correct.

As I have previously stated (S5.5), I consider that AVB-to-tube wear is not wholly dependent upon FEI activity.

S5.8 **Tube Wear Rates – Predicting the In-Service Period:** SCE presents the findings of its commissioned OAs in a positive light, claiming that at 70% power the restarted Unit 2 plant will maintain RSG tube integrity for 16 to 18 months of continuous running, that is considerably longer than the proposed 150 day inspection interval.

However, closer study of the OAs reveals that the reasoning behind important aspects of the deterioration period for the AVB effectiveness in Unit 2 is flawed, being overly dependent upon a number of uncertainties that I identify and expand upon in my affidavit. Some account of these uncertainties has been taken by AREVA in revising the TTW *time-to-burst* period down to 2.5 months which is well below the 150 days inspection interval but, without much justification, it determines and front-ends the *time-to burst* with a further 3.5 month AVB wear-in period, thereby delaying the onset of TTW and the unacceptable level of risk of tube burst to about 1 month longer than the proposed inspection period.

I have little confidence in the outcome of the AREVA and other OAs projection of the time period through which the Unit 2 nuclear plant could be reliably expected to operate without a) incurring a tube failure or b) running at a greater risk of a tube failure occurring. This is primarily because

i) it is generally accepted that Unit 2 is following along the same path of deterioration as Unit 3 (AVB wear and loss of effectiveness preceding TTW), although the reasons why it lags so much behind are not at all understood by SCE and, indeed, subject to disagreement between the OA consultants;

ii) moreover, the pattern of AVB breakdown is not clear from the more advanced TTW degradation of Unit 3, thus the extrapolation to Unit 2 is not robust – again, there is disagreement between the OAs on this; so, it follows,

iii) there is very little justification in adding to the *time-to-burst* for Unit 2 tubes a 3.5 month AVB wear-in period, this is particularly so because so there is no certainty of just where Unit 2 is presently at along the path towards TTW wear.

In account of these uncertainties, together with the uniqueness of the *in-plane* FEI in the SONGS RSGs that I will touch upon later, I consider that restarting Unit 2 to continuous running, even at 70%, will incur a great deal of *change*, *test* and *experiment*.

S5.9 **TTW Tube Performance during Design Basis Accidents (DBA):** I have also considered tube structural integrity performance when subject to additional forces during and following certain design basis accidents.

i) From the OAs it is not clear to me that this important nuclear safety prerequisite has been adequately reviewed and included in the SCE response to the CAL.

S6 RESPONSE TO ASLB'S ISSUES

I have been instructed to respond the Issues iv), v), vi), vii) and viii) raised by the Board in its Order of December 7, 2012.

S7 **Factual Issue iv) – Final Safety Analysis Report:** I give my interpretation of the SONGS *Technical Specification* that for normal operation the DBA event is limited to the burst of a **single** tube and that for all other design basis incidents (SSE, LOCA, etc) all tubes are required to maintain structural integrity throughout and following the incident.

That said, I conclude that the conditions, uncertainties and risks that will accompany the proposed restart and continuous running of Unit 2 significantly depart from those incorporated in the unrevised FSAR, particularly

i) since there are no means of monitoring tube wall thinning whilst the plant is in service, the risk of tube burst is wholly dependent upon the accuracy and reliability of SCE's

   a) Outage 16 inspection results obtained indirectly, using remote inference techniques, to predict the extant tube wear and, importantly, the condition and contact forces of many thousands of AVB-to-tube locations when in the cold and unpressurized state, and projecting these to the hot, pressurized service operational state; and

   b) using the data predictions of i)a) that are, in my opinion, drawn from uncertain and empirically unsound bases, to seed models of AVB-to-tube contact characteristics and tube motion, in order to determine the tube wall wear rate, tube wall thinning and, hence, the risk of tube rupture;

ii) certain of the wear patterns and tube thinning seem to be unique and have not, to my knowledge, been experienced in operational SGs elsewhere, so the rate(s) of tube wall thinning adopted by AREVA and the other OAs are largely hypothetical; and

ii) prediction of FEI activity, the placement and effectiveness of the preventatively plugged tube buffer zones in delaying the advancement of TTW are, to my mind, similarly founded on a great deal of uncertainty; and as I have previously noted

iii) there is disagreement over the extent of other (non-FEI) fluid excitation sources, particularly at the TSP and AVB contact points.

In other words, with such uncertainties prevalent, RSG tube integrity cannot be assured throughout the inspection interval proposed by SCE, thus previous studies and analyses contained within the present version of the FSAR would be invalid for the restart and continuous running of Unit 2.

S8/9 **Factual Issues v) & vi) – SONGS SG Comparison to Other Operating SGs:** I identify a number of issues with the representation of Figures 4-3 and 5-1 of the AREVA *Tube-to-Tube Report*, including:

i) it is not exactly clear which properties are being represented on the spider diagram for comparison with the other operational SGs; even so

ii) since it is most unlikely that AREVA has undertaken a comprehensive (ATHOS) simulation of each of the five nominated SGs, the comparisons drawn are likely to be between aggregate or bulk flows within the entire tube bundle of each SG;

iii) as acknowledged by AREVA, the SONGS RSGs are dominated by *in-plane* flow regimes whereas all other SGs are characterized by *out-of-plane* flow regimes; and

iv) none of the comparative SGs has been identified.

In other words, unless the spider diagrams of Figure 4-3 and 5-1 somehow, and I cannot reason how, are making a direct comparison of the complex two-phase fluid cross-flow situation in the SONGS and other five comparative plant steam generators, then these figures only provide the bases of a somewhat meaningless comparisons.

S10 **Factual Issue vii) – FEI SR = 0.75 Probability & 50% Confidence at 70% Power:** For the general and specific reasons that I expounded upon throughout my Affidavit, I do not agree that the claimed confidence level of 50% will satisfy the regulatory requirement.

A difficulty that I have with the AREVA and, generally, with the other OAs is that whereas the results of analyses, particularly relating probability and confidence, are often resolutely stated, very little of the analytical procedures arriving at the results are open to inspection. For example, I refer to my previous comments (S7) where, because of the uncertainties I very much doubt that in the present circumstances tube structural integrity could be guaranteed to satisfy the 95% probability at 50% confidence criterion but, that said, AREVA presents no substantial data that enables me to explore and possibly resolve these doubts.

S11-12 **Factual Issue viii) – Operational Assessment *-vs-* Test and Experiment:** For this I deploy the NRC guidelines on how these elements should be identified and evaluated in the context of eight NRC assessment criteria.

I generally find that

i) the requirement that an *Operational Assessment* must ensure that the RSG tubing will meet the criteria for structural and leakage integrity over the inspection period has not been satisfied – this is because there is too much uncertainty over a number of important respects that I have referred to earlier; and

ii) referring to the short section of the FSAR provided to me by SCE, which I understand is not to be amended for the Unit 2 restart

   a) there is no account of the *changes* that have been made in the evaluation of the tube structural and leakage integrity, that is from the stage of predicting those tubes at risk of TTW and other forms of wear, the tube thinning wear rates, through to the nature of the tube failure being unique to the type and extent of the wear pattern and tube thinning; and

   b) the methods of deducing, mainly by unproven inference, from the probe inspection results particularly to determine the *in-plane* AVB effectiveness, includes unacceptably large elements of *test* and *experimentation* that are inconsistent with the analyses and descriptions of the FSAR.

I provide a number of explicit examples where I consider that the circumstances and risks accompanying the proposed restart of Unit 2 will result in unacceptable levels of *test* and *experiment*.

S13 **IN CONCLUSION**

SCE's assertion that reducing power to 70% will at the best alleviate, but not eliminate, the TTW and other modes of tube and component wear is little more than hypothesis - the supporting Operational Assessments and analyses have not proven it to be otherwise. I am of the opinion that trialling this hypothesis by putting the SONGS Unit 2 back into service will, because of the uncertainties and unresolved issues involved, embrace a great deal of *change*, *test* and *experiment*.

The terms of the *Confirmatory Action Letter* of March 11 2012, are versed such that to meet compliance the response of SCE via its *Return to Service Report*,[11] must include considerable changes of conditions and procedures that are outside the reference bounds of the present FSAR – this is because the physical condition of the RSGs, and the means by which this is evaluated and projected into future in-service operation, have substantially and irrevocably changed since the current FSAR was approved.

The fact that SCE fails to satisfy the requirements of the CAL is neither here nor there, although it illustrates the scope and complexity of the response required. At the time of preparing the CAL, the NRC being well-versed in the failures at the San Onofre nuclear plant, surely must have known that the only satisfactory response to the CAL would indeed require considerable *changes*, *test*s and *experiments* to be implemented.

Put another way, the extensive and rapid rates of tube wear experience at the SONGS Unit 2 and Unit 3 RSGs, have necessitated an extensive raft of analysis, assessments and projections to qualify, or otherwise, that Unit 2 is fit for purpose. Not only is this prequalifying work unique to the San Onofre nuclear plant, much of it has never been undertaken before so, it follows, its inclusion in safety considerations must be a new and hitherto unconsidered component now required to be incorporated into an updated version of the FSAR.

Hence, I am of the opinion that, on a technical basis alone, the CAL must be considered to have been at the time of its preparation, a de facto license amendment.

JOHN H LARGE
CONSULTING ENGINEER
LARGE & ASSOCIATES, LONDON

1ST AFFIDAVIT OF JOHN LARGE

1 QUALIFICATIONS AND EXPERIENCE

1.1 I am **John H Large** of the Gatehouse, 1 & 2 Repository Road, Ha Ha Road, Woolwich, London, United Kingdom, SEI8 4BQ.

1.2 I am a citizen of the United Kingdom.

1.3 I am a Consulting Engineer, Chartered Engineer, Fellow of the Institution of Mechanical Engineers,[1] Graduate Member of the Institution Civil Engineers, Learned Member of the Nuclear Institute and a Fellow of the Royal Society of Arts.

1.4 I head the firm of Consulting Engineers, Large & Associates.

1.5 Based in London UK, Large & Associates provides engineering and analytical services relating to nuclear activities, systems failure and engineering defects.

1.6 Prior to founding Large & Associates, from the 1960s through to the early 1990s I was a full time, tenured academic in the School of Engineering of Brunel University (London) where, as a Senior Research Fellow, I undertook applications research on behalf of the United Kingdom Atomic Energy Authority (UKAEA) and other UK government agencies.

1.7 A résumé of my academic and professional consulting careers is available at the Large & Associates website.

1.8 I present myself as a Consulting Engineer with considerable experience of the nuclear industry worldwide, being qualified by education, experience and professional standing to provide expert opinion on this matter.

2 INSTRUCTIONS

2.1 On December 8 2012, I received instruction from Mr Shaun Burnie of Friends of the Earth (FoE) to prepare a response to the certain of Factual Issues that the Atomic Safety and

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1 The Institution of Mechanical Engineers (IMechE) is the UK equivalent to the American Society of Mechanical Engineers (ASME). A Fellow is the highest grade of IMechE membership and at the time of advancement a Fellow must be a corporate member of the Institution and have been responsible for significant engineering achievements and, typically, to have practised as a corporate member for at least 10 years, elevation to corporate membership usually takes about 5 to 10 or so years from graduation, depending on the workplace experience. In the UK professional engineers are separately Chartered via the CEng registration and have to have demonstrated the required level professional competence. The Institution of Civil Engineers and Nuclear Institute are also corporate, chartering bodies in their respective field of interests. The Royal Society of Arts is a learned society and election to fellowship is via recommendation of other Fellows.

Licensing Board (ASLB) had directed the parties involved in the matter of the San Onofre nuclear generating station.

3 LINKS AND REFERENCES

3.1 For ease of reference this evidence includes hyperlinks, thus TABLE A, which will relocate to a specific bookmark in this document.

3.2 Other hyperlinks, such as *Root Cause Evaluation*, will link directly to a reference document and, similarly, proprietary documents referred to thus *Attachment 4: MHI Document L5-04GA564* for reasons of the non-disclosure agreement, will link to the non-proprietary versions – for these links to function the host computer has to be internet connected.

3.3 Links to text locations in this document are shown as paragraph locations {¶2.1} and, similarly, specific text locations, but not links, in referenced documents are show thus [p11, ¶5].[7]

4 ATOMIC SAFETY AND LICENSING BOARD'S FACTUAL ISSUES

4.1 I received a copy of the Atomic Safety Licensing Board's (the Board – ASLB) December 7 Order[2] on December 8, thereafter on December 9 and 10, 2012 I compiled two lists of hitherto redacted or publicly unavailable documents that I believed necessary for me to consider and respond to the issues specified in the ASLB Order.

4.2 The first of these lists of documents for disclosure related to documents held by the Nuclear Regulatory Commission (NRC), mainly drawn from the reporting[3,4] of its Augmented Inspection Team (AIT) at the San Onofre nuclear generating station (SONGS). The second list related to the documents that Southern California Edison (SCE) had submitted to the NRC in response to the Confirmatory Action Letter (CAL),[5] including requests for undredacted copies of the main documents that had been previously publicly available.

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2 United States of America Nuclear Regulatory Commission, Atomic Safety and Licensing Board, Docket N° 50-361-CAL, 50-362-CAL, ASLBP No 3-924-01-CAL-BD01, December 7 2012.

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

4 *NRC Augmented Inspection Team Follow Up Report 05000361/2012010 & 05000362/2012010*, November 9 2012

5 Letter from Elmo E Collins (USNRC) to Peter T Dietrich (SCE), *Confirmatory Action Letter 4-12-001*, San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation, March 27 2012.

4.3 I understand that these lists were passed to the appropriate recipients, NRC and SCE, on or about December 10 and 11 respectively.

4.4 I received copies of the proprietary versions of the SCE documents that had been previously heavily redacted on December 19. Other than a short text extract from the San Onofre Final Safety Assessment Report (FSAR), none of the other documents requested of SCE have been provided. Similarly, none of the documents requested of the NRC have been provided to me.

4.5 I should note here that although non-disclosure of the documents requested from SCE and NRC has been somewhat irksome, it has not impaired the strength of my opinion and conclusions in this matter.

5 REPLACEMENT STEAM GENERATOR PROBLEMS AT SONGS

5.1 Prior to Mr Burnie's instruction {¶2.1}, Large & Associates had been engaged by FoE on November 8 2012, to identify and report on any nuclear safety issues that could arise from SCE's proposal to restart SONGS Unit 2. The final draft of this report[6] was delivered to FoE on December 7 2012.

5.2 In responding to the issues identified by the Board, I shall rely upon a number of items that are best presented by me recounting certain of the findings of this previous report – these are as follows:

5.3 BACKGROUND – FORCED SHUTDOWN OF UNIT 3

5.3.1 On January 31 2012, while the SONGS Unit 2 (U2) fuelling outage was in progress, the virtually identical U3 was forcibly shut down when an alarm alerted SCE operators that a breach had occurred with reactor primary circuit water leaking across the replacement steam generator (RSG) tube interface to the secondary steam circuit.

5.4 RSG TUBE INSPECTIONS

5.4.1 Thereafter post-shutdown, non-destructive inspection of all of the tubes in both U3 RSGs revealed significant incidence of tube wear. This wear not only ranged across the

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6 Review, *Proposal to Restart Unit 2 San Onofre Nuclear Power Plant*, R3216-A1, Large & Associates, Friends of the Earth, December 7 2012

retainer bar (RB), anti-vibration bar (AVB) and tube support plate (TSP) fretting modes but, also, included a considerable number of tube-to-tube wear (TTW) extrados and intrados incidences.

5.4.2 For example, each U3 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 RSG.

5.4.3 A total of 193 and 188 tubes in the U3 088 and 089 RSGs respectively had exceeded the wall thinning threshold of 35% above which tube plugging was mandatory - this severe wear was at both TSP and tube free-span localities – the incidence of tube wear in the U3 RSGs is summarized in TABLE A.

5.4.4 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, this resulted in 8 individual tube failures, all located in one of the U3 RSGs.

5.4.5 There is some ambiguity about the rigor and sequencing of the SCE tube inspections of the U2 RSGs: SCE stated that the first inspection, completed before the January 31 forced outage of U3, found no incidence of TTW in the Unit 2 RSGs [p11, ¶5],[7] although it was silent about any other mode of tube wear being present. During March SCE carried out additional inspections of the U2 using a *'more sensitive'* method [p5, ¶2],[7] thereafter reporting that two adjacent tubes had sustained shallow TTW but, again, it did not refer to the existence of other modes of tube wear.

5.4.6 In fact, it was not until June 18 that SCE first publicly acknowledged that *'12% of the* [U2 RSG] *tubes showed wear greater than 10% through-wall indication'*.[8]

5.4.7 It is now established that the first and additional rounds of U2 tube inspections[9] revealed about 2,000 and 2,700 tube wear indications, dispersed over a total of 734 and 861

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

8 NRC Public Meeting 18 June 2012. In fact NRC revealed the extent of the U2 tube wear incidence much earlier that SCE on February 4 or 5 2012.

