ATTACHMENT 2

Declaration of John Large

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE NRC STAFF

In the Matter of	
SOUTHERN CALIFORNIA EDISON COMPANY	Docket ID NRC-2013-0070
(San Onofre Nuclear Generating Station, Units 2 and 3)	May 16, 2013

A REVIEW OF THE NRC[1] PROPOSED DETERMINATION OF NO SIGNIFICANT HAZARD CONSIDERATION FOR SOUTHERN CALIFORNIA EDISON'S PROPOSED LICENSE AMENDMENT FOR SAN ONOFRE UNIT 2

DECLARATION OF JOHN LARGE

I, John Large, being duly sworn, state:

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, 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

1 Whereas I acknowledge that this declaration relates to NRC's finding on the no significant hazard consideration, I have channelled my comments through SCE because the request for the license amendment derives from SCE and the NRC proffers no statement of its own position.

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.

2 EXPERIENCE OF THE SAN ONOFRE NUCLEAR GENERATING STATION

2.1 I have previously prepared and submitted evidence in the matter of the San Onofre Nuclear Generating Station (SONGS) to the Nuclear Regulatory Commission (NRC) Atomic Safety Licensing Board (ASLB).

2.2 In my 1st Affidavit (January 2013) to the ASLB I provided opinion on the failings of the SONGS replacement steam generator (RSG) design, how this gave rise to unrestrained tube motion and excessive tube wear, and on the uncertainties of restarting Unit 2 at the proposed maximum limit of 70% rated thermal power (RTP). In my 2nd Affidavit (February 2013) to the ASLB I examined the RSG steamside thermal-hydraulic flow regime and how this determined the types and rates of tube and tube restraint component wear, particularly at the Southern California Edison (SCE) proposal to operate Unit 2 at 70% rated thermal power (RTP).

2.3 I have also prepared and submitted opinion (March 2013) to the NRC Petition Review Board in which I review the involvement of SCE and Mitsubishi Heavy Industries (MHI) in the specification and design of the RSGs.[2]

3 LICENSE AMENDMENT REQUEST AND NO SIGNIFICANT HAZARD CONSIDERATION (NSHC)[3]

3.1 SCE has submitted a license amendment request for a temporary change to the steam generator management program and license condition for maximum power, both being integral parts of the OL *Technical Specification* (TS). In short, the amendment applies for the duration of the fuel cycle (Cycle 17) in that power operation would be restricted to up to 70% rated thermal power rating (RTP) and that a tube inspection would be undertaken at 150 days of operation into Cycle 17. Other than the power reduction and tube inspection period, no other physical

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2 *Tube Wear Identified in the San Onofre Replacement Steam Generators Mitsubishi Reports UES-20120254 Rev.0 (3/64) and L5-04ga588(0) Together with Other Relevant Information*, March 2013 - this supplementary report was placed before the Petition Review Board of the NRC as part of the §2.206 process.

3 NSHC is required under 10CFR §50.91 and §50.92 and the *Regulatory Issue Summary* (RSI) *Attributes of a Proposed No Significant Hazards Consideration Determination* (March 29 2012) provides the public an opportunity to comment or request a hearing on the proposed amendment request via the published NRC Notice of April 16 2013 *Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generating Station, Unit 2*.

changes to the operation and/or detailed installation of the components of the plant were proposed.

3.2 A *No Significant Hazard Consideration* (NSHC) determination requires that operation of the facility in accordance with the proposed amendment would not

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

3.4 (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

3.5 (3) involve a significant reduction in a margin of safety.

4 EXTANT CONDITION OF THE UNIT 2 STEAM GENERATOR TUBING

4.1 My 1st Affidavit to the ASLB describes the processes that gave rise to the extensive degradation of the tubes in each of the two RSGs serving SONGS Unit 2 and, similarly, Unit 3 – the events, processes and the extent of tube wear degradation have been extensively reported by a number of sources.[4]

4.2 A summary of the Units 2 and 3 RSG tube wear is given in APPENDIX I of my 1st Affidavit being a true reproduction of Table 6-1 of the SCE response[5] to the CAL of October 2012 – detailed inspection data for Unit 2 tube wear is provided by the SCE Special Report.[6]

4.3 The OL TS sets out criteria stipulating the condition in which individual tubes have to be withdrawn from pressurized service by plugging.[7] Essentially, these tubes are:

4.3.1 a) First, in those instances where the tube wall thinning is equal to or greater than 35% of the original tube wall thickness (*TW Depth* - Column 1 of Table 6-1) the tube has to be withdrawn from service (by plugging).

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4 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, also the SCE, Enclosure 2, *Songs Return to Service Report*, October 3 2012 and 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 SCE, Enclosure 2, *SONGS Return to Service Report*, October 3, 2012

6 SCE *Special Report, Inspection of Steam Generator Tubes, Cycle 17, San Onofre Nuclear Generating Station, Unit 2,* Docket No 50-361, 10 CFR 50.4, April 10, 2013

7 Localised repair of degraded tubes by sleeving is not permitted by the TS.

4.3.2 b) Second, the TS requires that the operational assessment, made at the time of the tube inspection, has to provide assurance that no individual tube will wear beyond the 35% *TW Depth* limit in the following fuel cycle or until the next tube inspection. This means that the OA has to project the tube wear rate(s) forward over the next in-service cycle – for Unit 2 this forward projection applies to a period of 150 days into Cycle 17 being the tube inspection interval nominated by SCE as a condition of the restart.

4.3.3 c) In addition to those tubes that have experienced or are likely to develop excess levels of wall thinning (=>35%), SCE has also chosen to preventatively plug zones of tubes to reduce the risk and incidence of tube wear.

4.4 These three groups of plugged tubes make up the numbers of *defective*[17] tubes of the tube bundles of the two RSGs serving Unit 2. Those tubes that have sustained some degree of wear to a depth of less than 35% and which remain in pressurized service are referred to as *degraded*[17] tubes.

5 ADEQUACY OF THE OPERATING LICENSE TECHNICAL SPECIFICATION

5.1 The TS tube integrity criteria relate tube resilience (integrity) to the remaining cross sectional (wall) area of then thinned tube section and the radial stress active within the wall. The radial or membrane stress derives directly (and solely) from the pressure differential between the reactor circuit and lower pressure of the steamside of the RSG. The resilience of a degraded tube is evaluated against *allowable stress limits* (membrane plus some in-plane bending stress) by multiplying the stresses for a non-degraded tube by the ratio of the corresponding sectional properties (thinned) of the degraded tube.[8]

5.2 In my opinion this approach, adopted for the TS tube integrity criteria, is overly simplistic in that it provides little account for anomalies (aging, chemical deterioration, etc) of tube material and/or physical degradation of the tube geometry (in this case surface imperfections and flaws).

5.3 In effect, the *allowable stress limits* based on a pressure bursting failure mode provide, so it is assumed, sufficient *margin* to accommodate all other (undefined) processes and conditions that

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8 For the degraded tube case evaluation the minimum tube wall thickness required to meet the structural requirements of UNSCR R.G.1.121 is calculated by considering (1) wall thickness loss over the entire tube length, (2) wall thickness loss at the tube intersections with tube support plates (TSPs), and (3) wall thickness loss at the tube intersections with the anti-vibration bars (AVBs) in the tube bundle U-bend region and the minimum wall thickness is calculated for: (a) the fault condition, and (b) the normal operating condition. The more limiting of these two loading conditions determines the minimum allowable tube wall thickness for the tube not to burst under the conditions specified in R.G. 1.121.

could contribute to and/or accelerate failure of the tube – this is the basis of the TS criteria underwriting tube integrity.

5.4 In other words, the underlying premise is that at 35% thinning (for whatever reason and by whichever means), the tube remains a sound structure, there being sufficient margin in hand to safeguard against all other circumstances and conditions that might quite independently progress to tube failure. Moreover, the failure mechanisms of such other *'independent'* factors, for example plain fatigue cracking of a vibrating tube, might themselves be enhanced by the nature of the tube wear, either by the presence of surface flaws, abrasions, notches and/or areas of work hardening.[9]

5.5 Other than the *margin*, the TS tube integrity criteria do not provide for quantitative cross linkage of possible separate failure mechanisms to the condition of the tube surface as generated by the degradation processes (tube rubbing, abrasion and impacting) experienced in the San Onofre RSGs.

