Citizens' Oversight Projects (COPs)
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August 1, 2012

Gregory Werner
Nuclear Regulatory Commission
Chief, Plant Support Branch 2, Region IV
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(817)320-3984

Dear Mr. Werner:

I first want to thank you for conducting a great public meeting on June 18 and the NRC Augmented Inspection Team Report of July 18. I think the NRC is generally doing a good job of reporting to the public on this. There are a number of questions and comments our team has regarding the public meeting and the AIT Report.

Although I do not have significant nuclear industry training, I am an electrical and electronics engineer (Master's of Science degree from San Diego State University) which is surprisingly appropriate for this review due to the fact that electronics engineers do quite a lot of work with vibrations and oscillations in tuned circuits, and therefore, I have some intuition about what may be important to consider. I was the leader of our review team and submit this document as a start of what we hope will be a productive interaction for everyone.

In the attached detailed document, there are a total of 87 questions. Some of these are relatively simple to allow us to continue to research the topic, while many are follow-up questions from both the public meeting and from the AIT Report of July 18, 2012.

The public meeting was video recorded, available at: http://www.copswiki.org/Common/M1272 and a complete transcript of this meeting exists at this URL: http://www.copswiki.org/Common/NRCMeetingTranscript2012-06-18

The most important issue found by the COPS working group that was not adequately discussed by your report was the **thinning of the steam-generator tubes from 0.048 to 0.043 inch, by 10.4%, removing a total of about 11 tons of steel from the original 6530 tube bundle**. This massive change to the strength and integrity of the tube bundle may have made a significant contribution to the vibration and tube degradation. However, it is not explicitly mentioned as a contributing factor.

Your report mentions a tube plugging criteria of 35%. But with 10.4% thinner tubes to start with, the actual threshold should be about 27% to result in the same thickness of the tubes when plugging is required. This seems like a mistake.

This major change to the design should have required additional scrutiny during the design process, and is unacceptable from the view point of Citizens' Oversight. This, coupled with the removal of structural support of the stay cylinder and vibration suppression supports only attached to the tube bundle itself, allows the tube bundle to vibrate as a whole. It does not appear that vibration of the entire tube bundle, exacerbated by the thinner and weaker tubes has been adequately addressed.

The steam generator tubes provide the isolation from the radioactive side to the nonradioactive side, and thinning those tubes without flagging this as requiring approval of the NRC and a license amendment process is unacceptable. The dimensions of the tubes should be mentioned explicitly in the updated final safety analysis report because it represents a key safety feature. This was not mentioned in your report nor was this considered by the review of 50.59 compliance.

Also, we note that your summaries of the shutdown mention the leak as 75 gallon/day rate. But the reality is that rate was the estimate at the very start of the detection, but it increased 40% to 105 gal/day within an hour, and it was measured at 104 gal/day when the tube was tested. Therefore, your continued characterization of the leak as a 75 gal/day leak is incorrect and should be corrected in future documents to be a 104 gal/day leak, because that is the measured rate.

We have many other questions and comments, and we hope our contribution will help move us toward a safe solution for these poorly designed steam generators.

Please confirm your receipt of this communication.

Sincerely,

Raymond Lutz
National Coordinator, Citizens' Oversight Projects
(Electrical Engineer)

QUESTIONS FROM CITIZENS OVERSIGHT PROJECTS (COPS) REGARDING STEAM GENERATOR TUBE FAILURES AT SAN ONOFRE NUCLEAR GENERATING STATION

The AIT Report said “The release resulted in an estimated 0.0000452 (4.52 E-5) mrem dose to the public.” It also says “This unplanned offsite release of radioactivity was reviewed by Region IV health physicist inspectors who confirmed SONGS’ offsite dose estimate (see Section 10 for additional details).”
Q1 ---> How was the estimate of public radiation dose determined?

In the AIT Report, it said “The estimated leak rate was 75 gallons per day.”

However, according to the text, the report should state “The leak was first identified and estimated to be about 75 gallons per day, increasing at the rate of an additional 30 gallons per day in one hour (40% increase per hour)”

> On January 31, 2012, at 3:05 p.m. … Operations personnel determined the leakage to be about 75 gallons per day, using a mass balance calculation (.06 gpm), from steam generator 3E0-88. This leak rate was below the Technical Specification 3.4.13, “RCS Operational Leakage,” limit of 150 gallons per day for primary- to-secondary leakage through any one steam generator.
>
> At 4:10 p.m., operations personnel evaluated that the primary-to-secondary leak rate exceeded 75 gallons per day on steam generator 3E0-88 and that the leak was increasing at greater than 30 gallons per day per hour, and consequently, initiated a rapid power reduction to be ≤ 50 percent power in one hour and in Mode 3 within the next two hours per Abnormal Operating Instruction SO23-13-14. In accordance with Abnormal Operating Instruction SO23-13-14, when reactor power was less than 35 percent, operations personnel tripped the reactor at 5:31 p.m. to enter Mode 3.

[So at that time, the leak would have been over 105 gallons per day, and would exceed 150 gallons per day in another hour, if the increase continued.]

On page 16 of the NRC AIT Report, we see:

> Prior to being tested to failure, the tube that leaked during operation (row 106, column 78) exhibited a measured leak rate of 0.072 gallons per minute at a test pressure corresponding to normal operating conditions. This compares with a leak rate of 0.06 gallons per minute measured by SCE operating staff for SONGS Unit 3 when they made the decision to shut the plant down. The reported operational leakage was evaluated based on ambient conditions. Both the operational and test measurements are less than the applicable technical specification limit of 0.1 gallons per minute.