9 There are a number of chronological narratives of the events leading up to the withdrawal of all 4 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 and* SCE, Enclosure 2, *Songs Return to Service Report*, October 3 2012.

individual tubes affected in each U2 RSG respectively.[10] However, these additional inspections found only two adjacent tubes in one of the RSGs (2E-089) had moderately worn away (10 – 19% tube wall thickness) in the TTW mode. The incidence of tube wear in U2 is also summarized in TABLE A.

5.4.8 As a result of the additional inspection of the U2 RSGs, a total of six tubes required isolating (plugging) from the high pressure primary circuit because of excessive fretting wear of the thin-walled tube outer surfaces (in one instance, the tube wall thickness had reduced by 90%). Since this high incidence and dispersion tube wear was entirely unexpected, SCE plugged a further 192 tubes as a preventative strategy, including tubes that had exhibited wear from and/or were similarly at risk of fretting from contact with the RBs and AVBs, as well as at the TSPs, all of which serve to capture and restrain the individual tubes and tube bundles inside the U2 RSGs. Following further analysis of the U3 cause evaluation, a further 318 additional tubes were plugged in U2, bringing the total tube plugging to 205 and 305 tubes in 088 and 089 U2 RSGs respectively – these different modes of tube wear are listed in TABLE A and shown schematically by FIGURES 4A – TSP, 4B – AVB and TTW, 4C – RB.

5.4.9 The point to be stressed here is that the tube wear occurs not just at

5.4.9.1 i) the tube-to-tube or TTW free-span locations; but also at

5.4.9.2 ii) various tube restraint components, such as the RBs, AVBs and TSPs, all of which serve to restrain and/or provide points of fixity to the individual tubes.

5.5 CAUSES OF TUBE AND RESTRAINT COMPONENT MOTION AND WEAR

5.5.1 SCE and its consultants engaged to undertake the Operational Assessments (OAs), all generally agree that it was the presence of a thermal-hydraulic phenomenon called fluid elastic instability (FEI) activity in the higher regions of the RSGs that triggered and specifically resulted in tube motion and inter-tube fretting or TTW generated wear.

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10 Each RSG contains a single tube bundle of 9,727 individual tubes feeding up from the primary hot leg entry, traversing over a U-bend and down to the primary cold leg return – there is some variance on the reporting of how many tubes were plugged, for example Attachment 6 - Appendix A: SONGS U2C17 Outage – Steam Generator Operational Assessment reports that over 300 tubes were preventatively plugged in U2 RSGs - Attachment 6 – Appendix C: Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16 gives the U2 plugged tubes to be 2SG88 – 113 and 2SG89 – 214, that is a total of 327 plugged tubes - some of the plugged tubs have split wire-cable stabilizers installed.

5.5.2 From my study of the OAs, certain aspects of which I applaud for the attention to painstaking detail, I generally agree that FEI was the causation of the advanced TTW found in the U3 RSGs.

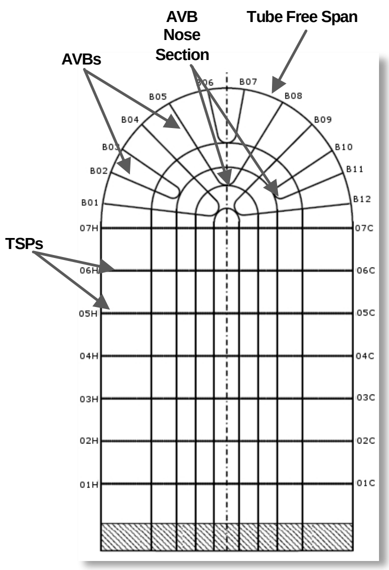
5.5.3 SCE summarizes its understanding of TTW in U3 to be [p11, ¶3]:[11]

5.5.4 *“. . . The mechanistic cause of the TTW in Unit 3 was identified as fluid elastic instability (FEI), caused by a combination of localized high steam velocity (tube vibration excitation forces), high steam void fraction (loss of ability to dampen vibration), and insufficient tube to anti-vibration bar (AVB) contact to overcome the excitation forces. The FEI resulted in a vibration mode of the SG tubes in which the tubes moved in the in-plane direction parallel to the AVBs in the U-bend region. This resulted in TTW in a localized region of the Unit 3 SGs. . .”*

5.5.5 That said, the root cause of how the RSG manufacturer Mitsubishi Heavy Industries’ (MHI) design permitted such vigorous levels of FEI activity has not been determined, even by MHI itself which continues to be at a loss to explain which feature(s) of its analytical and/or design processes was at fault.[12] Indeed, during the design stage, MHI went so far as to state [p21,¶4][7] that there was *“negligible possibility of fluid elastic vibration”*.

5.5.6 As I previously noted, at first SCE only reported on TTW in the U3 RSG, at that time making no reference to the other modes of tube and component wear found in the U2 and 3 RSGs.

5.5.7 MHI considered that FEI activity was suppressed in the region of the TSP localities (4A) and that this mode of tube wear arose not from FEI but via cross-flow induced random vibration of the tubes; the AVB-to-tube wear (4B) was also related to random vibration of the tubes which was exacerbated in some cases by the warped nose section AVB itself; and the AVB assembly retainer bar-to-tube wear (4C) arose because of the flow induced vibration of the retainer bar abrading directly against the outer row of tubes (ie no tube motion), perhaps with this contact



11 SCE, Enclosure 2, *Songs Return to Service Report*, October 3, 2012.

12 Conjecture is that the MHI flow distribution modelling software FIT-III was incorrectly adapted to model the triangular tube pitching from its previous square pitched tube array geometry.

being brought about by the thermally-related distortion (flowering) of the tubes in the U-bend region of the bundle.

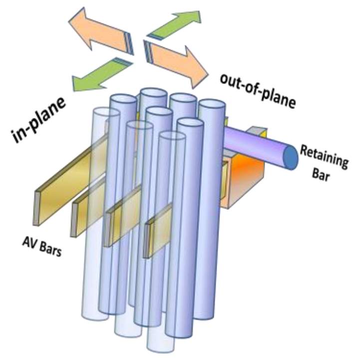
5.5.8 On its part for its *Confirmatory Action Letter* (CAL)[13] response, SCE's précis of MHI's analysis omits to echo the findings buried in MHI Appendix 10 concluding that AVB-to-tube wear (4B) in U2 arose in FEI inactive areas of the RSG tube bundle, it being excited by turbulent flow forces (vortex shedding, turbulent wake, etc) which may persist even when FEI is suppressed by the proposed reduction to 70% thermal power [p369, Appendix 10]:[14]

5.5.9 *". . . When **consecutive AVB support points are inactive** and in-plane **FEI occurs**, the tube vibrates to be in contact with the adjacent tube. The calculated wear depths at the contact point with the adjacent tube, AVBs and the top tube support plates are equivalent to the wear depths measured in Unit-3 SGs.*

> *When consecutive 6 or 8 AVB support points are inactive and in-plane **FEI does not occur**, the calculated tube wears at AVB support points **due to only the turbulent flow force** are equivalent to the **wear depths measured in Unit-2 SGs**."*
>
> my ***highlighting***

5.5.10 FIGURES 3A and 3B show the dispersal of the AVBs in the *in-plane* direction.

5.5.11 The AVBs act to restrain the tubes in the *out-of-plane* (OOP - side-to-side - across the *rows* of tubes) 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 inertia of the tube bundle.



5.5.12 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.

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13 Letter from Elmo E Collins (USNRC) to Peter T Dietrich (SCE), *Confirmatory Action Letter 4-12-001, San Onofre Nuclear Generating Station, Units 2 and 3, Commitments to Address Steam Generator Tube Degradation*, March 27 2012

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

5.5.13 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 13) short sections between the hot- to cold-leg top TSPs. This raises the 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).[15] 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.

5.5.14 In my opinion, this AVB geometry is clearly designed to cope only with *out-of-plane* tube motion since there is no *designed-in* resistance to movement in the *in-plane* direction. This because this type of AVB is designed to have a *'zero bar-to-tube gap'* functionality when in the hot, pressurized condition in order to minimize point contact with the tubes and the undesirable formation of dings and dents in the tube wall.

5.5.15 The wear scars at the AVB incident sites demonstrate this to be so, with the scars on both the tubes and AV bars being formed by *in-plane* relative movement in both vertical and horizontal directions – see {¶5.7.23}.

5.5.16 If, as it happened with the SONGS RSGs, the *in-plane* restraint (particularly in U3 following the flattening modification in the manufacture of the AV bars) is weak or non-existence (the intended design functionality) then *in-plane* random vibration of the AVB captured tubes will progressively reduce any residual (and unintended) AVB-to-tube contact force and, with it, the AV bar-to-tube friction, thereby freeing up and lengthening the free-span section of the tube to be excited at a lower resonance frequency with a larger amplitude vibration, thus promoting tube-to-tube contact and TTW.

15 A direct analogy here is with the fingerboard on a violin: holding down a string at a lower position on the fingerboard, say in the first position, produces a higher note, and moving up the fingerboard, produces a lower note of lower frequency of oscillation or vibration of the string. In effect, the finger pressing down and restraining the string is the equivalent to an active AVB restraint which determines the length of the free-span tube – if the AVB is effective, the free-span length is short and the natural frequency is high thereby rendering the tube less susceptible to the lower frequency FEI excitation forces. If the AVB is ineffective (ie the finger relaxed off the string), the free-span length is increased and the natural frequency lowered making the longer section of the tube more susceptible to excitation. The general rule is the greater the number of AVBs that are ineffective, then the longer the free-span and lower the excitation frequency.

5.5.17 This provides a logical explanation why the U3 tube wear was more advanced than in U2, even though U2 had operated in-service about twice as long as U3. Whereas with corrected and flattened AVB bars, the *in-plane* AVB-to-tube contact forces in U3 were low or zero as intended by the *'zero gap'* design strategy, the contact forces in U2 were higher due to the unintended residual clamping forces imposed by the uncorrected distorted AV bars.

5.5.18 Unintentionally, this 'improvement' in the manufacture of U3 led to a detrimental acceleration of the AVB-to-tube and TTW wear over the unmodified U2.

5.5.19 In fact, MHI's analysis of the various instances of tube wear quite specifically identifies the mechanisms in play [p81, Section 7]:[14]

5.5.20 *" . . . The conclusions regarding mechanistic causes of tube wear are as follows:*

- *The concluded mechanistic cause of the Type 1 wear {TTW} is tube FEI in the tube bundle U-bend region, which is caused by a combination of the SG secondary side thermal-hydraulic conditions (high fluid velocity and high void fraction) and **inactive AVB support conditions** in the in-plane direction.*
- *The concluded mechanistic cause of the Type 2 {AVB} and 3 {TSP} wear is random vibration of the tubes. The Type 2 {AVB} wear is caused by the tube motion due to high void fractions and high flow velocities. The Type 3 {TSP} wear is caused by high velocity flow across the straight leg sections of the tubes.*
- *The concluded mechanistic cause of the Type 4 {RB} wear type is vibration of the retainer bar, which is the same as in the Unit-2 SGs and is addressed in Reference 4.*

The tube-to-AVB contact forces of Unit-3 were more likely to be insufficient to prevent the in-plane motion of tubes and the Unit-3 SGs were more susceptible to in-plane tube vibration than Unit-2 SGs because the average contact force in the Unit-3 SGs was found to be smaller than the average contact force in the Unit-2 SGs. The difference in the contact forces between the Unit-2 and Unit-3 SGs was caused by the manufacturing dimensional tolerance variations, mainly due to improvement of AVB dimensional control.. . ."

my additional *{explanation}* and ***emphasis***

5.5.21 I can summarize MHI's findings as follows:

5.5.22 **TABLE 1 MHI'S SUGGESTED CAUSES OF VARIOUS MODES OF TUBE WEAR**[14]

TUBE WEAR MODE	MHI WEAR TYPE	FEI	TUBE RANDOM EXCITED VIBRATION	AVB ASSEMBLY	COMMENTS
TTW	1	■ in-plane		□ inactive AVBs tube in-plane direction	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 RB tubes exhibit no AVB/TSP or TTW

5.5.23 I consider these very important findings to have been overlooked by SCE in its understanding of the tube wear causal mechanism and, indeed, for its justification of the restart of U2 when responding to the CAL.

5.5.24 The three rudiments underpinning these finding are:

5.5.25 i) that degradation of the tube restraint localities (AVBs and TSPs) occurs in the absence of FEI activity;

5.5.26 ii) that TTW, acknowledged to arise from high *in-plane* FEI activity, generally occurs where the AVB restraint has deteriorated at one or more localities along the length of individual tubes; and

5.5.27 iii) that (from inspection of the U3 portions of TABLE A) the number of tube wear sites or incidences for AVB/TSP locations outstrips the TTW wear site incidences in the tube free-span locations.

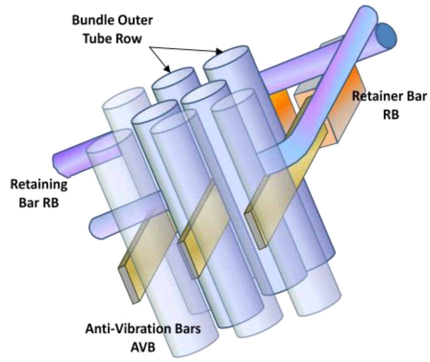
5.5.28 I find that the AVB assembly, which features strongly in the onset of TTW, is clearly designed to cope only with *out-of-plane* tube motion since there is little *designed-in* resistance to movement in the *in-plane* direction - because of this, it is just chance (a virtually random combination of manufacturing variations, expansion and pressurization, etc) that determines the *in-plane* effectiveness of the AVBs.

5.6 RETAINER BAR VIBRATION AND TUBE WEAR

5.6.1 TABLES A, 1 and 2 (see later) identify incidences in both U2 and U3 RSGs of tube wear which, although relatively low in number, the tube wall thickness reduction is quite severe, at one site in U2 reducing the wall thickness by 90%.

5.6.2 FIGURE 5 is typical of each of the 24 anti-vibration bar restraint assemblies located around outside of the U-bend region of the tube bundle as shown by FIGURE 3. The restraint assemblies, acting across the bridging plates (that run from top to bottom of the tube bundle), provide additional retention of the tube bundle during normal operation and abnormal fault conditions.

5.6.3 The continuous *retain<u>ing</u>* bar wraps around the tube bundle to which is fixed the outboard ends of the AV bars. The *retaining* bar is pulled in, wrapped around the tube bundle by the hairclip-like *retain<u>er</u>* bar, this being captured in situ by being threaded through the first two rows of tubes, and held in this position by friction between the *retainer* bar and the inboard top surfaces of the AV bars.



5.6.4 Because the tube-to-tube clearance tightens towards the apex of the U-bend, one-half of the total restraint assemblies require a smaller diameter *retainer* bar in order to fit between the tube rows.

5.6.5 In October 2012 MHI reported directly to the NRC[16] its safety concerns about the retainer bars:

5.6.6 *". . . The Steam Generator tube wear adjacent to the retainer bars was identified as creating a **potential safety hazard**. The maximum wear depth is 90% of the tube thickness. The cause of the tube wear has been determined to be the retainer bars' random flow-induced vibration caused by the secondary fluid exiting the tube bundle. Since the retainer bar has a low natural frequency, the **bar vibrates with a large amplitude**. This type tube wear could have an adverse effect on the structural integrity of the tubes, which are part of the pressure boundary.*

> *The plugging of the tubes that are adjacent to the retainer bars was performed. MHI has recommended to the purchaser* [SCE] ***to remove the retainer bars that would have the possibility of vibration with large amplitude*** *or to perform the plugging and stabilizing for the associated tubes. . ."*

my [addition] and ***highlighting***

16 *<u>Part 21 – Steam Generator Tube Wear Adjacent to Retainer Bars, October 5 2012</u>*, NRC Region 1, Defects and Non-Compliance, 10 CFR 21.21(d)(3)(i)

5.6.7 According to MHI, it is the lower resonance frequency of the smaller diameter *retainer* bars that is susceptible to turbulent two-phase flow exciting the bar into its prime resonance or some harmonic frequency thereof [p10, item 3].[14] Whatever, a number of the tubes capturing the *retainer* bar had sustained abraded wear from interaction with it. These tubes comprised six tubes in U2 and four tubes in U3, with seven tubes in total showing wear greater than the 35% limit of the tube wall thickness for which isolation from service is required by plugging with, as previously noted, an incidence site in one of U2 RSGs having worn through 90% of its wall thickness.

5.6.8 I agree with the findings of MHI that the tube wear at the retainer bar localities arises because of random flow induced (not FEI) vibration of the *retainer* bar itself, it being entirely independent of any tube motion excited from other sources.