5.6 For example, the 35% tube wall thinning threshold at or over which individual tubes should be plugged and withdrawn from pressurized service, must be drawn from operational and bench-testing experience of past tube failures. This is because the 35% threshold must provide a satisfactory margin to cover metallurgical and physical geometry features that serve to trigger various failure modes, such as stress corrosion cracking,[10] mechanical damage, wastage (thinning), denting, and vibration induced cyclic plain fatigue cracking.

5.7 However, the San Onofre tube degradation is acknowledged to be unique so, it follows, that the nature of the tube degradation scars are also likely to include unique features that are not replicated in the data bank of past tube failures at other nuclear plants and from bench-testing trials. If so, the use of the present single-failure mode TS tube integrity criteria (that relies upon

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9 The impact behavior and fracture response of Inconel 690 has not attracted that much research, although indications are that impacting results in the formation of localized shear bands which can prompt catastrophic failure - see Woei-Shyan Lee and Tai-Nong Sun, *Plastic Flow Behaviour of Inconel 690 Super Alloy Under Compressive Impact Loading*, Materials Trans, Vol. 45, No. 7, 2004

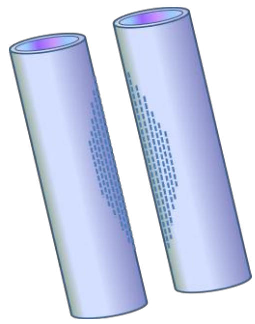
10 The are number of such modes of failure including, but the TS is heavily biased towards stress corrosion cracking which had become by the 1990s the principal degradation mechanism for SG tubing worldwide. For example outside diameter stress corrosion cracking (ODSCC) where the probability of failure is determined from proprietary coefficients obtained by bench-testing – for example, EPRI suggest the failure function for ODSCC to be $\Delta p_f(a) = A + B.\log10(a) + \mathcal{E}$ where A and B are proprietary coefficients and $\Delta p_f(a)$ is the burst pressure for a given flaw of *a* dimension. A common locality for ODSCC is at the TSP where debris, comprising corrosion sludge fills the TSP aperture providing conditions conducive to dry-out and adverse cation/anion being an accelerant to inter-granular SCC and crack linkage. Similar ODSCC is known to occur at dented TSP locations. Alloy 690 tube material is generally more (about 10 times) resistant to SCC than the earlier Alloy 600.

past experience of tube failures at other nuclear plants, etc) at San Onofre is inappropriate and introduced uncertainty.[11,12]

6 NATURE OF THE TUBE DEGRADATION AT SAN ONOFRE

6.1 In its reporting[13] to SCE, the manufacturers of the RSGs, Mitsubishi Heavy Industries (MHI), describes distinctly different patterns of wear at the TTW, AVB and TSP locations. The *in-plane* direction of the tube vibration, particularly at the AVB-to-tube fretting localities, is generally acknowledged to be unique to the San Onofre RSGs in that this mode of degradation has not been experienced at other US nuclear power plants wherein tube wear is dominated by *out-of-plane* motion.

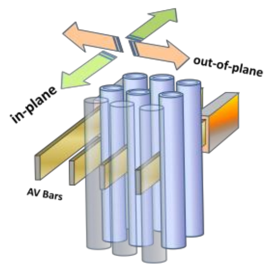
6.2 These variations in types of tube wear are described by MHI.[13]



6.2.1 **TTW**: This wear pattern occurs on the free span portion of the tubes (between the remaining effective AVB restraint points) in the U-bend region of the tube bundle. TTW produces long scars running in the axial direction of the tube as a result of continuous contact fretting or clashing of impacting tubes.

6.2.2 MHI conclude that the tube *in-plane* motion giving rise to TTW is caused either by random vibration and/or fluid elastic instability (FEI), favoring the latter on the basis that the amplitude of random vibration is small.

6.2.3 It is also possible that *out-of-plane* TTW occurs and that this lower frequency vibration mode is excited by low frequency flow induced forces from, for example, vortex shedding in the wake of tubes or groups of tubes. In this mode, even if the *in-* and *out-of-plane* FEI is suppressed in the 70% RTP restart of Unit 2, the tubes will remain vulnerable to excitation by flow induced fluid forces.



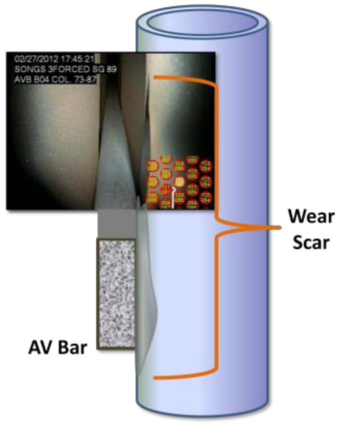
11 The *EPRI Steam Generator Examination Guidelines Revision 5* state that flaws in qualification data sets should produce signals similar to those observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise level.

12 For example, MHI describe a *zigzag* pattern wear scar – see ¶6.3.1 – which might provide a stress raiser in the axial tube direction in which the pressure membrane stress acts, although insufficient description of this type of wear scar is publicly available.

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

6.3 **AVB:** Wear at the tube-to-AVB contact points wear occurs in three distinctive patterns:

6.3.1 **In-Plane:** To generate this pattern of wear at the AVB the tube (shown right) moves relative to the AVB in the *in-plane* direction (up-and down). The resulting wear scar sits across the AV bar depth indicating relatively large *in-plane* motion amplitude.



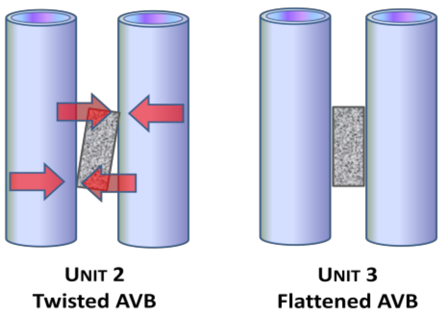
Adapted from MHI

6.3.2 This wear arises because the *'zero tube-to-bar gap* and *zero-preload* design functionality of the AVB provides no tube restraint the *in-plane* direction leaving the tube free to respond and slide (up and down) across the AV bar contact surface.

6.3.3 This pattern of wear scar is much longer than the typical case adopted in the *Updated Final Safety Analysis Report* (UFSAR)[14] for which the scar length is assumed to equal the AVB-to-tube contact length (ie the AV bar cross-section depth). In this case the UFSAR is overly optimistic in determining the permissible tube wall wear depth because it is generally accepted that tubes with shorter wear scar lengths exhibit higher burst pressures.

6.3.4 A variation of this *in-plane* motion is a *zigzag* or *saw-tooth* surface wear pattern suggesting a combination of *in-* and *out-of-plane* tube motion. This pattern of tube wear produces a distinctive line flaw orientated in the axis of the tube thereby presenting a weakness in the tensile direction of the principal tube stress arising from the pressure differential.

6.3.5 **AVB Dig In:** In this pattern of wear the misaligned or twisted AVB digs-in to the tube surface – the pattern is probably unique to Unit 2 because the Unit 3 AVBs were more effectively flattened by a modified manufacturing process – see ¶7.12.



Adapted from WEC

6.3.6 The resulting wear scar is a sharp notch or *'stress raiser'* in the surface of the tube.

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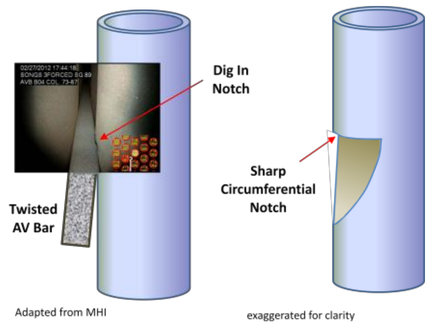
14 *San Onofre Nuclear Generating Station Unit 2 & 3 Updated Final Safety Analysis Report* Revised April 2011 – see Table 15.10.6.3.2-4 for the transient analysis summary results for a steam generator tube rupture.