Since there are 1440 minutes in a day, 0.072 gallons per minute is equal to 104 gallons per day, which is much larger than the often quoted 75 gallons per day. The NRC should discontinue describing the leak as 75 gallons per minute when it was measured to be 104 gallons per minute, and had increased to that rate in only an hour.

Q2 ---> Please make this correction in any future written description.

On page 16, the chart shows a column entitled "Leak Rate at Main Steam Line Break"
Q3 ---> Is this the leak rate after the tube failed?
Q4 ---> Since all columns say ">0.5" (720 gal/day), what was the actual leak rate after tube failure?

We keep hearing that Unit 2 was not experiencing the same amount of wear as Unit 3, but in every measure Unit 2 had more tubes plugged and more tubes with more than 10% wear than in Unit 3. From the report, we read about Unit 2

> Steam generator tubing inspections in steam generator 2E0-88 found wear on four tubes that required plugging in accordance with the technical specifications. Anti-vibration bars caused the wear on two of the tubes and retainer bars caused the wear on the other two tubes. Because of the unexpected wear, the licensee preventatively plugged 94 tubes in steam generator 2E0-89 and 98 tubes in steam generator 2E0-88. Fifteen of the tubes in steam generator 2E0-89 were stabilized prior to plugging, and 18 of the tubes in steam generator 2E0-88 were stabilized prior to plugging

But Unit 3 had fewer tubes plugged and fewer tubes with at least 10% degradation.

> the location of the leak in steam generator 3E0-88 as coming from the tube in Row 106 Column 78. No other tubes were found to be leaking. ...
>
> the licensee discovered unexpected wear in both steam generators, including wear at retainer bars (similar to the wear found in Unit 2 steam generators) and significant tube-to-tube wear in the freespan areas (u-bend area of the tubes). The inspections identified 56 tubes in steam generator 3E0-89 and 73 tubes in steam generator 3E0-88 that SCE performed in-situ pressure testing on to determine if they met the structural integrity requirements in plant technical specifications.

Wall thickness of the tubes was decreased

AIT Report Section 1.2:
The Unit 2 and Unit 3 replacement steam generators …

- nominal wall thickness of 0.043 inches (10.4% thinner than Original SG design)
- 9727 tubes within each steam generator, which are arranged in 142 rows and 177 columns.
- thermally treated Alloy 690 tubing
- u-bend configuration
- The new anti-vibration bar assembly is a free floating design that is supported by the tube bundle and is not attached to the tube bundle wrapper.

The original Model 3410 steam generators at Unit 2 and Unit 3

- nominal wall thickness of 0.048 inches
- 9,350 mill-annealed, Alloy 600 tubes

- combination of u-bend tubes and tubes with two 90 degree bends (also called square bends).
- lateral support was provided by a number of lattice-grid (i.e., eggcrate) carbon steel tube supports. Tube support in the upper bundle was provided by carbon steel diagonal bars (commonly called batwings) and vertical straps.
- contained a cylindrically shaped support structure beneath the center of the tubesheet (called the stay cylinder) that provided structural support to the large diameter tubesheet.

The Report says
"The design changes between the original and replacement steam generators noted above are commonly used in replacement steam generators today."

But the design of these steam generators is unique in the industry, and if these sort of changes are being applied to all nuclear reactors, with the same lack of oversight by the NRC, then we are in serious trouble, folks.

The AIT Report (1.4, Page 10) says:
Unit 2:

> A total of 2411 tubes were found with indications at the tube support plates and anti-vibration bar supports, the vast majority of which had a measured depth of less than 20 percent of the tube wall thickness. Only two of these indications, located at the anti-vibration bar supports, exceeded the Technical Specification 5.5.2.11.c repair limit of 35 percent of the tube wall thickness. The two affected tubes plus two additional tubes with 31 percent deep indications were stabilized and plugged.

But the tube wall were reduced by 10.4% from the original steam generators. Thus to meet the same wall thickness safety margin, the new repair limit should be 35% or the thickness of the original steam generator tubes, not the new (thinner) tubes. Since they were thinned by 10.4% in manufacturing, the new repair limit should be 27%.

Q5 ---> How many of the 2411 tubes with indications had indications greater than 27%? These would have exceeded the repair limit in the old steam generators. Are they not also past the repair limit for the new steam generators?

Unit 2 from AIT Report (page 11):

> The six tubes with retainer bar indications have been plugged and stabilized. In addition, the remaining 182 tubes (total for both Unit 2 steam generators) that intersect the retainer bars were plugged as a preventive measure. Twenty four of these tubes were stabilized prior to plugging to ensure that all 188 plugged tubes will not sever due to continued vibration of the retainer bar. The tubes that were stabilized are strategically located at each end and center of the retainer bars.

Q6 ---> What were the maximum wear indications in the 188 plugged tubes?

Unit 3 from AIT Report (Page 13)

> Four tubes were found with indications at retainer bar intersections, with measured depths ranging from 28 to 46 percent. At the time of the team's presence at the site, planned corrective actions with respect to tubes adjacent to the retainer bar were similar to those

completed for Unit 2. The four tubes with retainer bar indications were plugged and stabilized. In addition, the remaining 184 tubes (total for both Unit 3 steam generators) that intersect the retainer bars were plugged as a preventive measure.

According to this text, 188 tubes were plugged. However, since the tubes were manufactured 10.4% thinner than the original tubes, the threshold should be 27% rather than 35%.

Since the report only lists the number of tubes with greater than 20% (rather than 27%), we will work with that number, and it appears that about 134+247+372=753 tubes should be plugged, in addition to any that are plugged and stabilized in the area of the instability (which might mean an additional 184).

