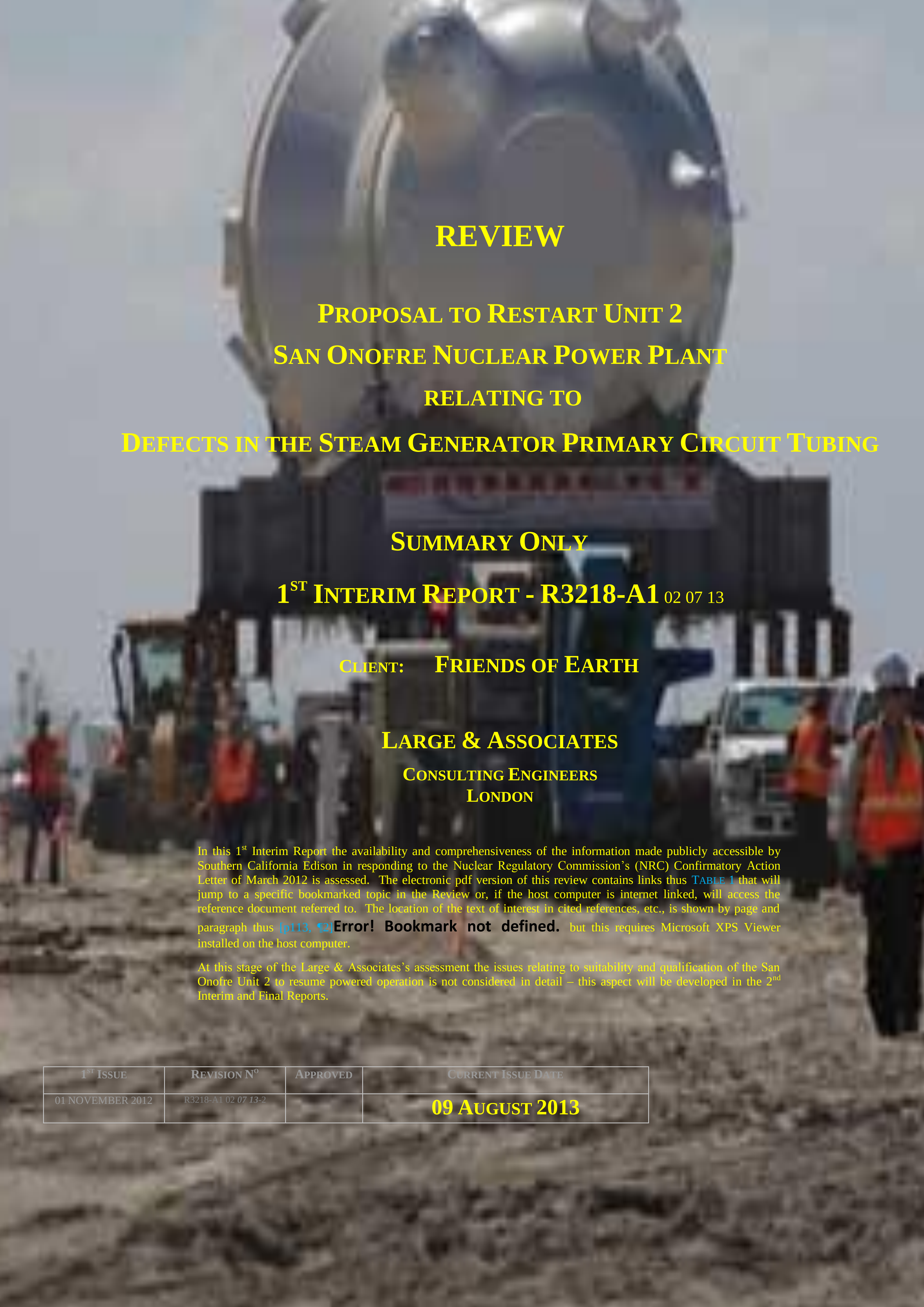

REVIEW

PROPOSAL TO RESTART UNIT 2

SAN ONOFRE NUCLEAR POWER PLANT

RELATING TO

DEFECTS IN THE STEAM GENERATOR PRIMARY CIRCUIT TUBING

SUMMARY ONLY

1ST INTERIM REPORT - R3218-A1 02 07 13

CLIENT: **FRIENDS OF EARTH**

LARGE & ASSOCIATES

CONSULTING ENGINEERS

LONDON

In this 1st Interim Report the availability and comprehensiveness of the information made publicly accessible by Southern California Edison in responding to the Nuclear Regulatory Commission's (NRC) Confirmatory Action Letter of March 2012 is assessed. The electronic pdf version of this review contains links thus TABLE 1 that will jump to a specific bookmarked topic in the Review or, if the host computer is internet linked, will access the reference document referred to. The location of the text of interest in cited references, etc., is shown by page and paragraph thus [p113, ¶2]**Error! Bookmark not defined.** but this requires Microsoft XPS Viewer installed on the host computer.

At this stage of the Large & Associates's assessment the issues relating to suitability and qualification of the San Onofre Unit 2 to resume powered operation is not considered in detail – this aspect will be developed in the 2nd Interim and Final Reports.

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PROPOSAL TO RESTART UNIT 2 SAN ONOFRE NUCLEAR POWER PLANT

COMBINED SUMMARY

This Summary combines two recent assessments undertaken by Large & Associates on the replacement steam generator (RSG) problems and proposals to restart at least one of the two nuclear power plants at the San Onofre Nuclear Generating Station (SONGS).

The first assessment, R3218-A1[1] completed in November 2012, considers the proposals of the plant operator and licensee, Southern California Edison (SCE), to restart Unit 2, operating the plant at 70% of its thermal rated power. This assessment draws on the information and data available from the heavily redacted or non-proprietary versions of SCE's submissions in response to the Nuclear Regulatory Commission's (NRC) *Confirmatory Action Letter* (CAL) of March 2012.

The second assessment was completed in the first-half of January 2013 in the form of an affirmed Affidavit[2] submitted in the current proceedings convened by NRC Atomic Safety Licensing Board (ASLB). These proceedings are judging whether the CAL should be considered to have been a de facto license amendment in account of the considerable degradation of the RSG tubing. For this assessment, access has been provided to all of SCE's substantive CAL proprietary submissions, together with sections of the SONGS operating license *Technical Specification* and *Final Safety Assessment Report* (FSAR), although a number of detailed reports and data have not been disclosed. In compliance with a non-disclosure order, only non-proprietary information of the CAL submissions is referred to in this Summary.

For brevity, sections of text, figures, etc., relied upon from the first assessment R3218-A1 are shown thus (p3, ¶6), and from the Affidavit by the paragraph number {5.4.31}. Similarly, sections of other sources referred to are show [p4, ¶6][15] with the reference citation given in the annotated footnote (F15). Text highlighted thus CAL will hyperlink directly to the reference section of text or the externally sourced document.

Large & Associates's Findings and Recommendations relating to the SCE's proposal to restart SONGS Unit 2 are shown thus.

REPLACEMENT STEAM GENERATORS

After 25 years of operation, the SCE replaced the original steam generators of Units 2 and 3 reactor power plants. Units 2 and 3 are virtually identical pressurized water reactors (PWR) built by Combustion Engineering (CE) and commissioned in 1983 and 1984; each original plant included two steam generators (SGs), both containing about 9,400 thin-walled tubes.

The four MHI replacement steam generators (RSGs) were installed and commissioned into service in April 2010 and February 2011 in the Unit 2 and 3 plants respectively. The RSGs increased the number of 0.75 inch diameter by 0.043 inch wall thickness tubes to about 9,700 with the design differing substantially in a number of important respects to the original CE SGs, particularly in the spatial distribution of the increased number of tubes in the central zone of the tube bundle, the change of tube material from Inconel 600 to Inconel 690, and the restraint means of the individual tubes and tube bundle in the top or U-bend region of the RSG.

UNIT 3 TUBE LEAK AND SUBSEQUENT INSPECTIONS

On January 31 2012, while the Unit 2 fuelling outage was in progress, the virtually identical Unit 3 was forcibly shut down when an alarm alerted SCE operators that a breach had occurred with reactor primary circuit water leaking across the RSG tube interface to the secondary steam circuit. This leak emanated from a single tube, although subsequent post-shutdown and reactor cool-down, non-destructive inspection of all of the tubes in the Unit 3 RSGs revealed very significant rates of tube wear in both RSGs. For example, each Unit 3 RSG exhibited approximately 5,000+ indications of wear localities, with many tubes having wear indications at more than one

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1 Large & Associates, Review - Proposal to Restart Unit 2 San Onofre Nuclear Power Plant Relating to the Defects in the Steam Generator Primary Circuit Tubing, R3218-A1, November 2012.

2 Large, John, 1st Affidavit of John H Large, United Sates 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, 14 January 2013.

locality and of differing degrees of wear severity, with a total of about 900 individual tubes affected in each RSG – the incidence of tube wear is summarized in TABLE A. Because of the depth and length of certain tube wear scars, a number of tubes were subjected to in situ hydrostatic pressure testing in March 2012, which resulted in 8 tube failures, all located in one of the Unit 3 RSGs.

There is some ambiguity about the rigor and sequencing of the SCE tube inspections of the Unit 2 RSGs: SCE stated that the first inspection, completed before the January 31 forced outage of Unit 3, found no incidence of *tube-to-tube wear* (TTW) in the Unit 2 RSGs, although it was silent about any other mode of tube wear being present. During March SCE carried out additional inspections of the Unit 2 RSGs using a *'more sensitive'* method, reporting thereafter that only two adjacent tubes had suffered shallow TTW but, again, it did not refer to the existence of other modes of tube wear. In fact, it was not until June 18 that SCE first acknowledged publicly that for Unit 2 *'12% of the tubes showed wear greater than 10% through-wall indication'*. SCE's taciturnity was not shared by NRC when much earlier, on or about February 4, it acknowledged that the 80% inspection of one of the Unit 2 RSGs had detected wear damage in over 800 tubes, although at that time little detail of the extent and nature of the wear was revealed.

