PUBLIC SUBMISSION

As of: May 17, 2013
Received: May 15, 2013
Status: Pending_Post
Tracking No. 1jx-85cl-jpbr
Comments Due: May 16, 2013
Submission Type: Web

Docket: NRC-2013-0070
Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination

Comment On: NRC-2013-0070-0001
Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generating Station, Unit 2

Document: NRC-2013-0070-DRAFT-0170
Comment on FR Doc # 2013-08888

4/16/2013
78 FR 22576
265

Submitter Information

Organization: DAB Safety Team

General Comment

Two Independent Nuclear Experts (Dr. Joram Hopenfeld and an Anonymous SONGS Insider) certify that MHI SG Tube Fatigue and Stress Calculations Assumptions are erroneous and based on faulty data – NRC is legally required to certify MHI's calculations to assure that San Onofre Unit 2 does not pose significant radiological risks at 70% normal steady state power operations during Anticipated Operational Transients and Design Basis Accidents.

Attachments

MHI San Onofre Fatigue Problem Statement

DAB Safety Team Part 4

DAB Safety Team Part 5

Reference: NUCLEAR REGULATORY COMMISSION [Docket No. 50-361; NRC-2013-00701, Application and Amendment to Facility Operating License Involving Proposed No Significant Hazards Consideration Determination; San Onofre Nuclear Generating Station, Unit 2]

Problem Statement: Two Independent Nuclear Experts (Dr. Joram Hopenfeld and an Anonymous SONGS Insider) certify that MHI SG Tube Fatigue and Stress Calculations Assumptions are erroneous and based on faulty data – NRC is legally required to certify MHI's calculations to assure that San Onofre Unit 2 does not pose significant radiological risks at 70% normal steady state power operations during Anticipated Operational Transients and Design Basis Accidents.

SCE Preliminary Response: Dr. Joram Hopenfeld Analysis concerning in-plane tube vibration is significantly flawed in that it applies an unreasonably high stress concentration factor based on solid body geometry rather than the more realistic stress concentration factors for a cylindrical geometry applicable to the SONGS steam generator tubes.

Introduction: The SG functions as a heat exchanger, by means of which the high temperature pressurized radioactive primary water on the inside of the tubes heats up the non-radioactive secondary water on the outside of the tubes, in order to generate the steam that turns the turbine which in turn generates electricity. In addition to providing a barrier (Reactor Coolant Pressure Boundary) to radioactivity and producing steam, a steam generator has many other important functions. It is the major component in the plant that contributes to safety during transients and/or accidents. A steam generator provides the driving force for natural circulation and-facilitates heat removal from the reactor core during a wide range of loss of coolant accidents. Proper steam generator operation is of major safety significance and therefore any changes to its design may have significant safety consequences.

Out-of-plane fluid-elastic instability has been observed in nuclear steam generators in the past and has led to tube bursts at normal operating conditions. However, the observation of in-plane fluid-elastic instability in steam generators of a nuclear power plant is a true paradigm shift. The combined effects of tube-to-tube wear and high cycle thermal fatigue cracks caused by fluid-elastic instability and/or flow-induced random vibrations have been witnessed as sudden tube ruptures in North Ana in 1987, MHI's SG in Mihama, Japan in 1991, three tube leakages in French SGs between 2004 through 2006, twenty tube ruptures/leakages in SGs between 1980-2000 in USA, and SONGS 3 in 2012.

Defect or Deviation: In San Onfre replacement steam generators, the relative motion between the tubes and the anti-vibration bars (AVBs), the tube support plates, and the retainer bars have resulted in tube wear and fatigue due to fluid elastic instability, flow-induced random vibrations, excessive fluid hydrodynamic pressures and Mitsubishi Flowering Effect. These adverse phenomena can produce relatively quick tube failures when the stresses generated during vibrations are sufficiently large. As described in Unit 2 Return to Service Report, Attachment 4, MHI Document L5-04GA564, Appendix 16, page 459 of 474, MHI used a finite element model ("FE"), to calculate that the tubes were subjected to a stress of 4.2 ksi (kilopounds per square inch) . Consequently, MHI concluded that the stress on the tube due to in-plane vibration is under fatigue limit (13.6 ksi) and the structural integrity of the tube is confirmed from the view point of fatigue due to in-plane vibration (page 470 of 474). As shown in the attached documents and described below, the MHI results are based on two erroneous assumptions. When these assumptions are corrected, the opposite conclusion is reached, which is that the tubes will be susceptible to failure from fatigue.

