ADDENDUM I

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 19 February 2013

NOTE: THIS IS A SWORN AFFIDAVIT OF MY PREVIOUS DECLARATION OF FEBRUARY 13, 2013

COMMENTS ON THE NRC AND SCE RESPONSES OF JANUARY 30, 2013

DECLARATION OF JOHN LARGE

I, JOHN **HENRY 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, London, United Kingdom, SEI8 4BQ.

1.2 I have given my qualification and experience in my 1st Affidavit of January 10, 2013.

2 **INSTRUCTIONS**

2.1 On January 31 2013, I received instruction from Mr Shaun Burnie of Friends of the Earth (FoE) to read through and prepare, as I believed appropriate, comments on the Responses of i) the Nuclear Regulatory Commission (NRC) and ii) Southern California Edison (SCE), both dated January 30, 2013.

3 In this respect I have read through the two responses and all of the attachments thereto.

4 **REFERENCING**

4.1 For ease of reference I shall refer to text sections of my 1st Affidavit thus {5.5.1}; the NRC Response [NRC p2, ¶3] and its attachments]NRC p6, ¶2[; the SCE Response [SCE p2, ¶3] and the Affidavit of Richard Brabec]SCE Brabec ¶16[; similarly to the Affidavit of Kenneth Karwoski]NRC Karwoski ¶9[; and to all other text references to other sources of information thus ¦p6,¶5¦.[6]

5 COMMENTS OF NRC AND SCE RESPONSES

5.1 In the following text I shall address only the substantive issues and topics that have been raised by NRC and/or SCE although, that said, my silence on any point and the fact that I have not addressed it in this rebuttal should not be taken as my agreement with that point.

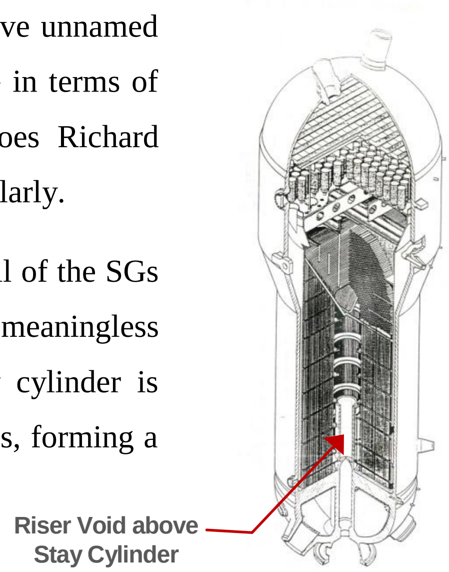
6 SONGS AND THE FIVE COMPARATIVE NUCLEAR PLANTS

6.1 In my 1st Affidavit {Section 8.01 to 8.20) I identified 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 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 or similar) simulation of each of the five nominated SGs, the comparisons drawn are likely to be between aggregate or bulk flows, both velocity and void fraction, within the entire tube bundle of each SG; and iii) that the SONGS RSGs are dominated by *in-plane* flow regimes, whereas all other SGs are characterized by *out-of-plane* flow regimes.

6.2 I concluded that unless the spider diagrams of the *Tube-to-Tube Report* 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.

6.3 SCE and Richard Brabec offer no further insight as to why the five unnamed nuclear power plants qualify for comparison to the SONGS RSG in terms of performance. Relying upon a few physical similarities, as does Richard Brabec, will not ensure that the various plants will all perform similarly.

6.4 For example, simply stating that one basis of comparison is that all of the SGs do not have a stay cylinder] SCE Brabec ¶16[is somewhat meaningless because it is how the space accessible above the omitted stay cylinder is utilized that is important. Typically, this space is left free of tubes, forming a



riser chimney which facilitates upward flow of the feedwater column, serving to maintain a high circulation ratio (CR) and regular dispersion of the heat transfer flux or profile through the tube bundle.