It is now established that the first and additional rounds of Unit 2 tube inspections revealed about 2,000 and 2,700 tube wear indications, with a total of 734 and 861 individual tubes affected in each RSG respectively – the incidence of tube wear in Unit 2 is also summarized in TABLE A. As a result of the additional inspection of the Unit 2 RSGs, a total of six tubes required isolating (plugging) from the high pressure primary circuit because of excessive fretting wear of the thin-walled tube outer surfaces (in one instance, the tube wall thickness had reduced by 90%) and, since this high incidence and dispersion tube wear was entirely unexpected, SCE plugged a further 192 tubes as a preventative strategy, including tubes that had exhibited wear from and/or were similarly at risk of fretting from contact with *retainer* and *anti-vibration bars* (RBs and AVBs), as well as at the *tube support plates* (TSPs), all of which capture and restrain the tube bundles inside the Unit 2 RSGs. Following further analysis of the Unit 3 cause evaluation, a further 318 additional tubes were plugged in Unit 2, bringing the total tube plugging to 205 and 305 tubes in 088 and 089 Unit 2 RSGs respectively – these different modes of tube wear are listed in TABLE A and shown schematically by (FIGURES 4A, 4B, 4C).

CONFIRMATORY ACTION LETTER OF MARCH 2012

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

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

A chronology of events since the Unit 3 forced shut down of January 2012 is given in ANNEX I.

PRESENT SITUATION

The present situation (February 2013) is that SCE, NRC and FOE continue in dispute under 10CFR§2.206[3] and FOE's Petition to Intervene via the Atomic Safety and Licensing Board (ASLB).[4]

On the ASLB matter, the NRC admits in its legal Brief of Response that [p60][5] *'it has not yet fully reviewed and approved'* SCE's response to the CAL, nor has it '*reached a position on whether to approve SCE's Return to*

3 10CFR§2.206 Requests for Action to Modify, Suspend or Revoke a Nuclear License.

4 Referred to the Atomic Safety and Licensing Board via the NRC decision CL1-12-20.

Service Plan', for which it has issued a series of *Requests for Additional Information* (RAI). The ASLB legal process may be further interrupted and delayed because, on its part, SCE [¶32][6] '*has not yet decided how it will respond to the RAI'* and *SCE* [¶37][7] '*has not yet identified long-term corrective actions for the steam generator tubes for Unit 2'.*

Other sources of further information and detail of the RSG detailed design and of the design process itself have recently become available in, amongst other sources, the voluminous attachments to the SCE legal Brief of Response;[8] Senator Boxer and Representative Markey released details[19] of a MHI Root Cause Analysis RCA) report[9] hitherto undisclosed on February 6 2013; on February 7 2013 the NRC convened a public meeting on steam generator tube degradation[10] at which details of the SONGS RSG design were revealed by a number of the nuclear industry presenters, including the Project Director of MHI's nuclear plant production division; and on February 8 2013 a press report[11] claimed to reveal further details of the MHI RCA report first partially disclosed on February 6 2013.

UNDERSTANDING THE ROOT CAUSE OF THE TUBE DEGRADATION

SCE nominates the root cause of the tube degradation to be [p4, ¶6]:[12]

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

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

In fact, the most probable underlying cause of the failure is rooted in SCE's reasoning that the RSGs were to replicate the original Combustion Engineering SGs as closely as practicable. This, it argued, would enable the RSG replacement program to proceed without having to seek an amendment to the Operating License via the 10 CFR §50.59 screening process.[13]

Representative Markey summed up the §50.59 tactic of avoiding a license amendment by quoting directly from the hitherto undisclosed text of the MHI RCA report in a press article, when he refers to the conclusion of the *'Anti-Vibration Bar Design Team'* after *'numerous technical and design review meetings . . . could lead to the tubes vibrating and subsequent wear damage'*:[11]

> *". . . "The team then 'considered making changes to the design' to mitigate the problems, but decided not to implement any of them," Markey said, quoting the document from Mitsubishi Heavy Industries. "The only specific reason cited in this document for the decision not to*

5 In the Matter of Southern California Edison Company, (San Onofre Nuclear Generating Station, Units 2 and 3), NRC Staff's Answering Brief in the San Onofre Nuclear Generating Station CAL Proceeding, Docket Nos. 50-361-CAL & 50-362-CAL, January 30 2013.

6 Affidavit of Richard Brabec, 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, January 30 2013.

7 Affidavit of Richard Brabec, 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, January 230 2013.

8 In the Matter of Southern California Edison Company, (San Onofre Nuclear Generating Station, Units 2 and 3), Southern California Edison Company's Brief of Issues Referred to by the Commission, Docket Nos. 50-361-CAL & 50-362-CAL, January 30 2013.

9 Mitsubishi Heavy Industries (MHI) document entitled *Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators of San Onofre Generating Station* (Report).

10 *Briefing on Steam Generator Tube Degradation*, (Public) 13:00 hours February 7 2012

11 E&E Daily, Nuclear Energy: *Before Leak, San Onofre owner formed 'anti-vibration' team that found risks, the rejected fixes – Markey*, Anne C Mulkern, Friday, February 8 2013

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

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

implement the potential design modifications was the desire to avoid a more lengthy license amendment process at the [Nuclear Regulatory Commission]," he said yesterday via email."

The *'license amendment process'* referred to is the 10 CFR §50.59 screening option which, essentially, enables the nuclear plant operator to replace nuclear safety related components on a *like-for-like* basis without having to seek a license amendment. There are a number of sources of this resolve of SCE to almost doggedly to avoid a license amendment via 10 CFR §50.59, with examples recorded in the *Certified Design Specification* (CDS) bound to the SCE-MHI contract of September 2004, and reiterated by SCE when presenting to the NRC at a public meeting in June 2006 [Slide 8].[14]

The rigidity imposed by SCE on the RSG designers, may have resulted in an encounter, or at least a pointer that FEI would be present in the in-service RSGs as early as 2005 at a time when the RSGs were at the design stage. This is partially admitted by SCE during questioning by the NRC at the November 30, 2012 Public Meeting,[15] responded specifically on the issue of void fraction, here alluding that it had not known of the high void fraction at the early stages of the design process (ie the *'2005 timeframe'*):

". . . Werner {NRC} - "Just so we are clear the underprediction of the velocity by FIT III was not recognised - the problem of the model when it was changed from square pitch to triangular pitch a number of years ago - but the void fraction even under FIT-III while not predicting 99.6% was predicting 95% which was still high and was a matter of concern back in the 2005 timeframe – I know that still being looked that was a matter of concern a number of feasibility studies were conducted to try to lower the void fraction before the steam generators were fabricated but apparently it was not - so - we will need to understand that better as we go forward"

. . . Palmisano {SCE} - "We have as well – we have asked MHI for a better explanation of that it and we are looking at it ourselves because as you say the void fraction was high it was not predicted as high 99.5% it was high it was questioned ultimately the calculations and the operating experience showed even with that void fraction the system should have been effective it was not – clearly thats a failure several reasons for that failure that have to be dealt with."

my *{clarification}*

Indeed, more recently SCE has denied any knowledge of FEI presence in the RSG during the design stages of around 2005. For example, SCE's response analysis to FoE's Allegation [p18, Appendix 1][16] states:

" . . . At the time the RSGs were designed, MHI performed analysis that demonstrated that the ***steam*** *in any area of the tube bundles* ***would be low enough to provide the required damping****, and that the quality of the steam in the vast majority of the secondary side of the steam generators would be even less. Furthermore, MHI analyzed the potential for fluid elastic vibration, and determined that conditions were stable.*

SCE's root cause evaluation has determined that FEI did occur. However, ***SCE had no evidence of that beforehand****."*

my ***emphasis***

As previously noted, on February 6 2013 Senator Boxer[17] and Representative Markey[18] published the contents of a letter[19] addressed to Allison Macfarlane, NRC Chair, referring to the undisclosed MHI *Root Cause Analysis* report.[9] The letter, noting that *'SCE and MHI were aware of problems with the . . . replacement steam generators before they were installed . . ,'* went on:

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

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

16 Docket Nº 50-361 and 50-362 Response to Friends of the Earth 10 CFR 2,206 Petition, January 9, 2013

17 Senator Barbara Boxer (D-CA), Chairman of the Senate Committee on Environment and Public Works (EPW).

18 Representative Ed Markey (D-MA), Ranking Member of the House Natural Resources Committee.

19 E-mail of February 6 2013 with letter attached, Chairman Boxer and Rep. Markey to Allison Macfarlane NRC Chairman asking for NRC to *Investigate New Safety Concerns at Southern California Nuclear Plan*t.