6.3.7 For failure analysis, account of this *stress raiser* is taken by assuming a *stress concentration factor* (k_t) determined by the dimensional geometry of the notch. Since the detection of this wear is blind from within the tube it is impracticable to determine the sharpness and depth of the notch so, it follows, the appropriate value of k_t cannot be chosen with absolute certainty.[15]



6.3.8 A variation of this wear pattern is where there is both *in-* and *out-of-plane* movement of the tube to produce the *zigzag* pattern described earlier (¶6.3.4), similarly producing an axial flaw that presents to the radial tensile stress in the pressurized tube wall.

6.3.9 **In-Plane AVB and Tube Wear:** This wear pattern is where both tube and AVB bar have both worn simultaneously or when the wear between tube and AV bar cannot be distinguished because visual access to the wear interface is not possible, although the left-hand tube in the example shown right, the tube motion has worn substantially through the width of the AV bar (~40%).

6.4 All modes of AVB-to-tube wear are provoked by fluid flow random vibration (ie turbulence) and, thus, the AVB-to-tube contact locations remain vulnerable induced excitation and wear even if the 70% RTP eliminates FEI.[16]

6.5 These two latter wear patterns (¶6.3.6 and ¶6.3.9 - as shown by photographs reproduced from the MHI inspection)[13] highlight the difficulty of accessing the wear scars to determine the extent of surface damage. Much the same applies to the locations of the TTW, where the close proximity of adjacent tubes practicably limits access for visual inspection.

6.6 In other words, although the eddy-current (ET) in-service through-wall inspection results provide a generally reliable measure of overall tube wall thinning, the assessment of the nature of individual incidences of tube surface damage (imperfections, etc) is uncertain.[17]

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15 For example the range of *stress concentration factor* kt is given in Chart 3.5 of Walter D. Pilkey, *Peterson's Stress Concentration Factors* Second Edition, John Wiley, Sons, Inc., 1997 for a tube under axial tension (but not pressurized).

16 Although not discussed here, I consider it likely that the TSP-to-tube wear is also driven by random fluid processes – this locality of tube wear is an important factor in considering the potential for the tubes that are effectively pinned at the top TSP but with successive AVB-to-tube restraint not active, to fail by high cycle fatigue.

6.7 So far I have considered the surface changes brought about by, for want of a better description, *'gouging'* of two adjacent parts (eg tube-to-tube, AVB-to-tube and TSP-to-tube) to form distinctive scars or stress raisers in the tube outer surface. Under tensile loading, deriving either from internal pressure, bending or plain cyclic fatigue, the stress concentration can develop cracking resulting in early tube failure from ductile tearing or brittle fracture.

6.8 Thus the presence of surface flaws produced in TTW, AVB- and TSP-to-tube wear may bring forward tube failure before that predicted by the TS tube burst criterion (3xΔP and 1.4xΔP for SIPC and AILPC cases respectively). Since the TS does not specifically refer to this and other types of tube failure, it vital to maintain the 3xΔP and 1.4xΔP margins to cover such contingencies.

6.9 There is another strength of materials phenomenon, referred to as fretting fatigue, occurring at the contacting and sliding surfaces of two adjacent parts under load and subject to slight relative movement by vibration or some other force. At very low stress levels and often after only a few thousands of cycles, fretting fatigue may initiate micro cracking in the rubbing surfaces that then become available to propagate into ductile/brittle failure zones (as in ¶6.7).

6.10 In plain fatigue (without fretting) the initiation and development of small cracks typically represents upwards of 80 to 90% of the total component life, but with a fatigue fretting contribution the plain fatigue strength or endurance limit can be reduced by as much as 50 to 70% during subsequent (or simultaneous) cyclic loading of the tube overall.[18,19]

6.11 As I previously noted in ¶6.6 through-wall ET may not have sufficient resolution to detect fine micro cracking between surfaces in contact (ie the tube and TSP or AVB or another tube) and, if so, the presence of established fretting fatigue may have passed unnoticed.

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17 The ET inspection system must detect tube wall internal and surface flaws at an acceptable level of detection reliability and it must also size the significant flaws. For the San Onofre degraded tubes ET must have acceptable reliability to detect and size flaws which are not necessarily significant but which might require action to mitigate further tube damage. This grading of flaws determines whether the tube is *degraded* but fit for continued pressurized service or *defective* because it contains a flaw of such severity that it is unacceptable for continued pressurized service until the next tube inspection outage.

18 ASM Handbook V19, *Fatigue and Fracture*, ASM International

19 Plain fatigue is where there is no direct contact, say where the pipe vibrates in a free span situation. In fretting fatigue there is contact between two slightly moving parts – the contact point stress gradients are likely to be very high due to the localised stress concentration at the contact and the magnitude of these stress gradients is usually much higher than those associated with typical design features of components, such as notches and holes. Also, loading is likely to be non-proportional in the neighbourhood of the contact with this feature caused by the non-linear nature of the friction at the contact interface. Localised surface damage at the asperity level may play a role in accelerating the initiation of cracks at the asperity scale.

6.12 Moreover, because surface imperfections contribute strongly to premature failure (ie failure within the TS allowable stress safety margin), it is absolutely essential that the wear surface flaws (either deriving from gouging, fretting fatigue, or just plain fatigue) be fully understood and taken into account when projecting the tube integrity for the Cycle 17 in-service period..

7 PROJECTING FURTHER TUBE DEGRADATION INTO CYCLE 17

7.1 Whereas the SCE license amendment request specifically applies to Unit 2 it is, nevertheless, important to consider the extent and nature of the tube wear degradation in the identical Unit 3.

7.2 This is because the tube bundle damage in both RSGs serving Unit 3 is universally acknowledged to be so severe and extensive that any level of return to powered operation of this nuclear plant would introduce further risk and lack assurance that Unit 3 could operate safely. This situation is confirmed by the requirement of the *Confirmatory Action Letter* (CAL)[20] issued by the NRC following the RSG tube failure that provoked the forced shutdown of Unit 3 in January 2012.

7.3 The CAL specifically requires SCE to undertake a number of *Actions* relating to any intention to restart Unit 3, including

7.4 *". . . 7. Prior to entry of Unit 3 into Mode 4, SCE will submit to the NRC in writing the results of your assessment of Unit 3 steam generators, the protocol of inspections and/or operational limits, including schedule dates for a mid-cycle shutdown for further inspections, and the basis for SCE's conclusion that there is a* ***reasonable assurance****, as required by NRC regulations, that the* ***unit will operate safely****."*

my added ***emphasis***

7.5 On its part SCE has chosen not to respond to this *Action 7* (and also four other CAL *Actions*) and no preparations have been made to restart Unit 3.[21] In my opinion, this failure of SCE to respond to *Action 7* is tacit acknowledgement that it is not possible to provide a *'reasonable assurance'* that Unit 3 with the present level of tube degradation will *'operate safely'.*

7.6 This brings me to consider Unit 2 which SCE proposes to restart, subject to the license amendment request being accepted and ratified by the NRC. The CAL also required SCE to undertake specific actions, these being:

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20 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.