Required Action: The NRC Chairman has publically stated that SCE is responsible for the work of MHI, Westinghouse, AREVA and Intertek. In light of SONGS Units 2 & 3 massive amounts of tube damage (wear) and tube failure in Unit 3, along with incomplete tube inspections for detection of circumferential incubating cracks in Unit 2, based on the review of attached reports and governing standards described below, NRC is legally required to check MHI Fatigue Calculations and post the results on its website before any approval of SONGS proposed New License Amendment for restart of Unit 2, to demonstrate:

That the proposed amendment (1) Would not involve a significant increase in the probability of an accident previously evaluated in the SONGS FSAR; or, (2) Would not create the possibility of a new or different type of accident previously evaluated in the SONGS FSAR; or, (3) Would not involve a significant reduction in the required margin of safety by operating Unit 2 at 70% power However, because of the wear damage previously sustained by Unit 2, some tubes may now be susceptible to rapid fatigue failure.

Based on the above review, NRC needs to provide a calculation justifying the engineering basis of the above statement to meet the ASME Code, NRC RG 1.121, the NRC Chairman and its own Standards. The calculation should be performed by a Licensed Mechanical or Civil Engineer and Independently verified by a Licensed Structural Engineer.

Governing Standards:

A. SCE design document states, "The Supplier shall perform fatigue analyses as required by ASME Section III, Division 1, "Rules for Construction of Nuclear Facility Components" to demonstrate that the RSG components are capable of withstanding all accumulated load cycles imposed by the operating and seismic conditions, which the equipment will be subjected to. The Supplier shall provide evidence and results of analyses, modeling, and tests (if applicable) performed to demonstrate that tube fatigue and tube support damage will not occur over the full range of operating conditions. The fatigue analysis must also be addressed by the Supplier and approved by Edison." To ensure Unit 2 safe operations at 70% power, ASME code Section III, Division 1, Appendix I, "Design Fatigue Curves" Figure I-9.2.2 requires that the fatigued RSG tubes not be stressed beyond their endurance limit (13.6 ksi) to avoid tube rupture.

B. Regulatory Guide 1.121, "Bases For Plugging Degraded PWR Steam Generator Tubes", Section C.3.b.(2) states, "The fatigue effects of cyclic loading forces should be considered in determining the minimum tube wall thickness. The transients considered in the original design of the steam generator tubes should be included in the fatigue analysis of degraded tubes corresponding to the minimum tube wall thickness established. The magnitude and frequency of the temperature and pressure transients should be based on the estimated number of cycles anticipated during normal operation for the maximum service interval expected between tube inspection periods. **Notch effects** (emphasis added) resulting from tube thinning should be taken into account in the fatigue evaluation."

C. General Design Criteria 14, "Reactor Coolant Pressure Boundary," and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. General Design Criterion 15, "Reactor Coolant System Design," requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Furthermore, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of critical areas to assess their structural and leak-tight integrity.

D. NRC regulations 10CFR50, Appendix B, Criterion 16 specify that for a licensee to maintain his operating license, such non-conformance must be promptly identified and corrected.

Observations:

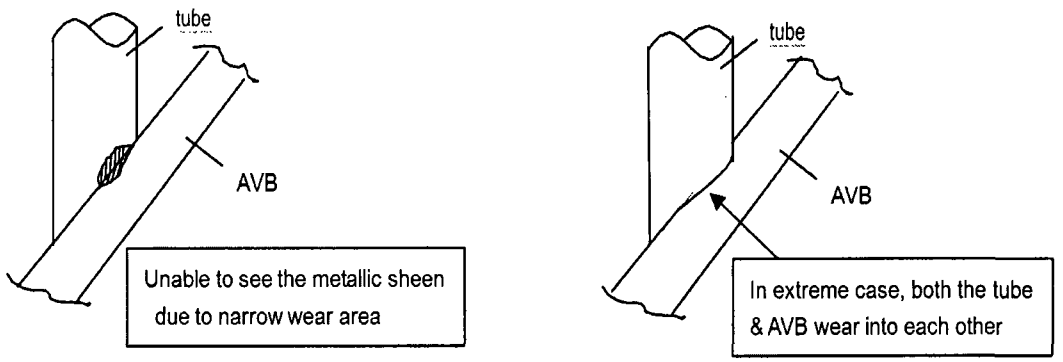
A. In Unit 2 Return to Service Report, Attachment 4, MHI Documents SO23-617-1-M1538, Revision 0, page 288 of 474, Appendix 7, L5-04GA564(9), page 7-9, Figure 2, MHI states, "Section 3.2, Wear Pattern-2 (Local Wear on Tube Surface Characteristics):