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

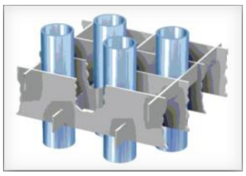
my *{addition}* and ***emphasis***

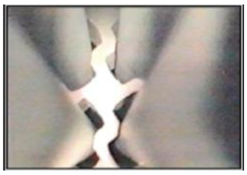
As noted previously, the latter statement referring to a more *rigorous license amendment* is referring to the 10CFR §50.59 process identifying whether new and/or replacement features introduced to a nuclear plant require amendment to the operating license. This bounding limitation on the RSG design process required the RSG designer/manufacturer MHI to conform to the overall dimensions of OSG design, facilitate the same reactor coolant circuit temperature, pressure and flow conditions and satisfy the *Technical Specification*[20] of the San Onofre Operating License. MHI hint at the strictures placed upon the design in its technical evaluation report of the tube wear [p10, Summary]:[21]

". . . {when the SONGS RSG order was placed, in September 2004} The SONGS RSGs were specified, designed and fabricated as replacements on a ***like-for-like basis*** *for the original steam generators in terms of fit, form and function with limited exceptions, and were replaced under the* ***10CFR50.59*** *rule. The CDS {Certified Design Specification} for the design and fabrication of the RSGs (SO23-617-01, Revision 3) specified the limiting design parameters and materials. Thus, replacement steam generator design with 3/4" tube diameter arranged in 1" triangular pitch, which was the same as in the original steam generators, and the larger heat transfer area than in the original steam generators, was optimal. The other parameters/materials not specified by CDS were established/ selected in the design process. The SONGS RSGs were designed and fabricated to achieve an {AVB} "effective zero gap" as required by CDS Rev. 3 in order to minimize its potential for tube wear. . ."*

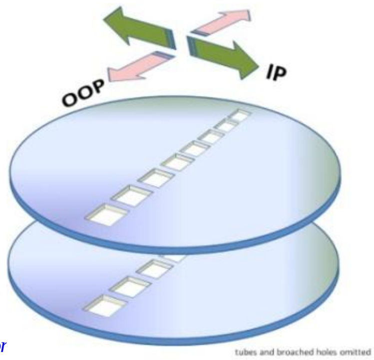
my ***emphasis*** and *{added explanation}*

However, adhering to the original OSG design involved the compromise of increasing the total tube heat transfer surface area, and hence a higher number of tubes, as a result of changing the tube material from Inconel 600 to Inconel 690 with its reduction of heat transfer coefficient of about 11%. This required a commensurate increase in the number of tubes from about 9,400 in each OSG to about 9,700 in each RSGs. Also, the physical restrictions on the overall dimensions of the RSG and, directly from this, maintaining the same secondary side flow area, required removing the tubesheet supporting pedestal or *stay cylinder* so that additional tubes could be accommodated in the central zone of the tube bundle. This meant that the cylindrical void above the tube support sheet could no longer function as in the traditional design of steam generator by providing a column of feedwater to rise up through the centre of tube bundle, thereby reducing the risk of excessively high void fraction or *dryout* in the U-bend region of the hot leg.





Another quite radical design departure from the OSG involved the replacement of the more conventional *'eggcrate'* or lattice horizontal tube supports (near right), with seven tube support plates (TSP) comprising solid steel plates broached with apertures through which the individual tubes passed (far right). This change has two significant outcomes in that compared to the *eggcrate* supports, the greater blockage to ascending flow presented by broached tube support plate, together with the loss of the flow column above the stay cylinder location, required slots to be cut across the centre *out-of-plane* direction of the tube bundle to facilitate the upward and circulation flow of feedwater (right).



Although SCE has not (publicly at least) explored how these and other smaller departures from the OSG and, generally, conventional steam

20 NRC *Attachment 1, Volume 7, San Onofre Nuclear Generating Station, Improved Technical Specification* System (RCS), c June 2010.

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

generator design, contributed the changes in the thermal-hydraulic environment of the secondary steamside circuit, very certainly the fluid flow within the RSGs was quite unique over the original Combustion Engineering OSGs in at least two important respects:

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

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

If and how the flow area and flow path changes to the steamside of the RSGs determined the *in-plane* dominance and the high void fraction has not been explained by SCE, although the high flow resistance of the broached TSPs may have resulted in a much lower circulation ratio (CR) of the steamside fluid body, so low as to produce very high void fraction (>90%) of zone of total *dryout* in the hot-leg U-bend zone, a fact that did not escape the NRC when questioning SCE at the November 30 2012 Public Meeting:[22]

> *". . . Werner - "Just so we are clear the underprediction of the velocity by FIT III was not recognised - the problem of the model when it was changed from square pitch to triangular pitch a number of years ago - but the void fraction even under FIT-III* ***while not predicting 99.6% was predicting 95%*** *which was still high and was a matter of concern back in the* ***2005*** *timeframe . . . I know that still being looked that was a matter of concern a number of feasibility studies were conducted to try to lower the void fraction before the steam generators were fabricated but apparently it was not . . ."*
>
> my ***emphasis*** and *{added explanation}*

According to Representative Markey (or his office)[11] SCE formed the *'Anti-Vibration Bar Design Team'* that was brought together *'early in the project . . . specifically for the purpose of minimizing the sort of vibration that [later] caused the recent reactor shutdowns'.* It is believed that at that time, in or about May to June 2005, SCE became more directly involved with MHI in the actual design and development of the tube bundle support structures, namely the AVBs. This SCE-MHI *'AVB Design Team . . . then 'considered making changes to the design' to mitigate the problems, but decided not to implement any of them'* meaning that it was probably tasked with, amongst other things, examining possible design changes that would reduce the high void fraction in the critical hot leg U-bend zone. Potential void fraction reduction design changes that would have been considered probably included increasing the number of AVBs, reducing the number of tube support plates and increasing the flow slot aperture size to increase the circulation ratio, and increasing the feedwater downcomer size.

The *'AVB Design Team'* feasibility study phase ran parallel with the ongoing detailed design of the tube bundle and outer shell components of the RSG and, not unlike most major engineering projects, the manufacturing and fabrication programs for these activities would have commenced relatively early in the design phase, say in or

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

about the half way through of 2005. It is possible to track back to the time[14] about when the first of the Unit 2 RSG tube bundles would have been assembled – this is in or about March 2006 – thus providing an effectively no turning back point forward of which no substantial modifications to the design could have been practicably implemented.

The Senator Boxer and Representative Markey letter[17] to the NRC, referring to the undisclosed MHI RCA report, provides insight into the difficulties facing SCE and MHI when trying to *'design out'* the problem:

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

In the February 7 press article[11] adds further by noting that the *'engineering team'* (ie the *AVB Design Team*) was *'brought together "specifically for the purpose of minimizing the sort of vibration that [later] caused the recent reactor shutdowns."'*

In other words, these latest revelations suggests that the present acknowledged situation of the strong possibility of FEI activity that gave rise to TTW in, at least, Unit 3 was known to SCE and MHI early on in the design process – it is now established that *'SCE and MHI were aware of serious problems with the design of San Onofre nuclear power plant's replacement steam generators before they were installed*.[{Boxer-Markey}]

With this knowledge, SCE and MHI set about modifying the RSG design via the *'potential changes'*[{MHI RCA}] needed to alleviate the FEI caused by the combination of high void fraction,[{NRC Werner}] together with the fact that the two-phase flow velocity was being underestimated in 2005,[{NRC Werner}] and that *'further safety modifications were found to have "unacceptable consequences" and were rejected'*,[{Boxer-Markey}] all suggest that the problem of FEI activity was unsuccessfully tackled in or around 2005-06.

The lack of success of the 2005-06 remedial design activity probably arose for two reasons:

i) First, *'SCE and MHI rejected enhanced safety modifications and avoided triggering a more rigorous license amendment and safety review process . . ."Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG design" without the requirement for a license amendment.'*.[{Boxer-Markey}]

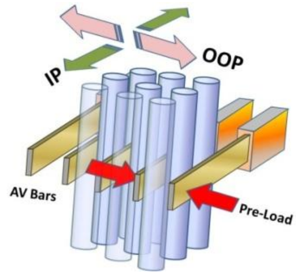
This is because MHI's hands were tied by SCE's contract requirements that the *'RSGs were specified, designed and fabricated as replacements on a like-for-like basis for the original steam generators in terms of fit, form and function with limited exceptions, and were replaced under the 10CFR50.59 rule'*,[{MHI}] and the *'only specific reason . . . not to implement the potential design modifications was the desire to avoid a more lengthy license amendment process at the [Nuclear Regulatory Commission]'.*[{Markey}]

ii) Second, it seems that during the 2005-06 design phase, both SCE and MHI failed to recognise that the RSG flow regime was dominated by the *in-plane* flow direction (now generally acknowledged by all parties) because the *'SONGS RSGs were designed and fabricated to achieve an {AVB} "effective zero gap" as required by CDS Rev. 3 in order to minimize its potential for tube wear. . ."*[{MHI}]

The irony of this second factor is that, although SCE and MHI must have known that the FIT-III was grossly underestimating the two-phase fluid void fraction, a strong contributor to fluid elastic instability and, hence, tube wear, they did not recognize that the FIT-III model also failed to predict the *in-plane* dominance of the flow regime in the hot-leg section of the tube bundle U-bend. It is now obvious that SCE and MHI put considerable, and some would opine, misguided effort into the design of the AVB assemblies (*zero-gap/zero preload*) that offered little or no restraint against tube motion in the *in-plane* direction – an effort with the outcome of accelerated and severe tube wear in the replacement SGs during in-service operation.

The TSPs (FIGURE 4A) and AVBs (FIGURES 4B and 4C) provide points of restraint that arrest tube motion. The arrangement of the TSPs and AVBs is shown by (FIGURE 3). The TSPs acting at seven locations along each of the hot- and cold-leg sections of the tube bundle capture the individual tubes, via a broached piercing which minimizes lateral and radial tube motions whilst allowing some flow continuity at that particular tube location. The twelve AVBs act to restrain the tubes in the *out-of-plane* (OOP) direction by the tubes reacting against the AV bar which, itself, reacts against the next and successive *rows* of tubes. In this way the system of sandwiched AVBs obtains stiffness and restraint via the collective inertia of the tube bundle. Normally, because the dominant direction of motion experienced in SG tubing is in the *out-of-plane* direction (that is the least stiff axis of the individual U-bend section of a tube), the restraint acting against tube motion in the *in-plane* (IP – along the *columns* of tubes) direction is considered of secondary importance.

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

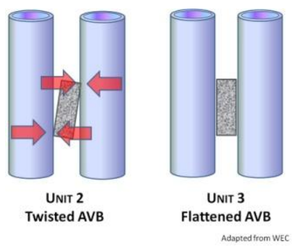


In effect, any significant *in-plane* fluid excitation force had the potential to excite individual tubes into motion in the *in-plane* direction because of the lack of *in-plane* restraint wore down the surfaces of the ABV and tube, leading to greater freedom to move in both *in-* and *out-of-plane*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

The second role of the AVB system is to curtail the free-span tube length between successive AVBs. Pinning down the individual tubes in this way, effectively chops the free-span U-bend into (here 13) short sections between the hot- to cold-leg top TSPs. This raises the OOP fundamental frequency of vibration of the tube free-span sections between each successive AVB restraint location with, in the optimum design, the resonant frequency being taken above any excitation frequency active in the fluid (turbulence, vortex shearing, etc). Even in situations where the OOP FEI is vigorous, the lower amplitude motion of the pinned short sections of free-span tube will tend to govern and inhibit tube-to-tube clashing and TTW.