21 In fact, the nuclear fuel of the reactor core of Unit 3 has been removed and placed in water pool storage.

7.7 "*. . . 1. Southern California Edison Company (SCE) will* ***determine the causes of the tube-to-tube interactions*** *that resulted in steam generator tube wear in Unit 3, and will implement* ***actions to prevent loss of integrity due to these causes*** *in the Unit 2 steam generator tubes. SCE will establish a protocol of inspections and/or operational limits for Unit 2, including plans for a mid-cycle shutdown for further inspections.*

2. Prior to entry of Unit 2 into Mode 2, SCE will submit to the NRC in writing the results of your assessment of Unit 2 steam generators, the protocol of inspections and/or operational limits, including schedule dates for a mid-cycle shutdown for further inspections, and the basis for SCE's conclusion that there is ***reasonable assurance,*** *as required by NRC regulations, that the* ***unit will operate safely****."*

my added ***emphasis***

7.8 In responding to the CAL, SCE presented a series of *Operational Assessments* (OAs) that it claimed justified restarting and operating Unit 2 for a trial period. The OAs of interest here are those by AREVA[22] and, independently, the latest revision of the OA by Intertek APTECH.[23]

7.9 First, it is of interest to note that each OA fails to *'determine the causes of the tube-to-tube interactions'* as stipulated by the CAL. This is because all of the OAs (including a third OA by Westinghouse)[24] skirt round and stop short of identifying the *root* cause, delving no further into the design features, peculiarities and processes of the RSGs that give rise, so it seems uniquely at San Onofre, to the tube motion *in-plane* excitation forces.[25]

7.10 However, both OAs recognized that

7.10.1 i) the tube wear degradation experienced in Unit 2 was less advanced than the tube wear in Unit 3; although, that said

7.10.2 ii) the wear locations (AVB and TSP) and number of incidences present in Unit 2 were very similar to those present in Unit 3, although Unit 3 had, in addition, a much greater number of TTW incidences; and that

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22 SCE, Attachment 6 – Appendix B: SONGS U2C17 - *Steam Generator Operational Assessment for Tube-to-Tube Wear*, AREVA

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

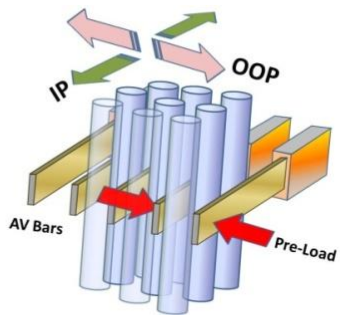
24 *Attachment 6 – Appendix D: Operational Assessment of Wear Indications In the U-bend Region of San Onofre Unit 2 Replacement Steam Generators*, Westinghouse Rev 3, October 2012.

25 In fact, as I discuss in some detail in my 1st ASLB affidavit, there is disagreement between the various OA consultants as to whether the tube motion excitation forces derive from random fluid processes (turbulence, downstream wake, etc.) and/or fluid elastic instability.

7.10.3 iii) the wear processes involved in these virtually identical RSGs related to the effectiveness of restraint provided by the anti-vibration bar (AVB) assemblies in the U-bend region of the tube bundle.

7.11 It follows that the tube wear mechanism is a two-stage process whereby, first, the AVB contact with the individual tubes is worn away by vibration of individual tubes excited into *in-plane* motion by local fluid flow forces. Second, loss of the AVB restraint, and successive points of AVB restraints, results in a lengthening of the unsupported or free-span sections of tube to the extent, again by fluid forces, that the tube vibrates at low frequency and at sufficient amplitude to enable tube-to-tube clashing and, hence, accelerated tube-to-tube wear (TTW)[26] to occur.

7.12 The TTW in the Unit 2 RSGs was less advanced than that of Unit 3 because of the omission in the manufacturing process of the Unit 2 AVB components that, quite fortuitously, resulted in the presence of an unintentional clamping or preload force across the individual tubes – this preload force delayed the loss of the *in-plane* (IP) AVB restraint and the onset of second phase TTW process in Unit 2.[27]



7.13 In other words, the tube wear extant in Unit 2 is representative of the first stage of the two stage mechanism that leads to accelerated TTW.

7.14 The disadvantages arising from the installation of the distorted (twisted) AV bars include i) notching and the formation of a stress raiser as shown at ¶6.3.7, and ii) the opportunity for fretting fatigue in the locality of the notch of i), generated whilst the AVB restraint was active with the preload force. Both of these features could contribute to localized ductile/brittle failure driven either by internal pressure and/or exceeding the modified plain fatigue cyclic loading endurance limit.

7.15 SCE's proposed new operating regime for Unit 2 at 70% RTP claims that the 1st and 2nd stages of wear will be slowed but not totally eliminated.

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26 For a fuller description of the tube wear processes see my 1st Affidavit to ASLB.

27 The intended design function of the AVBs was to provide a *'zero-gap-zero-contact-force'*, that is no preload, across the individual tubes thereby offering no effective restraint in the *in-plane* direction. Unit 3 achieved this design functionality following modifications in the manufacturing of the AVBs, whereas the unmodified AVBs of Unit 2 remained distorted (twisted) so much so that certain of the AVB locations applied a tube clamping preload force.

7.16 In fact, both AREVA and Intertek OAs agree that even at the 70% RTP reduction, the 1st stage process (AVB restraint loosening) will progress to the threshold at which the 2nd stage TTW commences in earnest, thereafter putting individual tubes at risk of structural failure (bursting) as a result of tube wall thinning.

7.17 Intertek's projection for rate of TTW for the proposed Unit 2 restart (Cycle 17) is summarized by the following graphic:[23]

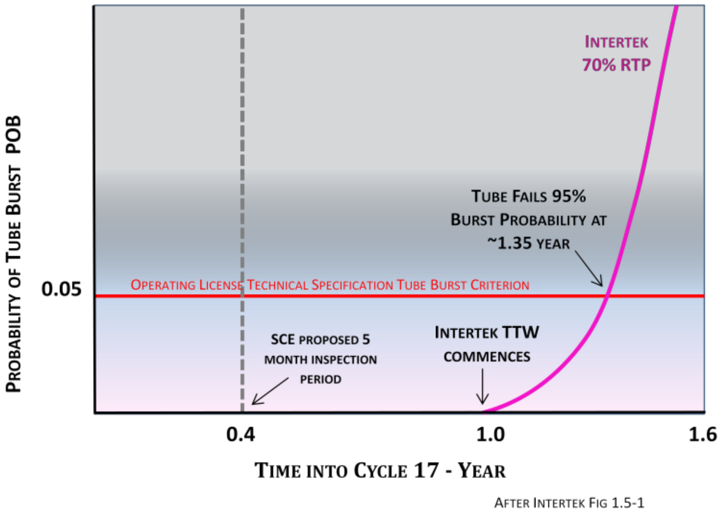

FIGURE 1 INTERTEK PROJECTION OF CYCLE 17 TUBE DEGRADATION

7.18 The 1st stage AVB-to-tube restraint loosening proceeds from the time of restart over a period of 12 months at which time a group of successive AVBs have little or no preload force in the *in-plane* direction.

7.19 From this point forward the 2nd stage wear process (TTW) commences following the characteristic (—).

7.20 The TTW wear rate is such that the tube passes the 95% tube burst criterion (—)[28] at about 0.35 years thereafter – this means that the tube is considered to have failed at about 1.35 years into Cycle 17.

7.21 SCE's argument is that the proposed Unit 2 shut down and RSG tube inspection at 5 months (¦) safeguards against tube failure by providing a sufficient time buffer ahead of the Intertek 1.35 year (16 months) period to exceed the 95% threshold.

7.22 However, Intertek's projections do not compare at all favorably with the timings evaluated by the AREVA OA[29] because:

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28 The 95% Probability, 50% Confidence criterion for an individual tube burst is specified in the Operating License Technical Specification for Unit 2 of the San Onofre nuclear plant.

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

7.23 for AREVA (shown thus), the AVB-to-tube restraint loosening period until the 2nd stage TTW commences is about is ~0.3 year compared to 1 year by Intertek; and similarly

7.24 the equivalent total time to tube burst (95% probability) projected by AREVA is in the range of 0.5 to 1.5 year compared to ~1.35 year Intertek.

