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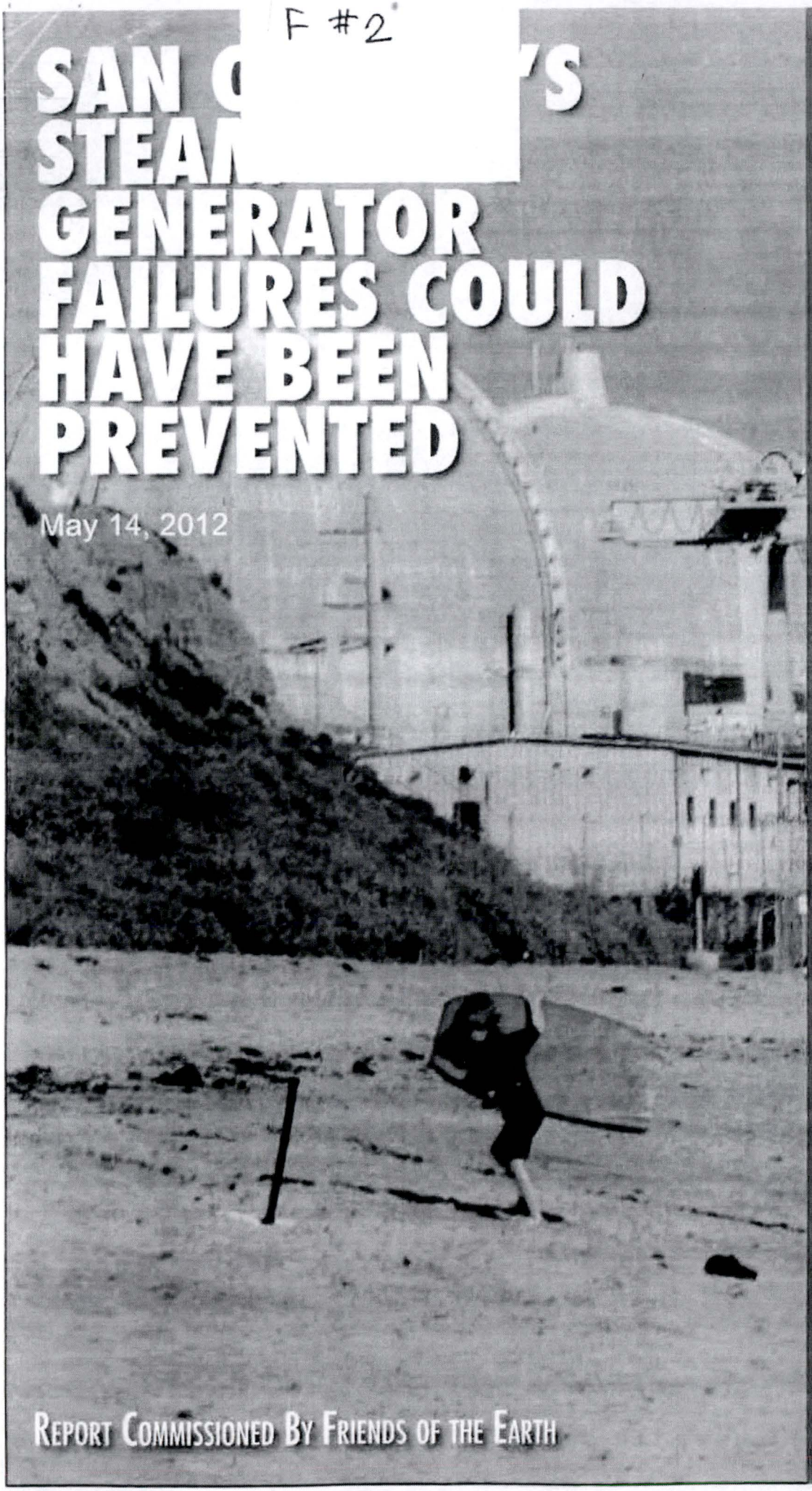
SAN O[illegible]'S STEAM[illegible] GENERATOR FAILURES COULD HAVE BEEN PREVENTED

May 14, 2012



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REPORT COMMISSIONED BY FRIENDS OF THE EARTH

San Onofre's Steam Generator Failures

Could Have Been Prevented

Summary

Southern California Edison's four replacement steam generators at their San Onofre Nuclear Generating Station failed in less than two years of operation, while the original equipment operated for 28 years. Fairewinds has been analyzing the data in order to determine how such an expensive investment could fail so quickly.

In June of 2006 Edison informed the NRC that the replacement steam generators to be manufactured by Mitsubishi would be fabricated to the same design specifications as the original San Onofre Combustion Engineering (CE) steam generators. According to Nuclear Engineering International, Edison has admitted that this was a strategic decision to avoid a more thorough license amendment and review process.[1]

> At SONGS, the major premise of the steam generator replacement project was that it would be implemented under the lOCFR50.59 rule, that is, without prior approval by the US Nuclear Regulatory Commission (USNRC). To achieve this goal, the RSGs were to be designed as 'in-kind' replacement for the OSGs in terms of form, fit and function.[2]

Fairewinds finds that there are numerous changes to the San Onofre steam generators that are not *like-for-like or "in-kind"*.

Furthermore, the facts reviewed by Fairewinds makes it clear that if Edison had informed the NRC that the new steam generators were not *like-for-like*, the more thorough NRC licensing review process would have likely identified the design problems before the steam generators were manufactured.

Finally, Fairewinds finds that tube plugging is not the solution to the vibration problem[3] and that the damaged steam generators will still require major modifications with repair and outage time that could last more than 18 months if Edison and Mitsubishi are even able to repair these faulty designed steam generators. However, Fairewinds finds that the safest long-term action is the replacement of the San Onofre steam generators.

Analysis

The requirements for the process by which nuclear power plant operators and licensees may make changes to their facilities and procedures as delineated in the safety analysis report and without prior NRC approval are limited by specific regulations detailed in The Nuclear Regulatory Commission's *10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, Section 50.59, Changes, Tests and Experiments*.

The implementing procedures for the 10 CFR 50.59 regulations have eight criteria that are important for nuclear power plant safety. (These eight criteria are provided in Table 1, footnote A below.)

These implementing procedures created for 10 CFR. 50.59 require that the license be amended unless none of these eight criteria are triggered by any change made by Edison at San Onofre. If a single criterion is met, then the regulation requires that the licensee pursue a license amendment process.

By claiming that the steam generator replacements were a *like-for-like* design and fabrication, Edison avoided the more rigorous license amendment process. From the evidence reviewed, it appears that the NRC accepted Edison's statement and documents without further independent analysis. In the analysis detailed below, Fairewinds identified 39 separate safety issues that failed to meet the NRC 50.59 criteria. Any one of these 39 separate safety issues should have triggered the license amendment review process by which the NRC would have been notified of the proposed significant design and fabrication changes.

As the NRC guidelines state:

> "(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as 1.187-A-1updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to § 50.90 only if: (i)A change to the technical specifications incorporated in the license is not required, and **(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section."**[4] [Emphasis Added]

In its previous reports, Fairewinds identified at least eight modifications to the original steam generators at San Onofre.

Table 1 below was designed to compare the eight major design modifications that Fairewinds identified in its analysis with the eight criteria the NRC applies to the license review process in order to determine whether or not a new license amendment process is required. The major design changes are located at the top of the table, and the NRC Criteria are listed in the left hand column of table. The term SSC stands for Systems, Structures and Components. A green *No* means that the *like-for-like* criteria were indeed met and that no license amendment was required. A red *Yes* means that Edison should have applied for a license amendment.

Table 1 shows that 7 out of 8 of the major design changes to the original steam generators meet a total of 39 of the NRC's 50.59 criteria requiring amendment to the license.

Table 1
Steam Generator Design Changes Identified By Fairewinds Compared With The NRC's Like-For-Like Criteria

50:59 Criteria (A)	(B) Remove stay cylinder	Change tube sheet	Tube alloy change	Add tubes	Change tube support	Add flow restrictor	Additional water volume	Feed water distribution ring
i – Accident Frequency Increase	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	No	No	No
ii – Increase in SSC Malfunction occurrence	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	No	No	No
iii - Accident consequent increase	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	Yes (2)	Yes (2,5,6)	No
iv - Increase in SSC consequence of malfunction	Yes (1)	Yes (1)	No	Yes (3,4)	Yes (3,4,8)	Yes (2)	Yes (2,5,6)	No
v - Create unanalysed accident	Yes (1)	Yes (1)	No	No	No	Yes (2)	Yes (2,5,6)	Yes (3,7,8)
vi – Create new malfunction	Yes (1)	Yes (1)	No	No	Yes (3,8)	Yes (2)	No	Yes (3,7,8)
vii – Alter fission product barrier	Yes (1)	Yes (1)	No	Yes (3)	No	No	No	No
viii – Change design basis evaluation method	Yes (2)	Yes (2)	No	Yes (2)	Yes (2,8)	Yes (2)	Yes (2,5,6)	No

Table Footnotes

A - The criteria listed in the left column in the table above refers to the criteria as laid out in the NRC Guidelines[5] which states as follows:

"(2) A licensee shall obtain a license amendment pursuant to § 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than

any previously evaluated in the final safety analysis report (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses."

B – The horizontal axis contains a list of design changes made by Edison and whether they meet or have not met the criteria as set out in 10 CFR 50.59.

1 – The Steam Generator Replacement Project modified the tube sheets and stay cylinder that are a containment barrier – The NRC was not informed nor did it specifically approve these changes to the containment barrier as they were apparently not addressed under Edison's analysis for the 10 CFR 50.59 process;

2 – The Mitsubishi thermo hydraulic code is inadequate to assess flow inside the Steam Generators that dramatically affect the ability to cool the nuclear reactor core in the event of an accident;

3 – The Steam Generator Replacement Project increases the consequences of a steam line break accident; how??