The effectiveness of this second role of *out-of-plane* restraint is strongly influenced by the *in-plane* degradation of the AVB when subject to the *'bumping'* phenomena. Progressive bumping loss of AVB effectiveness along a single tube results in longer free-span tube length, a lower fundamental resonance frequency and a higher amplitude vibration for that tube, leading to tube-to-tube clashing and TTW.

Incidentally, tube *'bumping'* explains the different time periods over which AVB-to-tube wear and TTW occurred in Units 2 and 3. The incidence of TTW was delayed in Unit 2 because the tube bundle in each to the two RSG had been assembled with distorted or warped AVBs, whereas the same AV bars of Unit 3 were corrected[23] to achieve the design intent of *zero-gap-no-contact force*. The warped AV bars effectively pre-loaded ➡⬅ the AVB-to-tube contact with an *in-plane* friction force, thereby unintentionally providing the AVB with effective *in-plane* restraint that has delayed, but not halted, the advance to TTW.



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

In summary, the trail leading to the rapid degradation of the SONGS RSG tubing is somewhat more complex and deep rooted that SCE's explanation that it derived from fluid elastic instability. Very certainly SCE itself seems to have played a strong role in contributing to the failure, in two distinct ways.

First, by generally requiring that the RSGs follow the original Combustion Engineering SG design on a *'like-for-like basis'* and, moreover, stipulating that the RSG were to satisfy 10CFR §50.59 and thus not require a License Amendment. Second, by SCE's specification of the design functionality of the AVBs, via the Certified Design Specification that it issued to MHI requiring [p10, Summary],[21] to be *'designed and fabricated to achieve an "effective zero gap" as required by CDS Rev. 3 in order to minimize its potential for tube wear'.*

Apparently and according to the Boxer-Markey revelations,[11,19] SCE must have been fully aware of the uncertainties then (in 2005 to 2006) developing in the thermal-hydraulic two phase flow analysis and, from this, the demands on the AVB restraint design, although significant corrective action seems not to have been implemented during this part of the design phase of the RSGs.

The NRC *Confirmatory Action Letter* CAL required SCE demonstrate understanding of the root cause of tube degradation, its corrective measures, and in account of the 70% power level proposed, if the restart of Unit 2 is justified and of acceptable risk and tolerable radiological consequences, should further tube failures occur whilst Unit 2 is at power operation. In its response, however, SCE does not share, generally nor in detail, the detailed reasons and design failure underlying the Boxer-Markey revelations.

In detail: As preconditions to restarting Unit 2, the CAL required SCE to:

- **Determine the Root Causes of Tube-to-Tube Wear in Unit 3 RSGs**

  All of the parties involved generally agree that it was the presence of a thermal-hydraulic phenomenon known as fluid elastic instability (FEI) activity in the higher regions of the RSGs that triggered tube motion and inter-tube fretting or TTW generated wear.

  That said, the root cause of how the MHI design (using its own FIT-III software suite) permitted such vigorous levels of FEI activity has not been exactly determined, even by MHI itself which continues to be at a loss to explain which feature(s) of its analytical and/or design processes was at fault (other than the in out error over triangulation). Moreover, the RSG assembly includes a number of distinctly different design features compared to the original SGs, for example the higher tube bundle density and smaller inter-tube gaps, and in the geometry of the restraint and anti-vibration bars acting at the U-bend section of the tube bundle. However, the ways in which these and other unique design features could have exacerbated and/or been susceptible to FEI, or other fluid flow sources of excitation, has not been fully explained by SCE or its consultants, and MHI {5.5.22-28}.

  Appropriate to this incomplete and, at places, inconsistent understanding of the root causes:

  1) *Until all of the potential factors contributing to the onset of tube motion and the tube wear are fully understood then the effectiveness of any remedial action cannot be certain.*

  2) *In this respect further work should be undertaken by SCE to determine why this particular design of steam generator, coupled to the characteristics of the primary and secondary (steamside) circuits, was and may remain vulnerable to FEI – until this root cause is fully understood both Units 2 and 3 should be held at a cold shutdown state.*

  3) *MHI is still performing (November 2012) an evaluation of the potential factors that contributed to the flow velocity and void fraction disparities generated by its FIT-III software modeling – until this MHI evaluation has been completed, published and demonstrated to be sound, SCE's understanding of the mechanistic process contributing to tube-to-tube and other fretting damage remains incomplete.*

- **Determine the Causes of Other Modes of Tube Wear in Units 2 RSGs**

  As previously noted, at first SCE only reported on TTW in the Unit 3 RSG, making no reference to the other modes of tube and component wear found in the Unit 2 and 3 RSGs.

  MHI considered that FEI activity was suppressed in the region of the TSP localities and that this mode of tube wear arose from cross-flow induced random vibration; the AVB-to-tube wear was also related to random vibration of the tubes which was exacerbated in some cases by twisting and distortion of the AVB itself; and the retainer rod-to-tube wear arose because of the flow induced vibration of the retainer rod abrading directly against the outer row of tubes (ie no tube motion), perhaps with this contact being brought about by the thermally-related distortion (flowering) of the tubes in the U-bend region of the bundle.

MHI's analysis of the TSP, AVB- and RB-to-tube wear is attached to the SCE response to the NRC CAL, but in its précis of MHI's work SCE omits to echo the findings buried in the heavily redacted MHI Appendix 10. One such MHI finding is that AVB-to-tube wear in Unit 2 arose in FEI inactive areas of the RSG tube bundle, it being excited by turbulent flow forces (vortex shedding, turbulent wake, etc) which may persist even when FEI is suppressed by the proposed reduction to 70% thermal power.

There remains acute disagreement between SCE's consultant AREVA and the RSG designer MHI as to the cause of the AVB wear and, particularly, how this will apply to Unit 2 if restarted {5.7.7-11}. AREVA claim that if the reactor power level is reduced to 70% then the cause of continuing AVB wear, *in-plane* FEI will cease, so AVB *in-plane* effectiveness will also cease to decline further, and TTW will be arrested. To the contrary, MHI reckons that the AVB wear is driven by random flow perturbations and not FEI, so the decline in AVB *in-plane* effectiveness will continue to advance even at the reduced 70% power level, thereby freeing-up longer free-span tube sections that are more susceptible to TTW.

4) *The MHI analysis considering TSP, AVB and RB wear modes for both Units 2 and 3 should be openly available in unredacted format; and, if appropriate,*

5) *the mechanisms and flow regimes inducing TSP, AVB and RB tube and component wear modes should be subject to benchmark simulation at the appropriate scale.*

- **Determine the Tube Motion Process**

Another area of considerable concern, but which is yet to be resolved, is the phased process through which the tube restraint systems evolve to become susceptible to tube motion and, as a result of this, tube-to-tube wear. It is now reasoned that the high rate of tube-to-tube wear is preceded by a period whilst the newly manufactured and tightly packed tube bundle *'slackens off'*. This general slackening off of the tube bundle progresses as certain of the AVB assembly retainer bars wear and, separately, as the contact/gap relaxes between the AVBs and individual tubes, the restraint conditions particularly in the U-bend region of the tube bundle, drift into a quasi-stable condition {5.7.6-7 & 5.7.33-35}. This re-conditioning of the tube bundle restraint systems provides conditions conducive for FEI tube motion to be triggered. The period over which this *slackening off* occurs is believed to relate to the initial inter-component tolerancing and, particularly, the more effective flattening of the AV bar distortion at the time of manufacture for Unit 3 compared to Unit 2, which possibly explains the lack of tube-to-tube wear damage, although high incidence of restraint and anti-vibration bar fretting and wear in the Unit 2 RSGs compared, because of its tighter assembly tolerances (ie the AV bar distortion was more effectively flattened out and, hence, less *in-plane* AVB-to-tube preload), to the high tube-to-tube wear incidence in Unit 3.

6) *SCE's argument that the different pattern of tube wear between Units 2 and 3 to be solely the outcome of different assembly tolerancing is unconvincing and not substantiated – not only does SCE overly rely upon the low incidence of TTW (2 TTW indications in Unit 2), but it ignores the strikingly similar pattern and rates of AVB-to-tube wear in all four Unit 2 and 3 RSGs that strongly suggest that AVB-to-tube wear is part of a transient process that leads to enhanced TTW.*

Comparison of the nature and advancement of the wear of various components in the Units 2 and Unit 3 RSGs,see [p24-26, Figures 3.2/3/4] shows strong correspondence in the first *slackening off* phase, with Unit 2 lagging along a virtually identical path of component wear to that of Unit 3, so much so that it suggests that before it was shut down for refueling Unit 2 was entering the phase in which high rates of TTW could be expected.

7) *Clearly, there is second factor that determines the phasing of the wear mechanisms but this has not been investigated further than the exploratory analysis of the Intertek APTECH Operational Assessment - until this wear phasing is fully understood the remedial measures to prevent further component wear cannot be considered to have been finalized.*

SCE's consultant AREVA recognizes that the spread of TTW through the high FEI active zone in the tube bundle relates directly to the number of AVB restraints that have been rendered in-plane ineffective, although it is unable to quantify the rate at which additional AVB restraints will be rendered ineffective [p44, ¶4][24]{5.7.15-17}.

8) SCE's application to restart Unit 2 centers around its claim that unlike Unit 3, there is little (tolerable) TTW incidence in Unit 2 – this line of reasoning gives no cognizance to the high incidence (like Unit 3) of AVB, RB and TSP wear also present in Unit 2, and that this component wear is likely to be a portend of entry into a phase of tube-to-tube wear.