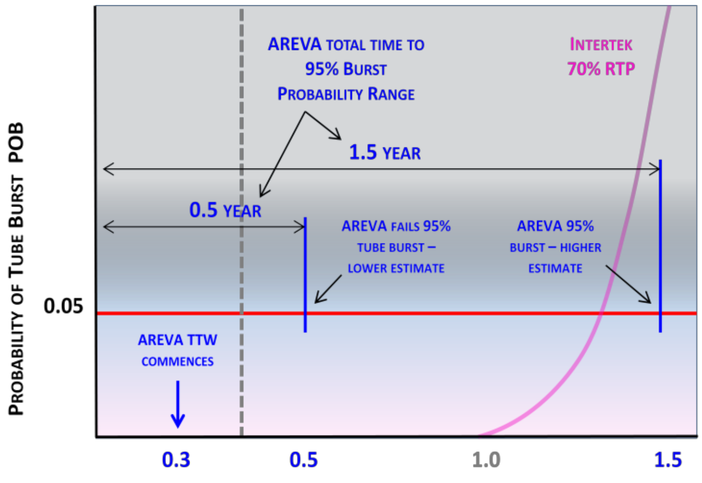

FIGURE 2 AREVA PROJECTION OF CYCLE 17 TUBE DEGRADATION

7.25 **In summary:** There are two important issues raised by the AREVA and Intertek OAs, these are, first, both of these independent analyses conclude that even at the reduced 70% RTP level of operation, the AVB-to-tube wear[30] and TTW will continue to be present in the RSG tube bundles of Unit 2; and second, each OA arrives at significantly different time projections for individual tubes in the tube bundles of the RSGs to reach and exceed the TS tube burst failure criterion.

7.26 On the second issue, it is the broad range of the projected periods for the start of TTW that reflects, for both OAs, the uncertainty about just how far along the path to complete loss of AVB *in-plane* restraint Unit 2 was at the end of Cycle 16. The difference of 0.3 (AREVA) and 1 year (Intertek) is such a disparity that, equally it might be reasoned, TTW could have commenced at some time towards the end of Cycle 16. If it had, then the periods for TTW to exceed the 95% tube burst probability could be shorter that SCE's 0.4 year (150 days) period over which the tubes would remain acceptably *degraded* and not *defective*.[17]

7.27 Also, it is established that a significant number of tubes in each of the two RSGs serving Unit 2 are *defective*; some of these tubes have exceeded the wall thinning limit and have been withdrawn from pressurized service by plugging; other tubes that are at risk of exceeding the wall thinning limit during the projected Cycle 17 period[31] have or are also to be withdrawn from pressurized service by plugging.

30 The other modes of tube wear shown in Table 6-1 of my 1st Affidavit to the ASLB are also likely to continue, these include wear at the tube support plates (TSP) and at certain of the retainer bars (RB), although at the vulnerable RB localities all of the tubes have now been plugged.

31 The Operating License Technical Specification (3.4-51 – 3.4.17 and Action A.1) requires verification of steam generator tube integrity shall be maintained until the next refuelling outage or steam generator tube inspection.

7.28 Undermining SCE's confidence that it is able to reliably schedule a tube inspection frequency (ie 150 days) is that SCE's OA consultants have each failed to identify the *root* cause that leads to tube wear degradation.[32,33] In the absence of this *root* cause understanding, any prediction of the tube wear rates and, from this, the operational time throughout which the tube will continue to be in a *degraded* state and not *defective*[16] condition, is uncertain and non-compliant with the TS tube integrity criterion that the tubes shall remain serviceable (*degraded* and not *defective*) through until the next inspection outage.[31] Such uncertainty casts considerable doubt over the reliability of SCE's *no significant hazard consideration*.

7.29 **In summary:** This knowledge of the extant condition of the Unit 2 RSG tube bundles, together with the OA projections (and uncertainties and differences in these) of the future wear rates at the proposed 70% RTP level, enable me to consider the no significant hazard consideration.

8 NO SIGNIFICANT HAZARD CONSIDERATION (NSHC)

8.1 Previously I have considered the 10CFR §50.92 criteria requirement that the proposed OL amendment would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

8.2 I should note here that the 'hazard' of the NSHC is a composite of the risk or probability of radiological release coupled to the ensuring radiation dose exposure to the most critical individual - the risk of release must be acceptable and the radiological consequences tolerable.

8.3 A SG tube rupture (SGTR) incident is a penetration of the barrier between the reactor cooling circuit (RCS) and the main steam system (steamside). The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube would allow for

32 The OAs generally commence investigation and analysis of the tube wear on the basis that fluid elastic instability (FEI) is active in the U-bend region of the tube bundle but there is little comment on why FEI is present and, particularly, why the flow regime results in the *in-plane* activity.

33 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, 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, or 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 the assertion of neither party is wholly or partly correct.

the transfer of radioactive reactor coolant into the steamside. Radioactivity contained in the reactor coolant would mix with water in the shel1 side of the affected steam generator and then transported by steam to the turbine and then to the condenser or directly to the condenser via the steam dump bypass system. Non-condensable radioactive gases in the condenser would be removed by the main condenser evacuation system and discharged to the plant vent stack and, hence, a radiological discharge to the environment.

8.4 A single tube SGTR is classified as a limiting fault. The radiological consequences for this design basis case (both with and without a pre-existing or accident induced iodine spike) is given in the UFSAR wherein the predicted radiological doses are compared to the NRC Standard Review Plan Acceptance Criteria.

8.5 1) **No Significant Increase in Probability/Consequence - Previously Evaluated Accident**

8.5.1 The previously identified accident[34] of immediate concern here is the failure of a single tube in the steam generator tube bundle thereby permitting the radioactive primary circuit to bypass the nuclear island containment via the normally isolated the steamside (turbine) circuit.[35]

8.5.2 Maintaining integrity of the barrier between the reactor coolant circuit (RCS) and the RSG steamside is significant from a radiological standpoint since a leaking steam generator tube would result in migration of radioactive reactor coolant into the steamside.[36]

8.5.3 The driving force, so to speak, for single tube failure is the differential pressure acting across the tube wall at the operating temperature. Operating at the proposed 70% RTP will not result in any significant change in the tube differential pressure and the peak tube wall temperature, so

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34 Here *'accident'* refers to postulated design basis accidents, including the internal and external events with which the plant must be able to cope (e.g., earthquake, flooding, turbine missiles, and fire) as described in the updated final safety analysis report (UFSAR).

35 See Operating License Technical Specification Docket N° 50-361, NPF-10 *San Onofre Improved Technical Specification 50-361 - Bypass* is defined (1.1-4 a.3) as the reactor coolant system (RCS) LEAKAGE through a steam generator to the secondary system (primary to secondary LEAKAGE) – *tube failure* is covered by the Steam Generator Program of the TS and relates to maintaining tube integrity (5.0-13/14 – 5.5.2.11) for both normal steady state full power (for which a revision to 70% RTP is required) and in account of additional loading conditions induced during the occurrence of design basis accidents and combination of accidents, with tube integrity defined in terms of pressure differentials, leakage rates, and tube wall flaw and/or thinning depth at or exceeding 35% of the nominal wall thickness, with tube integrity assured until the next SG tube inspection (5.0-15 – 5.5.2.11d).

36 Radioactivity of the reactor coolant would mix with steamside water in the affected steam generator and, during normal plant operations, some of this radioactivity would be transported through the turbine to the condenser where the radioactive materials would be released via the condenser air ejectors. Since the nuclear plant continues to operate for 15 or so minutes before the increasing levels of radioactivity initiates a trip, there arises an accumulation of radioactivity that is then dumped unfiltered to atmosphere by lifting of the steam generator safety valves.

the tubes will be subject to the much same forces (radial stress) and tube material strength response (ie the yield stress weighted in account of temperature) as experienced at 100% RTP.[37]

8.5.4 Similarly, the tube bundle gross displacement (so called *'flowering'*) will induce some elements of bending and shear stresses but these also would not be expected to radically depart from conditions at 100% RTP operation.

8.5.5 In other words, reducing the power level for 100% to 70% RTP will not reduce the pressure and temperature environment acting within and across the tube wall. In this important respect, the forces driving tube failure do not change (ie are not reduced) and remain essentially independent of the proposed reduction in RTP.

8.5.6 On the other side of the tube integrity equation is the nature and extent of the tube wear degradation at the TTW, AVB- and TSP-to-tube locations. I have described the uncertainties relating to the extent and nature of the tube wear (so far as it is available to me)[38] at the AVB-to-tube locations in SECTION 6 – I would expect similar uncertainties to apply to the TSP and TTW locations.