4 – The Steam Generator Replacement Project has already proven to increase the frequency of tube failure;

5 – The Steam Generator Replacement Project changed the volume of primary coolant because more tubes were added, which changes the Final Safety Analysis Report;

6 – The Steam Generator Replacement Project changed the flow rate of primary coolant, which changes the Final Safety Analysis Report;

MODERATE CHANGE only

7 – The Steam Generator Replacement Project changed the potential for water hammer. Given that the Mitsubishi thermo hydraulic code is inadequate, the potential for water hammer is increased; how so??

WATER HAMMER IMPROVED

8 – The Steam Generator Replacement Project created steam binding at top of steam generator. The steam generator is designed to remove heat in the event of an accident and its role has been compromised.

COMMENT NOT CLEAR

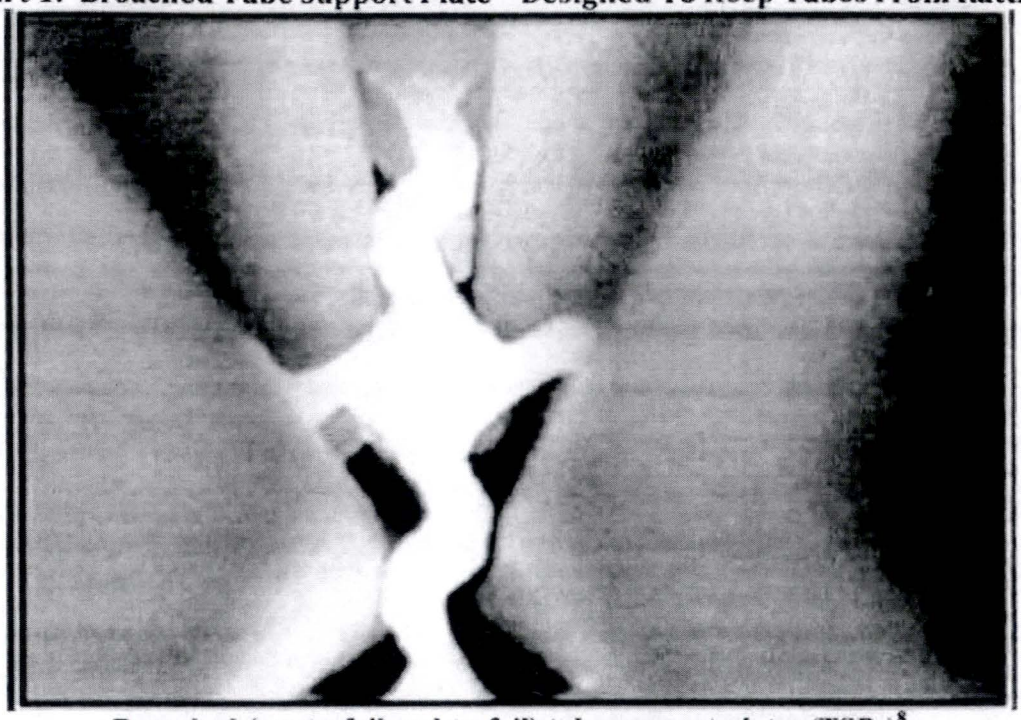
Ramifications Of Edison's Decision To Avoid The License Amendment Process

Edison's strategic goal was to avoid the process of license amendment according to the January 2012 article in Nuclear Engineering International NEI Magazine[6]. Had Edison notified the NRC that the new steam generators at San Onofre were not a *like-for-like* replacement, a more thorough review through the license amendment process would have been required. Given that scenario, it is likely that the requisite and thorough NRC review would have identified the design and fabrication inadequacies that appear to have caused the San Onofre steam generator tube failure.

More specifically, Fairewinds believes that the NRC would have identified the inadequacy of the Mitsubishi Heavy Industry computer code applied to validate the tube design and vibration pattern prior to fabrication. Mitsubishi's computer code was simply not capable of analyzing Combustion Engineering (CE) designs like San Onofre and was only qualified for Westinghouse designs that are not similar to the original CE steam generator design. In NRC licensing jargon, the Mitsubishi design codes were not benchmarked for the CE Design[7]. true

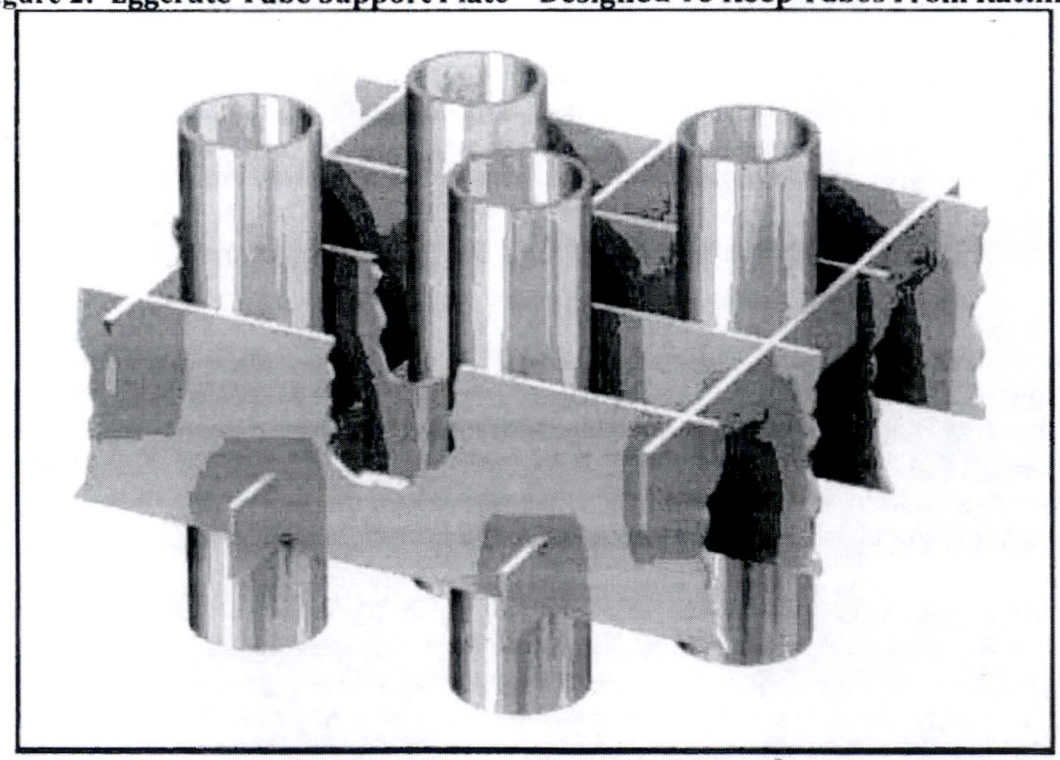
While Mitsubishi Heavy Industry has been supplying steam generators for many years in Japan, it did so under a specific license from Westinghouse for Westinghouse nuclear reactors. Although Mitsubishi made several incremental changes to the Westinghouse design, such as switching to alloy 690 tubing and the use of stainless steel broached plate tube supports, Mitsubishi has had very little experience with the tight tube pitch and the egg crate design used in the original CE design for San Onofre.

Figure 1: Broached Tube Support Plate – Designed To Keep Tubes From Rattling



Broached (quatrefoil and trefoil) tube support plates (TSPs)[8]

Figure 2: Eggcrate Tube Support Plate – Designed To Keep Tubes From Rattling



Horizontal Tube Supports (Eggcrate)[9]

The original steam generators designed and manufactured by CE for San Onofre were successfully operated 28 years. Moreover the original steam generators had a triangular tube pitch pattern, very closely packed U-tubes, and unique egg-crate tube supports that kept the tubes from vibrating and colliding. The pitch to diameter ratio of tubes in the original CE generators is dramatically different from any of the Westinghouse generators fabricated by Mitsubishi. Moreover, an NRC licensing review would have identified the fact that the Mitsubishi computer design code, which is based upon Westinghouse models, was not appropriate for design changes to the San Onofre replacement steam generators originally designed by CE.

Another problem with the San Onofre steam generators is that Edison and Mitsubishi made a very significant design change that magnified the San Onofre steam generator stresses and vibrations by removing the main structural pillar called the stay cylinder in order to fit an additional 400 tubes into the unique and already tightly packed design. Furthermore, this design is also bigger than anything Mitsubishi Heavy Industries (MHI) had ever fabricated or designed. The NRC license amendment review process would likely have identified these and other problems.

The Actual Steam Generator Problem

As water moves vertically up in a steam generator, the water content reduces as more steam is created. When the volume of steam is much greater than water then the flow resistance of the water/steam mixture passing through the tube supports accounts for one third of the total resistance at the top of the steam generator. Therefore to avoid vibration at the top of the tubes, Mitsubishi needed to specifically analyze the type of tube support to use in this unique application.

FLOW BLOCKAGE IS LESS FOR EGGCRATE DESIGN

The flow resistance of the Mitsubishi broached plate is *much higher* than that of the original Combustion Engineering egg crate design because the tubes are so tightly packed in the original CE San Onofre steam generators. By reviewing the documents thus far produced, it appears that due to Mitsubishi's fabrication experience with broached plates, both Edison and Mitsubishi missed this key difference in the design and fabrication of the new San Onofre steam generators.

Not only is Mitsubishi unfamiliar with the tightly packed CE design, but also Edison's engineers created so many untested variables to the new fabrication that this new design had a significantly increased risk of failure. As a result of the very tight pitch to diameter ratios used in the original CE steam generators, Mitsubishi fabricated a broached plate design that allows almost no water to reach the top of the steam generator.