(1) Local wear occurs on the tube but the wear surface is not exposed (cannot be seen),

(2) Unable to determine if wear occurs on tube or AVB or both,

(3) Unable to determine the direction of motion or vibration, and

(4) An extreme interpretation is that both tube and AVB are worn into each other.



(a) Case 1 (b) Case 2

Fig-2 Wear Pattern 2

B. In Unit 2 Return to Service Report, Attachment 4, MHI Documents SO23-617-1-M1538, Revision 0, Appendix 10, L5-04GA564(9), MHI states:

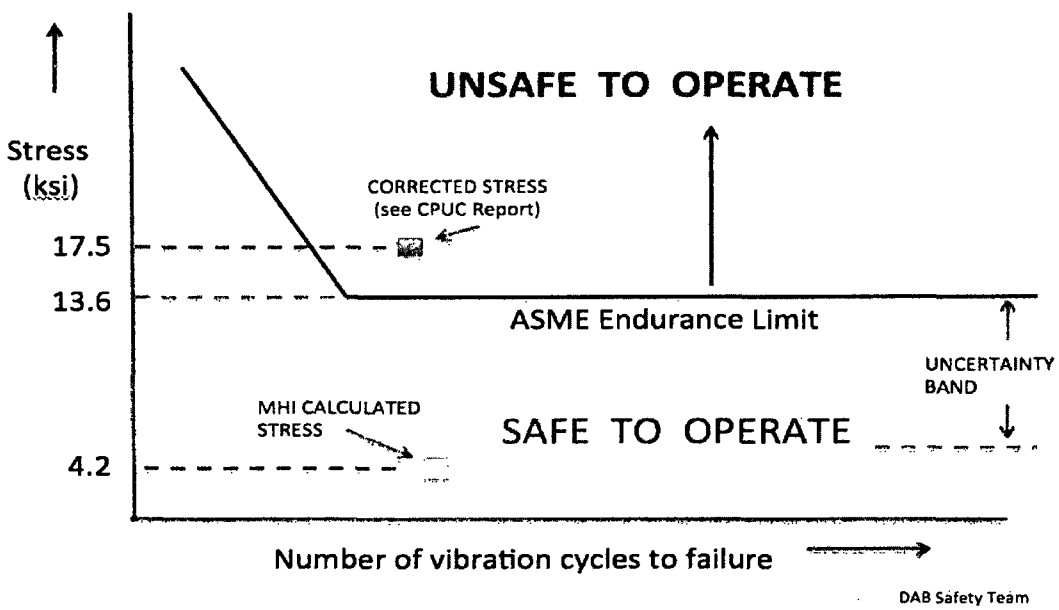
(Page 10-29, Item 7.1) "Simulation of Wear in Unit 3: It is consistent with the actual phenomenon of the tube wear at Unit-3. As shown in Fig 7-3 [Redacted by MHI], when the tube-to-AVB contact force is large, the in-plane vibration can be prevented."

Note: Contact force is the force in which an object comes in contact with another object. Some everyday examples where contact forces are at work are pushing a car up a hill, kicking a ball, or pushing a desk across a room. In the first and third cases the force is continuously applied, while in the second case the force is delivered in a short impulse. The most common instances of contact force include friction, normal force, and tension. Contact force may also be described as the push experienced when two objects are pressed together. The MHI-designed AVBs had zero contact forces in Unit 3 to prevent in-plane fluid elastic instability and subsequently, wear occurred under localized thermal-hydraulic conditions of high steam quality (void fraction) and high flow velocity. Large u-bends were moving with large amplitudes in the in-plane direction without any contact forces imposed by the out-of-plane restraints. The in-plane vibration associated with the wear observed in the Unit 3 RSGs occurred because all of the out-of-plane AVB supports were inactive by design in the in-plane direction. The Unit 3 tube-to-AVB contact force for the tubes with tube-to-tube wear (TTW) was zero. That is why they did not restrain the tubes in the in-plane direction (like a sports car moving with very high speed in freeway express lanes passing by a