Q7 --> How many tubes have wear indications greater than 27% the thinner tube thickness and have all those tubes been plugged?

On page 17 of the NRC AIT Report under SCE Cause Evaluation
Q8 ---> The report says that "No findings were identified." Do you plan to continue to work to identify findings and determine the cause?

Q9 ---> The list of "cause contributors" does not include the possibility that making the tubes 10.4% thinner probably made them weaker and more likely to vibrate in operation. Why was this cause contributor ignored? Specifically, we support the assertion that it is a key factor in determining stability or instability as follows

According to the reference, "VIBRATION IN HEAT EXCHANGERS" (http://www.thermopedia.com/content/1242/) they derive a natural frequency of tube vibration. The natural frequency of tube vibration depends on:

- E = the modulus of elasticity of the tube material
- I = Area moment of inertia, which depends on:
  - D = Outside Diameter
  - D_i = Inside Diameter
- m = Overall mass per unit length (including the mass of the tube itself, the mass of the tube side fluid, and the mass of the shell side fluid displaced in the vibration.
- Bn = a constant that describes the vibration modes.

In calculating *tube vibration*, it is important to find the *natural frequency* of vibration of the tubes. For a tube with pivoted ends vibration may occur according to mode shapes 1,2, and 3 as shown in **Figure 2.** The natural frequency of vibrations depends on both the mode shape and the physical characteristics of the tube, and the way its ends are fixed; it can be calculated by the formula

$$f_n = \frac{B_n}{2\pi l^2}\left(\frac{EI}{m}\right)^{0.5}, \tag{2}$$

where E is the modulus of elasticity of the tube material, I the area moment of inertia (= $\pi(D^4 - D_i^4)/64$ where D_i is the tube internal diameter), m is the overall tube mass per unit of its length (including the mass of the tube itself, the mass of the tube side fluid and the mass of the shell side fluid displaced in the vibration), 1 is the tube length, B_n is a constant depending on vibration shape and the manner of tube fixation in the heat exchanger. The constant B_n used for determining the frequency harmonic of natural vibrations in a quiescent fluid in the absence of axial forces is derived from tabulated data. For shell-and-tube heat exchanger with more than 4 baffles, and where the end spaces between the tube sheets and the nearest baffles does not exceed the baffle spacing by more than 20%, a value of $B_n = 10$ may be taken [Chenoweth (1983)]. Alternatively, the expression $B_n = \lambda_n^2$, where λ_n is calculated from the expression given in Table 1, may be used.

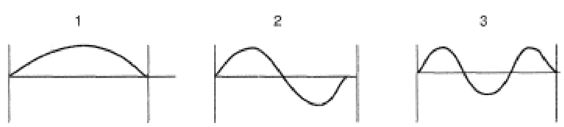


Figure 2. Mode shapes in tube vibration.

The reference "Fluid-Elastic Instability in Tube Arrays Subjected to Air-Water and Steam-Water Cross Flow" 2005 (http://boiling.seas.ucla.edu/Publications/d_mitra.pdf), the natural frequency of tube vibration is an expected key parameter determining stability vs. instability in fluid-elastic flow.

thus giving rise to the Connors' criterion that is still commonly used for presenting results of fluid-elastic instability. Such a correlation is suggested by a dimensional analysis of the governing parameters and takes the form

$$\frac{V_{cr}}{f_n D} = K\left(\frac{2\pi m \zeta}{\rho D^2}\right)^b \quad (1.1)$$

where V_{cr} is the critical velocity of the fluid and is typically the superficial pitch velocity or superficial velocity in the minimum gap space between the tubes, f_n is the natural frequency of the tube, D is the outer diameter, m is the mass per unit length of the tube, ζ is the damping ratio of the tube in vacuum, and ρ is the fluid density. K and b are

(K and b are coefficients obtained by fitting experimental data, 9.9 and 0.5 from Connors' experiments)
Therefore, the decision to use 10.4% thinner tubes may be an important change in the design and thus a contributor to unstable fluid-elastic flow.

Q10 ---> Did the computer models consider the change of resonant frequency of the steam generator tubes due to the 10.4% thinner walls?

Q11 ---> Did the computer models consider vibration of the entire tube bundle, and probably much lower natural frequency for the entire bundle vs. a single tube?

Q12 ---> Later, in "c. Conclusions" it mentions "the completed SCE Cause Evaluation". How can it be completed if there were "No findings identified."

The conclusions state

> The completed SCE cause evaluation identified the mechanistic cause of the tube-to-tube wear as fluid-elastic instability caused by a combination of localized high steam/water velocity, high steam void fraction, and insufficient contact forces between the anti-vibration bars and the tubes.

Q13 ---> But what caused the "high steam/water velocity, high steam void fraction, and insufficient contact forces"? (What was the ultimate cause?)

During the June 18, 2012 public meeting, R. Lutz asked the question:

> What is the cause of the excessive steam velocity?

And Mr. Warner answered:

> Actually, that question is outstanding. We have to understand that SONGS owes us that answer as far as what specifically in the design change in the steam generator causing the higher than expected velocity, and as they talked about steam with fractions. They still owe us that. That's been something that we've discussed since we have been on site.

Q14 ---> Should not the report state that this question is still outstanding?

Q15 ---> Have you received an answer to this question?

On page 18 of the NRC AIT Report, Section 2.2 describes the Mitsubishi Cause Evaluation

Q16 ---> Did Mitsubishi consider the fact that the tubes are thinner than those in the original steam generator by 10.4%
Q17 ---> Would thicker tubes increase the overall strength of the tube bundle, and therefore reduce the "flowering" and vibration?