The maximum quality of the water/steam mixture at the top of the steam generator in the U-Bend region should be approximately 40 to 50 percent, i.e. half water and half steam. With the Mitsubishi design the top of the U-tubes are almost dry in some regions.[10] Without liquid in the mixture, there is no damping[11] against vibration, and therefore a severe fluid-elastic instability developed.

— TRUE —

In response to the Edison/Mitsubishi steam generator changes, the top of the new steam generator is starved for water therefore making tube vibration inevitable. Furthermore, the problem appears to be exacerbated by Mitsubishi's three-dimensional thermal-hydraulic analysis determining how the steam and water mix at the top of the tubes that has been benchmarked against the Westinghouse but not the Combustion Engineering design.

The real problem in the replacement steam generators at San Onofre is that too much steam and too little water is causing the tubes to vibrate violently in the U-bend region. ~~The tubes are quickly wearing themselves thin enough to completely fail pressure tests. Even if the new tubes are actively not leaking or have not ruptured, the tubes in the Mitsubishi fabrication are at risk of bursting in a main steam line accident scenario and spewing radiation into the air~~.

This Tube Damage Cannot Be Repaired

Edison claims that the proximate cause of these U-tube failures at San Onofre is high vibration, and it has embarked upon a process of plugging some of these damaged tubes in hopes of quickly restarting one or both units. Fairewinds believes that this damage is occurring on the outside of the tubes where they collide with each other, while access to the tubes for repair and/or plugging can only be conducted from inside the tubes. Space limitations due to the tight fit of the 9,700 tubes (19,400 holes in the tube sheet) in each steam generator have made it impossible to access the outside of the U-tubes for inspection where the wear is actually occurring.

Presently, the Edison approach is to plug tubes in the most heavily damaged zone of each steam generator. Plugging the tubes only eliminates the radioactive water inside the tubes, but it does not eliminate the vibration, so the plugged tubes will continue to vibrate and damage adjacent tubes. More than 500 tubes have already been plugged in Unit 2 and more than 800 tubes have been plugged in Unit 3.[12] The number of plugged tubes is still considerably smaller than the number of tubes already ascertained as damaged in both steam generators. To date, Edison has not provided adequate data to compare damaged tubes to plugged tubes.

Initially, in March 2012, Edison claimed that as part of the Electric Power Research Institute's (EPRI) criteria used in the in-situ pressure testing of the Steam Generators, it was required to plug about one dozen tubes in the San Onofre steam generators. However, in May 2012, Edison announced it had plugged 1300 tubes, more than one hundred times the number of tubes required by the EPRI criteria. According to the industry steam generator experts interviewed by Fairewinds, Edison did not plug these additional tubes because they had failed, but rather Edison needed to plug these particular tubes because they would likely fail during a main steam line break accident.

If a steam line break accident were to occur, the depressurization of the steam generator caused by the steam line break coupled with the lack of water at the top of the steam generators would cause cascading tube failures, involving hundreds of tubes. The cascading tube failures would pop like popcorn and the cascading failures would cause excessive offsite radiation exposures. In an attempt to avoid a severe steam line break accident Edison prophylactically plugged additional tubes.

~~SG~~ MSLB $\Delta P = 2560$ psi

Fairewinds investigation has found that plugging the tubes is not a sure solution, because it fails to deal with the root causes of a failed design and it relies upon the incorrectly applied Mitsubishi 3-Dimensional steam analysis to determine which tubes should be plugged. Realistically, the 3-D steam analysis is not accurate enough to apply to such important safety-related determinations. To make such mathematical risk 3-D analysis, a very large margin of error must be applied, and that has not been done. For example, if the 3-D steam analysis determines that plugging 100 tubes is a solution, then plugging ten times that number might be the appropriate solution due to the mathematical errors in the 3-D analysis being applied by Edison and Mitsubishi.

Fairewinds concludes that plugging the tubes will never solve the underlying problem because vibration is the result not the root cause of the steam generator problems at San Onofre. The actual problem is a variety of design changes that have caused too much steam and too little water at the top of the steam generators. Plugging tubes cannot repair these design changes created and that are causing the tubes to collide with each other.

The tubes that Edison has already plugged on the inside will continue to vibrate because they are being pushed by steam and water from the outside. Therefore Fairewinds concludes that Edison's solution of plugging the inside of the tubes will not lessen the risk of an accident or stop the ongoing vibrational damage that is occurring to the inaccessible outside of the San Onofre steam generator tubes. —tubes are plugged are stabilized

Options For Continued Operation Of The San Onofre Reactors

Complete Replacement

The ongoing plugging of the tubes will not eliminate the vibrational failure mechanism causing tube failures. Over time, the damaged tubes that are plugged will in turn damage more tubes. Therefore, Fairewinds believes that the only sure solution to this significant safety issue is to once again cut open the reactor containment and install new steam generators that replicate the original CE design.

Due to the significant risk of a steam generator tube rupture accident in such a highly populated and vulnerable area, both San Onofre Unit 2 and Unit 3 should remain shut down until such a significant safety threat can be mitigated with the fabrication of new *like-for-like* steam generators adhering to the original CE design. If all the appropriate steps are taken in design and fabrication of new CE replica steam generators, and the proper procedures are taken to repair and reseal the San Onofre containment coupled with requisite NRC oversight, Fairewinds estimates that the entire process might take Edison approximately four years and cost in excess of $800,000,000, not including replacement power while the Units remain shut down.

Repair In Place

While technically this would be an extremely challenging repair process, it may be possible to cut the steam generators apart while still inside the containment. Such a process would take approximately 18 months to make repairs and then weld the steam generators back together again without cutting the containment open. Cutting the top off the steam generators would allow construction personnel access so that additional supports could be inserted into the U-tube region. Smaller replacement packages would fit through the existing equipment hatch and the containment would not be compromised another time. The cost for these repairs would be less than completely redesigning and manufacturing new steam generators and replacement power costs would be less. However, it is still reasonable to estimate that this cost would exceed $400,000,000 without replacement power, not including replacement power while the Units remain shut down.

There are two possible alternatives, both of which would require an additional analysis of the overall steam generator flow patterns to ensure that no new problems are created in the process of attempting to mitigate the damage from these design flaws and fabrication errors. The two alternatives are:

1. Because too much steam and too little water in the U-bend region cause the vibration problems, it might be possible to reduce the steam/water mixture qualities in the U-bend area by changing the internal structures to divert some of the internal recirculating flow into the U-bend region.
2. Another possible solution would require replacing the steam-water separators. The Mitsubishi separators require a water level that is quite low in the steam drum, and cannot be raised. Changing the separators to a different design may allow more water to reach the top of the tubes and thereby stop the tube vibration and wear.

Power Reduction

Reducing power does not provide a remedy for the underlying structural problems that are creating the vibration that has damaged and will continue to damage tubes deep inside the San Onofre steam generator. Edison has suggested that plugging tubes and operating at indeterminate reduced power levels for the remainder of the life of the plant may be a *solution* to the San Onofre tube vibration problem. Unfortunately this course of action would leave San Onofre operating with a significant safety risk if the NRC were to allow the reactors to restart.

The concept of reducing the power output from the San Onofre reactors will not change either the inside steam generator tube water temperature or the steam temperatures outside of the tubes. Reducing the power output will also not change the 2200-pound per square inch pressure within the tubes or the 1,000-pound pressure outside the tubes. Operating at reduced power will not prevent previously damaged tube supports and plugged tubes from vibrating and damaging surrounding tubes and tube supports, and it will worsen the existing damage.

More importantly, Fairewinds concern is that operating the San Onofre reactors at a lower power and flow rate might actually create a resonate frequency within the steam generators at which some of the tubes will vibrate as bad or worse than they did originally.

Because the plugged tubes are now filled with air their weight has changed, and therefore the plugged tubes will vibrate with a different amplitude and frequency. The inaccuracies in the Edison and Mitsubishi computer code do not allow Edison and Mitsubishi to conduct a resonant frequency analysis proving that such a problem will not occur.

It is impossible to determine exactly what is happening inside an operating steam generator. For example, at Millstone 2, a smaller CE reactor, the steam generator tube supports began to disintegrate due to vibration, and there was no method to alert the operations staff that such deterioration was occurring. This challenging problem was finally detected when the Millstone 2 was shut down for a refueling, and small cameras meant to inspect the steam generator found rubble on the tube sheet at the base of the tubes.

Historical evidence from other operating nuclear reactors that have attempted to mitigate vibrational damage by using power reductions rather than solving the resonant frequency issues have in fact compromised other nuclear safety related components by operating at reduced power.

- In 2002 the Exelon Quad Cities Nuclear Power Plant in Illinois operated its Unit 2 reactor at reduced power in order to eliminate vibrationally induced damage causing high moisture carryover in its steam dryer. While the power reduction temporarily reduced moisture carryover, the problem reoccurred and a shutdown was ordered causing an extended unplanned outage. Vibrationally induced severe cracking was discovered in the steam dryer and repaired. Following an analysis and subsequent repairs, Exelon claimed to have rectified the Quad Cities Unit 2 problems only to be forced in 2003 to once again attempt operation at a reduced power level when vibrationally induced steam dryer moisture carryover became excessive. Following this second attempt to operate the reactor at a reduced power level, pieces of the dryer as large as a man broke off and damaged nuclear power safety related components, and a second unplanned extended outage ensued. Once again, vibration was determined to be the cause of the gross failure and another unplanned and forced outage. Finally, following years of analysis and two damaged steam dryers, Quad Cities made major piping modifications that are alleged to have eliminated harmonic frequencies, prevented further component damage, and allowed Unit 2 to eventually return to full power production.[13]

- A second example of a failed attempt to reduce power to solve vibrationally induced resonance frequency problems occurred at the Susquehanna nuclear plant in Pennsylvania. During the mid 1990s, a vibrationally induced failure in the jet pump sensing lines occurred at Susquehanna. This failure was attributed to the vane passing frequency from the recirculation pumps causing harmonic vibration of the lines. Like Quad Cities, Susquehanna attempted to implement a power reduction in order to minimize the harmonic vibrations. Unfortunately, the resonant vibration issues continued to damage systems after the power was reduced thereby forcing an unplanned outage and extensive modifications and repairs.[14]

Conclusion

In conclusion, the NRC has stated that nuclear power plants like San Onofre cannot risk compromising critical safety systems and possible radiological contamination in an effort to return to operation before a thorough root cause analysis, modifications, and subsequent repairs are adequately reviewed by the NRC and implemented. Historical evidence has proven that power reductions do not solve underlying and serious degradation problems, resonance frequency issues. Rather, power reductions can significantly increase the risk of unplanned, forced outages during times of peak demand and can cause significant risk to public health in the event of a single tube rupture or a series of ruptures if the main steam line were to break.