In January 2012 a NRC *Nuclear Notification* was raised because of unexplained alarms triggered by 30 or so acoustic activity events in the Unit 3 RSGs. The subsequent evaluation concluded that the installed vibration and loose part monitoring system was obsolete and, particularly, that the sensors were too remote for reliable monitoring of the upper U-bend region for FEI activity. For the sake of argument, should Unit 2 re-enter operational service at a derated power level, then this should be conditional upon

9) SCE should be required to install a reliable RSG monitoring system, particularly with capability to closely monitor the upper region of the RSG tube bundles – this should be a mandatory requirement and the monitoring system should be shown to be effective and reliable before Unit 2 is permitted to restart.

It is of interest to note here that little or nothing was done in response to the unexplained alarms in Unit 3 over the 11 months of in-service operation before the forced shutdown. This strongly suggests that it was not only limitations in the installed monitoring equipment but, also, shortfalls in the management approach by SCE and the degree of regulatory oversight by the NRC.

- **Implement Remedial Measures to Unit 2 RSG**

At this time the proposal is to preventatively plug a zone of tubes to act as a buffer to protect those tubes believed to be most susceptible to FEI activation and wear. The uncertainty here is that Unit 2 has little TTW incidence (ie it had yet to finish *slackening off*), so there are few clues for the appropriate placing of the preventatively plugged tube buffer zones. At its best, preventative plugging is somewhat of a *hit-and-miss* approach, so the added uncertainty of extrapolating further from just the wear of the restraint components piles uncertainty on uncertainty.

10) SCE should undertake and publish further studies with aim of reducing the present uncertainty of locating the most appropriate preventively plugged tube buffer zones – there are clearly difficulties with the computer modeling so, importantly, this further substantiation work should be supported by a program of hydraulic trials that simulate real-time thermal-hydraulic conditions in the Unit 2 RSG at 100% and 70% thermal power rating fluid flow and void fraction distributions.

Quite distinct from FEI triggered tube motion, it is clear that both AVBs and the retainer bar (RB) assemblies were susceptible to flow-induced vibration, identified by MHI to arise between the low resonance frequency of the bracing retainer bars interacting with the turbulent secondary fluid flows leaving the tube bundle. The component motions excited by this fluid interaction resulted in quite harsh wear rates between the tubes and the retainer bars, at the AVB and, possibly, at the TSP localities.

11) Much of the remedial work has focused on the FEI activity and TTW and a similar level of analysis and remediation action now needs to be given to measures that would effectively dampen out further flow induced vibration and wear to the anti-vibration bar assemblies – it is not clear, nor has it been demonstrated by the OAs that the AVB and TSP wear modes correlate directly to levels of FEI activity and, instead of or as well, other fluid excitation mechanisms may be in play – this work should be undertaken in conjunction with Recommendation 8) foregoing.

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

These other forms of fluid excitation of tube motion may not necessarily weaken with a reduction in power rating and, indeed, such may be entirely independent of the dynamic pressure cross-flow interaction ($\sim\rho v^2$), as exampled by the RB sympathetic resonance vibration found in both Units 2 and 3.

12) Particular regard will need to be given to understanding potential excitation fluid mechanisms such as vortex shearing, turbulent wake, etc., as these apply to the particular component designs of the SONGS RSGs, and the fluid and two-phase fluid flow distributions within and across the tube bundles at the proposed reduced power level.

In October 2012 MHI communicated directly to NRC that the *'tube wear adjacent to the retainer bars was identified as creating a potential safety hazard'*, recommending that the retainer bars that have the possibility to vibrate with large amplitude should be removed {5.6.6-12}. This restraint system locates the anti-vibration bars in place during normal operation and, also, serves to contain the tube bundle geometry during a main line steam break (MSLB) design basis event.

13) This ad hoc approach to managing nuclear safety risk is entirely unsatisfactory – modification to any component of a nuclear system, especially one providing a safety function during a design basis event, should be evaluated in terms of the plant as a whole rather than, as it seems to be the case here, in isolation to other components of and without regard to the overall safety function of the RSGs themselves.

If the quite fortuitous event of the Unit 3 in-service tube leak had not occurred whilst Unit 2 was shut down for refueling then the likelihood is that, since all of the Unit 2 RSG tubes had satisfactorily passed the first inspection (since it was not reported otherwise by SCE), Unit 2 would have been returned to full power operation without any modification being undertaken to its RSG tube bundles. It was only the second inspection of Unit 2, that seems to have been prompted by the Unit 3 tube leak and the subsequent acknowledgement of the high tube wear incidence, first by the NRC and the months later by SCE, that held back Unit 2 from returning to full power operation.

14) The failure of the first Unit 2 inspection (in January 2012 prior to Unit 3's forced shutdown) to detect (or at least publicly acknowledge) any significant incidence of tube wear damage should be considered most unsatisfactory, particularly in that based on the findings of the first inspection alone, Unit 2 would have been returned to full power operation without any of the preventative plugging and (now proposed) reduction in the power rating required for its (claimed) safe operation.

It is not clear why, particularly in account of the high alarm rate generated in Unit 3 [see *9)* above] and, obviously, because this was the first operational spell for both Unit 2 and Unit 3 RSGs, that the earliest opportunity for tube inspection (of Unit 2 RSGs) was so *'limited'* and incapable of detecting the Unit 2 TTW and high incidence of tube wear in the anti-vibration bar localities. Indeed, if the Unit 2 damage had been acknowledged immediately after the first inspection then, on this basis alone, Unit 3 would have been shut down immediately, that is before the tube leak developed. Again, this strongly suggests poor management by SCE and lax oversight by NRC.

15) The ambiguity over the reporting of the findings of the first inspection of the Unit 2 RSGs by SCE, which seems to be contradicted by the NRC February 4-5 statements, should be resolved via publication of the 10 CFR TS 5.5.2.11a) 'as found' condition evaluation report submitted by CSE.

- **Operational Limits**

The NRC requires SCE to specify a power level at which the plant is to operate for a period of 150 days before shutting down for inspection of the RSGs.

SCE has nominated Unit 2 to operate at 70% of its full thermal power rating. It has made this nomination in the absence of any (published) justification and, similarly, there is no assessment of the Unit 2 nuclear plant performance, and any nuclear safety implications thereof, that might arise as a result of continuous running at this reduced power level.

Again, for the sake of argument, should Unit 2 re-enter operational service at a derated power level, then this should be conditional upon

16) *Very certainly, a nuclear safety case should be prepared and made publicly available before this new regime of nuclear operation is implemented – in this respect alone the Final Safety Analysis Report (FSAR) should be updated.*

17) *Similarly, the Technical Specification (an integral component of the Nuclear Operating License) should be comprehensively reviewed and revised where appropriate – this is particularly relevant to those parts of the Steam Generator Program dealing with tube integrity because the types of tube wear, particularly at the AV Bar-to-tube contacts points, appear to be unique for which a tube failure data base has not been established.*

- **In-Service Period**

Note that although the FEI activity will, on average, be reduced at 70% power, there can be no guarantee that some level of FEI will not exist from the onset of restart or develop thereafter to initiate tube motion and progression of tube wall wear.

In its arguments for restarting Unit 2 at 70% reduced power level, SCE claim that the independent *Operational Assessments* demonstrate that Unit 2 will be able to operate for 16 to 18 months without significant tube wear – these claims are shown in the SCE Table 3-1 reproduced here [p19, Table 3-1]:[25]

SCE TABLE 3-1: OA Approach and Results Comparison source SCE

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

The claim that tube-to-tube (TTW) wear rates will be tolerable at 70% power rating (ie an in-service period of 16 to 18 months) is based on SCE's unwarranted assumption that FEI activity will be so reduced through the tube bundle that the free-span unplugged tubes will no longer be susceptible to debilitating rates of tube-to-tube wear.

However, this claim is based on the assumption that FEI susceptibility will be entirely eliminated at the reduced 70% power level, whereas the AREVA Operational Assessment reasons that FEI induced tube wear could also be expected to occur at 70% power. In AREVA's opinion, the intervention of FEI at proportionally reduced vigor for the intended 70% power rating will lead to, in the worst subset case, exceeding the tube bursting integrity criteria within 2.5 to 5 months once that the FEI tube wear process has initiated. To this time-to-burst period is added a handsome (and somewhat unjustified) *slackening-off* period of 3.5 months, so the overall time-to-burst is between 6 to 8.5 months which is uncomfortably close to the 150 days (5 months) required by the NRC for Unit 2 to be shut down for inspection of its RSGs {5.8.16}.

The extremes of the range of tube wear burst time extracted from the Operational Assessments is quite contrary to SCE's Table 3-1 results above:

CASE	SLACKENING OFF PERIOD	TUBE WEAR PERIOD	TIME TO BURST
U3	7	2.5 to 11	9.5 to 18
U2	3.5	2.5 to 11	6 to 18
$U2^{dynamic}$	3.5	2.5 to 5	6 to 8.5
$U2^{static}$	3.5	4.5 to 8	8 to 12

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

Even so, the protraction of the time-to-burst period by the addition of a nominal *slackening off* period is unjustified: first because its mechanism is not sufficiently understood for even a broad band of time periods to be allocated to it and, second, there is no reliable means of determining just what stage of the *slackening off* process had been reached in each of the Unit 2 RSGs at the time of the Unit 2 refuel shutdown.

18) *In view of these uncertainties the time-to-burst period should be conservatively assumed at 2.5 months and not at SCE's claimed 16 to 18 months – this minimal period of 2.5 months is well short of the 5 months continuous running period proposed for Unit 2 before shutdown for inspection***.**

Overall, in respect to the CAL Unit 2 restart prerequisites: SCE has not demonstrated a comprehensive understanding of the causes of all modes of tube wear in Units 2 and 3; its understanding of the tube motion processes, particularly the phasing of wear to the various tube bundle restraint systems (TSP, AVB and RB) preceding the onset of TTW is patchy; and it seems to have turned blind eye to the findings of one of its Operational Assessment consultants that the 2.5 month time-to-burst for certain tubes is well short of the NRC required 150 day period at power before the first in-situ inspection of the Unit 2 RSGs.