stalled police car with empty guns and disabled communication systems). In-plane fluid elastic instability did not happen in Unit 2 because of operational differences, so therefore double contact forces and better supports is just conjecture in Unit 2 and a pre-planned and ill-conceived SCE reason to justify the restart of an Unsafe Unit 2. SCE's baseless contact force theory is based upon hideous statistical data and manufacturing simulations which are supposed to show that the design is capable of stopping the super high velocity in-plane vibrations. This theory is refuted based upon an in-depth review of the speculative and incomplete SCE Root Cause Evaluation, Dr. Pettigrew's 2006 Research Paper, Westinghouse, AREVA, John Large and the earlier version of MHI Reports.

C. The source of MHI's error described below resulted from how they calculated the increase in the local stress at geometrical discontinuities (notches), which are formed when two metal surfaces come in contact during vibration. Since the worn surfaces of the tubes inside the steam generators as described above cannot be seen, MHI made two incorrect key assumptions, which are inconsistent with the observation that both the tube and the supporting bar are worn into each other. First, MHI assumed that the ASME endurance limit could be applied directly to the notched tube surfaces. Since it is commonly known that surface roughness significantly reduces fatigue life and since the ASME data is for smooth polished surfaces, this assumption would underestimate the amount of fatigue damage. Second, when using the Peterson chart, MHI assumed an unrealistically large fillet radius and consequently derived a low concentration stress factor. Large radii would decrease the local stress and cause the tube to fail at a higher level of stress, thereby increasing its fatigue life. Only by using these two, arbitrary, non-conservative-assumptions was MHI able to conclude that Unit 2 did not suffer any fatigue damage.

FIG 1 – COMPARISON OF THE MHI CALCULATED STRESS WITH THE STRESS CALCULATED IN THE CPUC REPORT



As depicted in the MHI drawings above (Figure 2), the support bar and the tube form a sharp discontinuity at the contacting surface, therefore the appropriate geometry for calculating the stress concentration is an abrupt geometry change (very small radii or zero radii of a notch in accordance with RG 1.121), not a large radius shoulder fillet that was assumed by MHI. When a correction is made to account for the sharp notch, the corrected stress indicates (See Figure 1 above) that the tubes have used up their fatigue life during the first cycle of operation. Structures with sharp notches can fail catastrophically when subjected to high cycle fluid elastic instability and flow-induced random vibrations. (MHI redacted their assumption so the exact value of the radius they used is unknown.)

C. In Unit 2 Return to Service Report, Attachment 4, MHI Documents SO23-617-1-M1538, Revision 0, , Appendix 10, L5-04GA564(9), MHI states:

- (Page 10-2, Section 2) "Conclusion: When consecutive 6 or 8 AVB support points are inactive and in-plane FEI does not occur, the calculated tube wears at AVB support points due to only the turbulent flow force are equivalent to the wear depths measured in Unit-2 SGs."

- (Page 10-20, Item (7) "Wear Calculation: Experimental correlations between the metal loss and the work rate are used to calculate tube metal volume reduction. The relation between wear volume (V) and wear depth (h) is represented as follows: (a) Tube to AVB wear and Tube-to-Tube wear: The tube thickness reduction is then calculated using the wear properties and work rate parameter as assuming the wear configuration as shown in Fig.6-4. The wear width of tube to AVB wear is assumed to be the same as the width of AVB.