8.5.7 I should also note here that reducing to 70% RTP will not necessarily result in a lessening of the radiological consequences in the event of a failure of the fission product boundary represented by the RSG tube surface area. The radiological consequences in the public (off-site) domain are dominated by the radioactive inventory of the reactor cooling circuit water, which varies over time and which, under reactor fault conditions, could include high levels of fission products released from damaged or melted fuel, in combination which other factors and circumstances, such as atmospheric stability, energy of the release, and so on.

8.5.8 On this basis I am able to respond directly to SCE's 10CFR §50.92 response:[39]

8.5.9 SCE claim that

8.5.10 ". . . *The proposed change to reduce the power level will not affect the probability of any accident initiators because the only effect on plant operations is to lower the allowable power level. . ."*

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37 Other than small pressure variations as a result of power transients, the RCS steady state operating pressure will be at the pressurizer setpoint pressure which for 70% RTP would not be expected to significantly change from 100% RTP setpoint.

38 As noted in SECTION 6, SCE and its consultants are unlikely to have that much greater detailed knowledge of the surface damage.

39 NRC, *Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination*; San Onofre Nuclear Generating Station, Unit 2 - Notice by the Nuclear Regulatory Commission on 04/16/2013.

8.5.11 Presented in this way, the SCE claim implies that the proposed reduction in power level will be accompanied by a reduction in the forces driving tube failure. This is incorrect, because the tube failure driving forces remain sensibility the same at 70% RTP as at 100% RTP.

8.5.12 As previously noted (¶14) both the AREVA and Intertek OAs[22,23] acknowledge that the RSG tubing will continue to degrade even with the plant operating at 70% RTP. Each of these OAs justifies the Unit 2 plant restart on the basis that the tube inspection will occur (at 150 days into Cycle 17) before the tube resilience has ventured into the unacceptable failure regime of 95% probability (the 0.05 threshold of FIGURES 1 and 2).

8.5.13 It follows that (¶7.24) Unit 2 operation into Cycle 17 will incur further and progressive degradation of the RSG tubes with this degradation being accompanied by an increasing probability of tube failure.

8.5.14 In terms of the TS, the probability of tube failure becomes *significant* once that the threshold of 95% probability has been passed. However, each of the consultants relied upon by SCE projects significantly different periods of time to reach and pass this safety threshold (as different as 0.5 and 1.35 year) which, essentially, shows that the underlying data and/or methodology of the predictions is fundamentally flawed.

8.5.15 So, on the basis of such uncertainty and unreliability, little assurance can be placed with SCE's confidence that its Cycle 17 inspection period of 150 days (~0.4 year) will pass without encountering a significant increase in the risk of tube failure.

8.5.16 This rationale applies to both the structural integrity performance criteria (SIPC) and the accident induced leakage performance criteria (AILPC),[40] although it is worthwhile noting that in a separate assessment of the Unit 3 TTW and TSP wear profiles, AREVA identified a number of tube wear modes, wall thickness wear depths and specific locations that failed AILPC[41] with a *'pop-through'* failure mode.[42]

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40 AILPC 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

The design basis to consider a coincident event involving either the LOCA, MSLB or FWLB with a SSE

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

8.5.17 For normal operation, the radiological consequences previously referred to in ¶8.4 would, all things being equal, remain much the same for each separate single tube burst event. Should a tube burst be provoked by some internal event in the nuclear plant, for example a loss of coolant accident (LOCA) or main steam line break (MSLB), then in certain circumstances and if the fuel core was damaged by the triggering event, then the radiological consequences arising from a single tube burst could be severe.

8.5.18 **In Summary:** In my opinion, in its NSHC analysis SCE has not taken due account of all of the uncertainties relating to i) the nature and severity of the tube degradation extant; ii) the reliability of the predictions of the period for further AVB restraint and TSP deterioration leading to TTW in order to set the first Cycle 17 inspection period (150 days); and, more generally, iii) the premise that the tube failure is predictable is unreliable because of the uniqueness of the tube wear in the AVB, TSP and TTW modes.

8.5.19 For these reasons I cannot agree with SCE that there will be no significant increase in the probability of a previously evaluated accident, namely a single tube burst under both SIPC and AILPC situations.

8.6 Create the Possibility of a New or Different Kind of Accident

8.6.1 SCE claim that

8.6.2 *“. . . **The proposed changes** do not adversely affect the method of operation of the steam generators **nor introduce any changes to existing design functions of systems, structures or components** that could create the possibility of a new or different kind of accident from any previously evaluated. Also, the proposed change will not introduce any significant changes to **postulated accidents resulting from potential tube degradation**. Because SONGS Unit 2 will operate at or below 70% Rated Thermal Power, the change **will continue to ensure that tube integrity is demonstrated** over the range of power levels at which the plant will operate. Therefore, there is no significant increase in the probability that the tubes will fail or leak during the period . . .“*

my added ***emphasis***

8.6.3 Relating to ***‘continuing tube integrity’***, as I have previously noted (¶8.5.3) that at 70% RTP the driving force of tube bursting remains sensibly unchanged (determined by the pressurizer

42 SCE does not seem to have applied this Unit 3 finding to setting a limitation on the acceptable tube wall thickness wear for the Unit 2 restart on the basis of AILPC alone which, for TSP and TTW modes of wear, will equally apply in Unit 2 during an MSLB design basis event.

setpoint pressure) so the only outcome of reducing power is, according to the Intertek and AREVA OAs, to slow the rate of AVB slackening and, once triggered, the rate of TTW.

8.6.4 I have noted previously, however, the Intertek and AREVA projections for the timing of each leg of the 2-stage process are at such variance that neither can be considered reliable.

8.6.5 On ***'changes to existing design functions of systems, structures or components'***, the original design functionality of the AVB was that the contact force or *'preload'* acting between tube and AV bar would be absolutely minimal because the *'zero tube-to-bar gap'* geometry. As previously discussed (¶7.12), the AVB installations in Unit 2 did not comply with this design prerequisite because of a manufacturing omission that resulted in, quite fortuitously, various levels of preload acting in the *in-plane* direction – this unintended preload served beneficially in Unit 3 to delay the onset of the second stage of tube wear or TTW.

8.6.6 Clearly, returning Unit 2 to service and, particularly, the time offset afforded by the unplanned presence of the AVB preload is a *change in the design function*.

8.6.7 Moreover, there is great uncertainty as to which of the many thousands of AVB-to-tube locations there is an active preload force and, where it is active, the magnitude of the preload depends on the extent of a dimensional distortion that was never recorded at the time of manufacture of the individual AV bar.

8.6.8 First, this gives rise to uncertainty about the proposed restart performance of Unit 2 – for example, how long before the preload force at each AVB-to-tube contact point will remain effective, if there will arise a sequence of loss of preload over a number of adjacent and successive AVB contact points on any one individual tube,[43] and so on – all of which places considerable doubt of the reliability of the Intertek and AREVA projection periods for AVB-to-tube slackening and the onset of TTW (¶7.25).

8.6.9 Second, since the preload force was unintentional and not considered in the original AVB-to-tube design it would not have been subject to the design approval and verification process which would have taken in account not just the immediate AVB functionality but, also, the *extended functionalities* and how these aspects of the AVB interacts with and impinges upon

43 The loss of successive AVB restrain lengthens the tube free span length making the tube vulnerable to lower frequency and higher amplitude motion.

other functions (and hence safety) of the overall steam generator assembly, both in part and as a whole.

8.6.10 To my knowledge, this expedient adoption of and reliance upon the unintended role of the AVBs – at those undefined AVB contact points where a preload does exist – has not been thoroughly thought through and, hence, its inclusion has not been justified by the Design Authority, be this SCE and/or MHI.

8.6.11 The *extended functionalities* of the AVBs apply in both SIPC and AILPC senses,[40] with the latter involving a Safe Shutdown Earthquake (SSE) coincidence event during and following which the RSG tubes and AVB assemblies would all be subject to material stresses in addition to the acting pressure differential (primary membrane) stresses in each tube wall.[44]

8.6.12 **Multiple Tube Failure:**[45] SCE should have undertaken further analysis on the possibility of a multiple tube failure which would, in terms of off-site radiological consequences, greatly exceed the design basis accident of a single tube burst.