Finally, if a steam-line accident were to occur, vibrationally induced tube damage at San Onofre could cause an inordinate amount of radioactivity to be released outside of the containment system compromising public health and safety in one of the most heavily populated areas in the entire United States.

Note:
This report represents the opinion of Fairewinds. Industry insiders, who have had lengthy careers in steam generator design, fabrication, and operation, and who have chosen to remain anonymous, have assisted Fairewinds with research for this report, but are not responsible for its content.

Endnotes

[1] Improving Like-For-Like Replacement Steam Generators by Boguslaw Olech of Southern California Edison and Tomouki Inoue of Mitsubishi Heavy Industries, Nuclear Engineering International, January 2012, page 36-38. http://edition.pagesuite-professional.co.uk/launch.aspx?referral=other&pnum=36&refresh=K0s3a21GRq61%20&EID=af75ecb1-5b23-49be-9dd6-d806f2e9b7b5&skip=&p=36

[2] Ibid.

[3] Vibration source: March 27, 2012, Confirmatory Action Letter – San Onofre Nuclear Generating Station, Units 2 And 3, Commitments To Address Steam Generator Tube Degradation

[4] See, 1.187-A-1, ibid, ttp://pbadupws.nrc.gov/docs/ML0037/ML003759710.pdf

[5] Ibid.

[6] Improving Like-For-Like Replacement Steam Generators by Boguslaw Olech of Southern California Edison and Tomouki Inoue of Mitsubishi Heavy Industries, Nuclear Engineering International, January 2012, page 39. This article was based on a paper published at ICAPP 2011, 2-5 May 2011, Nice, France, paper 11330. Boguslaw Olech, P.E., South8fn California Edison Company, 14300 Mesa Rd., San Clemente, CA 92674, USA, Email: bob.olech@sce.com.
Tomoyuki Inoue, Mitsubishi Heavy Industries Ltd. (MHt), 1-1 Wadasaki-cho 1-Chome, HyogoKu, Kobe, Japan 652 8585, Email: tomoyukiJnoue@mhi.co.jp.
The authors wish to acknowledge all Edison and MHI personnel involved in the SONGS steam generator replacement project for their efforts to make this project a success.

[7] ~~This statement is based upon Fairewinds analysis and confirmed by two independent industry steam generator experts who wish to remain anonymous.~~

[8] http://westinghousenuclear.com/Products_&_Services/docs/flysheets/NS-ES-0073.pdf

[9] Figure 13-7 http://www.kntc.re.kr/openlec/nuc/NPRT/module2/module2_6/2_6.htm

[10] With the Mitsubishi design the top of the U-tubes are almost dry in some regions. Fairewinds research and four independent industry experts, who wish to remain anonymous, substantiate this statement.

[11] Damping [**dam**-ping] ***noun*** *Physics.*

1. a decreasing of the amplitude of an electrical or mechanical wave.

2. an energy-absorbing mechanism or resistance circuit causing this decrease.

3. a reduction in the amplitude of an oscillation or vibration as a result of energy being dissipated as heat.

http://dictionary.reference.com/browse/damping?s=t

[12] http://www.songscommunity.com/docs/SONGS19_CALFactSheet_050712.pdf

[13] http://pbadupws.nrc.gov/docs/ML0609/ML060960338.pdf

[14] http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1995/in95016.html

Introduction

Replacement Steam Generators (RSGs) are designed and manufactured in accordance with licensee Design Specification (Ref. 5) and a Procurement Specification in accordance with licensee-specified applicable year issue of Section III of the ASME Code. The RSGs have enhanced materials and maintenance features but are, based on the licensee 50.59, a like for like replacement of the original steam generators (OSGs).

Like the OSGs, these are recirculating type generators with inverted U-tubes where reactor coolant enters through the primary inlet nozzle into the channel head that is divided into two leak-tight compartments by a partition plate. The reactor fluid flows through the U-tubes heating the feed and recirculation water mixture and then exits through the outlet channel head nozzle. On the secondary side, feedwater enters through a nozzle and is distributed via a feedwater pipe ring equipped with inverted J-tubes. Feedwater sprays down into the downcomer region periphery where it enters the tube bundle at the top of the tubesheet. A steam /water mixture is produced in the bundle boiling section where it is diverted to separation equipment to make dried steam (first stage of cyclone separators and a second stage of dryers) while the separated recirculating water returns to the downcomer to pass through boiler region again. Dry steam exits to the main steam system through the steam outlet nozzle.

The tube bundle, comprising 9727 U-tubes, is supported by a set of seven tube support plates (TSPs) which are maintained and spaced by a network of tie-rods. The ends of the U-tubes are welded onto the tubesheet lower face cladding and are full depth expanded in the tubesheet holes. The U-bends are supported by a set of 6 anti-vibration bars (AVBs), having a maximum of 12 contact points, i.e., in the center of the bundle. For shorter tubes near the periphery, a fewer number of AVBs are present.

One of the major enhancements of the RSG is use of Alloy-690 tubing versus Alloy-600 for corrosion resistance. However, Alloy-690 has lower heat conductivity so, to achieve same power, tube heat transfer surface area must be increased by at least 10%. This requires more tubing and a more tightly compacted tube bundle. Other operational and physical comparisons of the RSG and OSG, contained in the licensee 50.59 (Ref. 8), were reviewed and no significant differences were noted.

T/H Assessments

The RSGs' thermal hydraulic operation and design responses are based on its geometric characteristics and the flow and temperature of the reactor primary fluid and the secondary feedwater fluid. Calculations are performed for 0% to 100% power steady-state operational BOL (beginning of life) conditions and for EOL conditions considering limiting tube plugging and fouling. Important operating parameters are saturation pressure, circulation ratio, steam flowrate, tube and shell side pressure drops, water and steam inventories, and the global heat transfer coefficient. MHI used the SSPC code (Ref. 1) to compute these operational parameters, the FTI-III code (Ref. 2) to determine 3D fluid flow conditions, and the FIVATS code (Ref. 3) to compute tube stability ratios. In addition, the ABAQUS code was used compute stress and natural vibration frequency, and a code called IVHET was used for tube wear analysis. Of

these, the key design code for tube bundle design and vibration analysis is FTI-III since it computes the forcing function, ρV^2.

The MHI acceptance criteria for vibration was to avoid fluid-elastic instability (FEI) of tube spans insuring that calculated stability ratios do not exceed 1 using suggested approach given in the ASME code Appendix N-1330 and to avoid natural frequencies of the tubes that may interfere with RCP dynamic frequencies.

As a general rule, for optimum design, maximum stability ratio should never exceed 0.75 and should generally be less than 0.5. The licensee design spec (Ref. 5) does not address specific criteria for stability ratio and does not mention FEI. Although the design values generated by MHI appear to meet these criteria (calculated design values ≤ 0.5), the design spec should have mentioned the potential of this mechanism.

During design specific FIV analysis are performed for select U-bend tubes exposed to the greatest vibration risk, generally those with longest unsupported length under most limiting operating condition, i.e., lowest steam pressure (EOL design conditions). The phenomenon of FEI of tubes is characterized by cross-flow velocity (for out-of-plane mode) and normal or axial velocity (for the in-plane mode) where the local gap velocity exceeds a critical velocity value (given via Conner's Equation). The accuracy of this equation however is limited based on user input constants that are best determined by design-specific mockup test data. MHI did not perform mockup tests but used generally accepted test data, and some were based on MHI specific test riggs.

If operating velocities reach this critical value vibration amplitudes can increase rapidly and FEI forces can lead to rapid damaging of tubes. Turbulent random excitation can cause fretting wear between a tube and its support structures but its vibrational amplitudes are relatively small. Design should consider both mechanisms and traditionally, in-plane modes have not been considered by vendors since it was felt that the rigidity of the tube in this direction precludes it. This event at SONGS is the first industry occurrence of in-plane FEI in the US operating fleet sufficient to cause tube-to-tube contact and wear in a U-bend.

MHI mentioned in meetings that two conservatisms are added to their bundle vibration analysis (1) FTI-III gap velocities are multiplied by 1.5 and (2) 1 of 12 AVB contacts are assumed inactive. However, from MHI documents, Ref. 3, the 1.5 multiplier is not a conservatism but a requirement, needed to match test data results. Based on NRC request, MHI provided a comparison of their ATHOS results (see Table 2-1, Ref. 6) to their FTI-III design calculations, and the MHI ATHOS velocities are 2.5 to 3 times higher than the FTI-III velocities with the 1.5 multiplier applied. This implies that the FTI-III code computes non-conservative results even with the multiplier applied, and that possibly both the Unit 2 and 3 RSGs are under-designed in regard to FIV and FEI loads at normal operation. Note that stability ratios will increase by about the same factor, which will likely cause many design tube values to exceed 0.5 and several others to exceed the optimum design of 0.75. This suggests that there is much less margin to onset of FEI than the designers intended.