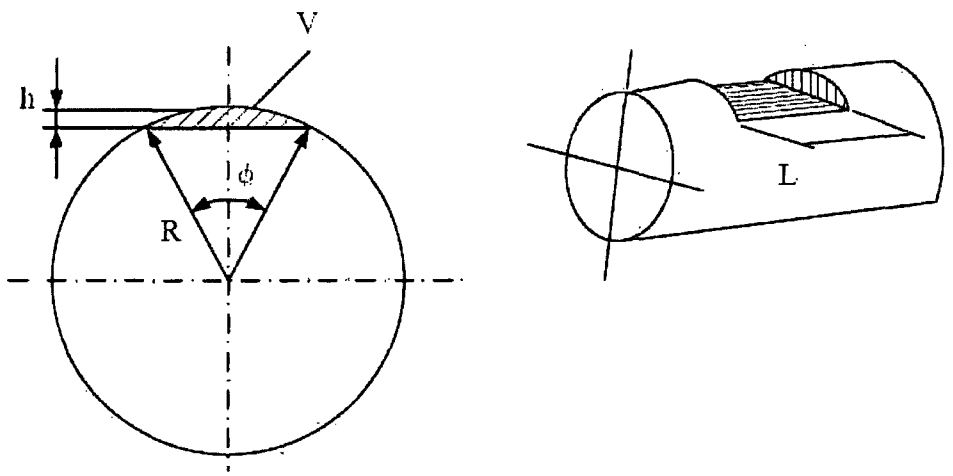


Fig. 6-4 Wear shape of tube at the contact point with AVB

Note: Both the impact and the sliding motions play a part in the tube/AVB interaction. These factors reduce tube strength because of material loss and loss of fatigue strength.

Consequences and Impact: Based on Dr. Joram Hopenfeld's evaluation of the tube wear data and the in-situ leak test results, restarting Unit 2 and/or Unit 3 would compromise public safety. The new components in the replacement steam generators ("RSGs"), constituted a major change to the original SGs, which lead to vibrations and the unusual rapid tube wear. The components causing the wear would have to be replaced and the SONGS license amended before either reactor can be restarted.

The above assessment also applies to SCE's proposed five-month test of Unit 2 at 70% of licensed power. After correcting for errors in the SCE stress calculations, the the analysis shows that because of the wear damage previously sustained by Unit 2, some tubes will be susceptible to rapid fatigue failure. These tubes will exceed their allowable fatigue life by 22 to 29% during the next operating cycle. The risk of tube rupture increases with operating time but the analysis is not capable of quantifying it in terms of operating time. Unit 2 should not be permitted to operate until SCE provides a thorough assessment of the fretting fatigue issue discussed above and in the attached report.

Because the uncertainties in predicting how fatigued tubes can propagate failures, it is impossible to assess safety risks quantitatively. The present assessment was therefore limited to a discussion of discrete accident scenarios without assigning a probability to any specific scenario. It is believed that a

main steam line break (MSLB) represents a bounding case. A conservative estimate of the probability of a large early release of radiation with containment bypass would be 10E-4 per year for any operating cycle. Such risk exceeds NRC's safety goals. SONGS Unit 2 has the highest risk of tube rupture related core damage of any power plant in the US Nuclear Fleet.

Background: Even with the improved SG inspection programs, operating experience provides examples of tube flaws that were not detected by in-service tube inspections. These flaws were later discovered after accidents and did not satisfy the required structural and accident leakage integrity margins as observed in SONGS Unit 3. There have been four such occurrences from 1987 to 2012:

□□On July 15, 1987, a steam generator tube rupture event occurred at North Anna, Unit 1 shortly after the unit reached 100% power. The rupture extended circumferentially 360° around the tube. The cause of the tube rupture has been determined to be high cycle fatigue and flow-induced vibrations.

□ Indian Point 2—SGTR event in February 2000. This represented a failure to meet structural and leakage integrity performance criteria.

□ *Comanche Peak 1—Failure to meet structural and leakage integrity performance criteria in Fall 2002,* as determined by in-situ pressure testing during condition monitoring.

□ Oconee 2—Failure to meet structural integrity performance criteria in fall 2002, as determined by in-situ pressure testing during condition monitoring.

□ SONGS 3 —Failure to meet structural integrity performance criteria in January 2012, as determined by in-situ pressure testing during condition monitoring.

Of these events, only the tube that leaked under normal operating conditions at SONGS 3 likely would had ruptured with 2 additional tubes, if an MSLB event had occurred during a several-month period preceding the SGTR event in January 2012. This experience indicates that the frequency at which SONGS 2 tubes may be vulnerable to rupture (or leakage from multiple tubes) under MSLB may be above the conditional probability value of 0.05 assumed in Westinghouse, AREVA and Intertek Operational Assessments.