On page 21 of the NRC AIT Report, there is a description of accelerometers installed on the steam generators:

> Two separate accelerometers were installed on each of the steam generators. The location of these instruments are on the steam generators' lower supporting structures and provide acoustic information about loose parts impacts specifically on the reactor coolant or primary side of the steam generators. The vibration and loose parts monitoring system real time functions consist mainly of impact alarm validation of suspected loose part events and recording acoustic data. Long term vibration monitoring and loose part event trending were done by engineering personnel using recorded data.

We were told that it is not possible to monitor vibration in the steam generators:

In the June 18 Meeting, MR. Warner said:

> "They do not measure steam flows within the steam generators. There is not that capability."

and later

> "Actually, there is no current way right now that you can evaluate vibration with the unit is running. It's actually being looked at as a potential method in the future."

but we note that the report talks about vibration monitors.

Q18 ---> were the vibration monitoring accelerometers installed in the steam generators during the failure?
Q19 ---> if so, was the data collected from those sensors reviewed?
Q20 ---> If accelerometers can be placed in the lower area, why not also place them in the upper section to detect vibration of the tube bundle and flowering?

On page 22 of the NRC AIT Report:

> Additional review and follow up will be required of the vibration and loose parts monitoring system alarms, including evaluation and disposition of Unit 3 alarms and then determine whether this issue represents a performance deficiency or constitutes a violation of NRC requirements. This issue is identified as URI 05000362/2012007-02, "Evaluation of Unit 3 Vibration and Loose Parts Monitoring System Alarms."

Q21 ---> What "NRC Requirements" are referred to here?

In the review of the difference between Unit 2 and Unit 3, at the June 18 Meeting, Mr. Deitrich said:

> Rough numbers, rough percentages on Unit 3, nine percent of the tubes in the Unit 3 steam

> generators, 19,454 tubes in the Unit 3 steam generators, nine percent of them showed wear with greater than 10 percent through wall indications. Nine percent.
>
> On Unit 2, 12 percent of the tubes showed wear greater than 10 percent through wall indication.

According to this disclosure, Unit 2 exhibited more wear than Unit 3.

Q22 ---> were these statements by Mr. Deitrich correct?

On page 24, it says:

> Fluid-elastic instability was found not applicable to the retainer bar because this mechanism did not apply to a single tube in cross flow.

However, this is not a single tube, but a retainer in an array of tubes, and so this assumption does not seem valid. Indeed, it appears that this initial assumption was later discarded.

On page 17, it says:

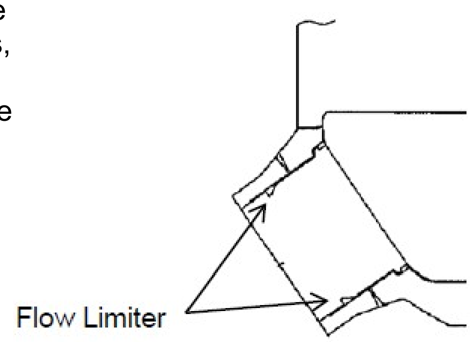
> Additionally, Mitsubishi identified that the Unit 3 replacement steam generators had better dimensional controls during the fabrication process. This determination was based, in part, on the results of pre-service and in-service eddy current examinations, and fabrication data from Unit 2 and Unit 3 replacement steam generators. The correlation of dimensional controls with the failure mechanism was that improved dimensional controls for Unit 3 replacement steam generators resulted in less variability of as-built critical dimensions such as anti-vibration bar thickness, tube roundness, and gaps between tubes and anti-vibration bars.

It is always assumed that better dimensional control means that the system will perform better. But in terms of fluid-elastic instability, having tubes with exactly the same natural frequency will mean that the whole thing will start to vibrate like mad at one key frequency, whereas poor dimensional controls may results in natural frequencies of vibration that are not exactly the same, and thus individual tubes might vibrate, but the additive effect of having many tubes with the same natural frequency may make it MORE subject to fluid-elastic instability rather than the other way around.

Q23 ---> Did the team consider that better dimensional control may lead to vibrations that reinforce each other and result in a unstable pole that may lead to catastrophic failure?

On page 30 of the NRC AIT Report, there is a discussion of the flow limiters:

> Flow Limiter for Primary Inlet Nozzles – The replacement steam generators were designed with a flow limiter located in the primary inlet nozzle (see figure below) in order to make the reactor coolant system flow similar to the flow rate of the original steam generator and not exceed the maximum allowable reactor coolant system flow rate. The licensee's evaluation for the engineering design package determined that although the original steam generators had a number of plugged tubes, the reactor coolant system flow rate of the original steam generators was near the design requirement. Because the



replacement steam generators has 377 more tubes than the original steam generators, and contained tubes with u-bends versus “square bends”, the pressure drop of the replacement steam generators with no plugged tubes would be much less than the original steam generators resulting in a higher flowrate.

The flow limiter was designed to ensure the total “best estimate” reactor coolant flow rate with the replacement steam generators installed would not exceed 106.5 percent of the design volumetric flow rate of 396,000 gallons per minute at a reactor coolant system cold leg temperature of Tcold = 540.9°F. For Unit 2 replacement steam generators, the flow limiter diameter to nozzle inner diameter ratio was 0.94 while the ratio for Unit 3 steam generators was 0.915 due to Unit 3 reactor coolant pump replacement. The flow limiter dimensions resulted from a scaled model test performed by Mitsubishi and it was designed to be machined as part of the nozzle base metal.