8.6.13 There are a number of situations with potential for a multiple tube failure that should have been evaluated by SCE, including:

8.6.14 AVB BREAK UP: The MHI photograph alongside ¶6.3.9 shows a situation where a Unit 3 tube has abraded and worn away a significant part of the AV bar. Although it is not possible to distinguish between tube wall and AV bar wear depths, it is obvious from the photograph that the AV bar wear greatly exceeds the tube wall thickness – on this basis, the AV bar wear depth is about 40% of the original bar thickness.

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44 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 – induced SSE tube loading is highest at the top of the U-bend.

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 and AVBs may also be subject to shaking loads caused by the LOCA break hydrodynamics and reactor coolant circuit motion.

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.

45 Rupture of a single tube is the design basis event. A multiple tube rupture would be a very much more hazardous event because reactor coolant water could be expelled rapidly through the ruptured tubes, resulting in water inventory for the emergency core cooling system being depleted followed by fuel core meltdown.

8.6.15 The tubes are arranged in a dense triangular grillage so *in-plane* motion of two or more tubes in adjacent rows could result in deep wear simultaneously occurring on opposite sides of the AV bar, and/or any *in-plane* tube motion could extend the AV bar wear into an extended crescent substantially weakening the AV bar in its cross section. Conceivably, the upshot of this situation would be a section of the AV bar detaching and acting adversely on a number of tubes in or nearby its original location in the tube bundle.

8.6.16 There are a number of situations that could challenge and possibly physically detach sections of such a worn down AVB, including seismically induced loading on the tube bundle, the immediate aftermath of a LOCA,[46] and, quite possibly, the dynamic fluid forces triggered by a MSLB.[47]

8.6.17 The presence of such a large unrestrained object within the tube bundle could impose a threat to a number of tubes. Hence, SCE should have included in the NSHC for the possibility of a worn down section of AV bar detaching under SSE, LOCA and MSLB fault event conditions, leading to a multiple tube failure.

8.6.18 <u>PLUGGED AND IN SERVICE TUBE FAILURE AND BREAK UP</u>: It is possible that both pressurized and plugged tubes could fail locally and dislodge shrapnel into the tube bundle, thereby providing opportunity for multiple in-service tube failure.

8.6.19 There are a number of mechanisms for this, including:

8.6.20 a) In a situation where the degraded tube surface is heavily scored and/or includes one or more notched stress raisers – see the diagram of ¶6.3.7 - this type of degradation scar can be generated on both in-service, pressurized and plugged tubes,[48] leading to a pressure driven brittle failure of one or more tubes.

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46 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.

47 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 multiple tube failure.

48 Plugging a tube does not remove it from the risk of it being excited into motion by fluid forces and/or by impact from an adjacent tube or tubes and, of course, a faulty plug may enable an individual plugged tube to become pressurized but with near zero flow through it.

8.6.21 UFSAR (S5.4.2.31.3) considers integrity of a degraded tube with wall thinning at TTW, AVB and TSP locations for SIPC and AILPC cases. The analysis, although not presented in detail in the UFSAR, arrives at the stresses in the degraded tube cross section by weighting the stresses in the non-degraded tube by the ratio of the corresponding sectional properties of nominal and degraded tubes.

8.6.22 This comparative approach, as it is described in the UFSAR, omits to give any regard to the tube surface damage and flaws that I have described in ¶6.2.1, ¶6.3.1 and ¶6.3.5 – if surface scarring and notching was taken into account (weighting via k_t), particularly for the AVB *Dig-In* case, then it is possible then the allowable stress limit of 74.3ksi would exceeded.[49]

8.6.23 b) I assume that in its NSHC SCE has considered the possibility of tube failure via *in-plane* high cycle vibration mode – if it has done so then it may have relied upon the previous MHI analysis.[50]

8.6.24 Where a tube is pinched at a particular location (say at the top TSP), displaced laterally from its zero load state (such as in *flowering*) thus inducing bending and shear stresses, its excitation into oscillatory motion (vibration) renders it subject to i) high cycle plain fatigue and some element of the separate phenomenon ii) fretting fatigue. A similar combination of plain and fretting fatigue can also occur at the AVB-to-tube contact point, particularly at the *Dig-In* locations where there is AVB-to-tube contact force and high inter-surface (Coulomb) friction exists.

8.6.25 Both in-service and plugged, cable stabilized tubes could also be vulnerable to this combined i) and ii) mode of fatigue failure with the resulting shrapnel, or the severed tube itself, resulting in high rates of wear and possible multiple failure of nearby in-service tubes.

8.6.26 Failure of a plugged tube and associated wear damage of two adjacent in-service tubes at TMI-1 in October 2001, suggests that high cycle plain fatigue can run its course to failure

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49 The AVB degraded tube locality represents the worst case where the degradation length is assumed to be equal to the average tube-to-AVB contact length, the minimum wall thickness is 0.01526 inch. Therefore, the degraded tube minimum wall thickness is conservatively taken as 0.01923 inch, which corresponds to 55.17% tube wall thinning for 0.0419 inch nominal wall thickness.

50 *Enclosure 3, Part 2 MHI Document L5-04GA564 Tube Wear of Unit-3 RSG – Technical Evaluation Report Appendix-16 Fatigue Evaluation of the Tube due to In-Plane Vibration* MHI Proprietary Class B

within a single fuel cycle and, although there was surface fretting evident, the post-event analysis did not consider a fretting fatigue contribution to the tube failure.[51]

8.6.27 The MHI analysis[50] that SCE probably relies upon contains proprietary information that I am obliged not to reveal in a public document such as this declaration.

8.6.28 That said, I consider the *in-plane* fatigue model adopted by MHI to have a number of shortcomings, particularly in that the TSP restraint should be clamped and not freely pinned and that the derivation and final selection of the stress concentration factor k_t is somewhat conservative. Also, it is not clear to me that MHI has given any regard to the quite possible contribution of fretting fatigue at the TSP location where some TSP-to-tube sliding motion would have occurred – as I have previously noted (¶6.10) the combination of plain and fretting fatigue can result in a substantial reduction of component life, that is bringing on unexpected early bursting of the tube.

8.6.29 In ¶6.3.5 I refer to the notch-like flaw caused by what I term to be *AVB Dig In* which develops a radial notch-like flaw in the tube. This provides a weakness for brittle failure in the *out-of-plane* direction in which the individual tubes and, indeed, sections of the tube bundle, are susceptible to excitation by fluid forces. Once again, because the interface between tube and the AV bar is hidden and cannot be reliably determined by through-wall ET, the determination of the stress concentration factor k_t is very uncertain.

8.6.30 I believe that it would be prudent for SCE to review the MHI analysis of tube fatigue limit of 13.6ksi,[52] together the selection of the stress concentration factor k_t and restraint conditions for both TSP and AVB *Dig In* cases, and in combination with this, if the possible contribution of fretting fatigue has been properly accounted for.

8.6.31 I understand that another party is to provide more detailed expert opinion on this aspect of tube vulnerability to high cycle plain fatigue failure.

8.6.32 SEISMIC LOADING: I have already commented on the possibility that seismically induced loading on the tube bundle could detach a worn through AV bar component (¶8.6.16).

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51 In October 2001 a plugged tube severed at TMI-1 and inflicted wear on two adjacent tubes at such a high rates that the structural integrity of both tubes was challenged within one fuel cycle. The circumstances that apparently contributed to the severed tube included tube swelling and flow-induced vibration leading to high cycle fatigue failure – the SG at TMI-1 was a once-through and not the recirculation type as at San Onofre.

52 ksi – kilo pounds force per square inch.

8.6.33 Here I note my surprise that in answering the NSHC SCE has chosen not to explore, apparently neither generally nor in the detail that it merits, seismic loading of the overall tube bundle, taking into account the degraded and defective tubes and components (including both inactive and preloaded AVBs).

8.6.34 Indeed, I am surprised that SCE has not been required to undertake a seismic response evaluation for the entire RSG assembly, particularly now that it has been established by the ASLB Hearing[53] that the CAL was in effect a de facto license amendment, meaning that the steam generator replacements at San Onofre were not like-for-like replacements and therefore should have qualified for the rigors of a full safety evaluation, including for response to seismic events.