NRC ran some independent preliminary ATHOS calculations to compare SONGS Unit 2 and Unit 3 operating cycle differences and it was determined that the differences in T_{hot} and steam flow were negligible. NRC results indicated also that higher void fractions existed in the bundle than was computed by the MHI ATHOS and MHI design FTI-III analyses. This implied that NRC computed gap velocities in the U-bend would also be higher than those of the MHI ATHOS analysis.

NRC ran preliminary ATHOS calculations to compare SONGS RSG lower and upper bound T_{hot} case U-bend velocities to other similar CE plants including both RSG and OSG designs. The results, shown in Table 1, demonstrate that SONGS predicted velocities are indeed higher. It is not apparent which features in the RSG design is causing the void fraction and effective bulk velocities to be considerably higher than other similar design generators. One factor is the lower steam pressure (i.e., for the low T_{hot} case). But even for the higher T_{hot} case, the bulk field velocities are considerably higher than both the Plant 1 and Plant 2 results.

Table 1

Parameter	Plant 1 RSG	SONGS Low Thot	SONGS High Thot	Plant 2 OSG
Heat Transfer Area (ft**2)	-	115500	-	-
Design Power (MWt)	~1350	1729.00	-	~1700
No. Tubes (0% TP)	9000	9727	-	9350.00
Layout	U-Bend, Triangular	U-Bend, Triangular	-	Sq-Bend, Triangular
Dome Pressure (psia)	**909.5**	**840.0**	**942.0**	**900.0**
**Dome Temp. (Deg F)	533.2	523.8	537.4	531.9
Density Of Sat. Liq. (lbm/ft**3)	47.0	47.6	46.8	47.1
Density Of Sat. Vapor (lbm/ft**3)	2.02	1.85	2.10	2.00
*Feedwater Flow Rate (lbm/sec)	820.8	1053.9	1058.3	1050.3
*Prim. Flow Rate (lbm/sec)	10528.8	11081.8	10872.0	10277.5
*Bundle Flow Rate (lbm/sec)	2797.3	3523.7	3483.5	3584.5
*Steam Flow Rate (lbm/sec)	820.4	1058.7	1062.6	1050.3
Circulation Ratio	**3.42**	**3.33**	**3.28**	**3.42**
Downcomer Liquid Level (in)	411.3	451.6	-	438.6
Mass in Shroud (lbm)	25272.9	32371.8	33959.35	42629.5
Exit Steam Quality (%)	99.8	99.8	-	99.8
Calc. Power (MWt)	1352.4	1724.0	-	1708.0
**Feedwater Temperature (Deg F)	435.0	442.0	442.0	445.0
Primary Inlet Temp. (Deg F)	**598.9**	**599.8**	**611.6**	**611.0**
Primary Outlet Temp. (Deg F)	**552.2**	**542.1**	**554.9**	**552.2**
Max Void	0.980	0.998	0.998	0.970
Max Quality	0.670	0.945	0.940	0.610
**Max Field Velocity (ft/s)	12.139	21.325	19.685	14.764
- (m/s)	3.700	6.500	6.000	4.500

* Note for 1/2 Generator
** in Effective Area

Most of the work in two-phase flow for FEI has been carried out for air-water flows, however accepted industry data has shown that in staggered array (triangular pitch) bundles the onset of tube instability for modern steam generators (pitch/diameter=1.33) can start at tube gap velocities of about 6 meter/sec. Although mixture densities are lower, the ρV^2 forcing function is very much driven by high steam velocities. The NRC ATHOS analysis results indicates that there is a substantial localized region in the lower hot side of the RSG U-bends where velocities exceed 6 meter/sec and these are highest when operating at lower T_{hot} condition.

The NRC preliminary ATHOS calculations were also used to compare with MHI gap velocities computed both with the ATHOS code and their FTI-III code (Ref. 6). The NRC analysis included gap velocities results for the following tubes:

R1 C 89
R106 C 78
R142 C 88

Since the tube R142C88 is common for each of the analyses, it can be used as basis for general comparison. The effective velocities in meter/sec are 2.5 (FIT-III, raw w/o 1.5 mult. applied, see reference Figure 3.3-1), MHI ATHOS 5.6 (see reference Figure 4.3-1), and NRC ATHOS, 5.2. The NRC result is very close to the MHI ATHOS result. It should be noted that the result from reference Figure 4.3-1 is not consistent with that shown in their reference Table 2-1 so it is likely the table results are not correct or the wrong tube was used in their figure. Regardless, the results confirm that ATHOS computed gap velocities are significantly higher than values computed with FTI-III. It should also be noted that the NRC model does not explicitly model the AVB locations however this difference is not expected to appreciably affect the overall results.

The NRC U-bend gap velocities are shown in Figure 1a for tube R142C88. In addition, the results of the leaker tube (R106C78) are shown in Figure 1b and as indicated the gap velocities at 0 to 30 degrees segments up the U-bend are above 6.1 meter/sec (20 ft/sec).

These results strongly suggest that high velocities are a "primary" factor in the tube FEI failure and the excessive wear patterns observed.

NRC ATHOS Modeling

- Two cases (lower bound T_{hot}=598, upper bound T_{hot}=611)
- Boundary conditions from MHI Performance Analysis Report (Ref. 4)
- Both at BOL (100% power, 0% TP)
- 18x20x54 node Grid (w/ polar coordinates)

- Tubes and Tie rods modeled explicitly
- AVBs estimated (4 V bars only)
- TSPs adjusted due to model limitations to represent flow holes
- Ave blockage factor 0.272 used

- Algebraic slip with default Lellouche and EPRI78 HT package
- Boundary type (calc. T_{hot})
- Blowdown not considered, Feed=Steam
- Wrapper entry form loss tuned to MHI CircR of 3.33 (Ref. 4)

Figure 1a

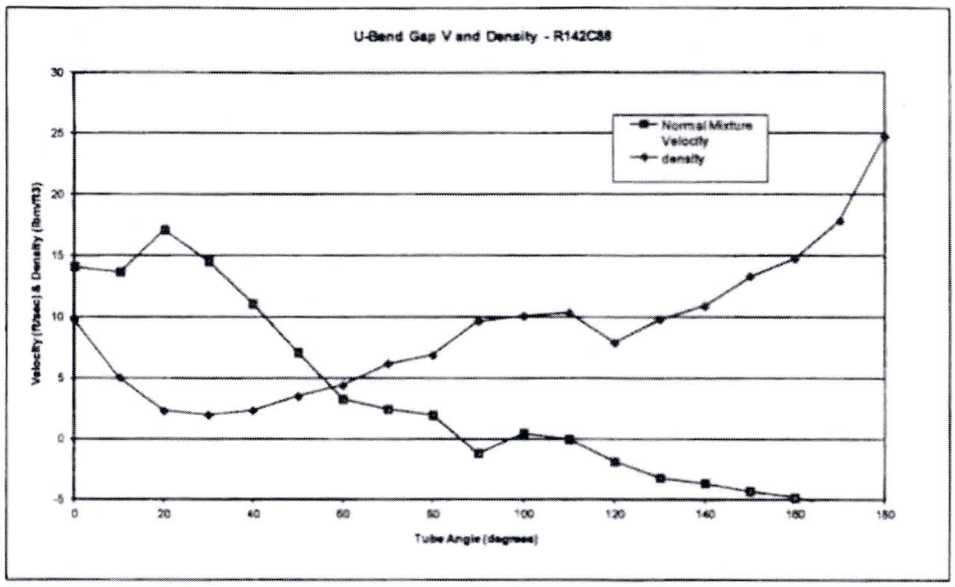
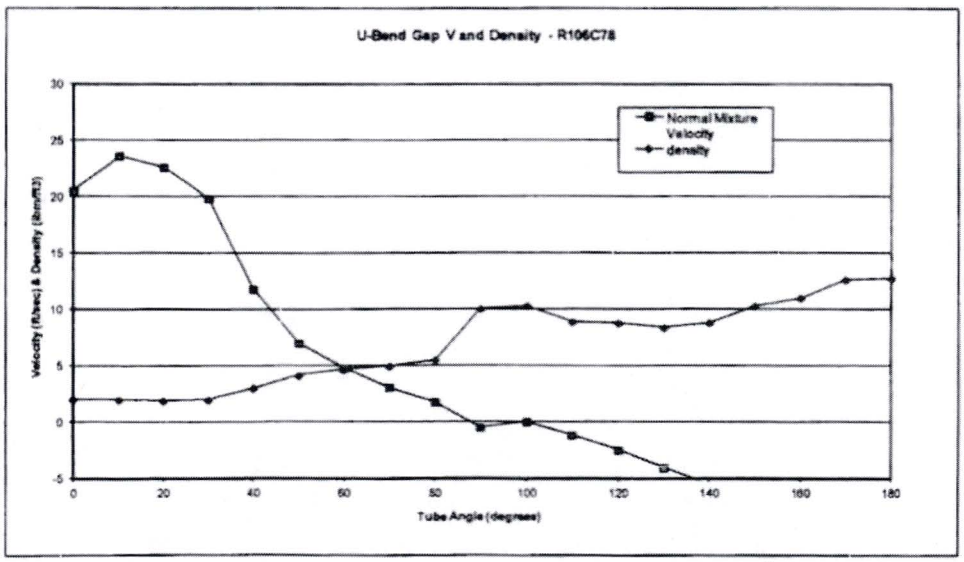


Figure 1b



Since the ATHOS version used cannot represent TSP flow holes, the effective areas are located in the circular center for bottom 2 TSPs, and in long narrow gaps for the upper 5 TSPs. The second approximation is that the AVBs are represented by 4 full across V bars (making 8 contact points). The geometry ATHOSGPP model is ran first and checked. The geometry results are shown in Figure 1c. Note ATHOS only models half of the steam generator hot side and half of the cold side, i.e., 180 degrees split by the open tube lane.