Q24 ---> What were the actual reactor coolant flow rates in the steam generators?

According to this description, no adjustment was made due to the change of inside diameter of the tubes (because the thickness was decreased by 10.4%)

Original inside radius = (0.75/2 = 0.375) – 0.048 = 0.327; Area = 0.3359 sqin

Redesigned inside radius = (0.75/2 = 0.375) – 0.043 = 0.322; Area = 0.3257 sqin

Increase = .3359 / .3257 = 103.13% => reduce flow by 0.9696

In other words, flow is increased by 3.13% due to thinning of the tubes alone.

Assuming the figures DID NOT take this into account, the flow limiters should have been sized to decrease the flow by an additional 3.13%. They should have been:

Unit 2: 0.94 * 0.9696 = 0.911
Unit 3: 0.915 * 0.9696 = 0.887

Q25 ---> Did the calculations for the flow limiters take into account the 3.13% increase due to thinning of the tube wall thickness by 10.4%?

At the June 18 Public meeting, we were old that:

> (GREG WARNER) The team identified the primary cause of the unexpected tube wear was higher than expected flow velocities in the steam generators. Early in our inspections we independently developed a simplified mathematical thermal hydraulic computer simulation model of the steam generators in units 2 and 3. Using this, we determined that the computer simulation used by Mitsubishi during the design of the steam generator had under-predicted velocities of steam and water inside the steam generator by factors of 3 to 4 times. San Onofre also had 3 other steam generator vendors conduct computer simulation. The results of their computer simulation also showed significantly higher steam velocities and confirmed our results.

On page 35 of the NRC AIT Report:

> The team noted that a key methodology for the design of the replacement steam generators was the thermal-hydraulic code used to model the flow conditions in the steam generators. Mitsubishi’s FIT-III thermal-hydraulic code was accepted by SCE for the design of the replacement steam generators. The team noted that the updated final safety analysis report did

not describe the thermal-hydraulic code used for the design of the original steam generators and therefore the use of the FIT-III thermal-hydraulic code did not constitute a change in methodology or a change in an element of a methodology described in the updated final safety analysis report. The updated final safety analysis report did describe the computer code CRIB as the code used to analyze overall steam generator performance. As described in the updated final safety analysis report, CRIB was used to establish the recirculation ratio and fluid mass inventories as a function of power level in the original steam generators.

If the updated final safety analysis report did not describe the thermal-hydraulic code used in the original steam generators, you can't say there was not a change in methodology, because there was nothing to compare with. Apparently, this really was a key deficiency in the redesign, and the fact that it was not described in the updated final safety analysis report is a important error.

Also, you report that:

> With regard to the major design changes between the original and replacement steam generators, the updated final safety analysis report did not specify how the original steam generators relied on special design features such as the stay cylinder, tubesheet, tube support plates, or the shape of the tubes to perform the intended safety functions.

Q26 ---> who is responsible for preparing the "updated final safety analysis report" (UFSAR)?
Q27 ---> Why was the thermal-hydraulic code used in the original design not described?
Q28 ---> Why was there no description of the stay cylinder, tubesheet, tube support plates, or shape of the tubes in the UFSAR?
Q29 ---> Did the UFSAR describe the tube wall thickness? It was decreased by 10.4% and should be considered as a key element in the change of the natural vibration frequency of the tubes.
Q30 ---> Please provide the updated final safety analysis report description of the original steam generators.

You report that:

> Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” November 2000, allows the use of NEI 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1 for methods that are acceptable for complying with 10 CFR 50.59. Per NEI 96-07, changes affecting structures, systems, or components that are not explicitly described in the updated final safety analysis report can have the potential to adversely affect structure, system, or component design functions that are described and thus may require a 10 CFR 50.59 evaluation. Consistent with this guidance, SCE’s 50.59 screening evaluated the differences in subcomponents between the original steam generators and replacement steam generators as to whether the differences adversely affected the design function (reactor coolant pressure boundary) of the steam generators.

Q31 --> Did the NRC review the differences in subcomponents and the methodologies used, such as the thermal-hydraulic code used to model the steam generators? Leaving this to SCE to review their own work is hardly a good idea.

On page 41 of the AIT Report, it states:

> The original steam generators installed throughout the domestic fleet of pressurized water reactors, including SONGS, experienced widespread corrosion of the tubes and tube support plates, stress corrosion cracking of the tubes, and wear at tube supports. These problems led to the replacement of nearly all of the original steam generators, in most cases well before the end of their design lifetime. For SONGS, the design of the replacement steam generators included a number of design changes to correct life limiting problems with the original steam

> generators, based in part on consideration of SONGS-specific and industry-wide operating experience. This included use of more corrosion resistant materials for the tubing and tube support plates to mitigate corrosion and stress corrosion cracking issues experienced in the past.

However, you do not specify that the tube wall thickness was decreased by 10.4%. It seems intuitive that you would want to increase the wall thickness of the tubes to reduce stress corrosion cracking and allow more wear at tube support before the tubes must be plugged.

Q32 ---> Why were the tubes decreased in thickness.

My estimate indicates that about 8 tons of steel were removed from the tube bundle, significantly reducing they strength and probably a major contributor to weakness and therefore vibration.

Area of metal in cross section = pi * Ro^2 – pi * Ri^2 = pi * (Ro^2 – Ri^2)
Reduction in area factor = pi * (Ro^2 – Ri(new)^2) / pi * (Ro^2 – Ri(old)^2)
= (Ro^2 – Ri(new)^2) / (Ro^2 – Ri(old)^2)
= (0.375^2 – (0.375-0.043)^2) / (0.375^2 – (0.375-0.048)^2)
= (0.140625 – (0.332)^2) / (0.140625 – (0.327)^2)
= (0.140625 – 0.110224) / (0.140625 – 0.106929)
= 0.030401 / 0.033696 = 0.9022
= 0.033696 - 0.030401 = 0.003295
= 0.003295 / 0.033696 = 9.778% reduction in metal area.