SONGS Unit 3 RSGs' unprecedented tube failure and massive tube and AVB/TSP degradation occurred due to fluid elastic instability, flow-induced random vibrations, Mitsubishi Flowering Effect and high cyclic fatigue under the following unique circumstances:

(1) U-tube bundle areas with high dry steam, double in-plane velocities (> 56 feet/sec, Dr. Pettigrew and others, 2006-2011) compared with out-of plane velocities assumed (28 feet/sec) to have been used in William Krotiuk 2002 Report NUREG-1919 TH calculations and predicted by Outdated Out-of-Plane Westinghouse /NRC /MHI /AREVA ATHOS Computer Models,

(2) Lack of positive in-plane restraints and zero damping,

(3) Large number of SONGS Unit 2 RSG U-bends with tube clearances of only 0.05 inches (Design 0.25 inches, Industry Norm > 0.25 inches),

(4) Excessive number of tubes with narrow tube pitch to tube diameter,

(5) Low in-plane frequency tubes and retainer bars compared with MHI SGs' higher in-plane frequency tubes and retainer bars,

(6) SONGS' tubes being much longer than Westinghouse Model 51 steam generators (Average length of heated tube = 730 inches) and other MHI SGs,

(7) MHI RSGs' unique floating tube bundle with degraded Retainer Bars can collapse due to 100% tube uncovery for 10 minutes under MSLB SG Depressurization, Multiple SGTR SG over-pressurization and lifting of SG Relief Valves, Combination of MSLB and SGTR Conditions, Release of 100% RCS Iodine to Environment,

(8) Large amount of uncertainties and unverified assumptions in MHI, AREVA, Westinghouse and Intertek's contact force (zero for in-plane vibrations), wear rate and tube stress calculations (4.6 ksi versus 16-17 ksi) and computer modeling, and,

(9) Incomplete tube inspections in SONGS Unit 2. Incubating macroscopic circumferential cracks caused by fluid elastic instability, flow-induced random vibrations and high cycle thermal fatigue are extremely difficult to detect and be accurately sized by nondestructive evaluation techniques including X-ray, ultrasonic, and eddy current based bobbin coil probes, mechanically rotating pancake coil (RPC), etc., which have been used in 170,000 SONGS Tube inspections. State-of-the-art systems: Zetec MIZ-80 iD system, Tecnatom TEDDY+, Circular TE and TM, transmit-receive eddy current array probe C-3 and other specialized radiographic probes capable of detecting sub-surface cracks caused by high cycle thermal fatigue have not been used in the 170,000 SONGS Tube Partial and Limited Inspections as shown below for Unit 2 due to access problems in the most problematic innermost sections of the U-Tube Bundle, the high cost, lack of availability of highly specialized tools and contractors, radiation doses, and time considerations in a rush to start Unit 2. The inspection scope defectively designed and degraded SONGS Unit 2 RSGs should have covered 100% hot leg and cold leg tube inspections, 100% of dents or dings, 100% of tube inspections in the tight radius U-bends, 100% area of the Top of the Tube Sheet and Tube Support Plates.

Enclosed: Appendix A – Fatigue Analysis
Attachments:
1. Dr. Joram Hopenfeld, An assessment of San-Onofre Steam Generator Tube Failures (25 Pages)
2. Dr. Joram Hopenfeld, Testimony before the Public Utilities Commission of the State of California (13 Pages)

APPENDIX A - FATIGUE ANALYSIS

The relative motion between the tubes and the anti-vibration bars (AVBs), the tube support plates, and the retainer bars result in tube wear and fatigue. This can produce relatively quick tube failures when the stresses generated during vibrations are sufficiently large.

As described in Attachment 4, MHI Document L5-04GA564, MHI used a finite element model ("FE"), to calculate that the tubes were subjected to a stress of 4.2 Ksi (page 16-2), which is smaller than the endurance limit stress of 13.6 ksi (page 16-2). Consequently, MHI concluded that the tubes would not fail from fatigue even if they were subjected to infinite number of stress cycles (page 16-13).

The MHI results are based on two erroneous assumptions. When these assumptions are corrected, as discussed below, the opposite conclusion is reached, which is that the tubes will be susceptible to failure from fatigue.