The ATHOS T/H model is then run to achieve the steady-state solution. The form loss input at the wrapper entrance is adjusted to obtain proper recirculation ratio in the bundle. After the model is tuned, the results are checked against the operation parameters from MHI (Ref. 4).

NRC ATHOS Results

Figures 2 through 5 show Case 1 results, with low T_{hot} and low steam pressure and present a 3D isometric encapsulation of steam qualities (red scale) at and above 0.9 (i.e., void fraction > 0.99) and field velocities at and above 6.5 meter/sec (white scale). Different angles of view are shown in Figures 2 through 4, and Figure 5 shows the result with expanded field velocities at and above 6.0 meter/sec.

The ATHOS model results of predicted high steam quality and field velocity does appear to closely align with the area of concern (tube-to-tube wear) in the Unit 3 RSGs and appears overlap dominant trends of significant wear noted in the U-bend AVB locations. In Figure 5a, the code predicted regions of high void fraction and high steam velocities (vertically located z-axis cut at about 20 inches above the 7th TSP) are superimposed with tube-to-tube wear indications from Unit 3 SG 88.

It is reasonable that the tube-to-tube wear indications align with AVB wear indication trends, however, the AVB wear patterns appear to indicate a more square-like essentially rectangular behavior that suggests that there is also a mechanical, fabrication, assembly, and/or material contribution to the SONGS wear degradation.