5.6.9 However, MHI's advice to either plug the local tubes and/or remove the retainer bars at risk raises two issues unique to the retainer bar and its sub-assembly:

5.6.10 i) Plugging of the at-risk tubes is not a satisfactory solution because it is the retainer bar that vibrates via random fluid flow processes at sub FEI critical velocity levels - these are likely to continue in play or, indeed, exacerbate at the proposed U2 restart at 70% power, leading to through-tube abrasion, the detachment of tube fragments, lodging at other unplugged and in-service tube localities, resulting in the so-called *'foreign object'* tube wear;

5.6.11 ii) MHI's recommendation that those retainer bars at risk of large-amplitude fluid flow excited vibration should be removed is, of course, dependent upon reliable analysis to identify the *at-risk* assemblies; and, importantly,

5.6.12 iii) this restraint system probably also serves to contain the tube bundle geometry during a main line steam break (MSLB) design basis event, so any change or removal of the retaining bar assemblage would require a full safety justification.

5.6.13 Since the tubes worn by the retainer bar do not exhibit any of the other wear modes (ie TTW, AVB and/or TSP), I share MHI's quite reasonable assumption that the retainer bar excitation and the resulting tube wear is independent of and does not contribute to the AVB/TSP-to-tube and TTW sites.

5.7 PHASING OF AVB-TSP WEAR -*vs*- TTW

5.7.1 Now I shall examine the phasing of the wear mechanisms active in the RSGs.

5.7.2 To reiterate: in the U3 RSGs both TTW has developed in the FEI active regions of the tube bundles and AVB/TSP-to-tube wear has occurred at localities where FEI is inactive. In the U2 RSGs, apart from a single two-tube incidence of TTW and setting aside the separate cause RB-to-tube wear, all of the tube wear is at AVB and TSP locations.

5.7.3 At its simplest this is the causality dilemma of the *chicken-or-the-egg*?

5.7.4 *The Chicken*: That is did FEI forced motion of the free span sections of the tubes (where the TTW commonly occurs) lead to wear and hence relaxation of the restraint localities?

5.7.5 *The Egg*: Or was it the other way round, with the AVB/TSP localities freeing-up and, hence, providing the free-span tube sections with the degree of freedom enabling relatively large amplitude oscillatory motion?

5.7.6 I reason that the high rate of tube-to-tube wear is preceded by a period whilst the newly manufactured and tightly packed tube bundle wears in or *'slackens off'*. This general slackening of the tube bundle progresses as certain of the TSP localities wear and, separately, as the unintended contact relaxes between the AVBs and individual tubes, the restraint conditions, particularly in the U-bend region of the tube bundle, drift into a quasi-relaxed condition.

5.7.7 In its analysis of the FEI conditions in the U2 RSGs, AREVA[17] recognizes and develops an understanding of the interaction between the tubes and AVBs, noting that the restraint against *in-plane* motion of the tubes offered by 'new' AVBs declines as the AVB-to-tube contact surfaces fret and wear away. I generally agree with AREVA that this AVB-to-tube slackening off process results in a decline in the AVB effectiveness as an *in-plane* restraint, although I consider AREVA not to have demonstrated that FEI is the root cause.

17 Attachment 6 – Appendix B: SONGS U2C17 - Steam Generator Operational Assessment for Tube-to-Tube Wear, AREVA – ASLB's *Tube-to-Tube Report*.

5.7.8 MHI[14] come to much the same conclusion on the AVG-to-tube wear progression and accompanying loss of *in-plane* effectiveness, although it considers the causation is via random vibration excited by two-phase flow perturbations that are not at FEI levels.

5.7.9 Identifying the actual cause of AVB-to-tube wear is crucial to understand and model the timing of the second phase involving TTW.

5.7.10 a) Put simply, the AREVA postulate leads to the approach that if the reactor power level is reduced to 70% then FEI will cease, so AVB *in-plane* effectiveness will also cease to decline further, and TTW will be arrested.

5.7.11 b) To the contrary, if MHI is correct then, driven by random flow perturbations, decline in AVB *in-plane* effectiveness will continue to advance even at the reduced 70% power level, thereby freeing-up longer free-span tube sections that are more susceptible to TTW.

5.7.12 I shall return to this issue {¶5.8} later, but here I wish to explore how AREVA[17] approached its determination of the AVB-to-tube wear rate.

5.7.13 To this end, simulations of U3 and U2 RSGs compare the AVB-tube contact force distributions as these progress through their respective in-service periods – after 22 months of in-service operation the severity of AVB (and TSP) wear in the U2 RSG 2-089 is similar to that found after 11 months of in-service of U3.

5.7.14 The implication of this comparison is that the central U-bend tube region of U2, at 22 months, could be about to respond to FEI forces and hence commence a period of heightened TTW activity. In some respects, this may be supported by MHI's changes in the manufacturing flattening process of the AV bars between U2 and U3 RSGs {5.5.17}, although not I suggest for the precise reasons given by MHI.

5.7.15 AREVA concludes that [p15, ¶3][17]

5.7.16 "*. . . the location and orientation of the two shallow TTW indications in Unit 2 are consistent with the behavior observed in Unit 3 and indicates that* ***in-plane fluid-elastic instability in Unit 2 began shortly before the end of cycle 16 operation after 22 months of operation****.*"

my ***emphasis***

5.7.17 The other but independent OA, by Intertek APTECH,[18] approaches this from a statistical viewpoint [p24 - 26, Figures 3.2/3/4][18] showing that the incidence pattern for the first phase of wear at the tube supports is virtually identical and complete for U2. A sense of the progression of U2 towards the second phase involving TTW is given by inspection of TABLE A.

5.7.18 The 2nd, 3rd and 4th columns of TABLE A show the incidences of AVB- and TSP-to-tube wear and TTW for the U2 and U3 RSGs and for each table segment, the successive higher rows show the bands of increased wear depth. It is possible to present the data of TABLE A in a more comprehensive form, for example by linking the number of AVB wear sites to individual tubes, but for my purposes the present tabulation will suffice.

5.7.19 The first comparison to be made is the incidence of AVB/TSP to TTW by comparing across the columns, thus:

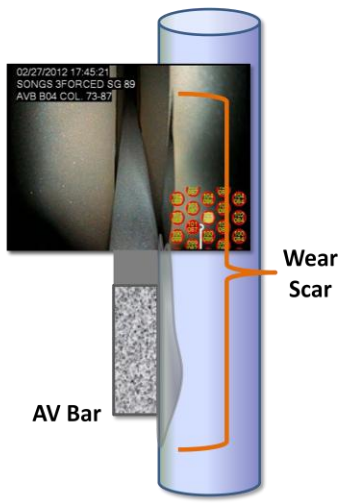
5.7.20 i) for U2, this shows that AVB/TSP-to-tube wear is occurring in the absence of TTW (ignoring the two tube incidence in 2E-089);

5.7.21 ii) for U3, in which there is an increased number of AVB/TSP-to-tube wear locations on tubes in which high incidence of TTW is established, the suggestion is that AVB/TSP -to-tube wear continues to arise; and

5.7.22 iii) the implication being that new AVB/TSB sites are seeded and continue to develop as a function of the in-service hours of the RSG.

5.7.23 The second observation, ii), suggests that adjacent tubes are repeatedly impacting in the *in-plane* direction with a resulting physical displacement of the tube at its AVB restraint location. Evidence of this *in-plane* movement, both horizontally and up-and-down {¶5.5.14}, is clear from the elongated AVB-to-tube wear scars as the *in-plane* effectiveness of the AVB is degraded.

5.7.24 In this way, an initially stable neighboring tube may be 'bumped' into instability as its AVB *in-plane* restraint is virtually worn away.



Adapted from MHI

18 Attachment 6 – Appendix C: *Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16* – Intertek APTECH for AREVA

This impacting mechanism leads to a growing region of instability in TTW in the free-span section of the affected tubes along a *column* of tubes.

5.7.25 Referring back to TABLE A, the second comparison is with the total number of tubes that have one or more wear sites, shown in the R/H column, suggesting that:

5.7.26 iv) since the range of individual tubes with one or more wear indications (734 to 919) is not particularly broad, the incidence of TTW is related to tubes in which the restraint systems have already been committed to AVB/TSP-to-tube wear.

5.7.27 The similarity of the depth of AVB wear between U2 and U3 RSGs as this relates to the projected steam quality and void fraction with this, separately in-plane velocity, is shown by MHI Figures 5.1-4/5, strongly suggesting that U2 is following the same AVB deterioration process as A3 [p62-63, Figure 5.1-4/5].[14]

5.7.28 MHI provides a generalized summary of the relationships between TTW and the restraint localities AVB/TSP/RB [p18, Table 4.1.1-1]:[14]

5.7.29 **TABLE 2 MHI'S WEAR TYPE LOCATIONS** (MHI TABLE 4.1.1-1)

WEAR PATTERN		WEAR LOCATIONS			
		FREE- SPAN	AVB	TSP	RETAINER BAR
TYPE 1	TTW	YES	YES	(YES)	NO
TYPE 2	AVB WEAR	NO	YES	(YES)	NO
TYPE 3	TSP WEAR	NO	NO	YES	NO
TYPE 4	RB WEAR	NO	NO	NO	YES

YES wear indication found
(YES) wear indication may be present since some tubes with AVB wear have no indications at TSP locations
NO no wear indication

5.7.30 In other words, in my *chicken-and-egg* quandary the outcome is not quite so straightforward.

5.7.31 First, the *egg* arrives - that is slackening of the tube bundle restraint systems, particularly at the AVBs.

5.7.32 I agree with MHI {5.7.11} that the fluid mechanism involved is the *in-plane*, two-phase fluid excitation of random, small amplitude vibration of individual tubes fretting away the friction grip of the AVB surface interface to the tube.

5.7.33 A rough and ready guide to the number of AVBs that are so slackened by this mechanism is given by the AVB incidences (in the absence of TTW) in the U2 RSGs (ie 1757 and 2591) of TABLE A.

5.7.34 Second, the relaxation of the AVB restraints provides conditions conducive for *in-plane* tube motion in the free-span sections to be triggered. This lower frequency, higher amplitude *in-plane* motion is sufficient for the unrestrained tube to impact against its restrained neighbor. This impacting motion and force is transmitted to the stable tube's AVB point of restraint that, with repetitive impacts, deteriorates to ineffectiveness allowing that same tube to be excited in its modified (lengthened) free-span section.

5.7.35 Again, a rough and ready guide to the number of AVBs that are so slackened by this tube-to-tube bumping mechanism is given by the AVB incidences (in the absence of TTW) in the U3 RSGs (ie 3357 and 3149), that is about one-third increase over the tubes slackened by the first phase of degradation {¶5.7.31}.

5.7.36 Tube-to-tube bumping transmits the instability along the columns of tubes, that is in the *in-plane* direction running along the AVBs as if these were tramway tracks. It does not, however, account for migration of free-span instability in the *out-of-plane* direction, being across the rows of tubes.

5.7.37 This row to row transfer could be via *out-of-plane* FEI or, more likely in my opinion, providing that the *in-plane* fluid activity is sufficiently dispersed then it, alone, could be enough to initiate and follow through removal of AVB (and TSP) restraint effectiveness across a number of tube columns and, as reported by MHI {TABLE 2}, the fluid mechanisms involved are not necessarily FEI.

5.7.38 AREVA acknowledges [p19, ¶4][17] that < *redacted proprietary information*

. . . >

5.7.39 *" . . . < redacted proprietary information*

>. . .*”*

my ~~*correction*~~ and *addition*

5.7.40 On this basis AREVA reckon [p44, ¶4][17] that

5.7.41 “. . **.< *redacted proprietary information***

>”

5.7.42 However, whereas previously AREVA acknowledged that < *redacted proprietary information*

. . .> *”* [p44, ¶4]:[17]

5.7.43 *“* . . . < ***redacted proprietary information***

>. . .*”*

my ***emphasis***

5.7.44 As I have previously noted, the effectiveness of the *in-plane* capture of the columns of tubes by the AVBs relates to the AV bar-to-tube clamping force and, hence, friction force between the tube and AV bar surfaces. For SCE, via its consultants and MHI, to establish how many of the presently active AVBs will advance to ineffectiveness via tube-to-tube bumping, it has to reliably predict the AV bar-to-tube clamping and friction forces for all of the remaining active AVBs in each U2 RSG, it has to do this ‘blind’ via remote *‘see-through-the-tube-wall’* probes and by inference when the RSG is cold and depressurized.[19]

5.7.45 It is worthwhile pausing to note that, probably, the greater number of AVBs in the U3 RSGs were in the design intended (*zero-gap*) inactive condition, hence the accelerated TTW incidence. To the contrary, because of the distortion and twisting of the AVBs of the U2 RSGs, the state of the AVB restraints was random, being inactive (as design intended) or active with various degrees of contact force in force.

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19 Bobbin probes tend to be subject top large signals generated by geometrical tube-wall distortions at the U-bend transition locations, creating significant reduction in their detectability, which is probably why the second round of inspections deployed, according to SCE [p5, ¶2][11] a ‘more sensitive’ method.

5.7.46 To model the AVB effectiveness approximately < ***redacted proprietary . . .***> gap elements need to be considered for each U2 RSG and, even by rationalizing this down by a quarter model of the tube bundle, the computer resources required are < ***redacted . .*** >[p56, ¶2].[17,20,21]

5.7.47 I have considerable reservations about the reliability of such modeling, particularly:

5.7.48 i) setting up the model relies upon the consistency of manufacture and assembly of the various components of the tube bundle as delivered by MHI, a challenge in itself but which has been cast into doubt by recent findings of non-conformance by the NRC;[22]

5.7.49 ii) there is insufficient explanation and justification of the AREVA[17] model of projecting AVB wear for its outcome to be adopted in extrapolating to the AVB wear rates and distributions, both of which are key to projecting short term TTW and other tube wear sites in the restarted U2 RSGs – AREVA's explanation as to how this achieved [p57, ¶2][17] is, for the most part, confusing;

5.7.50 iii) the reliance upon the tube non-classical ding and dent locations,[23] determined by eddy current transducing is not a sufficiently robust as a final check of the viability of the AVB contact force magnitude and distribution; and there are similar reservations associated with the non-access to a relatively large number of plugged and inaccessible tubes in the regions of interest;

5.7.51 iv) there is unwarranted confidence that the spatial geometry of the tubes and components in the cold, unpressurized condition (ie that deduced from the

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20 This is a finite element model which resolves the sum of forces, including the AVB contact forces, to zero. So far as the AVBs relate, the gap distribution is determined by consideration of i) the tube diameter, ii) AVB thickness, iii) TSP hole location, iv) AVS twist or warpage, v) AVB flatness and vi) tube flatness. In the manufacture of U2 RSG AVB the distortion caused by bending the AVB bars into the 'hairpin' shape left a residual twist compared to the U3 AVBs which resulted in a greater capture force, as evidenced by the increased number of 'dents' in the tubes located in the nose section of the AVBs.

21 Westinghouse also acknowledged that *'software limitation'* confined the analysis of wear progression to no more than three wear scars' [p25, ¶2].[50]

22 NRC Inspection Report No 99901030/2012-201, *Notice of Non-Conformance*, November 20 2012 – this report found failures in the MHI quality assurance program, including lack of dimensional control over the tubes used in a mock-up SG tube bundle being developed to explore and prove AVB modifications to the SONGS RSGs. During the manufacture of the RSG tubing a stop notice was placed on the tubing manufacturer with respect to quality assurance procedures.

23 The fact that the AVB contact is denting the tubes shows that at some localities (towards and at the return nose section of the AVB) the AVB design is doing exactly the converse of what was intended, this being that the AVBs would not touch and clamp the tubes. Now, however, AREVA [p63, ¶2][17] depend upon the unintended outcome of this design failure to show the ABV 'effectiveness' which is, some might argue, clutching at straws.

probe inspections) will reliably transpose to the same in the hot, pressurized condition – this is important when the tube-AVB geometry is offset, where the AVB is twisted with respect to the tube,[24] etc.. and

5.7.52 v) as I have previously noted {¶5.5.14}, this type of AVB assembly was not designed to provide effective restraint in the *in-plane* direction so, it follows, no specific contact and friction force levels were specified at the onset[25] or, put simply, *in-plane* tube motion was not foreseen at the design stage so nothing was put in place to counter it.

5.7.53 To arrive at these findings I have concentrated my assessment on the AREVA *Tube-to-Tube Report*[17] following its prognosis that

5.7.54 *". . . the TTW in the SONGS steam generators was caused by in-plane tube movement due to in-plane fluid-elastic instability (FEI)."*

5.7.55 And

5.7.56 *". . . However, given identical designs, Unit 2 must be judged, a priori, as susceptible to the same TTW degradation mechanism as Unit 3 where 8 tubes failed structural integrity requirements after 11 months of operation [12]. Indeed, the location and orientation of the two shallow TTW indications in Unit 2 are consistent with the behavior observed in Unit 3 and indicates that* ***in-plane fluid-elastic instability*** *in Unit 2 began shortly before the end of cycle 16 operation after 22 months of operation.*

. . . The argument that incipient in-plane fluid-elastic has developed in Unit 2 is considered a more logical explanation for the observed TTW . . ."