6.5 In the SONGS RSGs the riser chimney was packed with tubes ¦p9, R-H slide¦,[1] to the effect that the CR would have reduced and, with the additional heat transfer flux from these central zone tubes, this would have most likely promoted the earlier occurrence of undesirably high void fraction and dryout and, quite possibly, nucleate boiling on the hot side of the tubesheet.

6.6 In the comparison tabulation]SCE Brabec ¶15[, Richard Brabec promotes the inclusion of broached tube support plates (TSP) as a determining factor but it is the effect of the TSPs, particularly the upward flow resistance presented by the TSPs and its outcome on the CR, that are the important comparative parameters.

6.7 Thus, for a meaningful comparison in this aspect of the design alone, it is the number of TSPs and the specific design (free flow area) of the broached quatrofoil apertures of each comparative plant that are important.

6.8 Unusually, the SONGS RSG design deployed seven TSPs [SCE Attachment 5, Figure 6.3-1, p79], so this geometry presents a greater overall flow resistance than, say, a six TSP design – unless compensated by some other aspect of the SG design, added TSPs will result in a reduced CR. The different SG manufacturers are known to zealously guard details of the quatrofoil apertures so, unless these details are known for each of the comparative plants, simply stating that the plant performance will be similar because each is fitted with broached TSPs is somewhat meaningless.

6.9 Similarly, the maximum number of AVBs per tube is only meaningful if the design approach (zero-gap, preload or no preload, and tube free span length) is comparable; there are missing parameters that should be compared, for example the circulation ratio (CR), the magnitude and location of the two-phase fluid *'pitch'* velocity, the void fraction and the dominant flow direction in the most critical regimes of the tube bundle, that is *in-plane* (IP) or *out-of-plane* (OOP) or a combination of both, etc..

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1 SCE, *San Onofre Nuclear Generating Station, Steam Generator Replacement Project Overview*, slide presentation to the NRC, public meeting June 7 2005

6.10 With respect, Richard Brabec seems to misunderstand the mechanics of fluid induced excitation of tubular structures, assuming as he does]SCE Brabec ¶17[that only when conditions are conducive to fluid elastic instability (FEI) is there risk of tube motion and damage. In my 1st Affidavit I consider in some detail the different findings {5.7.61-62} arrived at by SCE's consultants, particularly when identifying the fluid sources inducing tube and support structure motion and the different outcomes {5.5.7-9} and {5.7.9-11}.

6.11 The point that Richard Brabec misses is, as found by Westinghouse {5.7.58} and Mitsubishi {5.5.20-23}, that tube motion and wear can occur in the absence of FEI, either IP or OOP or both. Thus, tube motion and wear can be induced by 'random' fluid processes that are just as likely to occur in a RSG operating at 70% of the rated thermal power (RTP).

6.12 Finally in this matter of the applicability of the so-called comparative nuclear plants and SGs, it is interesting to note that the NRC Staff, in its response [NRC p61, ¶3], admits that it *'does not presently know the identity of the plants compared . . . therefore, the Staff cannot provide an assessment of the similarity of the tube material, tube spacing, and support structures'.*

7 TUBE WEAR MODES AND RATES

7.1 Edison is particularly coy about the tube wear, claiming [SCE p6] that the AVB and TSP wear was *'expected for the RSGs . . . although the extent was higher than the industry average'*. SCE's judgment that the tube wear was *'higher than industry average'* is somewhat of a dubbing down of the 10,284 indications[2] in the two Unit 3 RSGs after just 11 months in service. This statement is clearly not borne out by the comparisons with the data presented by authoritative bodies such as EPRI.[3]

7.2 Moreover, SCE obviously misunderstands the two-phase slackening-off process of the AVBs leading to TTW [SCE p91] when comparing Unit 2 to Unit 3. This is because it considers that *'This difference in operating experience indicates that the experience of the Unit 3 RSGs is not applicable to Unit 2. Furthermore, since Unit 2 experienced only one instance of TTW in 21 months of full power operation, operation at 70% power for 150 days will incur even less likelihood of any TTW'.*

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2 Table A, *1st Affidavit of John Large* taken from Attachment 4 of the SCE, Enclosure 2, *SONGS Return to Service Report*, October 3 2012

3 Electrical Power Research Institute, Benson James, *Overview and Analysis of Historical Steam Generator Degradation Mechanisms*, NRC Meeting, February 7, 2013.