Figure 1c

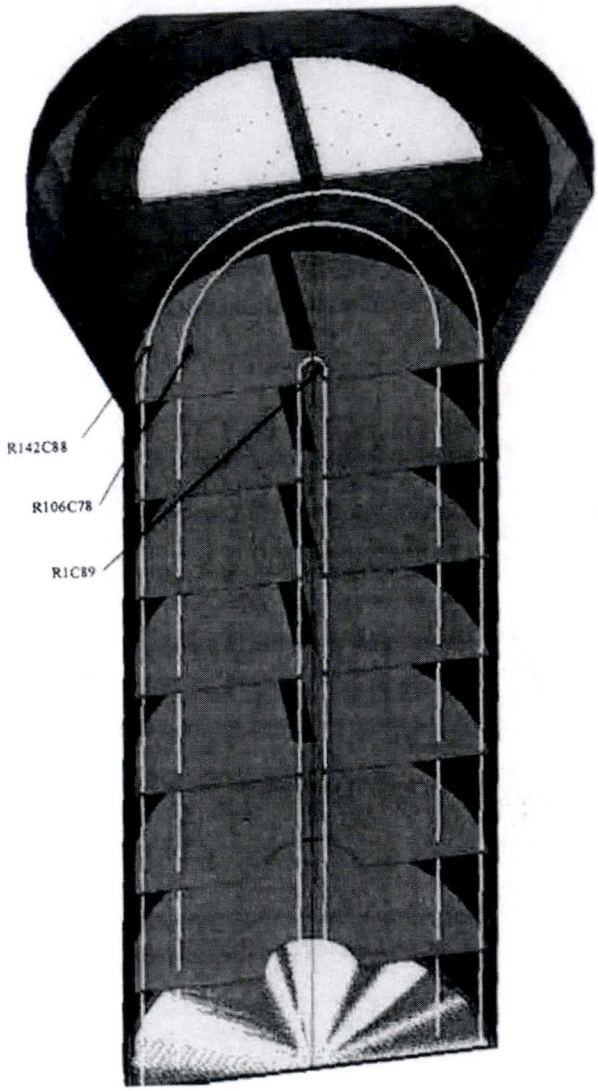


Figure 2

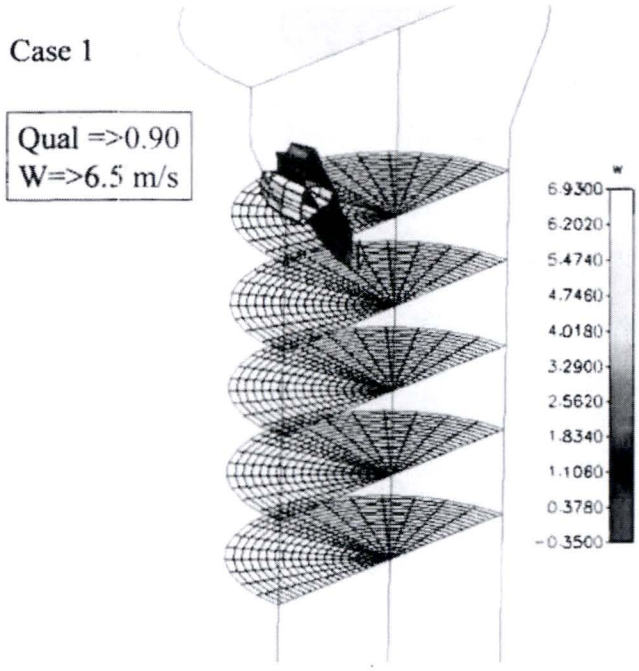


Figure 3

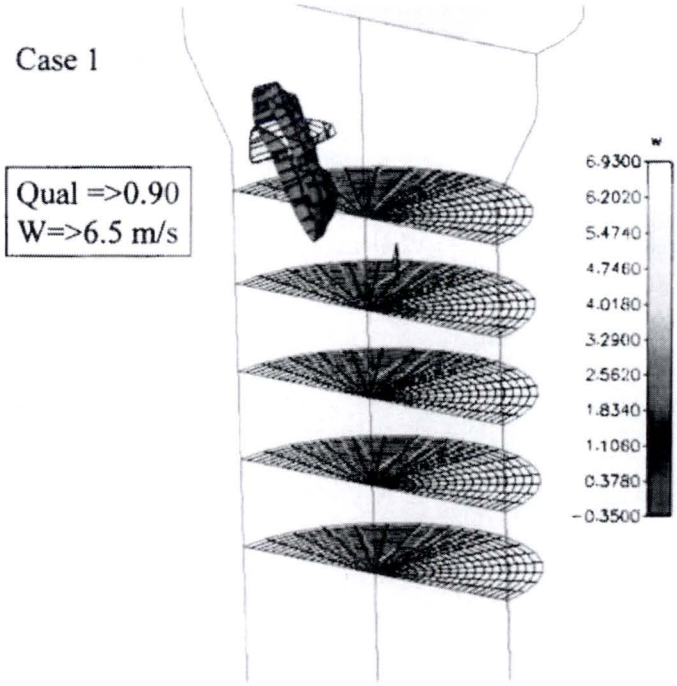


Figure 4

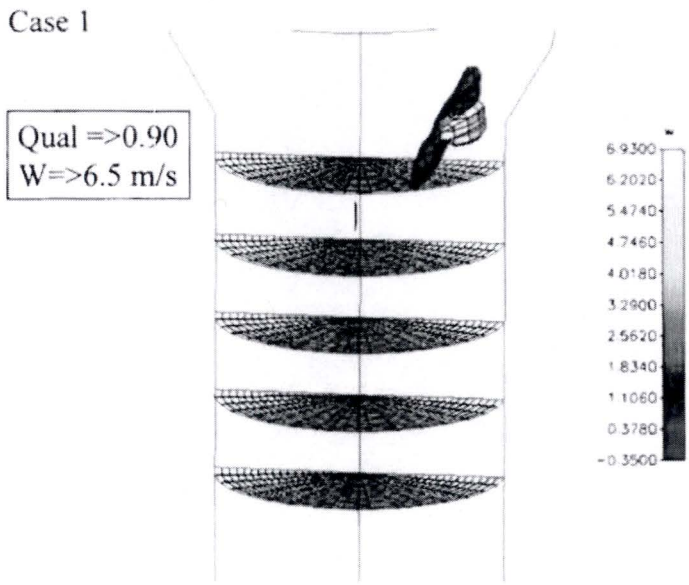


Figure 5

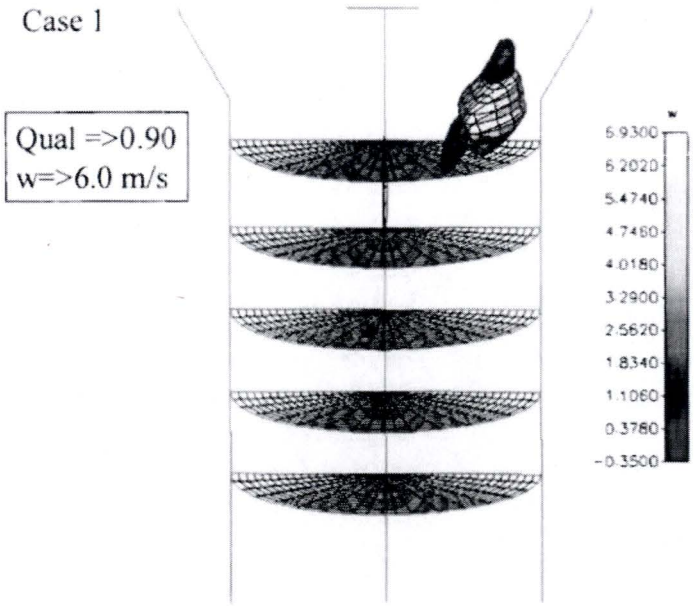
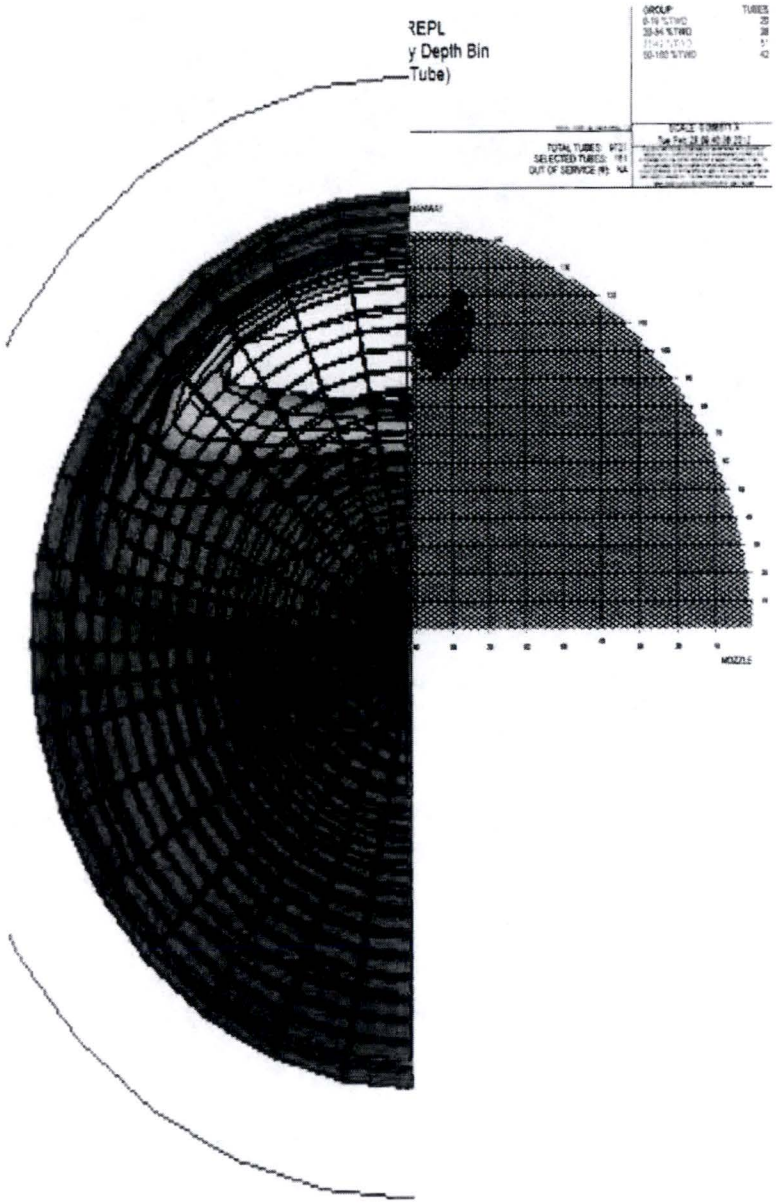


Figure 5a


REPL
y Depth Bin
Tube)
GROUP
TUBES
TOTAL TUBES
SELECTED TUBES
OUT OF SERVICE (#)
MANWAY
NOZZLE

Figures 6 through 8 show Case 2 results which has higher T_{hot} and consequently 100 psia higher steam pressure. The same isometric encapsulation of steam qualities (red scale) at and above 0.9 and field velocities at and above 6.0 meter/sec (white scale) are shown. The peak field velocities are about 10% less that the low T_{hot} case. Figure 8 shows a view with velocity vectors added. It indicates the proportional variation of axial velocities radially across the bundle and vertically up the bundle and at the wrapper transition region. The wrapper transition widening allows rapid acceleration and this is common characteristic of all re-circulating type steam generators. Note that vectors are included on the isometric surface (from Figure 6) and TSP locations added for comparison.

Figure 6

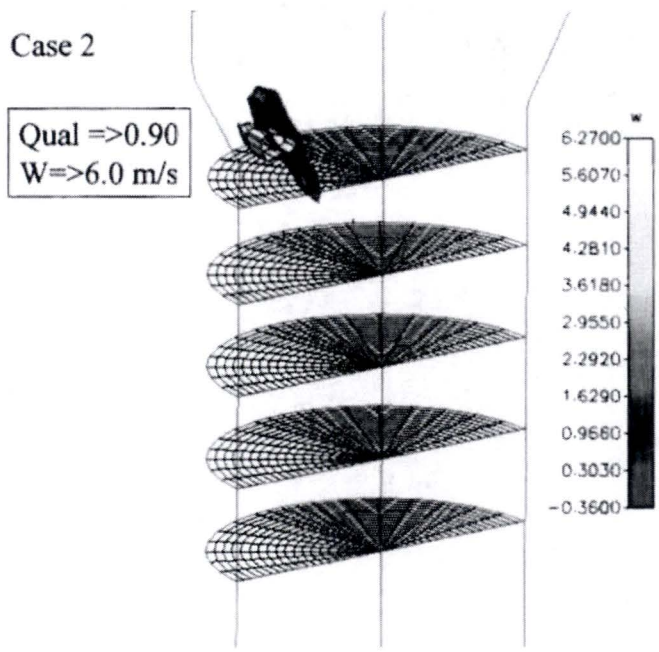


Figure 7

Case 2

Qual =>0.90
W=>6.0 m/s

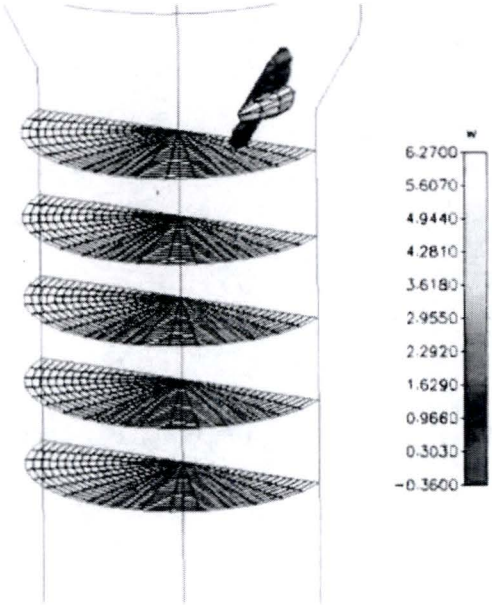
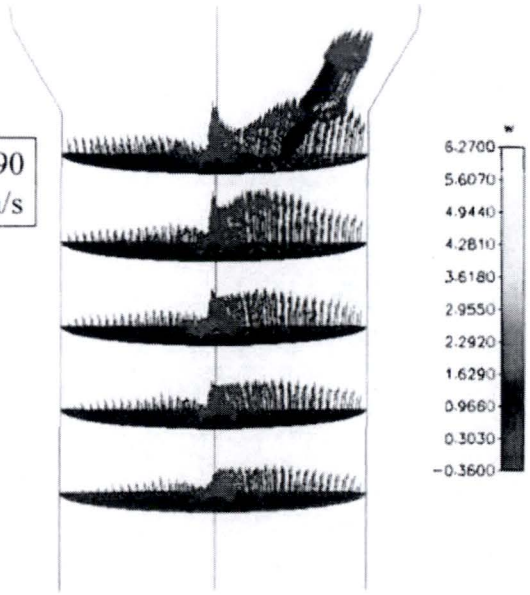


Figure 8

Case 2

Qual =>0.90
W=>6.0 m/s



The above analyses apply equally to Units 2 and 3 and so do not explain the accelerated FEI wear damage in Unit 3. This thermal-hydraulic model predicts bulk fluid behavior based on first principals and empirical correlations so it is not able to evaluate mechanical, fabrication, or structural material differences that may be unique to each steam generator.

T/H Conclusion

It appears that the SONGS RSGs thermal hydraulically may not be designed with adequate margin to onset of FEI. High fluid velocities in localized regions in the U-bend that border or exceed the critical velocity will continue to cause excessive tube wear and potentially accelerated wear that could result again in tube leakage. The design deficiencies appear to be related to MHI's FTI-III 3D thermal hydraulic code having predicted non-conservative low velocity results. These results lead MHI to conclude that margins to instability were significantly larger than they apparently are based on eddy current data, by NRC ATHOS analysis, MHI ATHOS analysis, and other on-going analyses (Westinghouse and AREVA) that are anticipated to reach a similar conclusion.

If the bundle designs are marginal to FEI, the best solution is to conservatively plug and stabilize the affected areas. Taking these tubes out of service should reduce the potential for localized fluid velocities reaching critical. This was done with the SONGS OSGs for batwing wear for a similar degradation mechanism based on Combustion Engineering analysis and mockup testing, Ref. 7, (for Cycles 1 and 2) and it proved successful.

151
174

This NRC analysis also supports licensee limited Unit 2 rotating (+Pt) probe ECT analysis scope to confirm existence of tube-to-tube wear in Unit 2 RSGs. The proposed area includes all rows above R79 between columns 70 to 110, and appears to reasonably bound AVB wear indication trends and those predicted based on analysis.

273
286

References

1. MHI Doc L5-04GA510 Rev. 5 "Thermal and Hydraulic Parametric Calculations".
2. MHI Doc L5-04GA521 Rev. 3 "3D Thermal and Hydraulic Analysis…".
3. MHI Doc L5-04GA504 Rev. 3 "Evaluation of Tube Vibration".
4. MHI Doc L5-04GA021 Rev. 3 "Performance Analysis Report".
5. SCE Doc SO23-617-1 Rev. 4 "Conformed Specification for Design and Fabrication of Replacement Steam Generators for Unit 2 and Unit 3".
6. MHI Response to AIT #181 handout "Comparison of Evaluation of Tube Vibration Based on Flow Characteristics obtained by ATHOS and FTI-III".
7. SCE Response to AIT #189 C-E Report "Evaluation of Steam Generator Tube and Diagonal Spacer Strip Interaction and Wear", March 1985.
8. SCE Response to AIT #066 Engineering Change Package NECP 8000717703 "Steam Generator Replacement, # Unit 3".