Based on a scale drawing of the original steam generators, we can estimate the height of the tubes by comparing to the tube sheet, which dimension is 28.19 inches. The measurement of the tubes from bottom to top is 4.35" on the drawing.
0.31" = 28.19" ==> scale = 90.935
4.35 ==> 395.57 about 400 inches for our estimate.

Can estimate metal in the tubes by assuming all tubes are full length with no turns. Should average out to about the same thing if all turns and changes of length are taken into account.

(Used http://www.onlinemetals.com/calculator.cfm assuming Stainless 301 alloy)

1 400 inch tube (0.043 thickness) = 10.9645 pounds
1 400 inch tube (0.048 thickness) = 12.1529 pounds
12.1529-10.9645=1.1884 pounds. There are tubes running up and down, so we 2*9350 tubes = 22223 pounds per steam generator, or about 11 tons of steel removed from the tubes in each generator (considering only the original tubes). Total weight of the added 370 tubes would be ABOUT 370*10.9645=4056 pounds (about 2 tons added back in), and thus total removed metal weight from tubes is about 8 tons.

Q33 ---> Does the elimination of 8 tons of steel in the tubes seem like a significant change that should be mentioned?

Page 47 of the AIT Report:

> One of the major enhancements of the replacement steam generators was the use of Alloy-690 tubing versus Alloy-600 for corrosion resistance. Alloy-690 has lower heat conductivity so, to achieve the same power, the heat transfer surface area must be increased by at least 10 percent. This required more tubes to be used in the replacement steam generators. The increased number of tubes resulted in a more tightly compacted tube bundle and elimination of the stay cylinder. The increase in the number of tubes could lead to increases in primary reactor coolant flow through the steam generators. Orifices were machined as part of the steam generator inlet nozzles to ensure maximum allowed primary system flowrates were not exceeded.

The fact that the alloy-690 has lower heat conductivity has not been mentioned prior to this point in the report. It says the heat transfer area must be increased by at least 10%. But only 370 tubes were added in each steam generator, which is 370/9350 = 3.95%.

Q34 ---> Why does the report not mention the fact that the tubes were also thinned by 10.4%, removing 11 tons of steel from each steam generator (before additional tubes were added)?

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page 48:

> the anti-vibration bar support structure is not connected to the wrapper for lateral or vertical support; instead the anti-vibration bar system structure is only supported vertically by resting on the tubes.
>
> Other operational and physical comparisons of the replacement steam generators and original steam generators were reviewed by the team and no significant differences were noted.

This is a MAJOR problem when you couple it with the fact that 8 tons of steel were removed from the tubes, making them much weaker, and no central support due to the stay cylinder.

Q35 ---> Did the team completely miss the fact that thinner tubes means they are much weaker? This is never mentioned, although the dimensions of the thickness are indeed mentioned.

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> Mitsubishi used the approach given in the ASME code Section III, Division 1, Appendix N-1330, "Flow-Induced Vibration of Tubes and Tube Banks," to calculate stability ratios and they also avoided natural frequencies of the tubes similar to the reactor coolant pump dynamic frequencies.

Q36 ---> Did the team check the calculations of the natural frequencies of the tubes?
Q37 ---> Were the natural frequencies determined by calculation or by a physical test (or both)?

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> The accuracy of calculating fluid-elastic instability is limited based on inputs that are best determined by design-specific mockup test data. Mitsubishi did not perform design-specific mockup tests, but used generally accepted test data, and other data based on Mitsubishi test rigs that were not specific to the SONGS replacement steam generator design.

This sort of design should never be allowed for components used in nuclear reactors because it is too easy to make mistakes by using "generally accepted test data" that do not apply to these steam generators.

Q38 ---> Did the NRC review and approve this method of design?

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> Traditional design of anti-vibration bar systems have not considered in-plane fluid forces since

> it was accepted that the rigidity and dampening strength of the tube in this direction was adequate to preclude it. This event at SONGS is the first US operating fleet experience of in-plane fluid-elastic instability, sufficient to cause tube-to-tube contact and wear in the U-bend region.

Of course, with anti-vibration bars only connected to other tubes, the tubes thinned by 10.4% (removing 8 tons of steel from the tubes) they became so weak that this phenomena can easily occur, and since no mockup tests were performed, this allowed the mistake to creep into the design.

Q39 ---> Do these steam generators comply with "traditional design" in other respects, such as anti-vibration bars only tied to other tubes, thinner tube thickness, and removal of the stay cylinder?

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> The team noted that Design Specification SO23-617-1 did not address specific criteria for stability ratio and does not mention fluid-elastic instability. The team did find that the Mitsubishi calculated design values for stability ratios did not exceed 0.5. It is important to note, that each steam generator manufacturer has different design values for maximum stability ratios; therefore there is no standard value. The smaller that the design stability ratio is (has to be less than 1), the more margin to fluid-elastic instability.

Q40 ---> It was previously stated that the design complied with "traditional design." What is the "traditional" maximum stability ratio? (i.e. how does 0.5 compare with other steam generator designs?)

Q41 ---> Since fluid-elastic instability is a key factor to cause tube-to-tube wear and has been recognized for decades, why did the Design Specification not mention it?
Q42 ---> Did the NRC approve this design specification without the mention of Fluid-Elastic Instability?

