March 11, 2015

MEMORANDUM TO: Gloria J. Kulesa, Chief
Steam Generator Tube Integrity and
Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

FROM: Alan T. Huynh, Materials Engineer ***/RA/***
Steam Generator Tube Integrity and
Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF THE FEBRUARY 12, 2015, CATEGORY 2 PUBLIC MEETING WITH THE STEAM GENERATOR TASK FORCE TO DISCUSS STEAM GENERATOR ISSUES

The industry's Steam Generator Task Force (SGTF) met with U.S. Nuclear Regulatory Commission (NRC) staff on February 12, 2015, at NRC Headquarters in Rockville, MD. The attendees discussed a variety of steam generator issues. The topics are shown in the industry's slides, which are available in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML15043A610. (Note that Slide 29 has been replaced, as discussed during the meeting.) The enclosure to this letter provides a list of people who attended the meeting, in person, and by phone. This meeting was noticed as a public meeting and the agenda is available in ADAMS under Accession No. ML15026A710.

During the meeting, industry and NRC participants made presentations. Information exchanged is summarized below:

Acronyms used and not defined in the industry slides include:

- 3dP – three times the normal operating differential pressure
- BPV – Boiler and Pressure Vessel
- FIV – Flow-Induced Vibration
- gpm – gallons per minute
- ISI – Inservice Inspection
- NDE – Nondestructive Examination
- R# – Revision #
- SCC – Stress Corrosion Cracking
- TAG – Technical Advisory Group
- Tech Spec – Technical Specifications
- TSTF – Technical Specification Task Force
- TT – Thermally Treated

CONTACT: Alan T. Huynh, NRR/DE/ESGB
(301) 415-1137

While discussing their response to the degradation observed at San Onofre, the industry indicated that it could not do a formal lessons learned evaluation since much of the information is not publicly available (i.e., it is proprietary); however, they did indicate that testing regarding the major cause of degradation (i.e., in-plane fluid-elastic instability) was warranted and was being pursued. The industry will be working with various vendors to determine an appropriate test matrix with a targeted completion date for this matrix in March 2015. The testing will be done in Canada at facilities where the Canadians are doing some of their own testing on this phenomenon. The series of tests proposed by Canadian Nuclear Labs is a phased project and will be finished in approximately 3 years.

Industry indicated it would review their level of involvement on the American Society of Mechanical Engineers (ASME) Task Group on flow induced vibration and would formally respond via a letter to the NRC's August 13, 2014 letter (ML14206A841). This letter inquired about the industry's actions in response to the steam generator operating experience at San Onofre Nuclear Generating Station.

Regarding the design of the tube-to-tubesheet joint, the industry indicated that there are no explicit requirements for the steam generator tube-to-tubesheet weld in NB-3000 of Section III of the ASME Boiler and Pressure Vessel Code, and that they were pursuing an inquiry regarding whether the tube-to-tubesheet welds must meet the stress limits of NB-3000. The industry stated that each licensee is responsible for determining whether they met the design requirements regarding the tube-to-tubesheet welds. The NRC staff indicated it would be evaluating the appropriate means for engaging those units where the tube-to-tubesheet welds were not evaluated against the stress limits of NB-3000. Following the meeting, industry clarified that the total number of units where the tube-to-tubesheet welds were not designed as the pressure boundary and were not evaluated against the stress limits of NB-3000 is five (rather than the 6 implied on slide 13). In addition, the industry clarified following the meeting that there are 13 units where the tube-to-tubesheet welds were designed as the pressure boundary, but did not perform a complete stress analysis per NB-3000. Of these 13 units, 9 performed some stress analysis and 4 performed no stress analysis per NB-3000. The NRC staff also observed that in many of the early steam generators, the tubes were not expanded for the full depth of the tubesheet, thereby implying that the welds must have served more than just a sealing function (i.e., they were structural welds). Regarding the requirements of the ASME Code, in general, the NRC staff indicated that it is normally preferable to inquire what the intent of the Code was prior to the design and fabrication of an item rather than waiting until the item is in service.

With respect to the workshop on automated steam generator tube eddy current data analysis, the NRC staff indicated that it would be evaluating the role of the regional inspectors when most of the data analysis is performed offsite and what information is needed for the regional inspectors to perform their inspections. The NRC staff encouraged the industry to consider this issue as well, as it prepares for the workshop and develops its guidelines for automated data analysis.

On the issue of thermocouple measurement uncertainty when thermally treating steam generator tubing, the industry indicated that data from at least one vendor indicates that the objectives of the thermal treatment process can still be met even when temperatures are outside the indicated range (e.g., at 710°C). The NRC staff concludes this issue can be closed, since the industry clarified that although measurement uncertainty is not included in the

specified temperature range, there is sufficient margin in both the temperature range and the thermal treatment time given typical measurement uncertainties that the objectives of the thermal treatment will continue to be met.

The issue of whether sufficient guidance exists for establishing action levels for units with primary-to-secondary leakage limits less than 150 gallons per day was discussed during the meeting. The NRC staff is still reviewing the information provided during the meeting.

For the units with no plans to revise their technical specifications to be consistent with TSTF-510 at this time, the industry indicated that there were some extenuating circumstances for some of these units (e.g., not currently licensed).

On the consistent measuring of flaw sizes, the NRC staff clarified that the issue goes beyond just the change in flaw signal from outage-to-outage. The industry stated that performance demonstrations have indicated that sizing is relatively consistent from person-to-person if each individual is given the same sizing procedure. The industry also indicated that if the noise is substantial, the analyst is expected to report that they are unable to size the flaw.