A. Stress Concentration

It is a well-established fact that geometrical discontinuities such sharp corners introduce high local stresses, which act as a site for crack initiation. A common engineering practice is to fillet or chamfer sharp corners to reduce stress concentrations and increase fatigue life. A large database is available to guide designers in selecting the particular fillet for a given application. MHI used a design chart, (Figure 2), for a tube in pure tension to determine the stress concentration factor Kt. Assuming an undisclosed value for the fillet radius and the value of the parameter t ,MHI concluded that Kt was less than 1.5 when $t/r = 1.33$. These numbers indicate that MHI used a value of t/h that exceed unity. Had MHI assumed smaller values for t/h, Kt would have exceeded 1.5 because Kt is sensitive to the assumed geometry of the fillet. MHI selected an arbitrary geometry, which is not valid, and for this reason only MHI obtained an unrealistically low value for Kt.

Figure 2 at page 17 is intended for applications when one is trying to minimize stress concentration. Visual examination of the contact between the AVB plates and the tubes do not suggest that the relative motion resulted in geometry with minimum stress concentration. On the contrary, as shown in Figures 3 and 4, the method in which the AVB interacted with the tubes allows for a formation of sharp corners at the intersection of the plate corner with the tube. MHI's own discussion is not consistent with their application of Chart 5 in Figure 2. The observation that the "tube and the AVB are worn into each other" and the fact that the AVB plate has sharp corners suggest that Chart 3.5 at page 17 does not apply to observed wear pattern.

The model shown in Figure 5 at page 19, represent more closely the wall-thinned geometry than that of the one used by MHI in selecting the stress concentration factor. Since Figure 2 does not provide data for fillets with very small radius, it is necessary to

consider a similar geometry giving Kt values for small radii. In Figure 6 (a special case of Figure 2, when di = 0), Kt is plotted for very small radiuses for bending, (Kt values in tension are similar.)

Using a reported wear of 35%TW, Kt is calculated as follows:

$t = (1 - \%TW)\,T = (1\text{-}0.35)\ 0.043 = 0.028in$

$d/D = D/D\text{-}2t = 0.750\,/0.750 - 0.056 = 0.750/0.694 = 1.08$

$Kt = 5$ when $r = 0.0014$

for $Kt = 5$, $r = 0.002\ (0.694) = 0.0014$

(Theoretically the chamfer radius of a sharp corner is zero, and therefore Kt will tend to be very large For a finite but small radius of 0.0015, which is close to describing a sharp corner, Kt exceeds 5.)

B. Loss of Wall Thickness (wall thinning)

The effective wall thickness Teff. of the geometry in Figure 4 can be expressed as:

$Teff = t \times (2\ \theta)/360 + T \times (360 - 2\ \theta)360$

$\theta = 2\ \mathrm{Cos}^{-1}\,(d/2 + t)/(d/2 + T)$

For a 35 and 70% TW wear, θ equals 44.6 and 44.1 degrees respectively; the corresponding effective tube thickness equals 0.0360 and 0.0345, respectively.

C. Surface Finish

Fatigue life and therefore the endurance limit is strongly affected by surface finish. Studies (Reference 11) show that fatigue life can be reduced by as much as a factor of 2 to 3 when a smooth surface is roughened to about 1.6 microns.

The data (Edison Attachment 6- Appendix D, pages 130-131) indicates that the fretted tube surfaces do not maintain their original surface finish; instead they are severely scarred. Such scars are sites for the formation of micro-cracks.

Bounding calculations would require that the ASME design stress used by MHI (13.6 ksi) be lowered to account for surface finish. It is not clear, however, that the introduction of both a stress concentration and surface finish correction simultaneously would not be overly conservative. Since no data was found in the literature where both a sharp corner and adjacent rough surface, a surface finish correction was not included in the present assessment.

D. Corrected MHI Stress Calculation

Corrected stress = MHI stress x concentration correction factor C, x thickness correction factor Tc. = 4.2 x C x Tc

C = actual stress concern. factor / MHI concentration factor = 5/1.5 = 3.33

1/Tc = Decrease in wall thickness/original wall thickness = 0.036/0.043 for beginning of cycle 0.0345/0.043 at the end of cycle assuming the same wear rate.

Tc = 1.19 to 1.25

Increase in stress= 4.2 x 3.33 x 1.19 to 4.2 x 3.33 x 1.25 = 16.7 to 17.5

Actual increase over the endurance limit = 16.7/13.6 to 17.5/13.6 = 1.22 to 1.29.

Joram Hazanfe
April 15, 2012

Don Leichtling Sign out

Don Leichtling Sign out