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> The critical flow velocity is then calculated based on damping ratio, tube mass, tube outside diameter, averaged local cross flow gap velocity, and fluid density per selected tube.

Q43 ---> Why is the tube thickness (and therefore natural frequency) not considered a key factor in calculating the critical flow velocity?

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> Mitsubishi indicated that in their methodology two conservatisms were used in their bundle vibration analysis: (1) FIT-III gap velocities were averaged and multiplied by 1.5 and (2) one of 12 anti-vibration bars contacts were assumed to be inactive. The team noted that in Mitsubishi Document L5-04GA504, "Evaluation of Tube Vibration," Revision 3, the 1.5 multiplier was not an added conservatism but a requirement, needed to match test data results.
>
> The team developed an independent model of the new steam generators using the ATHOS thermal hydraulic code
>
> …
> The Mitsubishi ATHOS model fluid velocities were approximately 3 times higher than the FIT-III model velocities with the 1.5 multiplier applied. Other independent code calculations, including

an analysis by Westinghouse using their in-house modified version of ATHOS and an analysis by AREVA using their French code CAFCA4 showed similar thermal-hydraulic results (up to 4 times higher velocities than FIT-III) as those computed in the Mitsubishi ATHOS results and the NRC independent ATHOS calculations. Based on these comparisons, it was concluded that the FIT-III code and model results used for design were non-conservative even with the multiplier applied.

In other words, Mitsubishi did not perform the thermal analysis correctly, and the review process did not catch it.

Q44 ---> Did the NRC approve of the analysis?
Q45 ---> If SCE been required to undergo a license amendment process, would the normal review process likely detect this error?
Q46 ---> Was there any review of the design intended to catch errors of this type that were not employed?
Q47 ---> Why did NRC not develop an independent model during the pre-manufacturing review process instead of waiting until after a failure occurred?
Q48 ---> Did the team apply the ATHOS model to the old steam generators to see what the steam velocity is predicted, and then slowly make design changes to see which one was the key culprit in causing the higher steam velocities?

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Because of the limited information provided, the team could not determine the validity of the benchmarking of FIT-III.

Overall, the team determined that the validation and verification of the FIT-III code did not present overwhelming evidence that this code has been adequately benchmarked.

Q49 ---> Does that mean that more information is being requested, or does it mean that the team is finding that the benchmarking of FIT-III was inadequate?
Q50 ---> Since the code apparently under-predicted the steam velocity by three to four times, is the team unwilling to state that FIT-III is a bad computer model? How could it possibly be benchmarked adequately if it produces such bad results?
Q51 ---> Are there other steam generators that have been designed and/or approved by the NRC that use the defective FIT-III computer model?

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Based on the cause evaluation and corrective action plan, SCE determined that the best solution to prevent tube-to-tube wear was to conservatively plug and stabilize the affected areas.

Q52 ---> How did “SCE determine” that this is the “best solution”?
Q53 ---> What other solutions were evaluated?
Q54 ---> Has the solution been simulated?
Q55 ---> Previously, one shortcoming was the lack of mock-up testing to confirm simulation results. How much mock-up testing has been performed to validate this solution?

By taking the impacted tubes out-of-service, SCE determined that this should reduce the potential for localized fluid velocities reaching critical velocity.

Q56 ---> SCE says it “should reduce the potential” but how was this conclusion reached?
Q57 ---> Is it not possible that critical velocities will be reached in other areas of the tube bundle?

> In addition, in order to ensure sufficient margin to preclude the onset of fluid-elastic instability, SCE determined that reactor power would also have to be reduced. At this time SCE is still developing additional corrective actions to prevent tube-to-tube wear. The actions have not been finalized and no determination has been made concerning the appropriate power level. The NRC has not made any conclusions on the proposed corrective actions. Once the corrective actions have been finalized, they will be inspected as part of the Confirmatory Action Letter followup inspection.

Q58 ---> How did SCE determine this (i.e. that power reduction would provide a sufficient margin)?
Q59 ---> How was this modeled or confirmed (prior to restart)?
Q60 ---> How much % did SCE determine would be required to “preclude the onset of fluid-elastic instability”?
Q61 ---> Since none of this has been confirmed, how does SCE know this is the “Best Solution”?
Q62 ---> Has the “no restart” solution been considered as a possible “best solution” including all costs and risks? Certainly, if the plant is not restarted, no fluid-elastic instability can occur, while the current proposal is only a guess, at best.

Page 57: “Risk Assessment”

> Assuming that a steam line break would cause the degraded steam generator tubes to rupture during a “T/2” exposure period of 6 months yielded a change in the large early release frequency of 4E-6/yr.

Q63 ---> Where did this assumption come from?
Q64 ---> The “small leak” at the 75 gal/day rate increased in the first hour by 40% to 105 gal/day. Why would you suggest it would take 6 months for another one to break? The reality is that the “small leak” was increasing very quickly, and within hours, would be a full steam line break, which may have progressed to other tubes in the vicinity resulting in a full LOCA. Was this possibility considered and analyzed?
Q65 ---> Without changes to the steam generators, would not the frequency need to be much higher since ACTUAL failure occurred at the rate of 1 in 11 months? (I.e closer to >1/yr).

Page 64 on 50.59 compliance.

> The evaluation process used by the technical specialist included determining if the design changes to the replacement steam generators were a change to the facility or procedures as described in the updated final safety analysis report or a test or experiment not described in the updated final safety analysis report.