Information Availability: Other than the NRC Augmentation Team's first and final reports, SCE holds a virtual monopoly on all other information relating to the design and performance of the RSGs, the inspection records of the tube bundles and the subsequent analysis undertaken by itself and its consultants.

- **Comprehensiveness and Public Accessibility of SCE's Evaluation**

(APPENDIX II) lists a sample of documents containing information, evaluations and data that remain out of the reach of members of the public. The Operational Assessments that SCE deploys in justification of its proposal to restart Unit 2 are themselves so heavily redacted to the extent that often it is not possible to follow through the thread of the argument being presented.[for example see [p58-65, Attachment 4]]

19) *The redaction and non-availability of documents and data on the basis of the need to safeguard proprietary information should be subject to independent scrutiny and, in account of the high level of public concern, it should assumed that it is in the public interest that unfettered disclosure outweighs the need to protect propriety and/or trade information.*

For preparation of the Affidavit,[2] the ASLB granted access to the proprietary (unredacted) versions of the documents that SCE had submitted in response these are listed in (APPENDIX II). The detailed content and information contained within the unredacted SCE submissions are bound by a non-disclosure agreement so, it follows, it is not permissible to present any comment and/or finding on the general nature and management of what has and what has not been redacted from the non-proprietary versions of the SCE CAL documents.

That said, the scope and detail of the content of the Boxer-Markey extracts[19] of the MHI RCA leaked report when compared to the non-proprietary (redacted) SCE CAL documentation is revealing in the broadest sense.

In general: As previously noted, this interim and first reporting stage of the Review examines the comprehensiveness and public accessibility of the SCE submissions. In other words, has SCE undertaken a sufficient probing of the RSG faults to understand the root cause(s) and, if it has, has it demonstrated its own confidence that a restart of Unit 2 will be free of exceptional risk of intolerable radiological consequences?

- **Assurance that Unit 2 will Operate Safely**

SCE *SONGS Return to Service Report* of October 3 2012, together with its enclosures and attachments, provides the basis for the CAL inspection and NRC approval for the Unit 2 restart.

There are a number of important shortfalls in SCE's reasoning and justification for the restart of Unit 2, these arise on:

20) *SCE's lack of understanding of the root cause of the excessive tube failures and wear in the RSGs;*

21) *the inability to be able to realistically locate SCE's preferred scheme of preventatively plugging tube buffer zones in Unit 2;*

22) *the inadequate time margins between the proposed restart operating period of 5 months to shutdown and inspection, compared to the AREVA worst case projected tube burst failure of 2.5 months;*

23) *SCE's apparent lack of safety analysis of operating the Unit 2 nuclear plant at a continuous level of 70% thermal power rating; for which*

24) *particular regard should be given to the resilience of tubes at the TSP and TT locations of tube wear that are susceptible to pop-through failure when subject to MSLB design basis event conditions – this accident induced performance criteria (ie pressure + external forces) should be applied in addition to the structural integrity performance criteria (pressure alone).*

Moreover, because the SONGS RSG root cause of the failure, so far as this is understood at this present time, is unique and unprecedented, particularly in respect the number of tube wear incidences and the rapidity at which these have occurred, together with the now acknowledged MHI's flawed computer modeling at the RSG design stage, this

25) *invalidates SCE's over-reliance upon data and experience from other operating nuclear plants and, general, adaptation of computer modeling for its understanding and solution of the SONGS RSG failures – this approach, which might be best described as selective 'pick-and-mix', must be thoroughly substantiated and endorsed by physical benchmark testing.*

These shortfalls, omissions and lack of clear understanding of the SONGS RSG issues by SCE strongly advocate that permission to restart Unit 2, even at lower power rating and with an interim shut down inspection, should not be permitted to proceed – in short, restarting Unit 2 with so much uncertainty must be accompanied by a strong measure of experimentation.

In summary: Although this Review is at an interim stage, nevertheless, the work of the Review so far has established sufficient confidence in the following findings and observations:

- **Future Nuclear Power Operations at San Onofre**

  The design geometry of this type of U-bend steam generator severely limits access to all but the outer peripheral tubes of the 9,700 or so individual tubes comprising the tube bundle, there being no facility enabling inner tubes to be practicably in-situ repaired or replaced. The only practicable remedial action available is to isolate the damaged tubes from the reactor primary circuit by plugging. To protect these plugged tubes, and other pressurized in-service tubes that are susceptible FEI induced motion, from further wear it is necessary to create zones of preventatively plugged tubes to dampen out the FEI activity. Similarly, because of the compact design, there are similar difficulties in repairing and/or modifying the various tube restraint components that insert into the depths of the tube bundle, such as retainer bars, tube support plates and anti-vibration bars which, themselves, are also subject to FEI and other flow induced wear and fatigue phenomena.

  The Unit 3 inspection has shown a large number of tubes had worn to levels approaching the limits at which the tubes would have to be withdrawn from pressurized service by plugging. At this time nothing substantial has been published on any remedial scheme that would arrest the more advanced tube wear mechanisms and incidence within the Unit 3 RSGs and, moreover, doubts have been expressed on whether it would practicable and economically viable to do so.

  It follows that the wear incidence in the Unit 2 RSGs strongly suggests that a restarted Unit 2, even at a reduced power level, would continue along the same tube degradation processes experienced by the virtually identical Unit 3 RSGs. So, like Unit 3, a restarted Unit 2 would operate at much the same levels of nuclear safety risk associated with the heightened potential for in-service multiple tube failure. Also, reduction of thermal capacity associated with the need to progressively increase the numbers of preventively plugged tubes throughout the RSG life cycle, the more frequent inspections and greater outage times, and the loss of thermal efficiency arising from running continuously at lower power levels, will all further reduce the thermal, electrical generating and economic performance of the nuclear power plant.

APPENDIX I

TABLES, FIGURES AND DIAGRAMS

Table 6-1: Steam Generator Wear Depth Summary

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

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

TABLE A TUBE WEAR DEPTH SUMMARY – UNITS 2 AND 3

Source Table 6-1

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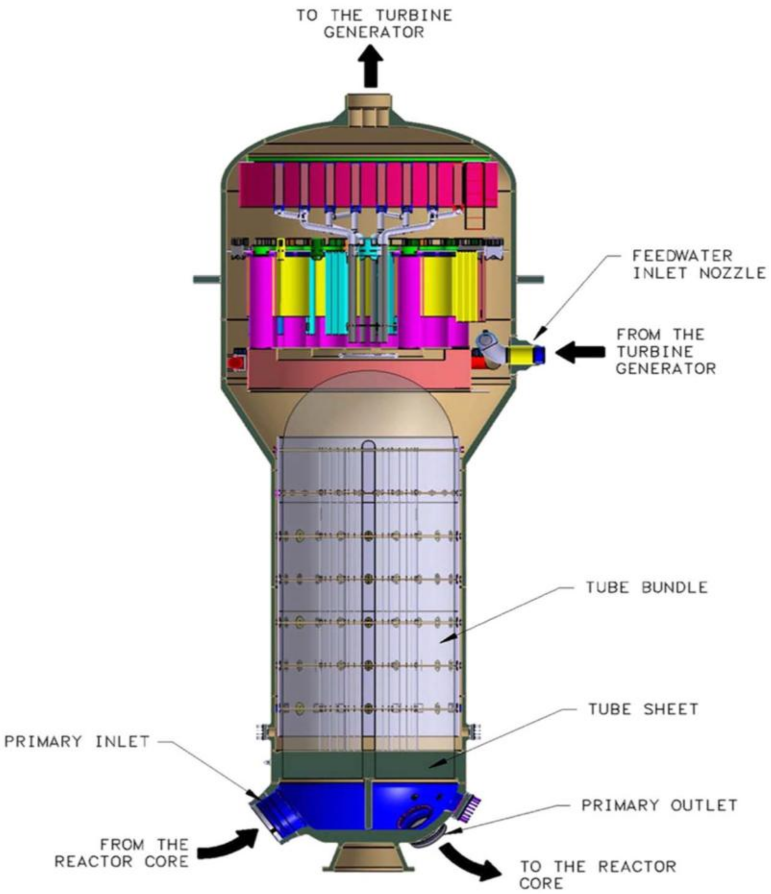

FIGURE 1A MHI REPLACEMENT STEAM GENERATOR

Source f**Error! Bookmark not defined.**

		PRIMARY CIRCUIT	SECONDARY CIRCUIT
OPERATING TEMPERATURE	T_{hot} °F T_{cold} °F	598 541	523 422 water
OPERATING PRESSURE	lbf/in^2	2,250	833
FLUID FLOW	gallons/lbs pm	209,880 water	126,470 steam
TUBE BUNDLE HEIGHT	ft	31.75	
N° OF TUBES		9,727	
TUBE DIAMETER/WALL THICKNESS	inch	0.750 - 0.0429	
TUBE PLUGGING MARGIN	sleeving repair not permitted	8%	
OVERALL HEIGHT	ft	65.5	
OUTER SHELL DIAMETER	UPPER ft LOWER ft	22 14	
OPERATING WEIGHT	TONS	691	