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

5.7.57 However, an entirely contrary argument is put by Westinghouse in its OA when accounting for the TTW of the two tubes in U2 [p87,¶3]:[32]

5.7.58 *". . . all available data suggest that the tube-to-tube wear in the U-bend free span* ***did not result from in-plane vibration*** *of the tubes. There is strong indication that it* ***resulted from out-of-plane vibration*** *of the two tubes in close proximity to the level of actual contact during operation."*

my ***emphasis***

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24 Westinghouse shows [p76, Figure 2-18][32] the relationship between *Wear Depth* and *Wear Volume* for various angles of twist with the notching effect of a AVB-to-tube wear scar with a 4^{o} twist angle being ~x3 deeper than for the untwisted case.

25 A criterion for the effectiveness of the individual AVB contact force was set at 3 Newton, based on the probabilistic base of the computed contact forces, and a total number of consecutive ineffective AVBs for each tube was set to establish whether the associated free-span section of the tube was stable or unstable – in the most limiting case the AREVA projection identified this to require a minimum of 4 effective AVBs [p104, ¶2].[32]

5.7.59 And [p88,¶1-2]: [32]

5.7.60 *"... Since the tubes were **stable in-plane** at 100% power, they will be **stable in-plane** at 70% power with additional margin. ... The evaluation showed that the **in-plane** stability ratios of all tubes in Unit 2 are **less than 1** at 70% power. Hence, **in-plane vibration will not occur in the Unit 2 SGs during the upcoming operating cycle at power levels up to 70%**. **Since all active tubes will be stable against in-plane vibration in the next cycle, tube-to-tube wear due to in-plane vibration in the U-bend free span, as has been observed in Unit 3, will not occur in Unit 2 during the next cycle of operation.** SG performance criteria will be satisfied for this degradation mechanism until the next inspection."*

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

5.7.61 I note here that there are three clear conflicts of findings between the OAs: From AREVA[17] that AVB-to-tube and TTW result from *in-plane* FEI, contrasted to Westinghouse[50] that there is no *in-plane* FEI but most probably it was *out-of-plane* FEI, and from MHI[14] {5.5.9} that certain AVB-to-tube wear results in the absence of *in-plane* FEI from just turbulent flow.

5.7.62 My opinion is that such conflicting disagreement over the cause of TTW reflects poorly on the depth of understanding of the crucially important FEI issue by each of these SCE consultants and the designer/manufacturer of the RSGs.

5.8 WEAR RATES - PREDICTING THE IN-SERVICE PERIOD

5.8.1 The overall objective of the three operational assessment (OA) commissioned by SCE was to gauge the structural integrity and accident induced leakage of individual tubes[26,27,28,29] of the U2 RSGs following a period in service at a pre-specified level of thermal power (70%).

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26 The fundamental OA structural integrity criteria is that the projected worst case degraded tube for each existing degradation mechanism must meet the limiting structural performance parameter with a 95% probability and 50% confidence

27 *Structural Integrity* is defined by the Electric Power Research Institute (EPRI) - *All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions . . . and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials . . . In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.*

28 *Accident-Induced Leakage - The primary to secondary accident leakage rate for the limiting design basis accident shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rates for an individual steam generator.*

29 See SG Tube Integrity [p3.4-51.¶3.4.17].[40]

5.8.2 SCE evaluated and interpreted the results of these independent assessments in its wrap-up report [p19, Table 3-1]:[30]

5.8.3 **TABLE 3 RESULTS OF THE OAS – COMPARISON** SCE TABLE 3-1

OA Description	Degradation Other Than TTW	TTW with No Effective AVB Supports	Traditional Probabilistic TTW	Deterministic TTW
Reference Appendix	A - 31	B - 17	C - 18	D - 32
Degradation Mechanisms	All but TTW	TTW	TTW	TTW & AVB Wear
Type	Probabilistic	Deterministic	Probabilistic	Deterministic
Thermal Power	100%	70%	70%	70%
Inspection Interval	18 months	18 months	16 months	18 months

5.8.4 The inspection interval of 16 to 18 months range at 70% thermal power (final row of TABLE 3-1) is based on elimination of susceptibility of FEI (SR=1),[33] within the tube bundle, but SCE's table does not present the full range of results for the inspection interval produced by the OAs.

5.8.5 SCE also evaluated the inspection interval if FEI was assumed to initiate at SR=0.75 rather than at SR=1 which reduces the inspection period, again at 70% thermal power, to 8 months. If, however, FEI tube motion activity is present immediately upon re-start (at 70% thermal power) then the time period for failure of an unplugged tube further reduces to 6 months – ie one month longer than SCE's proposed U2 in-service 150 day period before inspection. The 6 month unplugged tube failure is triggered by the higher risk of initiating FEI motion in a plugged tube causing TTW on a neighboring tube [p117, S10].[17]

5.8.6 If my premise {¶5.7.44} of tube bumping along the *in-plane* columns is accepted, then either *in-plane* or *out-of-plane* FEI, or *turbulent flow forces* alone {5.5.9), could advance the AVB-to-tube wear, and hence result in loss of AVB effectiveness, across successive <u>rows</u> of

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30 Attachment 6: SONGS U2C17 - Steam Generator Operational Assessment, SCE. October 10, 2012

31 Attachment 6 - Appendix A: SONGS U2C17 Outage – Steam Generator Operational Assessment, AREVA October 1, 2012

32 Attachment 6 – Appendix D: Operational Assessment of Wear Indications In the U-bend Region of San Onofre Unit 2 Replacement Steam Generators, Westinghouse Rev 3 October 2012

33 Fluid-elastic instability (FEI) is the interaction of two-phase fluid flow across a tube array, such as the liquid-steam flow across the SG tube bundle in the region of the U-bend. The individual tubes are excited into motion at a critical cross-flow velocity with each oscillating tube generating a pressure field acting on adjacent tubes which, in turn, respond in motion. This phased and coupled motion increases with increasing cross-flow velocity leading to, if the tube bundle geometry permits, tube-to-tube impacting and/or fretting with retainer bars, etc.. The onset of the unrestrained tube motion occurs at a *critical velocity*, determined by the fluid properties and tube array geometry, representing a point at which the amount of energy input to the tubes exceeds the amount of energy being dissipated by fluid damping, itself determined mainly by the voidage or fraction of steam-to-liquid make of the fluid – this is referred to as the stability ratio (ratio of *effective* to *critical* velocity - SR) where a unity value (SR=1) is the point at which FEI induced tube movement is expected to trigger – sometimes the term Excitation Ratio (ER) is used where the presence of restraints (ie AVBs are included in the system).

tubes. The location and interception of zones of plugged tubes, which have to be appropriately located, will only serve to delay but not halt this advance – the length of delay will be determined by the induced wear rate at the AVB-to-tube contact which, itself, is a function of the tube-AVB geometry, the contact and friction forces in play, and so on.[20]

5.8.7 I have previously touched upon the difficulty of determining whether or not any particular AVB is acting effectively as an *in-plane* restraint {¶5.7.52}, so to project the time period for that particular AVB to wear down to a relaxed or ineffective restraint compounds the difficulty with even more uncertainty. Moreover, the degree of restraint effectiveness, the wear rates and time taken to the eventual relaxation state for each of several thousand AVB-to-tube contacts, has to be mapped out for each of the U2 RSGs if, that is, the inspection interval for the U2 nuclear plant is to be determined to be free of unacceptable risk of in-service tube failure.

5.8.8 I have previously referred to this complex, uncertain and, most probably, non-linear process of loss of AVB *in-plane* restraint effectiveness, together with advance across rows of tubes in regions of *out-of-plane* FEI activity, as *'slackening off'* of the tube bundle.

5.8.9 AREVA's[34] probability analysis of the inspection records of the U3 RSGs claims to overcome these uncertainties, reckoning that this *'slackening off'* time t_{so} for the known instability zone expansion took about 7 months [p113, ¶2].[17] AREVA then, with much the same confidence, calculate the time t_{ttw} for tube-to-tube wear, or TTW, to arrive at the in-service tube time-to-burst T_{tb}, that is any one tube failing in accord with the tube structural integrity requirement.[35]

5.8.10 The reliability of AREVA's approach is very much dependent upon the siting of preventatively plugged tube buffer zone location. For the U3 RSG the TTW pattern is established (because TTW exists on many tubes), so there is greater certainty about where to locate the preventatively plugged buffer zone.

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34 In this Evidence I shall concentrate on the AREVA approach to determining the Inspection Interval somewhat at the neglect of the accounts of MHI and WEC. MHI, for example, approach this with an arithmetic scoring system comprising of the nine criteria relates to one of the following characteristics of in-plane fluid-elastic vibration: (1) tube-to-AVB friction, (2) vibration frequency, (3) in-plane tube motion, (4) high void fraction, (5) regional effect, and (6) coupling effect [p10, Table 1].[14]

35 The limiting structural integrity performance criterion (SIPC) for a tube burst is that the tube must meet x3 the normal power pressure differential ($\triangle P$) between the primary and secondary (steam) circuits with the plant at normal operating power (NOP). At 70% power, U2 would develop a $\triangle P_{NOP}$ of 9.1MPa (1,324 lbf/in^2), so tubes should be resilient up to a minimum $3x\triangle P_{NOP}$ of 27.3MPa. This structural integrity criterion is that the projected worst case degraded tube for each existing degradation mechanism must meet the limiting structural performance parameter with a 95% probability and 50% confidence.

5.8.11 Because there is just the one (of two adjacent tubes) TTW incident in one of the U2 RSGs (2E0-89), the appropriateness of the location of the preventatively plugged buffer zone for U2 is more *hit-and-miss.* In the absence of a multi-tube TTW pattern, AREVA had to interpret (by inference from the bobbin and eddy current *'blind'* inspections) the AVB-to-tube wear patterns and, from this, extrapolate the desired location of the preventively plugged buffer zones in the U2 RSGs. Without much explanation [p113, ¶2],[17] AREVA calculate the *slackening off* period t_{so} for FEI expansion in U2 to be reduced by half of the 7 month estimate for U3, that is 3.5 months.[36]

5.8.12 The final element of the TTW time-to-burst composite is, simply, the period t_{ttw} for the TTW action to reach and unacceptable level of wear depth to an in-service, pressurized tube. For this, AREVA admit that the uncertainties arising in dynamic (complex) modelling of tube-to-tube *impacts* are too great and so reverted to a simple estimate[37] yielding a range for t_{ttw} of between *"2.5 to 11 months"*. Thus, AREVA's best estimate for the period to *tube burst* ($T_{tb} = t_{so} + t_{ttw}$) in the restarted U2 at 70% power for the worst case flaw, to be between 6 to 18 months [p114, ¶3].[17]

5.8.13 **TABLE 4A TUBE FLAW BURST TIME – MONTHS FROM RESTART**

CASE	SLACKENING OFF TIME t_{so}	TTW TIME t_{ttw}	TIME TO BURST T_{tb}
U3	7	2.5 to 11	9.5 to 18
U2	3.5	2.5 to 11	6 to 18

5.8.14 Finally, AREVA arrives at another interpretation by considering a combination of two different extremes for the abrasion rate in account of dynamic loading [p125, FIGURE A-3].[17] This approach yields a further revision in the TTW time t_{ttw}:

5.8.15 **TABLE 4B WORST CASE TUBE FLAW BURST TIME EXTREMES – MONTHS**

CASE	SLACKENING OFF TIME t_{so}	TTW TIME t_{ttw}	TIME TO BURST T_{tb}
U2^{static}	3.5	4.5 to 8	8 to 12
U2^{dynamic}	**3.5**	**2.5 to 5**	**6 to 8.5**

5.8.16 AREVA has arrived at this final range of results for the time-to-burst T_{tb} on the basis of a number of physical uncertainties and assumptions, so much so there is little confidence that

36 It appears that the U3 result has simply been factored by x50%.

37 The *'simple'* approach adopts Archard's rule which, in terms of the wear volume, relates the product of a wear coefficient to the abrading rate and wear time

any form of reliable distribution would be expected to apply between the two sets of extremes of T_{tb}. Put another way, the tube burst could occur at any time (randomly) between, for the U2^{dynamic} case above, 6 to 8.5 months from restart of the U2 nuclear plant.

5.8.17 Nor can there be that much confidence in simply extending the time-to-burst T_{tb} by front-ending with the *slackening off* period t_{so} of 3.5 months, as shown in TABLE 4B. AREVA arrives at this period for the U2 RSGs simply by halving the U3 *slackening off* period t_{so} (TABLE 4A) and, similarly, to convert the static to dynamic loading for the U2^{dynamic} t_{ttw} case of TABLE 4B a simple x2 factor has been assumed.

5.8.18 AREVA collates the most optimistic outcome of its probability [p110, Figure 8-3],[17] (right) neglecting those of its considerations that results in the 2.5 month U2 restart in-service period to TTW.

< *redacted proprietary information* …>

5.8.19 I have little confidence in the outcome of AREVA's projection of the time period through which the U2 nuclear plant could be reliably expected to operate without incurring a tube failure or running at a greater risk of a tube failure occurring.

5.8.20 This is because:

5.8.21 i) the root cause leading to TTW is the omission in the design of the AVB features to cope with *in-plane* motion of the tubes – since this was not specified at the design stage, the *in-plane* effectiveness of the AVBs is, essentially, something acquired by chance, being highly variable and inconsistent from one AVB to another;

5.8.22 ii) the key assumption that the AVB tube capturing friction force can be reliably deduced indirectly by the eddy current probe reading of the AVB-to-tube gaps, for thousands of AVB-tube locations, is not at all robust;

5.8.23 iii) there are similar, indeed, if not greater difficulties in applying such measurements made when the RSG is cold and depressurized, to the RSG when it is in service, that is hot and pressurized, particularly when it is acknowledged that thermal-pressure distortion (flowering) is an acknowledged in-service phenomenon in the higher U-bend region of the tube bundle; and

5.8.24 iv) even if these uncertainties can be resolved, which in my opinion is highly unlikely, there must remain strong doubts about the quality assurance at the MHI manufacturing and procurement plants if, that is, the recent NRC Non-conformance Notice[22] relating to the tubing for RSG mock-ups currently being evaluated by MHI, were to have equally applied to the manufacture of the tubes installed in the U2 RSGs.

5.8.25 I am unable to go further with Intertek APTECH's[18] analysis of the TTW time because it seems to make a fundamental error in deriving the U2 wear time rate from the whole of the cycle length for the U3 wear [p38, ¶2][18] (ie ignoring the AVB slackening off period). That said, even in account of this (apparent) significant error the total time to burst remarkably closely coincides at 16 months with the other OA time to burst predictions of TABLE 3 {¶5.8.3}.

5.9 TTW TUBE PERFORMANCE DURING A MSLB EVENT

5.9.1 The AREVA and WEC OAs considered the limiting *structural integrity performance criterion* (SIPC - ie $3x\triangle P_{NOP}$). However, SIPC is one-dimensional in that only differential pressure forces are included, whereas other external forces apply to the tube bundle and RSG structural containment during certain fault sequences.

5.9.2 For example, during the design basis MSLB event individual tubes are subject to imposed bending force and stresses which add to the overall force composite acting in the tube wall so, for these circumstances, the *accident induced leakage performance criteria* (AILPC) applies.

5.9.3 In a separate assessment of the U3 TTW and TSP wear profiles, AREVA identified a number of tube wear modes, wall thickness wear depths and specific locations that failed AILPC.[38] Tubes found to be at risk included TTW [p69, ¶1][38] and TSP [p56, ¶5][38] with a *'pop-through'* failure mode.

5.9.4 SCE does not seem to have applied this U3 finding to setting a limitation on the acceptable tube wall thickness wear for the U2 restart on the basis of AILPC alone which, for TSP and TTW modes of wear, will equally apply in U2 during an MSLB design basis event.

5.9.5 I discuss further the additional forces acting on the RSG tubes during fault conditions later.[42]

6 RESPONSE TO THE ATOMIC SAFETY LICENSING BOARD'S FACTUAL ISSUES

6.1 My response to the ASLB's factual issues[39] is as follows, seriatim:

7 FACTUAL ISSUE iv) – FINAL SAFETY ANALYSIS REPORT

7.1 *". . . Does the Final Safety Analysis Report (FSAR) analyze a steam generator (S/G) tube failure event?*

If it does, how many tubes are assumed in the analysis and what is the primary-to-secondary leak rate?

What is a conservative rate?

Please provide a copy of this section of the FSAR."

7.2 I have not been able procure a copy of the amended FSAR other than a short extract that has been provided by SCE as Attachment 1 of its December 13 *Answer to the Petitioner's Motion* of December 11 2012.

7.3 The FSAR extract provided by SCE comprises (what seems to be) pages 177 through to 186, it is not dated and there is no indication if it is a complete and unredacted copy extract of the FSAR.