8.6.35 **In Summary:** In my opinion, in its NSHC analysis SCE has not considered that the extant condition (notching, fretting fatigue, etc) of the individual tubes, tube bundle and its restraint components could lead to new or different kinds of event.

8.6.36 SCE's proposal to derate the plant to 70% RTP may *'not adversely affect the **method** of operation'* but, on the other hand, the Cycle 16 operation resulted in substantial degradation and damage to the RSG tubes and restraint components, so much so, that the response of these components to normal and adverse operating conditions had not been accounted for in the original design case.

8.6.37 Moreover, it has been acknowledged by SCE that the degree of twisting of certain of the Unit 2 AVBs resulted in an unintentional tube clamping preload force (see ¶7.12), with result that these AVBs do not comply with the original design intent.[54]

8.6.38 In these respects alone, SCE cannot rule out that new or different kinds of event will occur.

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53 Atomic Safety Licensing Board, In the Matter of Southern California Edison Co, *Memorandum and Order*, May 13 2013

54 In this situation, in pressing against the tube the AVB-to-tube contact surface, which must sense the vibration of the up- or downstream free span portion in the form of slight relative movement (or vibration of the AV bar itself), is likely to be conducive the fretting fatigue. Thereafter if and when the AVB preload is loosened (a process acknowledged by both Intertek and AREVA), then the total plain fatigue life of the same but now freely vibrating tube may have its endurance limit shortened by as much as 50 to 70% - see ¶6.10 – in this respect the NSHC should have included account for fretting and plain fatigues modes of tube failure.

8.7 Involve a Significant Reduction in a Margin of Safety

8.7.1 The current regulatory practice (the TS) assumes a prescriptive approach under which tube plugging is required when certain conditions are met. The most notable of these is that degradation depth by any process cannot, in general, exceed 35% of the tubing wall thickness.

8.7.2 The 35% maximum wear was chosen to provide a factor x3ΔP safety margin against burst under operating conditions (SIPC) and x1.4ΔP against burst under postulated accident conditions (AILPC).

8.7.3 The safety margin takes in a number of uncertainties, including allowances for ET and other in-tube measurement errors, flaw growth between inspections, temperature compensation, various processes that contribute to tube material and geometry degradation, and so on.

8.7.4 When originally compiled the safety margin would have assumed that the RSG functionality was compliant with the design specification. Generally, this means any detriment arising from a design omission or design shortcoming, such as the inadvertent introduction of AVB preload, would not have been included for in the safety margin.

8.7.5 It follows that particular processes arising from such a omission or shortfall, in this case the occurrence of fretting fatigue at the AVB-to-tube contact point and its potential to substantially reduce the plain fatigue life of individual tubes, would also not have been include for in the safety margin.

8.7.6 At San Onofre the proposed Cycle 17 operation will include for continuing degradation of the tubes, TSP and AVB restraint points, and it will do so on the basis of largely uncertain data and information about the detailed extant condition of the tubes, TSPs and AVBs – see FIGURES 1 and 2.

8.7.7 Such intention to knowingly permit further erosion of the safety margin when it has to be acknowledged that the root cause and processes that have led to, and are continuing to degrade the tubing, are not fully understood nor can be reliably described in a quantitative way is, in my opinion, altogether unacceptable.

8.7.8 **In summary:** I consider that to permit the established safety margins to be reduced in ways and to an extent that cannot be precisely defined to be wholly unjustified and beyond the discipline of sound engineering design and practice.

9 ASPECTS RELATING TO THE POTENTIAL RADIOLOGICAL CONSEQUENCES

9.1 In this declaration I have focused on the opportunity for tube failures due to the degraded or, more properly, defective condition of the tubes themselves and how this if sufficiently severe, such as a multiple tube failure, could lead to malfunction of the nuclear plant. Of course, it is quite feasible that failure of a few defective tubes could trigger a major nuclear plant malfunction that, in itself, provokes the bursting of more degraded or defective tubes creating a very significant radiological release via a primary containment bypass. Also, there is the possibility that a major plant malfunction, such as a MSLB, could rapidly result in failure of multiple tubes already weakened in a degraded or defective condition.

9.2 The radiological potential of the release is generally set by the rapidity of the events. Thus, a rapid or near simultaneous multiple tube failure in a single steam generator could overwhelm the reactor safety systems, for example by outstripping the capacity of the emergency core cooling system (ECCS), so much so that the uncovered fuel core would proceed to high temperature melt that would greatly add to the radiological inventory en-route via the SG containment bypass for release to atmosphere.

9.3 The point here is that, by whatever cause, failed steam generator tubing presents a direct and rapid route for the release of radioactivity to the atmosphere for onward dispersion and deposition in the public domain. Because the release route bypasses the primary containment, the impact on the public is virtually immediate, there being little or no time to prepare mitigation in the forms of prophylaxis, sheltering and evacuation countermeasures.

9.4 In other words, failure of a number of steam generator tubes in either or both of the Unit 2 steam generators, could result in a significant radiation dose uptake and onerous health detriment being borne by members of the public in the San Onofre region.

10 CONCLUDING REMARKS

10.1 For my assessment of the NSHC I have been reliant upon the information and data provided by SCE and its consultants. Much of this information and data continues to be heavily redacted and many reports are simply not available in the public domain.

10.2 Nevertheless, I am confident that even though I have relied upon elements of qualitative judgment my overall opinion is sound.

10.3 I note that SCE's NSHC omits to apportion proper regard to all of the uncertainties deriving from the extant and future degradation of the Unit 2 RSG tubing, particularly the surface flaws; it fails to address how and to what extent the thermal-hydraulic flow regimes will lessen the excitation forces acting across the tube bundles and whether these forces are generated by fluid elastic instability, which it claims will be largely eliminated by a reduction in RTP, or if tube motion is driven by random two-phase fluid processes which are likely to remain even at reduced RTP; and, it has yet to address the root cause of why the tube degradation processes were so vigorous in the design of the San Onofre RSGs.

10.4 In these and other respects, I find SCE's NSHC evaluation insufficient in breadth and detail of examination, and lacking the substance that the all-important issue of nuclear safety merits.

10.5 Each finding of SCE's NSHC tripartite evaluation is in error because, amongst other things:

10.5.1 (1) it fails to recognize that the degraded condition of the tubing and restraint components is such that there will arise a greater probability of single tube failure before the proposed 150 day tube inspection outage is reached;

10.5.2 (2) it assumes that the quite fortuitous change of design functionality of the AVB-to-tube restraint (ie the preload) will not carry forward with it detriment that fosters opportunity for different kinds of accident (ie break-up of the AV bar) leading to, for example, multiple tube failure and the knock-on events (not considered in the UFSAR) that could lead to a very significant radiological event in the public domain; and

10.5.3 (3) it plans to knowingly diminish the all-important tube integrity safety margin by operating Unit 2 under such conditions that are acknowledged to foster continuing tube degradation (wear) and add to existing levels of flaws (surface notches) and adverse processes (fretting and plain fatigue).

10.6 In these and other crucial respects, the outcome of the SCE No Significant Hazard Consideration is incorrect and inappropriate.

10.7 Also, I am surprised that the SCE submission in this respect is so wanting in detail and depth. For such an important matter that bears heavily on public safety I would expect a very much more substantive consideration – in my opinion, the SCE submission of the *No Significant Hazard Consideration* does not pass muster and would fail any reasonable Public Interest Test.

10.8 Overall, I consider that in view of the extensive degradation of the San Onofre Unit 3 RSG, which I consider to be a portend of how further tube degradation would occur if Unit 2 was to resume operation then, like Unit 3, Unit 2 should not be considered fit-for-purpose for return to any level of nuclear power operation. Moreover, my opinion is that such is the uncertainty about the condition of the two Unit 2 RSGs and how these individually or in combination would contribute to and/or respond to an adverse event (for example a MSLB) then, on nuclear safety grounds alone, Unit 2 should not be permitted to restart nuclear operation.

10.9 I John H Large declare, under penalty of perjury, that the foregoing information and 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.

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