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FIGURE 2A MHI REPLACEMENT STEAM GENERATOR IN TRANSIT

Source SCE

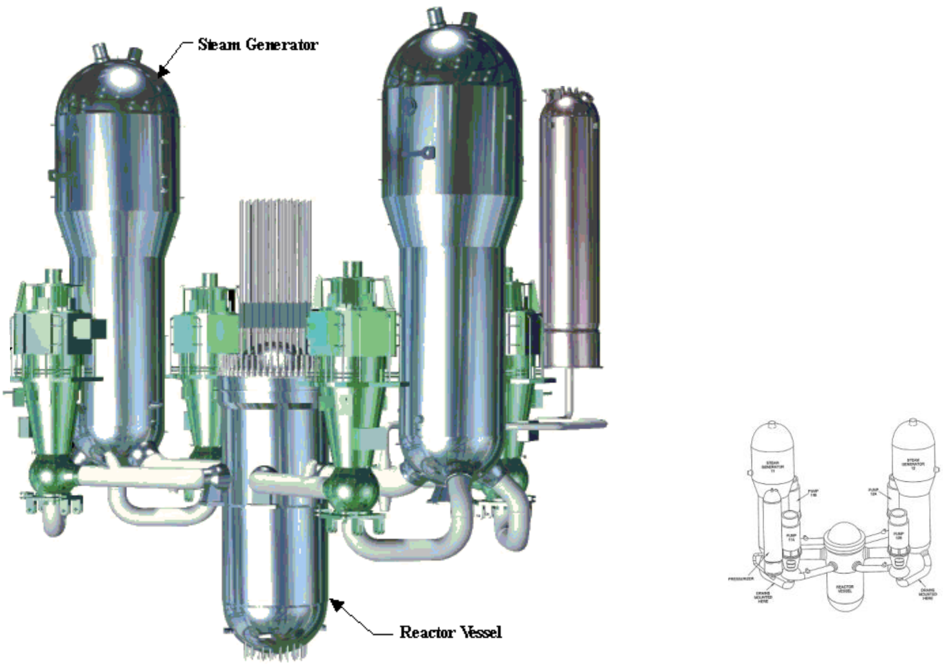

FIGURE 2B TYPICAL TWO-LOOP COMBUSTION ENGINEERING UNIT

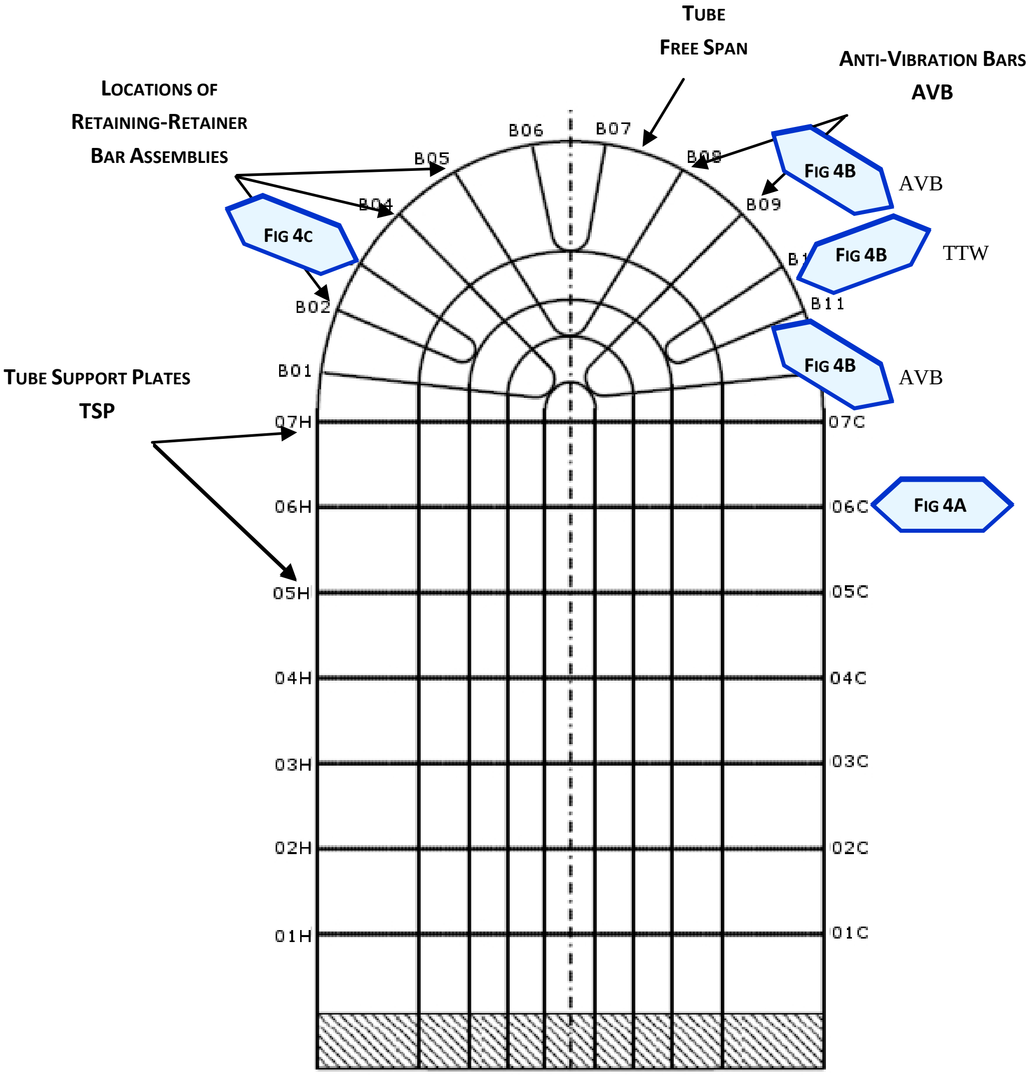

FIGURE 3 RSG COMPONENT LOCATIONS

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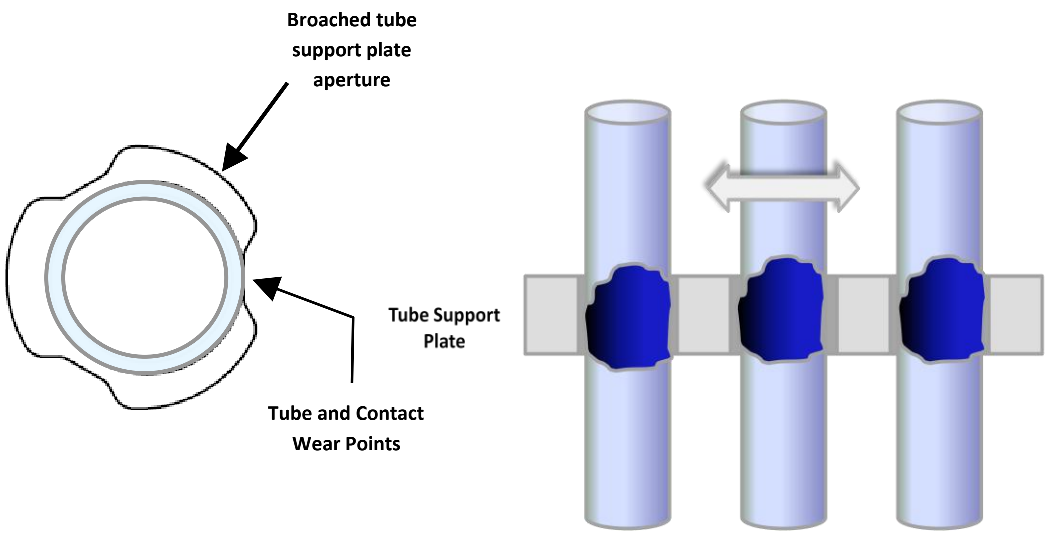
FIG 4A Typical locations of FIGURES 4