7.4 My instructing client FoE sought to obtain the most recent amended version of the FSAR, which I believe to be dated about April 2009, but without success. I also believe that the San Onofre FSAR was, like many other documents, being reviewed under the

38 Attachment 3: AREVA Document 51-9180143-001 - SONGS Unit 3 February 2012 Leaker Outage Steam Generator Condition Monitoring Report. AREVA October 1 2012

39 United States of America Nuclear Regulatory Commission Atomic Safety and Licensing Board, In the Matter of Southern California Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3). Order December 7, 2012

Sensitive Unclassified Non-Safeguards Information (SUNSI) arrangements and that it may be being withheld (or caught up in the system) as a SUNSI group *3) Security Related Information*.

7.5 FoE has, however, obtained a list of FSAR amendments from the NRC Public Documents Room (Washington DC) but this list does not go beyond May 1983. I assume that this list is not a complete record because I certainly expect that amendments to the FSAR would have been made since 1983, particularly in account of the installation and then proposed operation of the U2 and U3 RSGs in or earlier than 2010.

7.6 Therefore, at this time I am not able to provide the Board with a copy of those sections of the FSAR that relate to the issues that it raises.

7.7 However, I am able to outline my expectations of the requirements of the FSAR as follows:

7.7.1 The heat transfer area of the two RSGs in the SONGS Unit 2 comprises well over 50% of the total reactor primary system pressure boundary. This transfer area is entirely made up of the 9,700 or so individual tubes in each RSG, so the tubes, individually and collectively, represent an integral part of the nuclear plant barriers against fission product release to the environment. Of the two fission product barriers (fuel cladding and RSG tubing) the RSG tube surface area is the substantive barrier in defense in depth.

7.7.2 Failure of this barrier, via leakage of a single or multiple tubes, enables radioactive coolant water from the reactor coolant circuit to pass into the lower pressure steamraising circuit that feeds the turbo-generator machinery hall, thereby bypassing the primary containment of the plant's nuclear island.

7.7.3 I have used the term *'leakage'* in the context that it is defined by the Operating License [p1.1-4, ¶a.3], [p3.4-37, LCO 3.4.13].[40]

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40 United States Nuclear Regulatory Commission Washington D. C. 20555 Southern California Edison Company San Diego Gas and Electric Company the City Of Riverside, California the City Of Anaheim, California Docket N^{o}. 50-361 San Onofre Nuclear Generating Station Unit 2 Facility Operating License N^{o} NFP-10 as amended – *San Onofre Nuclear Generating Station, Unit 2, Improved Technical Specification based on NUREG-1432, "Standard Technical Specifications – Combustion Engineering Reactors*"

7.7.4 The performance criteria for RSG individual tube integrity comprises the three separate requirements of i) tube structural integrity (SIPC), ii) accident induced leakage (AILPC), iii) operational leakage - {¶5.9.1-5.9.2}. These requirements are set out in the operating license [p5.0-14, ¶5.5.2.11]:[40]

7.7.5 " . . . *Steam Generator (SG) Program (continued) b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.*

1. *Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the* ***full range of normal operating conditions*** *(including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state* ***full power operation*** *primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in* ***accordance with the design and licensing basis****, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.*
2. *Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.*
3. *The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE.*"

my ***emphasis***

7.7.6 I note here that unless there is an amendment granted by the NRC to the present Operating License [p5.0-14, ¶5.5.2.11][40] then, irrespective of SCE's proposal to operate Unit 2 at 70%, RSG tubes will still be required to meet the tube integrity criteria at the *'full power operation'* primary-to-secondary pressure differential.

7.7.7 My strict interpretation of TS 5.5.2.11b.1 is that unless the present Operating License is amended, then SCE is required to demonstrate tube structural integrity at the rated thermal power (RTP) level which is 100% (plus the instrument error margin of, typically, 2%) and not the 70% RTP proposed by SCE.

7.7.8 It follows, that any contributory factor that relates or contributes to tube structural integrity (eg tube wear rates, etc) will also have to be determined at the rated RTP of 100% unless, that is, a license amendment permits otherwise.

7.7.9 The RSG accident induced leak performance expectation of the FSAR is most probably that stated in the *Improved Technical Specification Conversion*[41] for SONGs [p510, ¶1-2]:[41]

7.7.10 *". . . The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a **double-ended rupture of a single tube**. . . . The analysis for design basis accidents and transients other than a SGTR assume the SG **tubes retain their structural integrity (i.e., they are assumed not to rupture)**."*

my ***emphasis***

7.8 With this information I am able to respond the Board's questions on Issue iv):

7.9 " . . . *Does the Final Safety Analysis Report (FSAR) analyze a steam generator (S/G) tube failure event?"*

7.10 My expectation is that the FSAR will reflect upon such analysis and this will be the limiting design basis SGTR event involving a **single** RSG tube bursting when the nuclear plant is operating.

7.11 I would, in *'**accordance with the design and licensing basis**'* {¶7.7.5}, also expect the design basis to consider a coincident event involving either the LOCA, MSLB or FWLB with a SSE.[42] During and following such an event, RSG tubes are required to

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

42 These design basis events are

Loss of Coolant Accident	LOCA (RSG tube crushing mode)
Main Steam Line Break	MSLB
Feedwater Line Break	FWLB
Inadvertent Safety Valve Dump	ISVD
Safe Shutdown Earthquake	SSE

maintain structural integrity and **not** rupture. These events will introduce stress loading to the RSG tubes in addition to the acting pressure differential (primary membrane) stresses acting in each tube wall.

7.12 Additional (mechanical stress) RSG tube loading from an SSE event (ie a horizontal shaking mode) would be expected to be at a maximum in the free-span tube sections in the top region of the U-bend of the tube bundle – ineffective AVB support would further heighten these SSE generated stresses.

7.13 The principal RSG tube loading during a LOCA is generated by the rarefaction wave initiated in the primary at the break location. This wave travels through the primary circuit and will generate a differential pressure across the hot and cold legs of the U-bend, resulting in *in-plane* movement that gives rise to significant bending stress across the U-bend tube sections and large *in-plane* reaction forces at the top TSP locations. The RSG tubing may also be subject to shaking loads caused by the LOCA break hydrodynamics and reactor coolant circuit motion.

7.14 MSLB, FWLB and ISVD events introduce secondary bending stresses in the lower portions of the RSG tube bundle. For the MSLB event very high, two-phase fluid cross-flow velocities would be expected to instantaneously develop in the U-bend region, triggering vigorous FEI that could, particularly if the AVB restraints are ineffective, promote violent tube to tube clashing and the potential for a multiple tube failure event.

7.15 I note that SCE's proposal to restart U2[11] does not, apparently, include a reassessment of the additional loadings and material stresses incurred in the RSG tubes during a coincident design basis accident event. If it is accepted that the rates of TTW and AVB-to-tube wear have been reliably forecast for the proposed in-service period – I do not accept this to be so – then a reassessment of compliance with the tube structural integrity criterion should be undertaken for the wear scars that are projected to develop throughout the in-service period.

7.16 Moving on to the Board's second item of ISSUE iv):

7.17 *". . . If it {FSAR} does, how many tubes are assumed in the analysis and what is the primary-to-secondary leak rate?*

7.18 All in-service, pressurized tubes of both RSGs should be considered in the analysis with, for the design basis SSE-LOCA, etc. coincident event taking account of tube position (ie particularly the higher U-bend tube sections) and the effectiveness of the restraint (AVBs).

7.19 As I have previously noted, in the SGTR event a **single** tube rupture is the limiting design basis, whereas in all other incidents SSE-LOCA, etc., all tubes are required to maintain structural integrity throughout and following the incident.

7.20 The Operating License does not specify the permissible leakage rate for the single tube SGTR event {¶7.10}. In any event, action would be required to reduce the leak or to bring the nuclear plant to MODE 3/5 states if the *'operational'* leakage exceeded the 150 gallons per day level specified in the Operating License [p3.4-37, LCO 3.4.13][43,44] and, of course, the presence of (radio)activity in the steam raising circuit[45] would be detected and alarmed at, for example, the condenser air ejector monitoring point.

7.21 For accident induced events {¶7.11} the leakage is not to exceed 0.5 gallon per minute per RSG and 1 gpm through both RSGs.

7.22 The Operating License also stipulates that account of degradation RSG tube structural integrity should be evaluated ahead of the next 'run time' or inspection interval [p5.0-15, ¶5.5.2.11d]:[40]

7.23 *" . . . In addition to meeting the requirements of d.l, d.2, and d.3 below, the* ***inspection scope, inspection methods, and inspection intervals*** *shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An* ***assessment of degradation*** *shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations."*

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43 United States Nuclear Regulatory Commission Washington D. C. 20555 Southern California Edison Company San Diego Gas and Electric Company the City of Riverside, California the City Of Anaheim, California Docket N^{o} 50-361 San Onofre Nuclear Generating Station Unit 2 Facility Operating License N^{o} NFP-10 as amended.

44 The Operating License states [p3.4-37][43] this to be '3.4.13 RCS operational LEAKAGE shall be limited to: . . . d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG)'.

45 For SG tube rupture), the source term in the primary coolant consists primarily of the levels of *Dose Equivalent 1-131* radioactivity levels calculated for the design basis accident. This, in turn, is based on the limiting values in the *Technical Specifications* and postulated iodine spikes. For accidents in which the source term in the primary coolant consists of the *Dose Equivalent 1-131* activity levels, the SG tube rupture yields the limiting values for radiation doses at offsite locations. In the calculation of radiation doses following this event, the rate of primary to secondary LEAKAGE in the intact SGs is set equal to the operational LEAKAGE rate limits [LCO 3.4.13]. For the ruptured SG, a double ended rupture of a single tube is assumed.

my ***emphasis***

7.24 And, similarly, [p3.4-51, ¶LCO 3.4.17]:

7.25 *". . . A.l Verify tube integrity of the affected tube(s) is maintained until the* ***next refueling outage or SG tube inspection****."*

my ***emphasis***

7.26 As I previously discussed {¶5.8}, the methodology and data available for predicting both the conditions conducive to AVB wear, the wear rate and the eventual loss of AVB effectiveness is not at all robust.

7.27 I have similar very serious doubts about the reliability of determining the TTW rate and, particularly, the period of time projected for individual tubes to reach a condition that would threaten the structural integrity of individual tubes – the basis of the methodology for arriving at the time-to-burst requires substantiation of the underlying assumption that the wear rate is a linear and not a non-linear phenomenon,[32,46] and that the local AVB geometry, eg where there exists a dominant tube-to-AVB offset to one side, reliably translates from the cold, unpressurized to the hot, pressurized condition.[47]

7.28 In other words, with such uncertainties prevalent, RSG tube integrity cannot be assured throughout the proposed inspection interval proposed by SCE.

8 FACTUAL ISSUE v) – SONGS SG COMPARISON TO OTHER OPERATING SGS

8.1 *". . . Figure 4-3 in the report entitled "Operational Envelope for Large U-bend Steam Generators, SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear" [hereinafter Tube-to-Tube Report] compares the velocity ratio and void fraction ratio to several successfully operating large S/Gs, and it notes that "[a]t 100% power, the thermal-hydraulic conditions in the u-bend region of the SONGS replacement [S/Gs] exceed the past successful operational envelope for U-bend nuclear [S/Gs] based on presently available data." Tube-to-Tube Report at 17.*

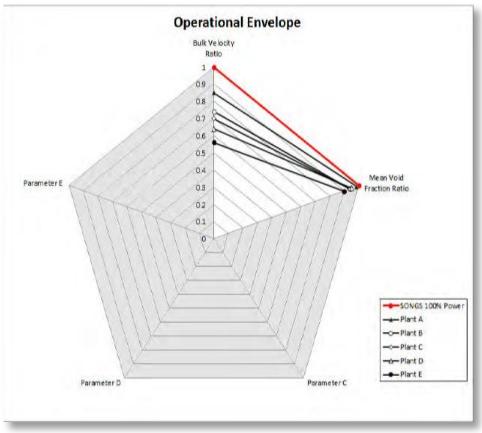
How similar to the SONGS S/Gs are these other S/Gs?

46 Benchmark trials should also be undertaken to confirm the wear rate characteristics of the Inconel Alloy 690 specifically for the SONGS conditions, including steam side water quality. It may be necessary to take these trials further than the baseline tests undertaken by Westinghouse [p21, ¶3].[32]

47 WEC report that even a small tube offset differential (>10%) between the adjacent AVBs, the wear rate was determined by the nearest AVB, although it is not clear whether offset in the cold, unpressurized state directly translated to the hot, pressurized state [p22, ¶4].[32]

> *Do the other steam generators, for example, use alloy 670* (sic 690) *tubes and have similar spacing, similar support structures, etc.?* "

8.2 Figure 4-3 (shown right) of the *Tube-to-Tube Report* does not plot, as it purports, [p17, ¶3][17] *'many factors'* providing a periphery that *'defines the operational parameters'* for a particular plant.



8.3 As I see it, the diagram is endeavoring to portray the energy balance that determines the onset of fluid elastic instability (FEI). Essentially, FEI results when (via fluid dynamic forces acting the tube) the energy input exceeds the amount of energy that can be dissipated by that rate system damping available.

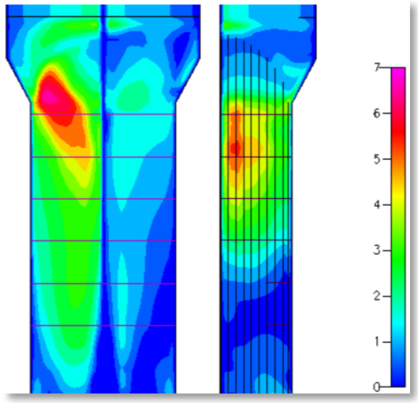
8.4 There are a number shortfalls with this depiction:

8.5 The input energy is the dynamic velocity ($\sim v^2/\rho$) of the two-phase fluid impinging on the tube. The energy dissipation is via damping which is strongly related to the two-phase mix of the fluid, here water and steam as described by the void fraction. Increase in steam content, a greater void fraction, reduces the damping and, correspondingly, the increased volume results in an increase of the impinging velocity.

8.6 The diagram represents only two groups and not *'many'* factors referred to by AREVA. The two groups represented are described as the *Bulk Velocity Ratio* and *Mean Void Ratio* that, logically, represent the factors making up input and output energy transfers respectively. Of these, the *Mean Void* is straightforward being a linear outcome, whereas the *Bulk Velocity* is the energy outcome of a force deriving from the square of the impinging velocity ($\rho\, v^2/2$).

8.7 It is not clear if the vertical axis of the Figure 4-3 represents the velocity (v) or, more correctly, the square of the velocity ($\rho\, v^2/2$), that is being more representative of the comparison between the SONGS and other SGs.

8.8 The next difficulty I have with Figure 4-3 is what exactly is meant by the *Bulk Velocity* and *Mean Void* parameters?



8.9 For example, consider the *Bulk Velocity*: SCE shows [p43, Figure 8-3][11] the velocity contours (v) predicted active in the SONGS U2 RSG at 100% power. The plot (right) shows the greatest velocity ■ present in the top section of the hot-leg in the U-bend region with the dispersal of this velocity represented *out-of-plane* (LH diagram, left side) and *in-plane* (RH diagram) - these are regions are where FEI is predicted to have been most active in the tube bundle.

8.10 AREVA states that [p17, ¶5]:[17]

8.11 *"... At 100% power, the thermal-hydraulic conditions in the u-bend region of the SONGS replacement steam generators exceed the past successful operational envelope for U-bend nuclear steam generators based on* ***presently available data****."*

my ***emphasis***

8.12 The inference here is that Figure 4-3 is comparing like-with-like, but that would require AREVA having undertaken an ATHOS flow analysis[48] for each of the comparative SGs. This I consider unlikely because for this AREVA would have required access to very detailed information on the design geometry and flow paths throughout the comparative SG tube bundles – being a designer/manufacturer of steam generators itself, I very much doubt that AREVA would have had access to such proprietary information from competitor manufacturers.

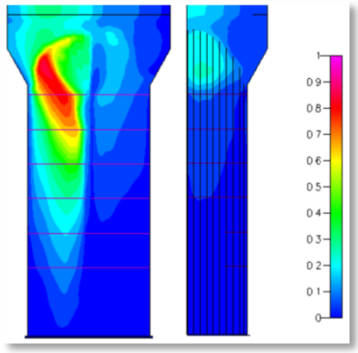
8.13 So since it is unlikely that AREVA would have carried out an ATHOS computer simulation for each of the five (A to F) comparative nuclear plants, then Figure 4-3 is unlikely to be directly comparing two-phase fluid flow velocity distribution in the critical FEI regions of the SONGS and comparative plant SG tube bundles.

8.14 I can only surmise that the Figure 4-3 comparison is between the mean or average velocity within the overall tube bundle for SONGs and each of the comparative plants. Moreover, since the velocity distributions within each of the comparative plants, because of different design geometries, flow areas, etc, will not be identical, it is very

48 Of course, if no presently operating SGs have experienced *in-plane* FEI [p16, ¶4][17] then it may be that computational routines such as ATHOS have never been tested in this respect. However, I have insufficient experience of ATHOS to comment further on this detailed aspect.

unlikely that the mean or average velocity presented in Figure 4-3 provides even a crude basis of comparison of the FEI potential of the SONGS RSGs.