Q66 ---> Please provide (or a link to) the “updated final safety analysis report” (FSAR)
Q67 ---> Why were so many key features omitted from the FSAR?
Q68 ---> Who creates the FSAR?
Q69 ---> Apparently, if something does not exist in the FSAR, then it is impossible to know if a changed occurred. Apparently, this allows significant changes to occur by producing an incomplete FSAR. Why were the following not included in the FSAR? (or were they?)

- Number of tubes
- Tube wall thickness
- Detailed description of the Anti-vibration support structures, including the fact they were not just attached to other tubes
- Stay Cylinder
- Modeling programs

The following questions were mentioned in the June 18 public meeting and not fully answered, or additional questions have surfaced from our working group.

The release was only a very small percentage of the release limits allowed by the plant license.
Q70 ---> What is the release limits allowed by the plant license?
Q71 ---> What was the percentage of this release?

A divider plate separates the hot and cold bowl areas.
Q72 ---> This seems very inefficient because it seems there will be a transfer of heat from the hot inlet side to the cold outlet side. Why is this inefficiency not a concern? Should not there be an air gap or other insulation to stop this thermal leak?

Mr. Bower said "as a result of the shut down on January 31st.... We have improved our leak detection capability."
Q73 ---> How was the leak detection improved, and why?

Mr. Pommassano and Mr. Dietrick of SCE said "we have determined the cause of the unexpected tube-to-tube wear." Mr. Blant said "The cause evaluation has been completed by SONGS and they are working on additional actions to prevent to tube-to-tube wear from occurring again."

But we know that only the primary cause (fluid-elastic instability and high steam velocity) has been determined. We do not know what the ultimate cause is, such as changes to the design of the steam generator, like adding 370 tubes, thinning the wall thickness by 10.4% and removing 8 tons of steel from each steam generator, designing anti-vibration stays that did not connect to any stable structure and only to the other tubes, removal of the stay cylinder, etc.

Q74 ---> Which one of these changes caused the higher steam velocity and unstable tube bundle?

Q75 ---> Until this question is fully and completely answered, SCE should avoid saying "we have determined the cause."

Mr. Blant said: "The N.R.C. does plan to have additional public meetings to keep you informed of our activities."
Q76 ---> Exactly what meetings are planned, when and where?

Gene Stone asked: "I would officially ask Elmo for the next meeting to be a Category 3 meeting so that we can actually discuss everything that the public wants to discuss with no limited time on that meeting."

Q77 ---> Is such a meeting planned?

Gene Stone: “How is it that 39 design changes did not trigger a complete review by N.R.C. and complete public hearings as is required by law? Has the law been broken by either California Edison, Mitsubishi or the N.R.C.?

MR. WARNER: Well, the 50.59 processes the regulation and by regulation, they were -- they were allowed to do what they did. Now to say that it wasn't reviewed, portions were reviewed by the N.R.C. Actually, there were two changes that required license amendments that were reviewed by N.R.C.

Q78 ---> What portions were explicitly reviewed by the NRC?
Q79 ---> Mr. Warner said there were two changes that required license amendments and were reviewed. Did you grant those license amendments in advance of the steam generator replacement project?
Q80 ---> Did you complete the correct process in completing the license amendments?

Jim Cummings, retired Southern California Edison employee asked: “why [was] the design changed on the steam generators from the initial construction (PMQ8)”

MR. WARNER: I'll take that question. Of course the steam generators were different than what was originally put in because the original steam generators had to be replaced. So they had issues with the original generators across the industry and from a lessons learned standpoint, the numerous changes that had been incorporated in the new generators.

Q81 ---> The question was why the DESIGN was changed, not whether the steam generators were replaced with new ones. It was not a requirement that the steam generators would be replaced. This was a desire by Edison. There were other options, such as to save $670 million and not replace them. It was not a requirement that the design was changed. But your answer does reveal that indeed they were changed, and those changes were OPTIONAL. The steam generators could have been built to the exact specifications as the original design to avoid the need for additional modeling and opening up the design to mistakes, such as those that were indeed made in this case.

I would respectfully request that you do not state things that are not actual facts. It is not true that the steam generators “had to do be replaced.” This was a desire by Edison and other options existed. The steam generators were changed in MANY MORE WAYS than in “lessons learned” across the industry.

Joe Holtsman asked: Was there an failure mode effect analysis done on these designs before construction was started? … The silence is deafening.

MR. WARNICK: Like Greg said, as part of the inspection process, we have a procedure that we implement for replacement of steam generators. We reviewed in part the 50.59s associated with the replacement steam generators. We did not review it to the level of detail to determine if the failure mode analysis was done.

Q82 ---> Was a failure mode analysis done?
Q83 ---> Why are you not reviewing it to the level sufficient to answer this question?

Mr. Collins said “That's two reports. There's the team report and then there's this report that was prepared by the other engineers that we brought in to challenge the conclusions.”
Q84 ---> Where is the report that was prepared by other engineers?

Well, as part of the N.R.C. process we do a risk assessment and we'll look at the possibility of the multiple tubes failing. That's being conducted right now.

Q85 ---> Was this completed? What is the chance that a broken steam tube might cause another to fail?

Dan Hirsh asked: “Will the N.R.C., before A decision is made on whether or not to permit restart of either unit, hold a formal, full adjudicatory, evidentiary hearing in which parties, not just Edison and the N.R.C. participate, but whereby experts who are critical of both of you testify with cross examination, discovery and a full evaluation of whether it is safe to restart?”
Q86 ---> When is this hearing scheduled?

Q87 ---> What are the exact numbers of tubes with wear, and how much? (Is there a web site with the complete information of the wear indications and where in each tube?”