7.3 However, SCE does accept the two-stage wear model when it suits, for example [SCE p98], but it simply picks the most favourable figures for separate tube wear onset and tube wear periods. Furthermore, SCE is incorrect, if not disingenuous with its allegation that *'FOE's statements quoted above ignore the time required for instability zone expansion (i.e., time for an in-service pressurized tube to be driven to instability with consequent development of TTW)'* because all of the various delay and wear periods are presented and explained in my 1st Affidavit {5.8 et seq}.

7.4 For example, at {5.8.13-15} I faithfully reproduce and explain the best estimates of the AREVA *Tube-to-Tube* report, viz

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

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**

7.5 I argue that the rudimentary basis of AREVA's calculation of its 3.5 month *slackening-off* time[4] t_{so} {5.8.16-18} does not instil confidence, so much so that it should not be used to front-end (ie incur a delay to) the onset of TTW {5.8.17-24}, noting that there is no certainty of just where Unit 2 is presently at along the path towards TTW wear {S5.8 i)-iii)}.

7.6 Considering the separate retainer bar vibration issue:- On the wear rate projected for the tubes adjacent to the smaller diameter retainer bars that are excited into large amplitude vibration by random fluid processes (not FEI), SCE's understanding appears contrary to fact. This is because SCE's response to the NRC's *Request for Additional Information* (SCE Attachment 27, p2) is incorrect in assuming that *'The integrity of the non-stabilized, preventively-plugged tubes is ensured by limiting the wear resulting from retainer bar*

4 SCE refer to the *slackening-off period* as the *instability zone expansion time*)35(.

vibration. The limited vibration amplitude of the tubes and retainer bars, combined with stabilizer deployment, prevents developing a displacement/wear geometry that could sever any of the tubes adjacent to retainer bars, either in the short term or long term'.

7.7 So far as I can ascertain there is no remedial action planned to inhibit the vibration of the retainer bars should Unit 2 return to service.

7.8 Since the retainer bar vibration is induced by random fluid processes, there can be no guarantee that this and other non-FEI sources of excitation will be eliminated by reducing the power level to 70% RTP. In other words, even though the tubes local to the retainer bars are to be plugged, fretting wear is likely to continue with risk that parts of the tube wall could detach to damage adjacent tubes and/or lodge elsewhere in the tube bundle as abrasive foreign bodies. I note that the rate of tube wear at the retainer bar locality can be high, as example by the Unit 2 wear indication where the tube wall thickness had been worn away by 90% of the tube wall thickness.

8 95% PROBABILITY AND 50% CONFIDENCE

8.1 In my 1st Affidavit I expressed strong doubt {10.6} over the reliability of the AREVA *Tube-to-Tube Report* probabilistic prediction of individual tube motion (ie the onset of FEI), and I detailed the reasons for my doubt {5.7.46-52}.

8.2 I noted 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 that are open to inspection {10.8}. A situation that does not at all help to explain, and I admit to being puzzled by, the SCE and Brabec statement]SCE Brabec ¶43[that *'the calculation performed at a stability ratio of 0.75 was performed to "demonstrate margin," not to determine reasonable assurance.'*[5]

8.3 I am also puzzled why the 95/50 criterion has been applied to the FEI stability ratio (SR) for individual tubes. This is because this EPRI recommendation ¦p3-1,¶1¦[6] applies to the limiting structural integrity requirements for SG tubing, ie the resilience against bursting. This being based on a study of the ratios of the probability of meeting a 3ΔP pressure

5 The NRC Staff notes that *'the regulations do not contain a preset confidence level for determining reasonable assurance'* [p81, footnote 312] .