The industry presented some results on the effect of broached supports on the burst pressure of flawed tubes. The wear scars that were used in the testing had burst pressures close to three times the normal operating differential pressure. The flaws used in the tests were as deep as 66 percent through-wall and as long as the width of the tube support plate. The diametral gap refers to the difference between the local outer diameter of the tube and the diameter of the hole (at the lands) in the tube support plate. There was some discussion on whether the tubes were pinned in their location and whether heavy deposit loading would affect the results. The presenter indicated it should not have a significant effect. The industry agreed to provide the final results in a future presentation and the results will be published in an Electric Power Research Institute (EPRI) / Steam Generator Management Program (SGMP) report.

Although none of the steam generators in the USA use Alloy 800, several members in the SGMP, who are located outside of the USA, have Alloy 800 steam generator tubes. The results of recent EPRI testing have found that as long as the amount of lead in the steam generators is limited (e.g., via sludge lancing), the potential for stress corrosion cracking of Alloy 800 steam generator tubes is limited.

Feedring support damage was observed in a steam generator. A contributing cause was leakage through an inspection port which is bolted to the ends of the feedring. This feature is unique to this type of steam generator. The vendor for this steam generator has been working with other potentially affected units. In addition, the industry will provide this operating experience to other vendors to ensure it will not happen in future designs (e.g., by locating the inspection port in a region where this type of event could not occur).

The NRC staff indicated that it would start using the summaries from the meetings between the Steam Generator Task Force and the NRC to track issues where the industry has indicated that it would consider revising their guidelines to address topics discussed in prior meetings. These issues include: (1) inspections to monitor for channel head degradation (discussed in August 2013 meeting); (2) guidance for when to perform inspections following design basis accidents (discussed from February 2012 through February 2013); (3) consistent measuring of flaw amplitude (August 2013 through present); (4) probe variability between vendors and

Examination Technique Specification Sheets (August 2013); and (5) need for modifying frequency of rotating probe for certain size tubing (being evaluated as part of phase 4 of the Examination Technique Specification Sheet study) (August 2014).

The NRC staff encouraged the industry to review the operating experience associated with a primary-to-secondary leak that occurred in Korea in 2014 to determine if there may be some lessons learned applicable to domestic plants (e.g., calibration of detectors, clogged sample lines, etc.).

The NRC staff informed the industry that the NRC staff's review of the lessons learned at San Onofre should be complete in March and that there would most likely be a meeting with the Advisory Committee on Reactor Safeguards meeting on this topic. The NRC staff indicated that the industry may want to attend this meeting.

The industry recently submitted a report to the NRC for information entitled: "Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly" dated October 2014 (ML14323A221). The staff is reviewing the information contained in this report and information provided in public meetings on this topic. As a result of the review to-date, the NRC staff indicated that it would be useful to have a public meeting where the industry summarizes all the work that has been done on this topic and for the industry to summarize their position/plans for an aging management program for tube-to-tubesheet welds and the divider plate. The industry indicated it would be amenable to such a meeting. The NRC staff indicated that it would provide the industry a list of specific topics they would like to discuss during the meeting. Specific topics identified to-date include: (1) basis for removing certain data because of the tungsten weld head positioning and the sensitivity of the results to this data; (2) the types of inspections that are currently being done as discussed in Section 7.3 (e.g., an acceptable aging management program); (3) the inspection results from the Z-welds and whether the performance of these inspections may give an opportunity to detect through-wall cracking at the triple point if it were to occur (e.g., by being in the vicinity of the triple point, if through-wall leakage was occurring could it be detected during the performance of the Z-weld inspections); and (4) a summary of prior studies on whether the tubesheet may be overstressed as a result of divider plate cracking (including whether these results are still valid since some steam generators may now rely on the divider plate to reduce the stresses in the tubesheet). The industry indicated that it would desire having feedback on this report by 2016. This would give industry sufficient time to develop and qualify techniques to perform more robust inspections of the divider plate tube-to-tubesheet weld, if they are needed.

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Enclosure:
Attendance List

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ADAMS ACCESSION Nos.:
Meeting Summary: ML15068A400
Meeting Notice: ML15026A710
Industry Slides: ML15043A610

OFFICE	NRR/DE/ESGB	NRR/DE	NRR/DE/ESGB
NAME	AHuynh	KKarwoski	GKulesa
DATE	03/09/2015	03/10/2015	03/11/2015

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Attendance List
February 12, 2015, NRC Public Meeting with the
Steam Generator Task Force to Discuss Steam Generator Issues

Note: The list of phone participants may not be all-inclusive

SGTF/Industry Participants
Viki Armentrout, Dominion
Jesse Baron, Westinghouse
Jim Begley, TCA Solutions
James Benson, EPRI
Helen Cothron, EPRI
Alex Cullu, Scientech
Robin Dyle, EPRI
Patrick Fabian, PSEG
Jeff Fleck, Areva
Stephen Fluit, B&W
Carl Lee Friant, Exelon
Rudy Gil, NextEra Energy
Greg Kammerdeiner, FENOC
Edward Korkowski, NextEra Energy
Ray Kuyler, Morgan Lewis
Richard Maurer, Westinghouse
Dan Mayes, Duke
Scott Redner, Xcel Energy
Mark Richter, NEI
Phil Rush, MPR Associates
Heather Sturgeon, Xcel
Damian Testa, Westinghouse

NRC
Mica Baquera
Allen Hiser
Alan Huynh
Ken Karwoski
Gloria Kulesa
John Lubinski
Greg Makar
Seung Min
Emmett Murphy
Cheng (John) Wu

Phone Participants
Brent Capell, EPRI
Clayton Webber, Westinghouse
Ryan Wolfe, EPRI

Members of the Public Phone Participants
Marvin Lewis, Public
Donna Gillmore, Public

NRC Phone Participants
Aloysius Obodoako

ENCLOSURE