From: Gibson, Kathy
Sent: Monday, April 30, 2012 11:17 PM
To: Thurston, Carl; Hoxie, Chris; Elkins, Scott
Cc: Scott, Michael
Subject: Re: Have you heard of FIT-III code?

Thanks Carl.

----- Original Message -----
From: Thurston, Carl
To: Gibson, Kathy; Hoxie, Chris; Elkins, Scott
Cc: Scott, Michael
Sent: Mon Apr 30 18:45:25 2012
Subject: RE: Have you heard of FIT-III code?

I'll contact Jose March-Leuba, but I suspect Grady Yoder is the right person to start with. I'll update you when I know more.

Carl

From: Gibson, Kathy
Sent: Monday, April 30, 2012 2:04 PM
To: Thurston, Carl; Hoxie, Chris; Elkins, Scott
Cc: Scott, Michael
Subject: RE: Have you heard of FIT-III code?

Thanks, Carl. Brian has asked us to contact DOE, are you the right person to do that and do we know who at DOE to contact?

-----Original Message-----
From: Thurston, Carl
Sent: Monday, April 30, 2012 2:03 PM
To: Gibson, Kathy; Hoxie, Chris; Elkins, Scott
Cc: Scott, Michael
Subject: RE: Have you heard of FIT-III code?

Yes, it is the 3D thermal hydraulic code used by MHI to compute secondary side fluid conditions in the SONGS replacement steam generators. We believe this code is giving non-conservative results and may have contributed to excessive tube wear (and leak) being found at SONGS.

I heard from R-IV on friday that Brian Sheron had been asked to contact DOE regarding some previous review assessments they may have done for this code where deficiencies were found.

As a part of the AIT, we asked SONGS to confirm deficiencies in FTI-III as compared to ATHOS and the french code CAFCA4.

If previous assessments are found it helps our case. I am at NC A&T for student presentations today and tomorrow, back in the office on Wednesday.

Carl
(b)(6)

From: Gibson, Kathy
Sent: Monday, April 30, 2012 9:11 AM
To: Hoxie, Chris; Elkins, Scott
Cc: Scott, Michael; Thurston, Carl
Subject: Have you heard of FIT-III code?

MHI proprietary T/H code

DOE mentioned in relation to SONGS S/Gs

Please let me know whatever your guys know or can find out about this code.

Thanks!

From: Gibson, Kathy
Sent: Saturday, May 12, 2012 12:28 PM
To: Hoxie, Chris; Scott, Michael
Subject: Re: FIT-III Presentation by MHI and Expert Panel

Did we run any calcs with ATHOS? And what were the results?

From: Hoxie, Chris
To: Scott, Michael; Gibson, Kathy
Sent: Sat May 12 12:19:45 2012
Subject: Fw: FIT-III Presentation by MHI and Expert Panel

Good summary of status.

From: Werner, Greg
To: Thurston, Carl; Murphy, Emmett; Rivera-Ortiz, Joel; Ortega-Luciano, Jonathan; Johnson, Andrew; Reynoso, John
Cc: Lantz, Ryan; Blount, Tom; Bloodgood, Michael; Louden, Patrick; Vias, Steven; Kulesa, Gloria; Hoxie, Chris; Warnick, Greg; Werner, Greg
Sent: Sat May 12 00:43:00 2012
Subject: FIT-III Presentation by MHI and Expert Panel

Good Evening Team,

Today, both Carl and I observed the Expert Panel on FIT-III. It was very interesting to say the least, but the information provided was not that useful. MHI did NOT investigate the area of concern. My perception was they were intent on proving that FIT-III was ok and it was NOT the problem. I will let Carl add or modify what I understood from the presentation as describe below.

The following are some of the key points that I took away.

MHI Presentation Items of Interest:

1. MHI compared FIT III and ATHOS using a 6 to 8 tubes. None of these tubes were in the area of interest and did not exhibit ANY tube-to-tube wear. Based on the comparison of data, MHI concluded that FIT-III model was appropriately used during the design.
2. ATHOS yielded velocities 3 – 5 times higher than FIT-III and void fractions of approximately 0.995 (very high).
3. MHI technical lead acknowledged that their highest void fraction calculated of 0.95 was VERY high.
4. Using both the outputs from FIT-III and ATHOS, all stability ratios (SR) for the tubes selected (see Item 1), were less than 1. SRs calculated using FIT-III outputs yielded results significantly lower than ATHOS outputs. The highest SR using ATHOS outputs was 0.60, well below the limiting value of 1.0.
5. Tube stability based on ATHOS output was NOT calculated with one missing AVB. However, when asked, MHI personnel stated that if they had, there would have been a number of SRs > 1.0. MHI indicated that this was NOT a design consideration, but it was used by MHI as a check.
6. Margins to instability using ATHOS results are lower than using results from FIT-III (in most cases, 2 times lower).
7. Use of FIT-III results during design is not considered a direct cause for the tube wear.

Expert Panel:

1. They were not satisfied with the presentation from MHI.
2. Over 20 questions asked by the Expert Panel could not be answered by MHI and they owed answers to the Panel.
3. Not clear that MHI calculated SRs using the correct velocities in accordance with ASME code. Several Expert panel members thought the FIT-III velocities were extremely low.
4. MHI needs to calculate SRs using ATHOS outputs assuming one AVB inactive.

NRC Thoughts/Comments:

1. For calculated SRs using both ATHOS and FIT-III, the results to Carl and I were NOT surprising, because all of the tubes selected did NOT exhibit FEI.
2. Nobody was able to say which model is accurate. This is a requirement in order to use any model for justification for restart.
3. We told Pete Dietrich and Frank Gillespie, that SONGS needs to have MHI calculate velocities and void fractions using ATHOS and FIT-III on tubes that had tube-to-tube wear as a comparison because what was presented was of little use as described above – we would NOT expect any SRs to be greater than 1 because none of the tubes that they used in their comparison had any tube-to-tube wear.
4. Based on information presented during the AIT, if MHI does calculate the SRs using ATHOS with one inactive support, we strongly believe that a number of the original tubes used for their discussion WOULD have SRs >1.0.
5. The MHI presentation was of little value other than knowing and letting SONGS know that additional work is needed in this area.

Note – Frank Gillespie, former NRC Hqs SES, works for MHI (Senior Vice President) and was attending this meeting.

Greg Werner

From: Gibson, Kathy
Sent: Wednesday, November 07, 2012 9:09 PM
To: Scott, Michael
Cc: Hoxie, Chris; Thurston, Carl
Subject: Re: Proposed Review Schedule for SONGS CAL Response and Return to Service Report

Not aware. We should support if we can. Does NRR have someone they can give us on rotation to fill in for Carl?

From: Scott, Michael
To: Gibson, Kathy
Cc: Hoxie, Chris; Thurston, Carl
Sent: Wed Nov 07 18:04:14 2012
Subject: Fw: Proposed Review Schedule for SONGS CAL Response and Return to Service Report

Wanted you to be aware of this; will also be of interest to B/J. I discussed this matter with Tony Mendiola, Bill Ruland and Tom Blount this evening. Region wants NRR to provide review of licensee report that says ok to go to 70 percent power. Report references athos code. Region also wants Carl to input to nrr report on this code since he is sme. Sort of like Carl as Region person (his role here) advising nrr. I talked briefly with Chris and Carl about it. I think Carl is okay with doing this, though Chris has resource concerns (estimate another 1.5 person-months). Given priority, I think we have to provide, as Carl appears to be uniquely qualified to do the work. Chris and I also want to make sure Carl is comfortable with the tasking and what he is being asked to do.

There is likely going to be a call at div dir level tomorrow to further discuss path forward. I plan to be on call, and I think Chris and Carl will attempt to call in from camp. Will keep you posted.

Sent from my NRC blackberry
Michael Scott
(b)(6)

From: Hoxie, Chris
To: Scott, Michael
Sent: Wed Nov 07 14:23:15 2012
Subject: Fw: Proposed Review Schedule for SONGS CAL Response and Return to Service Report

Let's discuss.

From: Thurston, Carl
To: Hall, Randy
Cc: Broaddus, Doug; Murphy, Emmett; Parks, Benjamin; Paige, Jason; Hoxie, Chris; Jackson, Christopher
Sent: Tue Nov 06 17:52:43 2012
Subject: RE: Proposed Review Schedule for SONGS CAL Response and Return to Service Report

Randy,

I am in agreement with NRR here, we are not approving vendor codes. We're reviewing code analysis results, assessments, calculation methods, and may review code V&V documents but approval is not implied. I believe this is likely just a misunderstanding. Chris Hoxie and I are in our annual CAMP (code application and maintenance program) meetings being held in DC this Wednesday through Friday, we can breakout and support a call with you on Thursday at 10am.

Carl

From: Hall, Randy
Sent: Tuesday, November 06, 2012 3:11 PM
To: Thurston, Carl
Cc: Broaddus, Doug; Murphy, Emmett; Parks, Benjamin; Paige, Jason; Hoxie, Chris; Jackson, Christopher
Subject: RE: Proposed Review Schedule for SONGS CAL Response and Return to Service Report

Carl,

RIV has drafted a memo to NRR identifying the respective areas of review for the CAL followup inspection and the supporting NRR TER. NRR has provided comments on that draft memo in the attached email. Items 3.e and 3.f call for an evaluation of the ATHOS and FIV models and calcs, and the V&V of those models. There's been some discussion between NRR and RIV about the nature and scope of NRC review of those areas; NRR maintains that our review should not explicitly or implicitly approve the codes, while RIV thinks that we need to indicate that we've independently reviewed the licensee's actions, including the use of the modified codes. So, we need to reach agreement about what NRC will review and what findings we expect to make. Since you will be conducting those reviews, we'd like to talk with you and Chris Hoxie to make sure we are all on the same page. Please let me know when you'll be able to support a meeting or call.

Thanks,

Randy Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
USNRC
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