8.15 Much the same may be concluded for the *Mean Void* comparison of Figure 4-3. The SCE plot (right) of *steam quality* [p39, Figure 8-1][17] indicates the complexity of the two-phase fluid in the SONGS RSG tube bundle, again presenting the same uncertainties, if not impossibility, in drawing meaningful comparisons with the comparative plants.



8.16 In other words, unless the spider diagram of Figure 4-3 somehow, and I cannot reason how, is making a direct comparison of the complex two-phase fluid cross flow situation in the SONGS and other five comparative plant steam generators, then it only provides the basis of a somewhat meaningless comparison.

8.17 On the Board's issue of similarity between the SONGS and the other five comparative plants, I can provide no further information because the documentation[49] identifying the plants has not been disclosed to me.

8.18 Steam generator manufacturers now favor thermally treated Inconel *Alloy 690* over the earlier used *Alloy 600* because it has improved corrosion resistance, However, *Alloy 690* has a lower heat transfer coefficient than *Alloy 600* so to compensate for this replacement steam generators have more tubes to increase the net heat transfer surface area.

8.19 Indeed, this need to increase the heat transfer area (ie putting more tubes into the RSGs) and, with this, reducing the steamside flow area, may have been a strong contributory factor to the enhanced FEI activity in the SONGS FSGs. Moreover, the location of the additional tubing, particularly in what I would describe as the lower swirl space immediately above the tube support sheet, may have contributed to and/or determined the unique *in-plane* flow characteristics of the SONGS RSGs,

8.20 *Alloy 690* tubes are deployed in the all four SONGS RSGs.

49 SONGS Document 90200, Rev. 0. *Average and Maximum Thermal-Hydraulic Parameter Comparisons between Songs RSGs and Similar Plants* – Ref 18 of the *Tube-to-Tube Report.*[17]

9 FACTUAL ISSUE vi) – 70% POWER LEVEL APPLIED TO SONGS AND OTHER SGS

9.1 *“. . . Figure 5-1 in the Tube-to-Tube Report compares the same parameters as in Figure 4-3, but for operation at 70% power. It appears from Figure 5-1 that the bulk fluid velocity for SONGS is at the high end of the experiential range.*

Given the likely differences between the SONGS generators and those cited in the discussion, can one conclude that operation at 70% power is conservative? “

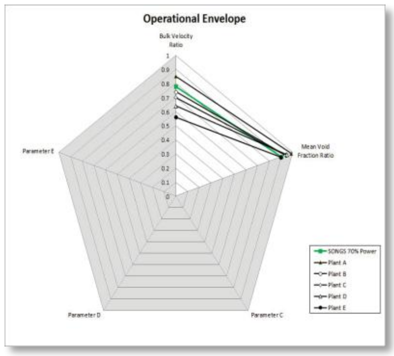
9.2 I have previously aired my reservations and doubts about the spider diagram representation of Figure 4-3 {¶8.2 to 8.17} - much the same applies to Figure 5-1.

9.3 AREVA claims [p43, ¶4][17] that Figure 5-1 demonstrates:

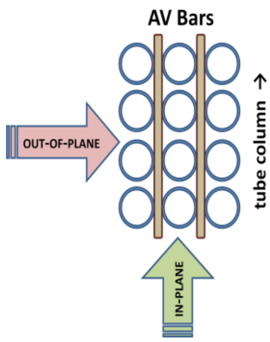
9.4 *“. . . A decrease to 70% power places the SONGS steam generators back inside the operational envelope of demonstrated successful performance relative to **in-plane** fluid-elastic stability of nuclear steam generators with large U-bends.”*

my ***emphasis***

9.5 Once again, AREVA is not comparing like-with-like. This is because the basis of the comparison being made by Figures 4-3 and 5-1 is with the parameters that determine the activity of FEI. As I have previously discussed, for FEI to result in tube motion (ie *instability*), as well as the appropriate levels of dynamic velocity and damping, the tubes have to be sufficiently unrestrained, particularly in the direction of the impinging two-phase cross flow.



9.6 In other words, FEI also has directional properties, these being *in-plane* or *out-of-plane*, or both.



9.7 In fact, AREVA discusses this at some length, concluding [p16, ¶4]:[17]

9.8 *“ . . . Prior to the observations at SONGS Unit 3, **no in-plane instability** had been observed in any U-bend nuclear steam generator.”*

my ***emphasis***

9.9 So, it follows, if none of the five comparative steam generators had previously experienced *in-plane* FEI, which is AREVA’s assertion, then there is no deduction to be drawn from Figure 5-1.

9.10 It is, I suggest, important to note the unique nature of the *in-plane* FEI that has affected the SONGS RSGs and no other steam generators. This could only have arisen from a difference or differences between the generally consistent designs of steam generators of other manufacturers and that of the SONGS RSGs.

9.11 Whereas I can confidently opine that the lack of any formalised AVB *in-plane* restraint effectiveness, ie *'left to chance'* {¶5.8.21}, has played a major role, there may be other secondary influences and factors that have rendered the SONGS RSG uniquely vulnerable to *in-plane* FEI.

9.12 A complete understanding of the causation of the *in-plane* FEI is essential to ensure that the SONGS Unit 2 plant is acceptably safe to restart and, once restarted, predictably safe to continue in operation over the proposed 150 day inspection interval. To the contrary, the understanding presented by SCE is neither comprehensive nor convincing.

9.13 In my opinion, simply sweeping the FEI issue under the carpet on the basis of (*in-* or *out-of-plane*) FEI will not reoccur at 70% power is not only disingenuous but foolhardy.

10 FACTUAL ISSUE vii) – FEI SR = 0.75 PROBABILITY AT 70% POWER

10.1 *". . . Section 8.0 in the Tube-to-Tube Report states that "[t]he desired margin is a projected maximum stability ratio of 0.75 with 0.95 probability at 50% confidence over the next inspection interval of 5 months." Tube-to-Tube Report at 104.*

Does a confidence level of 50% meet the reasonable assurance requirement in the regulations?"

10.2 For the general and specific reasons that I expounded upon throughout my Affidavit, I do not agree that the confidence level of 50% will satisfy the regulatory requirement.

10.3 Also, as I read it, the meaning of the first-half of the introductory paragraph the *Tube-to-Tube Report* [p104, ¶1][17] only to apply to FEI stability at the time of start-up of Unit 2, whereas to the contrary the second-half of the paragraph acknowledges that:

10.4 *". . . Some effective in-plane supports are needed to maintain a stability ratio of 0.75. In the most limiting case, 4 effective supports are required. This requirement applies to approximately 120 U-bends."*

10.5 In other words, the *Tube-to-Tube Report* acknowledges that ABV wear and effectiveness will continue as U2 progresses through the in-service period, that is the AVB wear advancing through the preventatively plugged zones as I have previously discussed {¶5.8}.

10.6 Again, I pause to reflect that in its reasoning AREVA requires at least four effective AVB-to-tube contact (clamping) points to safeguard 120 tubes. However, these active AVBs are only available by default because the original design *'zero-gap'* intent was not achieved (ie the AV bars distortion remained uncorrected). To undertake and commit to a probability based on a characteristic (the AVB being in-plane active) that was never part of the design intent is piling uncertainty upon uncertainty.

10.7 Also, I refer to Section 9 of the AREVA *Tube-to-Tube Report*, particularly [p114, ¶1][17] which predicts the in-service, pressurized tube-to-burst time of 2.5 months, being shorter than the proposed inspection interval of 5 months (150 days) – I consider this both generally {¶5.8} and in some detail {¶5.8.12 to ¶5.8.24} – this period for U2 seems to have been arrived at by quite unscientifically halving the same period for U3.

10.8 A difficulty that I have with the AREVA[17] and, generally, with the other OAs is that whereas the results of analyses, particularly relating probability and confidence, are often stated there is very little of the analytical procedures arriving at the results are open to inspection.

11 FACTUAL ISSUE viii) – OPERATIONAL ASSESSMENT -*vs*- TEST AND EXPERIMENT

11.1 *". . . Throughout the Tube-to-Tube Report, the term "operational assessment" is used.*

How is the term "operational assessment" different than or the same as the terms "test" and "experiment" used in 10 C.F.R. § 50.59?"

11.2 In its *Steam Generator Operation Assessment* SCE refer [p9, ¶2][50] to the *Songs Steam Generator Program* (undisclosed)[51,52] and which is likely to conform to the Nuclear

50 Attachment 6, *SONGS U2C17 Steam Generator Operational Assessment*, SCE, October 3 2012

51 SONGS Steam Generator Program, SO23-SG-1

52 SONGS Technical Specifications Sections 5.5.2.11, "Steam Generator (SG) Program," Amendment 204

Energy Institute's (NEI) *Steam Generator Program Guidelines*.[53] Under the general heading *Integrity Assessment*, the NEI guidelines state that [p11,¶3.3]:[53]

11.3 *". . . Licensees assess tube integrity after each steam generator tube inspection. The assessment includes: . . .*

- ***Operational Assessment** – A forward-looking assessment which **demonstrates** that the tube integrity performance criteria **will be met** throughout the next inspection interval."*

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

11.4 As I have previously identified {¶7.23}, the requirement for an operational assessment is also stipulated in the Operating License [p5.0-15, ¶5.5.2.11d].[40] To reiterate:

11.5 *" . . . In addition to meeting the requirements of d.l, d.2, and d.3 below, the **inspection scope, inspection methods, and inspection intervals** shall be such as to ensure that SG tube integrity is **maintained** until the next SG inspection. An **assessment of degradation** shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations."*

my ***emphasis***

11.6 And, similarly, [p3.4-51, ¶LCO 3.4.17]:

11.7 *". . . A.l **Verify** tube integrity of the affected tube(s) is maintained until the **next refueling outage or SG tube inspection**."*

my ***emphasis***

11.8 Clearly, the regulatory expectation is that an *'operational assessment'* is an objective exercise to demonstrate and verify the performance of the nuclear plant – in these respects the OAs should not rely upon elements of *'test'* and/or *'experimentat'*.

11.9 Moreover, in the steam generator case, the OA objective is quite specific, being to demonstrate that the structural and leakage integrity requirements of the tubing is compliant with limiting structural integrity performance criterion (SIPC), which does not rely upon any element of *test* and/or *experiment.*

11.10 On the face of it, the *Tube-to-Tube Report*[17] shares this objectivity, giving its basis for the operational assessment to be [p12, S1.0]:[17]

53 NEI 97-06, "SG Program Guidelines," Rev. 3, January 2011

11.11 *". . . an operational assessment (OA) must be performed to ensure that steam generator (SG) tubing will **meet established performance criteria** for structural and leakage integrity during the operating period prior to the next planned inspection. The OA projects and **evaluates** tube degradation mechanisms which have affected the SGs to date. . ."*

my ***emphasis***

11.12 But, put to the test, neither the AREVA nor any of the other OAs are underpinned by this basic prerequisite of objectivity.

11.13 NRC 10 CFR §50.59 *Changes, Tests and Experiments*[54] defines the key words of *'change'*, *'tests'* and *'experiments'* as follows:

11.14 *". . . (1) **Change** means a modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished. . . .*

*(6) **Tests** or **experiments** not described in the final safety analysis report (as updated) means any activity where any structure, system, or component is utilized or controlled in a manner which is either:*

*(i) **Outside the reference bounds** of the design bases as described in the final safety analysis report (as updated) or*

*(ii) **Inconsistent with the analyses or descriptions** in the final safety analysis report (as updated). . . "*

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

11.15 The NRC also publishes a guidance on how *changes*, *tests* and *experiments* (CTE) should be evaluated,[55] setting out eight evaluation criteria [p9-11, S4],[55] with each 10 CFR §50.59 evaluation considering the following *Evaluation Guideline* (EAG to EDG) [p8, S3]:[55]

EAG	*". . . - systems and components affected by the change (What is the effect of the change on their capability to perform their specified or intended functions?);*
EBG	*- parameters of the accident analysis affected by the change (Are all the relevant design basis accidents and transients identified?); and*
ECG	*- potential effects of system or component failure (i.e., the question, "what would happen if..." is explored and answered in the evaluation)*
EDG	*- how the evaluation criteria are met."*

54 19 CFR § 72.48 *Changes, Tests, and Experiments* NRC 10 CFR § 50.59

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

11.16 I can review each of the eight evaluation criteria for potential CTE impacts on the FSAR, although note {¶7.2}.

11.17 In the following tabulations I have **emphasized** those sections of the NRC guideline text[54] relevant to my response following each of the evaluation criteria, my response is versed in terms of SCE's *Return to Service Report*[11] and, overall, it is not intended to be far-reaching nor comprehensive.

11.18 **TABLE 5A CRITERION i) - CTE IMPACTS - 10 CFR § 50.59**

INCREASE IN ACCIDENT FREQUENCY
If the CTE would result in **more than a minimal increase in the frequency of an accident previously evaluated** in the FSAR (as updated). The intent of the criterion is to allow changes to be made without approval unless there is a discernible, attributable increase in frequency of an accident. There must be some reason to believe that the CTE would result in a more than minimal impact upon the accident frequency (as because **it affects the integrity of the reactor coolant system**, or the **ability of SSC to remove decay heat**, or makes an **initiating event more likely to occur**). Departures from the design, fabrication, testing and performance standards in the General Design Criteria are not compatible with a "no more than minimal increase" standard.

11.19 **Single and Multiple Tube Failure:** The individual *Operational Assessments* presented by SCE have individually failed to demonstrate a clear and proven relationship between reactor operational power level, the rate(s) of TTW and TSP- and AVB-to-tube wear and the *in*- and *out-of-plane* fluid flow forces that promote tube and other component motions that result in wear.

11.19.1 This renders the compliance with SIPC {¶11.33} uncertain, so much so that it remains doubtful that there is not an increased frequency of accident involving a single or multiple tube failure in normal operational and during and/or following design basis fault events.

11.19.2 The absence of a robust demonstration of tube integrity for both normal and design basis accident conditions includes EAG, EBG, ECG and EDG evaluation guidelines.

11.20 **TABLE 5B CRITERION ii) - CTE IMPACTS - 10 CFR § 50.59**

INCREASED LIKELIHOOD OF OCCURRENCE
If the CTE would result in more than a **minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated** in the FSAR (as updated). As for frequency, the intent is that there be some reason to conclude that the CTE has resulted in an increase in likelihood, rather than the licensee having to prove that it could not happen. In making these assessments, the licensee's evaluation should consider the effects of the proposed CTE on performance of all affected SSC and make a determination as to whether there has been an increase, and provide the basis for the determination. Specific guidance is included in NEI 96-07.

11.21 **Retainer Bars – Plugged Tubes Physical Integrity**: SCE consider that plugging the tubes local to the self-vibrating retainer bars [p32, ¶7],[11] to be an adequate solution to continuing tube wear and loss of tube integrity. However, since it is the retainer bars themselves that are vibrating, and not the tubes, then wear of the albeit plugged and depressurized tubes will continue with the possibility that free-debris will be generated and swept into and redistributed elsewhere in the RSG tube bundle.

11.21.1 SCE has not presented any analysis of the further degradation of tubes adjacent to and in contact with the smaller diameter retainer bars – this analysis should consider tube sear through, debris generation and the potential consequences of foreign object tube wear at other locations in the tube bundle.

11.21.2 In this respect, the potential effects of a component failure, the evaluation guidelines EAG and ECB have not been met – without supporting analysis of the entire retainer bar-to-tube degradation cycle, the outcome depends on an element of *experiment*.

11.22 **Preventatively Plugged Tubes:** The protection of in-service tubes identified to be at risk of FEI is via the preventative plugging of tubes to form buffer zones – these buffer zones are intended to delay (not necessarily halt) the advancing FEI during the proposed 150 day inspection interval for U2.

11.22.1 This scheme of things is illustrated by {5.8.18} and [p110, Figure 8-3].[17]

11.22.2 There are a number of unresolved aspects relating to this, particularly, that

11.22.3 i) in U3 the TTW is established and the incidence high, so it is relatively straightforward to optimize the location of preventively plugged buffer zones, but in the two U2 RSGs there is only one example of TTW, so the location of the preventatively plugged zones is crucially dependent upon accurately forecasting the AVB wear sites that have yet to develop;

11.22.4 ii) unlike AREVA[17] which considers FEI the driving fluid mechanism for AVB-to-tube wear, MHI[14] reckons the tube motion excitation source to be random, two-phase flow perturbations and not FEI – such a difference of opinion from two authoritative bodies suggests that a great deal of uncertainty about the cause of AVB-to-tube wear persists; and

11.22.5 iii) in either case of ii) above, it is necessary to model (*in*- and *out-of-plane*) FEI components to predict the TTW, and hence the tube structural integrity, but the accuracy and reliability of the ATHOS software to do so for a) such extensive buffer zone plugging, b) the geometric design of the SONGS RSGs, and c) at 70% RTP is unproven.