6 EPRI. *Technical Basis for Steam Generator Tube Integrity Performance Acceptance Standards*, December 2006.

differential to the probability of meeting the limiting main steam line break (MSLB) pressure differential ¦p3-1,¶1¦.[6]

8.4 Deploying the 95/50 criterion to the individual tubes of the whole tube bundle, starting at estimation of the incidental AVB pre-load brought about by a manufacturing shortfall – see ¶13.3.5 later – to determine a tube burst margin is, in my opinion, a step too far.

8.5 In other words, the SCE Response does nothing to allay my concerns.

9 TEST AND EXPERIMENT

9.1 Richard Brabec clearly refers to *experimentation* in]SCE Brabec ¶32[when referring to the 70% RTP running regime *'This administrative limit is temporary and may change based upon the results of inspections, further analyses and long-term corrective actions'*, at]SCE Brabec ¶33[where *'inspections is to confirm the effectiveness of the corrective and compensatory actions taken to address TTW in the Unit 2 RSGs'* and, further, at]SCE Brabec ¶37[*'SCE has not yet identified long-term corrective actions for the steam generator tubes for Unit 2'*.

9.2 Here Richard Brabec is acknowledging that the RSG tube bundle, which is an integral part of the structures, systems and components (SSC) crucial to nuclear safety, is being *'utilized in a new way'* ¦10CFR §50.59(a)(6)¦. In other words, the rate and outcome of tube degradation during and at the completion of the proposed in-service inspection period of 150 days, being compared to that predicted, is nothing more than *test* and *experiment*.

9.3 Perhaps tellingly, throughout its response SCE provides no opinion as to whether the San Onofre nuclear plants would be capable of operating at full power without comprising nuclear safety, as the present unamended operating license requires.

9.4 Similarly, no opinion is proffered as to whether the operational assessments (OAs) undertaken by the likes of AREVA prior to the proposed restart of Unit 2, forecast with certainty the rate and extent of tube-to-tube wear (TTW) degradation for the plant operating at 100% RTP or, indeed, at any level above 70% RTP – I refer to this further at ¶15.1 to 15.12.

10 COMMENTS THE AFFIDAVIT OF KENNETH KARWOSKI

10.1 Here I shall refer specifically to Attachment 1 of Mr Karwoski's Affidavit]NRC Karwoski ¶9[, particularly in respect to the claim that SCE's actions listed in the NRC's Confirmatory Action Letter (CAL) of March 27, 2012, may all be performed as part of the existing steam generator program and that no change to the technical specifications is needed to perform the steps outlined in the CAL.

10.2 Also, I shall comment on Mr Karwoski's claim SONGS Units 2 and 3 must continue to meet the technical specifications tabulated in Attachment 2 to his Affidavit

11 ATTACHMENT 1

12 For ease of reference I reproduce Attachment 1 as Addendum I of this Affidavit.

13 10 CFR §50 APPENDIX B – QUALITY ASSURANCE CRITERIA

13.1 The first item featuring in Kenneth Karwoski's Attachment 1 is 10CFR§50 Appendix B 1, being a quality assurance requirement applying to the design, fabrication, construction, and testing of the structures, systems, and components of the facility.

13.2 The *structures/components* involved are the replacement steam generators (RSG) generally and, specifically, the RSG tubing and its support structures (such as the AVB, TSP and RB components and assemblies).[7]

13.3 *Criterion III* of 10CFR§50 Appendix B stipulates that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. In this case and to my estimate, the *original design* for the RSGs would have been at or about the mid 2005 through to mid-2006 period.

13.3.1 My understanding is that the major design faults and shortfalls - vigorous FEI and lack of AVB *in-plane* restraint – would have arisen and most likely have been recognised by and known to SCE at this time.

13.3.2 Accordingly, my point here is twofold:

13.3.3 First: the design of the AVB, and hence the tube bundle overall, was faulty and not to specification from the onset; and

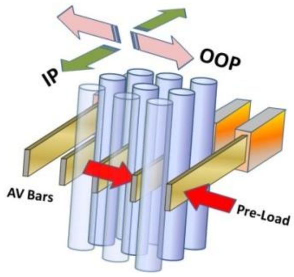
7 AVB – Anti Vibration Bar TSP – Tube Support Plate RB – Retainer Bar.

13.3.4 second: the SCE proposal is to return Unit 2 to powered operation with a faulty and non-compliant design in place without correction.

13.3.5 The first aspect means that a now acknowledged failure of the AVBs to perform the design function (arising from the warped AV bar problem in the Unit 2 components) constitutes a design change which has not, to my knowledge, been subject to approval by the design authority, being either SCE or MHI, or both. The second aspect means, as I contend in my 1st Affidavit, that returning Unit 2 will be accompanied by an unquantified risk of failure, particularly because of the uncertainties involved {10.6}.

13.4 There are similar requirements specified by *Criterion VI* on document control.

13.4.1 This is because in the original design, the AVB functionality centred on, according to MHI [p10, Summary],[8] a *zero-gap/zero-preload* strategy but, because of the AV bar distortion discussed above, an unspecified number (many thousands) of AVB-to-tube contact points each had, in varying degrees of magnitude, an unintentional pre-load force ➡⬅.



13.4.2 To my knowledge the magnitude and distribution of this unintentional pre-load force, although subject to prediction by AREVA in the *Tube-to-Tube Report*,[9] has not been mapped with the precision expected for a structure, system and component (SSC) crucial to nuclear safety.

13.5 *Criterion XVI* is the criterion for corrective action – it is the only criterion of 10CFR§50 Appendix B that Kenneth Karwoski has identified to the pertinent to the CAL.

13.5.1 It is worthwhile reproducing this quality assurance criterion here for ease of reference:

> ***XVI Corrective Action***
>
> *Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, **deviations**, defective material and equipment, and **non-conformances** are promptly identified and corrected. In the case of significant conditions adverse to quality, the **measures** shall **assure** that the cause of the condition is determined and **corrective action** taken to **preclude repetition**. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.*
>
> my ***emphasis***

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

9 Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," Appendix B, Revision 0, "SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear," prepared by Areva NP Inc. Document No. 519187230-000 (NP), Revision 0), October 2012. (ADAMS Accession Nos. ML 12285A267, ML 12285A268, and ML 12285A269).

13.5.2 In my opinion, the uncertainties over the magnitude and distribution of the pre-load at the AVB-to-tube contact points; the extent and nature of tube wall thickness wear extant at the AVB, TSP and retainer bar (RB) locations are all *non-conformances* and/or *deviations* that may be have been to some extent *identified*, but see ¶13.4.2 above, but which have not been *corrected*.

13.5.3 According to both SCE and Richard Brabec]SCE Brabec ¶37[*'SCE has not yet identified long-term corrective actions for the steam generator tubes for Unit 2'* so, it follows, no *corrective action* can taken to *preclude repetition* of AVB-, TSP- and RB-to-tube wear caused by random fluid processes exciting tube motion or the restraint (in the case of the RB-to-tube wear mode) ¦ ¶6.11¦.[10]

13.5.4 The proposal to reduce the incidence of FEI by reducing the power to 70% RTP is no guarantee that random fluid processes will not continue to excite tubes and support components in the AVB, TSP and RB localities. Continued oscillatory (vibratory) motion of tubes and structures at these localities will inevitably result in further tube wear.

13.6 As I have been able to determine from SCE's documentation (the OAs) and my deduction, SCE's actions listed in the CAL could not have been undertaken in a manner and in detail that is compliant with several of the quality assurance requirements of 10CFR§50 Appendix B.

14	**TECHNICAL SPECIFICATION**	**TS 3.4.17**	**TUBE INTEGRITY**
		SR 3.4.17.2	**TUBE PLUGGING**
		TS 5.5.2.11	**STEAM GENERATOR PROGRAM**
		LCO 3.4.27.B	**PASS TO MODE 5 STATE**
		LCO 3.0.4	**MODE LOCK DOWN**

14.1 Kenneth Karwoski provides a summary of what he considers to be the relevant Technical Specifications (TS) and these are discussed in further detail in my 1st Affidavit. Here I am considering tube integrity, tube plugging and the steam generator programme – the other two aspects (LCOs) are of not immediate concern being what I describe as *operational constraints.*

14.2 For any nuclear plant, the operating license (OL) typically includes a TS that is customized to that particular plant. The TS defines mandatory operating limits and other requirements and actions that must be taken to ensure the safe operation of the plant, protection of public health and safety and the environment. TS content must include: i) safety limits,

10 John Large, 2nd Affidavit, Comments on the NRC and SCE Responses of January 30, 2013, 13 February 2013.

limiting safety system settings and limiting control settings; ii) limiting conditions for operation; iii) surveillance requirements; iv) design features; and v) administrative controls.

14.3 At]NRC Karwoski ¶9[Kenneth Karwoski claims that all of the CAL actions imposed upon SCE may be undertaken within the requirements of the existing steam generator program (SGP) and *'that no change in the technical specifications is needed'* to perform the steps outlined in the CAL.

14.4 On the basis of my understanding of the technical and engineering issues involved, I disagree.

14.5 For example, 10CFR§50.59 establishes the framework under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval, and without submitting a license amendment request. A licensee may modify the plant and associated documents (procedures, drawings, updated final safety analysis report, etc.) without prior NRC approval unless, that is, the OL or TS must be revised to permit implementation of the modification, or the modification meets one or more of the eight criteria specified in 10CFR§50.59(c)(2) that are discussed in my 1st Affidavit {11.15 et seq}.

14.6 However, the basis of Kenneth Karwoski's claim of surety that all of SCE's actions are within the SGP and will not require modification of the TS is somewhat doubtful because [NRC p60]

14.7 *'. . . the Staff has not yet reached a position on whether to approve SCE's Return to Service Plan . . . NRC Staff is reviewing SCE's October 3, 2012 Return to Service Plan outside of this proceeding. As part of that review, the Staff will examine whether SCE's October 3, 2012 Return to Service Plan requires a license amendment and whether it provides a reasonable assurance that SONGS Unit 2 will operate safely after restart. . "*

14.8 SCE's current position compounds the uncertainty [SCE p50]

14.9 *". . . SCE has not yet decided how it will respond to the RAI. SCE will inform the Board and the parties of its position once it submits a response to RAI 32 . . "*

14.10 Interestingly, NRC introduces an element of chance to its reasoning [p60] stating that the CAL did not specify how SCE was to respond and, for this

14.11 *“. . . SCE could have taken a variety of approaches to establishing the requested actions, protocols, inspections, and operational limits, some of which may have been acceptable to the Staff and some of which may have required a license amendment . . .”*

14.12 In other words, NRC admits that whether or not the CAL is considered to be a de facto License Amendment depends upon **its** particular wording and the particular **response** of SCE. My understanding, again on the basis of the technical and engineering issues involved, is that on this basis the CAL would *‘modify the existing* license’ and thus become a legal entity in the licensing process.

14.13 Given that both SCE and NRC Staff have yet to complete the CAL submission and review processes, with the NRC issuing a number of requests for additional information (RAI) as late a December 26, 2012,[11] it is perhaps somewhat premature of Kenneth Karwoski to conclude [NRC Karwoski ¶10] *‘that the March 27, 2012 CAL has no effect on SCE’s licensing authority’*.

15 REQUEST FOR ADDITIONAL INFORMATION - RAI 32

15.1 This brings me to the recent round of requests for additional information (RAI) put to SCE on December 26 2012,[11] the most pertinent to this discussion being RAI 32.

15.2 I can précis RAI 32 being that it centres on whether, even if SCE proposed to operate Unit 2 not above 70% RTP, does the operating license and its integral TS still require the plant to demonstrate safe and compliant operation at full power rating (RTP)?

15.3 The technical and engineering components of this question are relatively straightforward, being:

15.4 SONGS Unit 2 TS 3.4.17 requires that steam generator structural integrity be maintained in Modes 1, 2, 3, and 4 (Power Operation, Startup, Hot Standby, and Hot Shutdown, respectively).[12]

15.5 The structural integrity performance criterion is described in SONGS Unit 2 TS 5.5.2.11.b.1 as follows:[13]

11 NRC San Onofre Nuclear Generating Station, Unit 2 – *Request for Additional Information Regarding Response to Confirmatory Action Letter* (TAC No, ME9727), Adams ML12361A065 December 26 2012

12 Maintaining SG tube integrity requires all steam generator tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program. The tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for LCO 3.4.17. There is also the surveillance requirement (SR) 3.4.17.1 that tube integrity shall be verified in accordance with the Steam Generator Program.

15.6 *“. . . 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, 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. . .”*

my ***emphasis***

15.7 The present unamended SONGS Unit 2 operating license states that SCE *‘is authorized to operate the facility at reactor core power levels not in excess of full power’*, which for Unit 2 is 3,438 megawatts thermal – RTP.

15.8 The SCE's operational assessment (OA) that evaluated tube degradation caused by mechanisms **other than** tube-to-tube wear ¦p15, ¶4¦,[14] concluded that *‘there is reasonable assurance that the SIPC and AILPC*[15] *for non-TTW will be satisfied for 18 months at 100% power’*.

15.9 In other words, this SCE OA reckons that the performance criteria of TS S.S.2.11.b.1 for the non tube-to-tube wear at the AVB-, TSP- and RB-to-tube localities will be met if Unit 2 were to operate for a full fuel cycle of about one and one-half effective full power years at 100% reactor power.

15.10 However, each of the three other SCE OAs[9,16,17] evaluated tube-to-tube wear (TTW) but these only considered structural integrity requirements at 70% RTP and **not** at 100% RTP. For example, the *Tube-to-Tube Report* concludes ¦p117,¶10¦[9]

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13 Songs Unit 2 *Technical Specification*.

14 Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," Appendix A, Revision 2, "SONGS U2C17 Outage -Steam Generator Operational Assessment," prepared by Areva NP Inc. Document No. 51-9182833-002 (NP), Revision 2), October 2012. (ADAMS Accession No. ML 12285A267)

15 SIPC - Structural Integrity Performance Criterion AILPC - Accident Induced Leakage Performance Criterion.

16 Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," Appendix C, "Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16," prepared by Intertek APTECH for AREVA, Report No. AES 12068150-2Q-1, Revision 0, September 2012. (ADAMS Accession No. ML 12285A269)

15.11 *". . . A 70% operating power level returns the Unit 2 steam generators to within the operational envelope of demonstrated successful operation of U-tube nuclear steam generators relative to in-plane fluid-elastic stability. Operation at 70% power assures in-plane stability (SR<1) without dependence on any effective in-plane supports for U-bends. Without the development of in-plane instability no TTW will occur and thus structural and leakage integrity requirements are met. . ."*

15.12 Therefore, the current, unamended OS SCE has not demonstrated compliance with TS S.S.2.11.b. for TTW at RTP. In other words, to meet with the CAL SCE has to rely upon compensatory measures, what it styles as an *'administrative limit'* (limiting reactor power to 70% RTP) which, to my understanding, requires an amendment to the current Operating License for the San Onofre Unit 2 nuclear plant.

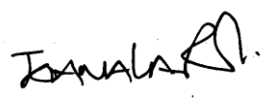
16 CONCLUDING REMARKS

16.1 In this 2nd Affidavit I have addressed the few substantive issues and topics that I consider germane to the technical and engineering reasoning present in my 1st Affidavit.

16.2 I find nothing, either in principle or of significant detail, that cause me to change any aspect or detail of the substantive or, indeed, any other of the issues and topics presented in my 1st Affidavit.

16.3 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.

Executed on 19 February 2013.



JOHN H LARGE
CONSULTING ENGINEER
LARGE & ASSOCIATES, LONDON

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17 Attachment 6 to Reference 1, "SONGS U2C 17 Steam Generator Operational Assessment," Appendix 0, "Operational Assessment of Wear Indications In the U-Bend Region of San Onofre Unit 2 Replacement Steam Generators," prepared by Westinghouse Electric Company LLC, Report No. SG-SGMP-12-10, Revision 3, October 2012. (ADAMS Accession No. ML 12285A269)

ADDENDUM I

SONGS Unit/RTS Plan Action #/Corresponding CAL Action #	Description (from RTS Action Plan Commitment List1)	Examples of Relevant Authority or Licensing Information
RTS Plan 3 **CAL Action 2** **Prior**	**Prior to entry of Unit 2 into Mode 2, SCE will, in a joint meeting, provide the NRC the results of our 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.**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity,** **TS 5.5.2.11, "Steam Generator Program."** **LCO 3.4.17.B (requiring Mode 5 if SG Tube integrity not maintained)** **LCO 3.0.4 (Prohibiting Mode Changes in Specified Circumstances)**
Unit 2 **RTS Plan 4** **CAL Action** **N/A**	**Both prior to and after entry of Unit 2 into Mode 2, the protocol and inspection time frames described in Action 2 above will be adjusted, as necessary, to account for the results of ongoing inspections and analyses of the causes of tube-to-tube interactions in the Unit 3 steam generators. NRC will be notified of any proposed changes to this protocol.**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity,** **TS 5.5.2.11, "Steam Generator Program."**
Unit 3 **RTS Plan 5** **CAL Action 3**	**SCE will complete in-situ pressure testing of tubes with potentially significant wear indications in accordance with the EPRI Steam Generator In-Situ Pressure Test Guidelines and will plug tubes in accordance with those guidelines**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity,** **TS 5.5.2.11, "Steam Generator Program."**
Unit 3 **RTS Plan 6** **CAL Action 4**	**SCE will plug all tubes with wear indications in excess of SGPR and EPRI guidelines as well as perform preventive plugging or take other Corrective Action to address retainer bar-related tube wear in Unit 3.**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity, SR 3.4.17.2 (verify tubes are plugged when criteria met) |** **TS 5.5.2.11, "Steam Generator Program."**
Unit 3 **RTS Plan 7** **CAL Action 5**	**SCE will determine the cause(s) of tube-to-tube interaction and implement actions in accordance with the Corrective Action Program to prevent recurrence of loss of integrity in the Unit 3 steam generator tubes while operating.**	**10 CFR 50, Appendix B Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity, TS 5.5.2.11, "Steam Generator Program."**

Unit 3 **RTS Plan 8** **CAL Action 6**	**SCE will establish a protocol of inspections and/or operational limits for Unit 3, including plans for a mid-cycle shutdown for inspections. The protocol is intended to minimize the progression of tube wear, and ensure that tube wear will not progress to the point of degradation that could cause tubes to not meet leakage and structural strength test criteria.**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity,** **TS 5.5.2.11, "Steam Generator Program."**
Unit 3 **RTS Plan 9** **CAL Action 7**	**Prior to entry of Unit 3 into Mode 4, SCE will, in a joint meeting, provide the NRC the results of our** **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 reasonable assurance, as required by NRC regulations, that the unit will operate safely.**	**10 CFR 50, Appendix B, Criterion XVI, "Corrective Action,"** **TS 3.4.17, "Steam Generator (SG) Tube Integrity,** **TS 5.5.2.11, "Steam Generator Program."** **LCO 3.4.17.B (requiring Mode 5 if SG Tube integrity not maintained)** **LCO 3.0.4 (Prohibiting Mode Changes in Specified Circumstances)**

ADDENDUM I

(Name)

Subscribed to and affirmed before me this 19th day of February, 2013
by JOHN HENRY LARGE
at 110 - 114 Norman Road
London England

Notary Public

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My commission expires: on death