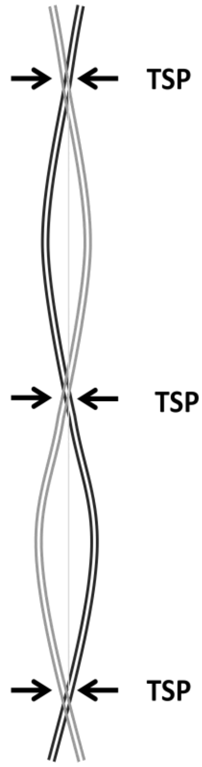


FIGURE 4A TUBE SUPPORT PLATE-TO-TUBE WEAR

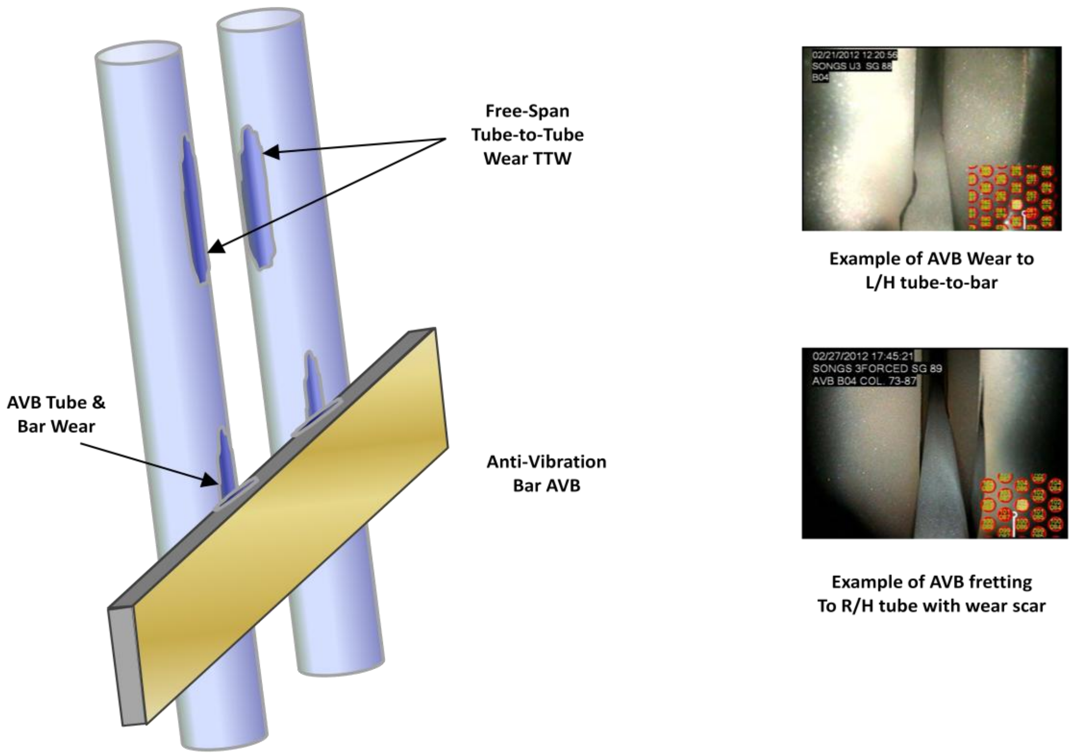


FIGURE 4B AVB WEAR & AVB-TO-TUBE WEAR

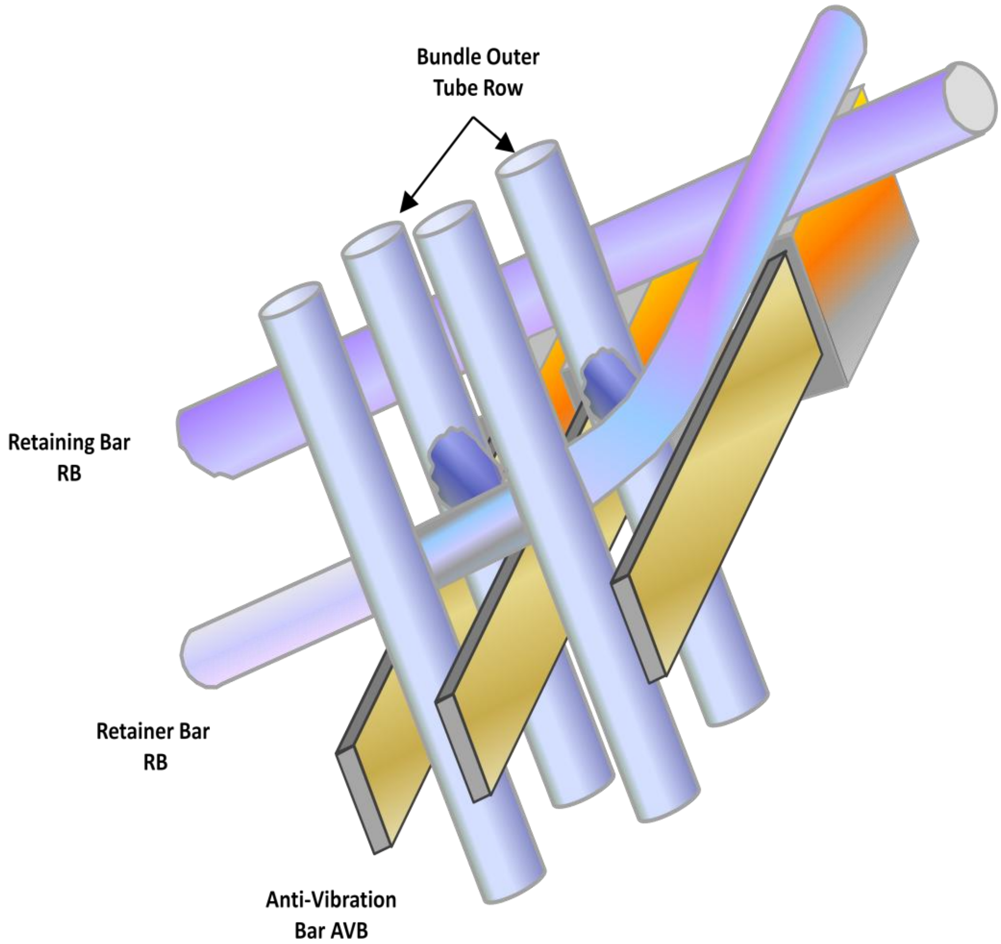


FIGURE 4C AVB ASSEMBLY RETAINER BAR-TO-TUBE WEAR

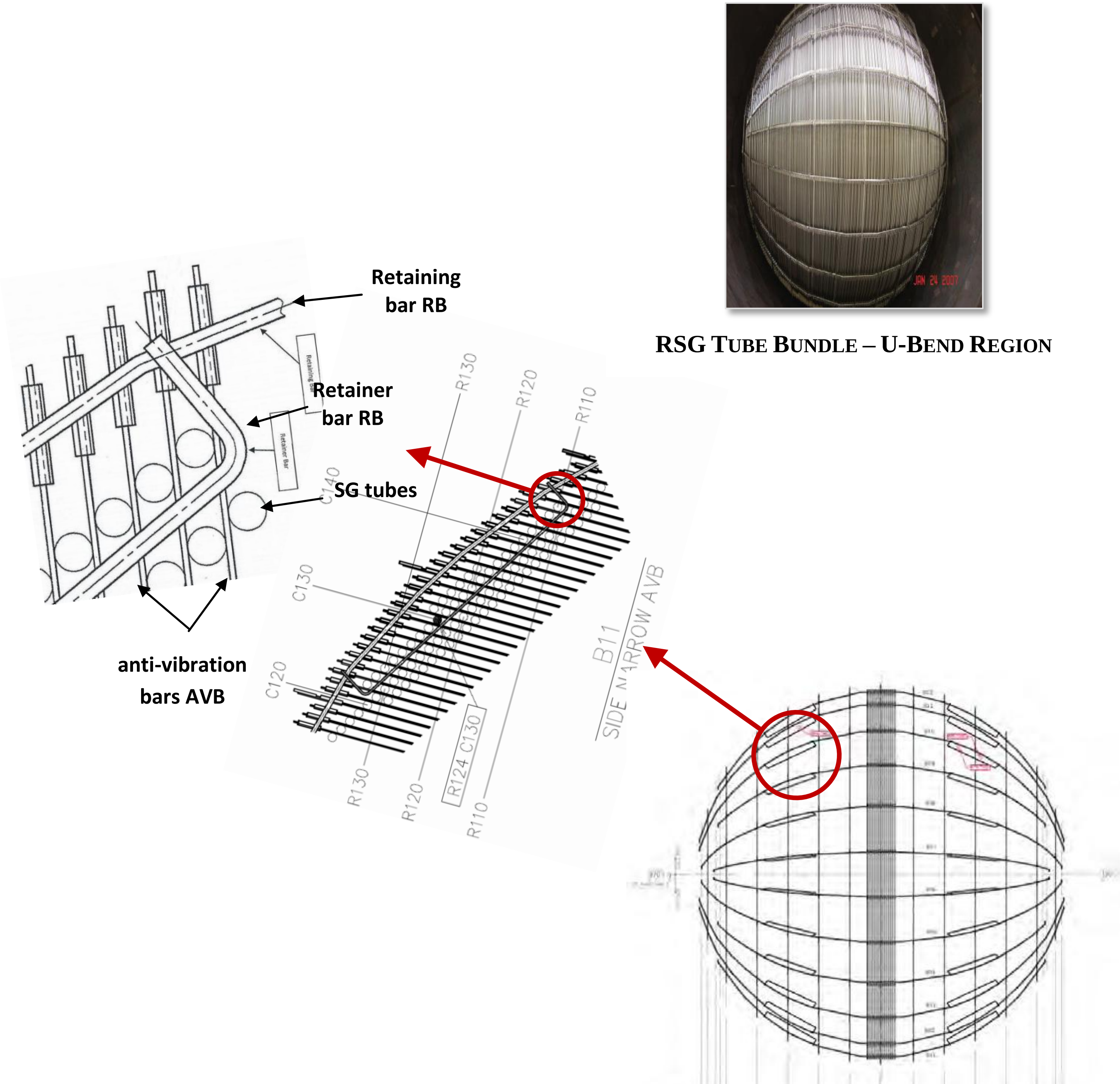


FIGURE 5 ANTI-VIBRATION AND RESTRAINT BAR ASSEMBLY

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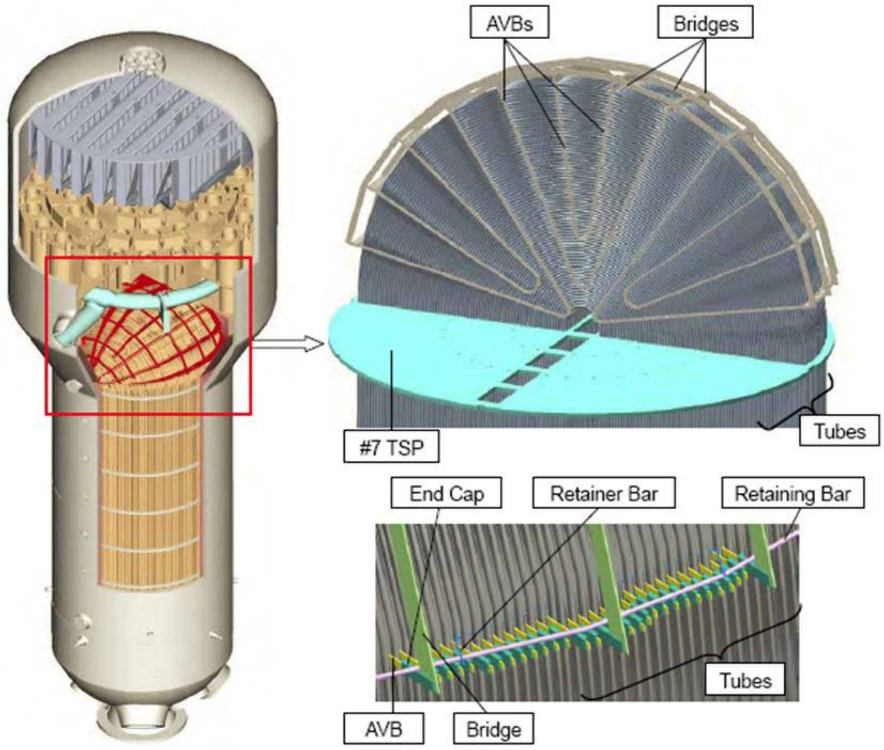

AVBs
Bridges
#7 TSP
Tubes
End Cap
Retainer Bar
Retaining Bar
Tubes
AVB
Bridge