11.22.6 The whole process of mapping out and quantifying the AVB (and TSP) –to-tube wear and the loss of AVB effectiveness, and then TTW, is wrought with uncertainty, so much so that the proposed application of its outcome must include a great deal of *test* and *experiment.*

11.22.7 In these important respects, SCE's proposal restart of U2 would not satisfy the guidelines of EAG, EBG, ECB and EDG.

11.23 TABLE 5C CRITERION iii) - CTE IMPACTS - 10 CFR § 50.59

INCREASED RADIOLOGICAL CONSEQUENCES
If the CTE would result **in more than a minimal increase in consequences of an accident previously evaluated** in the FSAR (as updated). The term "consequences" refers to radiological consequences, and consequences are with respect to **offsite release**, and onsite release, to the extent that onsite releases are evaluated in the FSAR for a particular accident or location (as for example, the control room). As discussed in the implementation guidance, a CTE involves no more than a minimal increase in consequences if the resulting dose (with the change) is no greater than the current licensee-established value plus ten percent of the difference between the regulatory value (specified in the regulations, e.g., GDC 19 or Part 100) and the current value, and provided that the result does not exceed the value established in the Standard Review Plan(SRP) guidance for the particular design basis event if applicable. Applicability is with respect to the particular type of accident, not whether the plant was specifically licensed using the SRP. Also as noted, the intent is to require NRC review of changes with more than a minimal increase in consequences. Consistent with a "minimal" concept, small changes in predicted dose (on the order of 0.1 rem) do not require prior approval, even if the above guidelines are not met. One special case of consequences concerns doses to operators outside the control room, as assessed under the Three Mile Island (TMI) action plan, where the applicable standard for "minimal" is whether the GDC 19 values would continue to be met.

11.24 **Multiple Tube Failure:** A multiple tube failure event, as described {¶11.19} would, for all phases of the reactor in-core fuel cycle, result in a significant increase in the off-site radiological consequences over the single tube burst event currently considered in the FSAR.

11.24.1 The greater frequency and potentially increased radiological consequences of such a failure justifies a revision of the FSAR.

11.25 **TABLE 5D CRITERION iv) - CTE IMPACTS - 10 CFR § 50.59**

INCREASED RADIOLOGICAL CONSEQUENCES FROM SSC MALFUNCTION
Similar to the third, and is if the CTE would result in **more than a minimal increase in (radiological) consequences of a malfunction of an SSC important to safety** previously evaluated in the FSAR (as updated). The above discussion as to understanding of "minimal" also applies to consequences of malfunctions.

11.26 **Multiple Tube Failure:** A multiple tube failure event, as described {¶11.19} would, for all phases of the reactor in-core fuel cycle, most likely result in a significant increase in the off-site radiological consequences.

11.26.1 {11.24.1} applies.

11.27 **TABLE 5E CRITERION v) - CTE IMPACTS - 10 CFR § 50.59**

CREATION OF DIFFERENT ACCIDENT TYPE
If the CTE would **create the possibility of an accident of a different type from any previously evaluated** in the FSAR (as updated). The intent of this criterion is to require review of changes that would create conditions that would have been viewed as design basis events **had the possibility existed before**. Thus, the assumptions typically used for design basis events, such as no credit for non-safety-related systems, postulated loss of offsite power, single failure, etc. are applicable. On the other hand, accidents that may be theoretically possible once the CTE is made if **multiple independent failures** were postulated would not be viewed as creation of an accident of a different type.

11.28 **Multiple Tube Failure:** A multiple tube failure event, as described {¶11.19} would, for all phases of the reactor in-core fuel cycle, most likely result in a significant increase in the off-site radiological consequences.

11.28.1 {11.24.1} applies.

11.29 **TABLE 5F CRITERION vi) - CTE IMPACTS - 10 CFR § 50.59**

INITIATE FAILURE OF NUCLEAR SAFETY SSC
If the CTE would create the possibility of a malfunction of an **SSC important to safety with a different result from any previously evaluated** in the FSAR (as updated). This criterion focuses upon the "effect" of the CTE, and whether the result of any malfunctions that might have been created by the CTE has already been analyzed or bounded by the analysis in the FSAR (as updated). Only if the **effect is different from those already considered** would this criterion require prior NRC approval for a CTE involving a new type of malfunction. Note that the likelihood of malfunction may be increased if new failure modes are introduced (even if the effects have been previously evaluated in the FSAR (as updated)), and this situation would have to be evaluated under criterion (ii).

11.30 **AVB Dig In Tube Wear:** SCE does not detail the nature of the various locations of tube wear [p22, ¶2].[11] However, via the OA undertaken by Westinghouse, a number of AVB-to-tube wear localities are identified where the preset twist of the AV bar

creates a wear scar that is up to x3 deeper than a scar produced by an untwisted AVB {footnote 24} and see [p76, Figure 2-18][14] and other examples where the AV bar has sharply cut into the tube.

11.30.1 This wear pattern, referred to as *Pattern 2* by MHI [p56, Figure 4.2-3],[14] is likely to include a distinct, work-hardened notch in the tube scar, being a defect that has not been previously considered in the FSAR. There are acknowledged difficulties in predicting ligament rupture pressures and leak rates for this type of tube wear.[56]

11.30.2 During normal operating conditions, or under design basis accident conditions, it is possible that fluid jets could produce damage in adjacent tubes via both droplet impact and cavitation erosion.

11.30.3 In this respect, the potential effects of a component failure evaluation guideline ECG has not been met - it is a *change* and its outcome depends on an element of *experiment*.

11.31 **Replacement Steam Generators:** Replacement of the original Combustion Engineering SGs with the MHI RSGs is likely to have influenced the plant response to a loss of coolant accident (LOCA). Moreover, the unexpected heat transfer characteristics shown by the ATHOS analysis, the extensive level of RSG tube preventative plugging and the subsequent modifications created by the proposed reduction to 70%, all could affect LOCA response, particularly in capacity and make-up rate of the emergency core cooling system (ECCS).

11.31.1 In this respect, the potential effects of failure of the RSGs to perform adequately in the event of a LOCA have not been demonstrated, so the EAG evaluation has not been met. - it is a *change* and its outcome depends on an element of *experiment*.

11.32 TABLE 5G CRITERION vii) - CTE IMPACTS - 10 CFR § 50.59

EXCEEDING OR ALTERING A FISSION PRODUCT BARRIER
If the CTE would result in a **design basis limit for a fission product barrier** as described in the FSAR (as updated) **being exceeded or altered**. As discussed in the implementation guidance, the determination of the need for NRC review is based upon whether the CTE **results in exceeding or altering one of the design basis limits**, established in the FSAR (as updated), for maintaining integrity of a fission product barrier. Effects of

56 NUREG, *Validation of Failure and Leak Rate Correlations for Stress Corrosion Cracks in Steam Generator Tubes*, Energy Technology Division, Argonne National Laboratory, December 2001

changes to SSC, including mitigation and support systems, need to be assessed with respect to whether the changes lead to exceeding or altering one of these limits. Depending upon the type of facility and its operational status, the particular fission product barriers and design basis limits may vary, but should be evident from the safety analyses presented in the FSAR (as updated). For operating power reactors, the barriers are the fuel clad, **reactor coolant system boundary**, and containment, and the design basis limits are the values for such parameters as DNB ratio, RCS design pressure, or containment design pressure. The parameters applicable to a specific facility should be ascertainable from review of the FSAR (as updated). Facility changes are judged in terms of whether the analysis results meet the criteria, such as not exceeding a design basis limit for any fission product barrier. There is not a "minimal" or amount of remaining margin standard to be applied. Effects under this criterion are to be judged using the methods described in the FSAR (as updated); methodology changes are evaluated using criterion (viii).

11.33 **SIPC:** This CET evaluation relates to that part of the reactor coolant system boundary formed by the RSG tubes and, primarily, determines if the limiting *structural integrity performance criterion* (SIPC) for individual tubes has been reached. SCE claims [p12, ¶1][11] that its submissions *"fulfill the TS requirements to demonstrate that SG tube integrity will be maintained"*.

11.33.1 This is not correct in that the OAs state but **do not** demonstrate by analysis open to inspection, that SIPC has been satisfied to 95% probability at 50% confidence for a) full (100%) power operation, see {¶7.7.7} and b) at the proposed 70% power operation.

11.33.2 In this respect evaluation criterion EDG has not been met.

11.34 **TABLE 5H CRITERION viii) - CTE IMPACTS - 10 CFR § 50.59**

DEPARTURE FROM A METHOD OF EVALUATION
If the CTE would **involve a departure from a method of evaluation described** in the FSAR (as updated) used **in establishing the design bases or in the safety analyses**. Unlike the other seven criteria for evaluating CTE, this criterion is specifically directed at changes to evaluation methods. The implementation guidance discusses the meaning of "evaluation method," and notes that the FSAR (as updated)(or documents incorporated by reference), must **describe the method**, and the change must affect this description, to **require evaluation**. Then, in accordance with criterion (viii), if the method is used in establishing the design bases, or in the safety analyses, prior NRC approval is required if there is a departure from the method as described in the FSAR (as updated). A departure occurs if some part of the method is changed, such that the result of the analysis using that method is not conservative or essentially the same. The "essentially the same" language is intended to allow licensees to benchmark revisions to methods for use without prior NRC review even if the results are not "conservative" when the changes are small, would have no effect upon the acceptability of the analysis, and the amount of change in the results is not used to justify that limits and requirements are met. "Conservative" is to be judged with respect to the results obtained from the method. If the result from the revised method is further from the established limit than under the| previous method, the revised method is in the non-conservative direction. When judging conservatism of a change in methods, a predicted result closer to an established limit is conservative, in that there is less opportunity for other changes without triggering the need for NRC review and approval. (In contrast, a facility change, which when evaluated (with no change in methods) results in a value further from the limit, is a "conservative" facility change. The difference is that it is the facility change that causes "more margin" in the actual expected result, as contrasted to an analytical result arising from a change to methodology). It is also not a departure if the licensee uses a different method that has already been reviewed and approved by NRC for the intended application, if used in accordance with the conditions and limitations specified in the approval. A different method must be used in its entirety to fall under this provision of the rule; changes to parts of methods are covered by the "essentially the same" standard noted above. Additional guidance for assessing whether a change to an evaluation method is a "departure" as defined in the rule is provided in the NEI 96-07 guidance. The elements of the evaluation method include such items as treatment of **uncertainties, correlations, and**

representations of phenomena. In contrast, items such as flows, temperatures, pressures, equipment response times that are physical characteristics of the facility are viewed either as facility changes or input parameters that are to be evaluated using the other criteria, not as "methods of evaluation." Changes to input parameters that are described in the FSAR (as updated), are to be evaluated as changes to the facility, and could be made without NRC approval as long as criteria (i) through (vii) and the TS are met.

Further, any changes to analyses and methods are also subject to design control process requirements in accordance with 10 CFR Part 50, Appendix B.

In sum, criterion (viii) is intended to preserve the basic assumptions of the evaluation method that provide the confidence that the analysis results are appropriately conservative, even if the results of the analysis are at the applicable limits or requirements.

Use of different methods without specific NRC review is acceptable only if those methods have been previously found acceptable by NRC for the intended application, or the results are conservative or essentially the same.

11.35 **70% Power Operation:** SCE proposes to return U2 to service and operate it at 70% of its thermal power rating.

11.35.1 Such a reduction in the continuous running output of the nuclear plant represents a considerable departure for current practice and, hence, will require a substantial re-evaluation of the nuclear safety case, particularly the reactor coolant circuit flows, reactor nuclear kinematics, and so on.

11.35.2 In this respect, the potential effects of this *change* on other systems and components does not seem to have been EAG evaluated and, similarly the accident sequence analysis has not been presented, so an EBG failure.

11.36 **Reactor Shut Down Procedure:** SCE state [p50, ¶9.4.1][11] that the plant operating procedures have been *changed* to enable operators to commence a reactor shutdown at a leakage level less than that allowed by the Technical Specification, although there is no statement of any assessment undertake to determine potential impacts of this revised procedure.

11.36.1 In this respect, the potential effects of this *change* on other systems and components does not seem to have been evaluated, so an EAG evaluation failure, similarly the accident sequence analysis has not been presented, so EBG failure, and the description of the *change* is ambiguous with respect to how the *change* is to be met by other requirements, such as operator training, additional actions necessary, etc., so evaluation failure EDG.

11.37 **In-Service Vibration Monitoring:** SCE refers [p52, ¶11.1][11] to upgrading the vibration and loose parts monitoring system (VLPMS) but it is not stated how this is to be achieved and how the transduced signal output (alarm points) are to be acted upon – referring to [p110, Figure 8-3][17] it is not at all clear at which stage of the slackening-off and/or TTW phases the alarm points will be calibrated.

11.37.1 Further information should be provided on the role and dependencies upon the VLPMS, so evaluation failure EDG.

12 **IN SUMMARY:** In my opinion,[57] the *changes*, *tests* and *experiments* (CTE) inherent in the SCE proposal to restart Unit 2:

12.2.1 a) involve a significant increase in the probability or consequences of an accident previously evaluated;

12.2.2 b) create the possibility of a new or different kind of accident previously evaluated; and

12.2.3 c) involve a significant reduction in a margin of safety.

13 SCE'S PROPOSAL TO RESTART UNIT 2 - CAL AND DE FACTO LICENSE AMENDMENT

13.2 In conclusion: SCE's assertion that reducing power to 70% will at the best alleviate, but not eliminate, the TTW and other modes of tube and component wear is little more than hypothesis - the supporting Operational Assessments and analyses have not proven it to be otherwise. I am of the opinion that trialling this hypothesis by putting the SONGS Unit 2 back into service will, because of the uncertainties and unresolved issues involved, embrace a great deal of *change*, *test* and *experiment*.

13.3 The terms of the *Confirmatory Action Letter* of March 11 2012, are versed such that to meet compliance the response of SCE via its *Return to Service Report*,[11] together with the OAs and other attachments, must include considerable changes of conditions and procedures that are outside the reference bounds of the present FSAR – this is because the physical condition of the RSGs, and the means by which this is evaluated and projected into future in-service operation, have substantially and irrevocably changed since the current FSAR was approved.

13.4 The fact that SCE fails to satisfy the requirements of the CAL is neither here nor there, although it illustrates the scope and complexity of the response required. At the time of

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57 This opinion is in accord with NRC, NRC Regulatory Issue Summary 2001-22 *Attributes of a Proposed No Significant Hazards Consideration Determination*, November 20, 2001 – Adams ML011860215

preparing the CAL, the NRC being well-versed in the failures at the San Onofre nuclear plant, must have known that the only satisfactory response to the CAL would indeed require considerable change to be implemented.

13.5 Put another way, the extensive and rapid rates of tube wear experience at the SONGS Unit 2 and Unit 3 RSGs, have necessitated an extensive raft of analysis, assessments and projections to qualify, or otherwise, that Unit 2 is fit for purpose. Not only is this prequalifying work unique to the San Onofre nuclear plant, much of it has never been undertaken before so, it follows, its inclusion in safety considerations must be a new and hitherto unconsidered component required to be incorporated into an updated version of the FSAR.

13.6 Hence, the CAL must, from a technical standpoint alone, be considered to have been at the time of its preparation, a de facto license amendment.

14 I John H Large declare, under penalty of perjury, that the foregoing facts are true and correct to the best of my knowledge and belief, and that the opinions expressed herein are based on my independent and best professional and personal judgment.

Executed on 22 January 2013.

JOHN H LARGE
CONSULTING ENGINEER
LARGE & ASSOCIATES, LONDON

APPENDIX I

UNIT 2 AND UNIT 3 TUBE WEAR

Table 6-1: Steam Generator Wear Depth Summary

SG 2E-088							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	0	0	1	0	1	1
35 - 49%	2	0	0	1	0	3	3
20 - 34%	86	0	0	0	2	88	74
10 - 19%	705	108	0	0	0	813	406
TW < 10%	964	117	0	0	0	1081	600
Total	**1757**	**225**	**0**	**2**	**2**	**1986**	**734***
SG 2E-089							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	0	0	1	0	1	1
35 - 49%	0	0	0	1	0	1	1
20 - 34%	78	1	0	3	0	82	67
10 - 19%	1014	85	2	0	0	1101	496
TW < 10%	1499	53	0	0	0	1552	768
Total	**2591**	**139**	**2**	**5**	**0**	**2737**	**861***
SG 3E-088							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	117**	48	0	0	165	74
35 - 49%	3	217	116	2	0	338	119
20 - 34%	156	506	134	1	0	797	197
10 - 19%	1380	542	98	0	0	2020	554
TW < 10%	1818	55	11	0	0	1884	817
Total	**3357**	**1437**	**407**	**3**	**0**	**5204**	**919***
SG 3E-089							
TW Depth	AVB Wear Indications	TSP Indications	TTW Indications	Retainer Bar Indications	Foreign Object Indications	Total Indications	Tubes with Indications
TW ≥ 50%	0	91**	26	0	0	117	60
35 - 49%	0	252	102	1	0	355	128
20 - 34%	45	487	215	0	0	747	175
10 - 19%	940	590	72	0	0	1602	450
TW < 10%	2164	94	1	0	0	2259	838
Total	**3149**	**1514**	**416**	**1**	**0**	**5080**	**887***

* This value is the number of tubes with a wear indication of any depth at any location. Since many tubes have indications in more than one depth category, the total number of tubes with wear indications is not the additive sum of the counts for the individual depth categories.
** All TSP indications ≥50% TW were in tubes with TTW indications.

TABLE A TUBE WEAR DEPTH SUMMARY – UNITS 2 AND 3

Source Table 6-1[58]

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58 SCE, Enclosure 2, *SONGS Return to Service Report*, October 3 2012

APPENDIX II

FIGURES AND DIAGRAMS

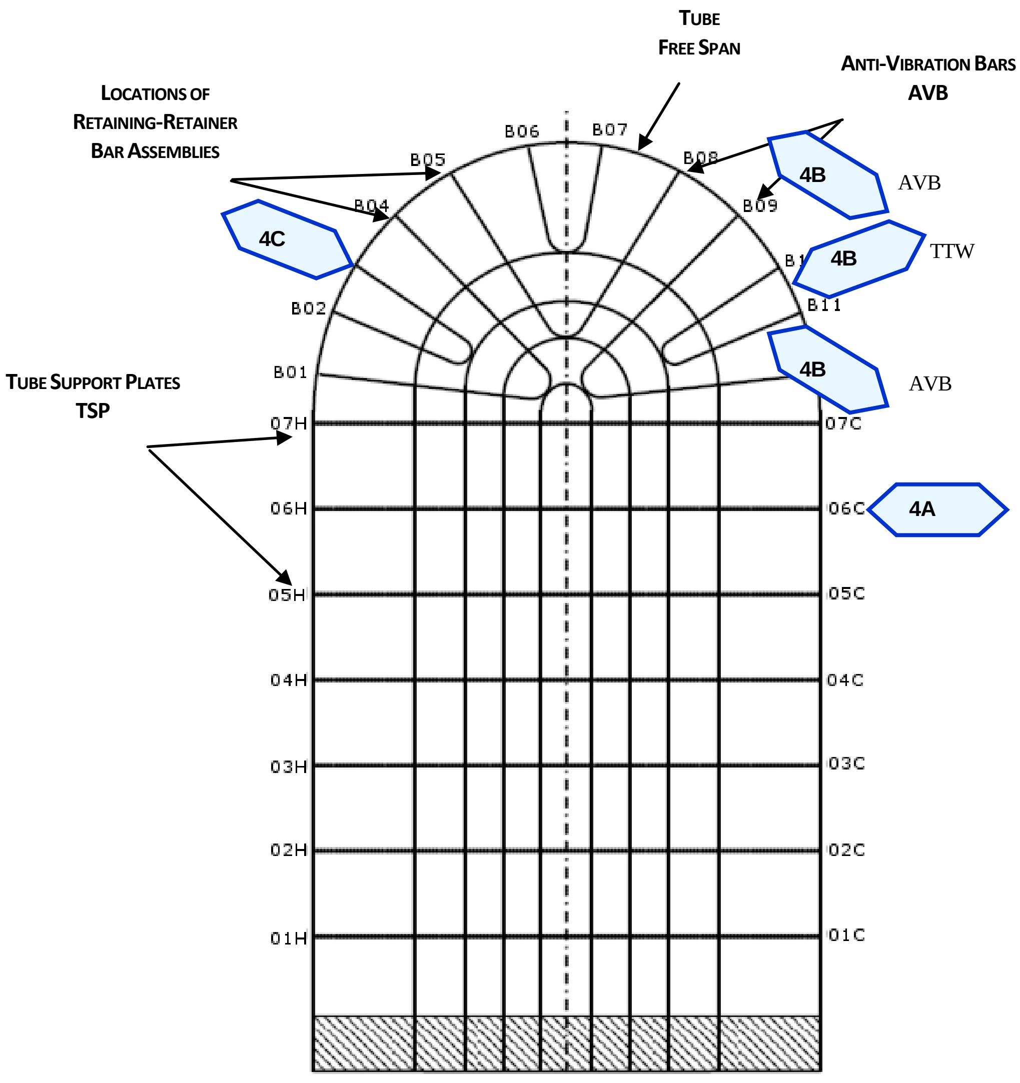
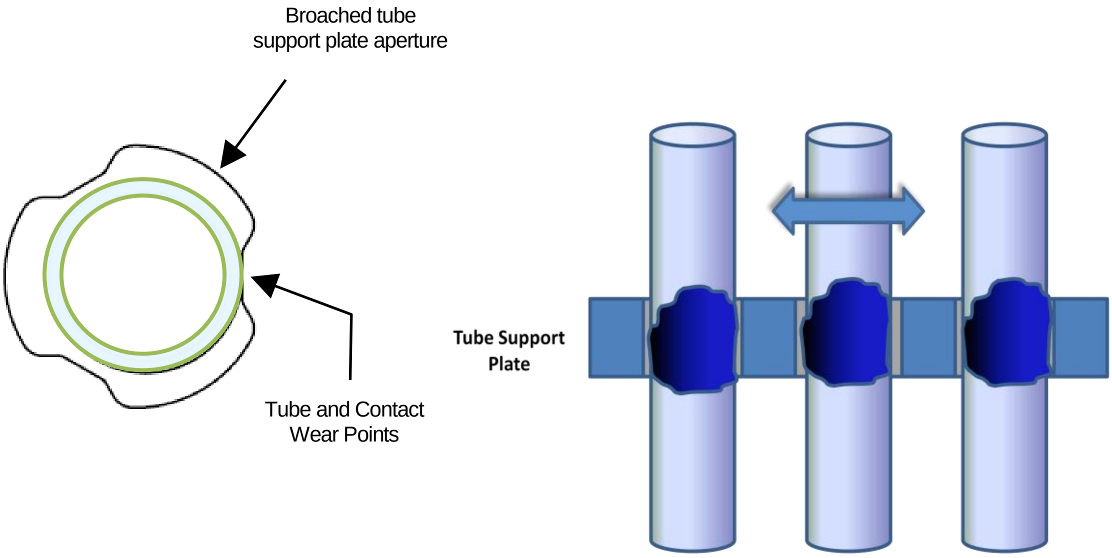


FIGURE 3A RSG COMPONENT LOCATIONS

source: f[17]

4A

Typical locations of FIGURES 4



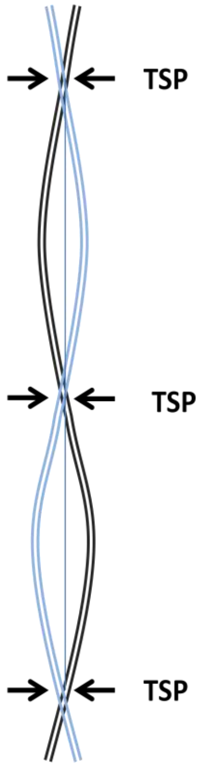


FIGURE 4A TUBE SUPPORT PLATE-TO-TUBE WEAR

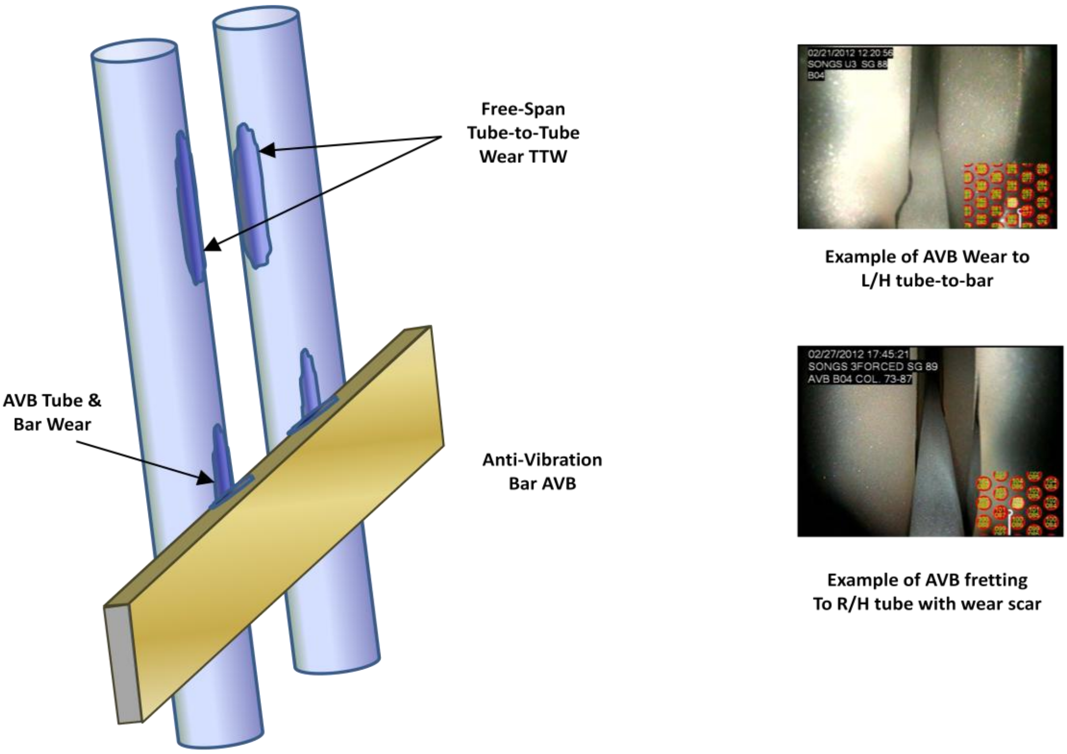


FIGURE 4B AVB WEAR & AVB-TO-TUBE WEAR

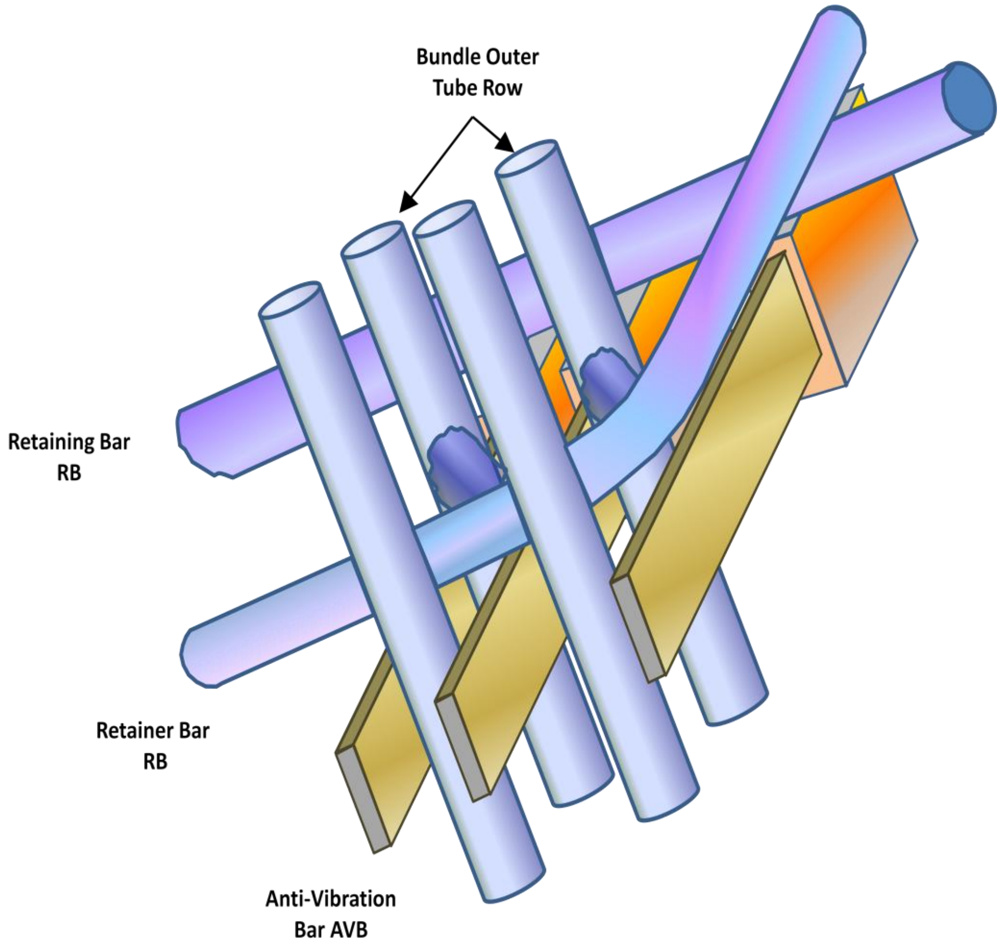


FIGURE 4C AVB ASSEMBLY RETAINER BAR-TO-TUBE WEAR

RSG TUBE BUNDLE – U-BEND REGION

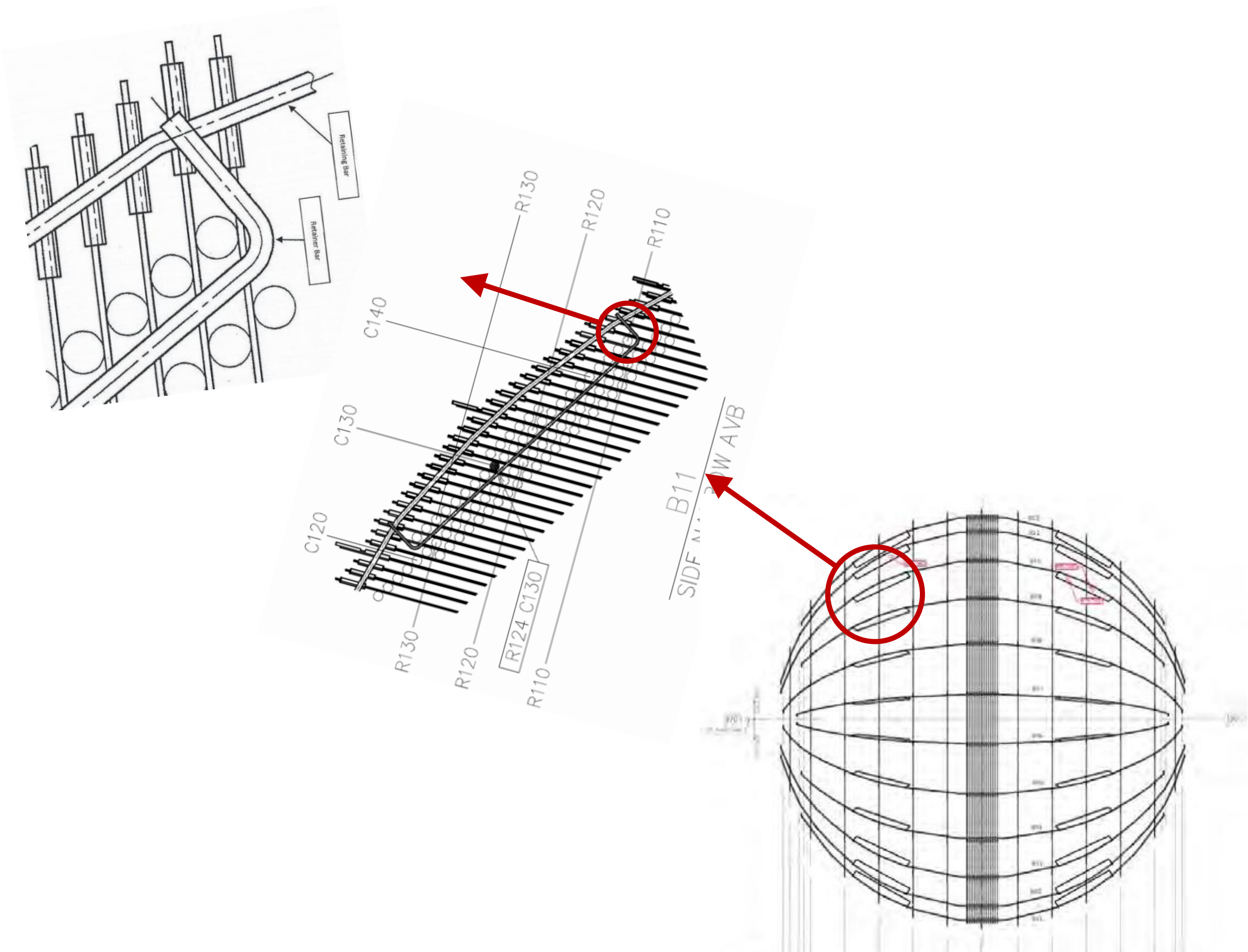


FIGURE 5 ANTI-VIBRATION AND RESTRAINT BAR ASSEMBLY

Source f7

(Name)

Subscribed to and affirmed before me this 10th day of January, 2013

Notary Public < *personal information redacted*

. . .>

My commission expires: