FENOC™
FirstEnergy Nuclear Operating Company

5501 North State Route 2
Oak Harbor, Ohio 43449

Raymond A. Lieb
Vice President, Nuclear

419-321-7676
Fax: 419-321-7582

September 27, 2013
L-13-302

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject:
Davis-Besse Nuclear Power Station
Docket No. 50-346, License No. NPF-3
Reply to Request for Additional Information Related to Steam Generator Inventory Change (TAC No. MF0536)

By letter dated January 18, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13018A350), FirstEnergy Nuclear Operating Company (FENOC) submitted an application for amendment of Operating License NPF-3 for the Davis-Besse Nuclear Power Station (DBNPS). The proposed amendment would revise Technical Specifications (TS) in support of the steam generator (SG) replacement scheduled for the DBNPS spring 2014 refueling outage.

By letter dated August 29, 2013 (ADAMS Accession No. ML13234A390), the Nuclear Regulatory Commission staff requested additional information in order to complete its review of the application. The FENOC response to this request is attached.

There are no regulatory commitments contained in this letter. If there are any questions, or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at 330-315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 27, 2013.

Sincerely,

Raymond A. Lieb

Attachment: Reply to NRC Request for Additional Information Dated August 29, 2013

cc: NRC Region III Administrator
NRC Project Manager
NRC Resident Inspector
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

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1. **As stated in Technical specification (TS) 3.4.17 Bases, Davis-Besse Nuclear Power Station (DBNPS), Unit 1, (FENOC, the licensee) uses the Nuclear Energy Institute (NEI) 97-06," Steam Generator [SG] Program Guidelines", and the Electric Power Research Institute (EPRI) guidelines referenced therein, to establish the content of the SG Program required by the DBNPS TS. The structural integrity performance criteria that are established to prevent tube burst, are addressed in Section 3.2 of the EPRI SG Integrity Assessment Guidelines. This section states, in part:**

   > **A structural limit is established from the mean (best estimate) regression relationship for tube failure for the conditions defined by the SIPC. The condition monitoring [CM] limit is obtained by modifying the structural limit to account for the uncertainties associated with the tube failure regression model, material properties, and the NDE [non-destructive examination] system. The repair limit and OA [operational assessment] limit are obtained by further modification to consider degradation growth, and require that flaws on tubes remaining in service at the beginning of cycle satisfy the structural integrity performance criterion over the next inspection interval.**

   Discuss how TSs 3.4.17 and 5.5.8 will ensure that the condition monitoring and OA examinations will account for uncertainties associated with the tube failure regression model, material properties, and the NDE system, in accordance with the SG Program that you are required to establish per TS 5.5.8.

FENOC Response

DBNPS Technical Specification (TS) Surveillance Requirement 3.4.17.1 requires steam generator tube integrity to be verified in accordance with the Steam Generator Program. TS 5.5.8 defines the Steam Generator Program requirements to be implemented. To satisfy these requirements, DBNPS procedures require that inservice inspection, repair

and assessment activities comply with industry guidance established by NEI 97-06, "Steam Generator Program Guidelines" and its referenced documents, consistent with the DBNPS response to NRC Generic Letter 97-05, Steam Generator Tube Inspection Techniques, provided by letter dated March 16, 1998.

NEI 97-06 provides guidance for the evaluation methods, required margins and adjustments, and the typical inputs and assumptions used to determine tube integrity. It stresses that the tube integrity assessments account for input variability and uncertainties to provide a conservative assessment of the condition of the tubing relative to the performance criteria. NEI 97-06 requires that these assessments follow Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines, which offer guidance and requirements for the evaluation methods, margin, and uncertainty considerations used to determine tube integrity.

For condition monitoring, the guideline identifies that tubing analysis assess: the burst model based on regression analysis of tube failure data, including uncertainty in the prediction of burst pressure for a given extent of degradation; the tube material strength information, including uncertainty in mechanical strength behavior due to material heat-to-heat and within-heat variability; and measurement uncertainty to the NDE sizing technique, including systematic and random components of sizing uncertainty due to technique and analyst variability. For operational assessment, the guideline also adds degradation growth during future operation.

Background:

In its application, the licensee stated that the new SGs have different dimensions, materials, and thermal performance from that of the original SGs, and because of these differences a revision to the TS is required. TS 3.7.18 requires SG water level restrictions based on preserving the initial condition assumptions for the SG inventory used in the main steam line break (MSLB) analyses presented in the updated safety analysis report (USAR). The four restrictions in TS 3.7.18 are based on the specific physical design characteristics and dimensions of the SGs. FENOC is proposing to revise this TS so that it addresses the new physical design characteristics of the new SGs.

In addition, the licensee stated:

> **The proposed new [TS 3.7.18] curve was developed using the same methodology that was used for developing the original curve, but the supporting analyses are based on the dimensions and thermal performance of the replacement SGs. The existing USAR MSLB analysis remains bounding. To provide the technical bases for the proposed LCO [limiting condition for operation] 3.7.18.b, c, and d, requirements, calculations were performed using the same methodology as that used in the original calculations.**

2. Describe the methodology and provide a summary of the analysis that was used to develop the new TS 3.7.18 curve.

FENOC Response

DBNPS TS 3.7.18.a requires the water level of each steam generator to be less than or equal to the water level shown in TS Figure 3.7.18-1, "Maximum Allowable Steam Generator Level," to ensure that the secondary side inventory in the replacement SGs when operating in Mode 1 or 2 is maintained within the limits previously analyzed for a main steam line break (MSLB) event. The figure provides an acceptable range of operation based on SG operating range (OR) level and main steam superheat temperature. The OR level is a measure from which inventory in the downcomer can be determined. A higher OR level is indicative of higher inventory in the downcomer. Main steam superheat is a measure from which the inventory within the tube bundle and the steam annulus can be determined. Lower superheat would be indicative of a higher tube bundle inventory, which causes a shorter zone for superheating. To maintain total inventory within the allowable limit, if the downcomer region inventory is higher, as indicated on the OR level, the inventory within the tube bundle region must be lower, which would be indicated by a higher steam superheat value. This relationship is defined by TS Figure 3.7.18-1.

The curve presented in the original TS 3.7.18, Figure 3.7.18-1 was developed using the 1983 version of the VAGEN computer code. VAGEN is a computer program designed to perform steady-state analysis of one-dimensional, tube-in-shell heat exchangers, including once-through steam generators (OTSG).

A revised curve is needed for the replacement OTSG (ROTSG) because of dimensional differences between the original OTSG and the ROTSG that affect the curve. The major difference is a thinner inner shroud, which increases the downcomer cross sectional area. Also the use of higher strength steels resulted in some of the ROTSG component dimensions, such as the tubesheets, being different from those of the same components in the original OTSG. These changes affect the mass inventory and its impact on steam superheat. The revised curve was developed to reflect these changes.

Changes in the operating conditions, such as increased tube fouling, increase the required boiling length due to lower heat transfer in the boiling zone. This increases the tube region mass inventory in the OTSG and reduces the tube surface area available for producing steam superheat in the upper region of the tube bundle, resulting in less steam superheat. This behavior was analyzed with VAGEN, using conservative inputs and assumptions, to predict the steam superheat that would be produced by the steam generator for a given mass inventory. The results of that analysis were used to produce a curve of steam superheat versus steam generator level to ensure the secondary side inventory does not exceed the value that was used in analyzing the MSLB event.

The proposed revision to the Technical Specification 3.7.18 curve was developed using version 1.8 of VAGEN, the current version of the same computer code that was used to produce the original curve. Because several changes were made to the VAGEN code between the 1983 version used for the original curve and the current version 1.8, a

benchmarking calculation was prepared to evaluate the impact of those changes on VAGEN's performance. The benchmarking calculation determined the combined net effect of all of the VAGEN changes by using version 1.8 to repeat the original calculation. This produced a new version of the original curve, which showed how the curve would be affected solely by the changes in version 1.8 of VAGEN. This benchmarking showed that VAGEN, version 1.8, produces results that are more conservative, although similar to the 1983 version. The version 1.8 superheat predictions also more closely match plant data. This results in the curve based on version 1.8 being more restrictive for plant operation. Based on this benchmarking, the current version of VAGEN was determined to be appropriate and more conservative than the 1983 version for use in generating the revised Technical Specification 3.7.18 curve.

The analysis that was performed to develop the revised curve uses the same technique that was used to produce the original curve, but also uses conservative inputs to bound plant operating conditions and to produce a more restrictive curve. These included conservatively high ROTSG heat duty, low reactor coolant system flowrate, reduced feedwater temperature, and high secondary side pressure. The conservative inputs, combined with the more conservative results produced by VAGEN version 1.8, result in a revised curve that is more conservative and restrictive than the original curve.

3. Describe the methodology and provide a summary of the analysis that was used to perform the calculations for LCO 3.7.18.b, c, and d.

FENOC Response

The limitations specified in the original TS LCO 3.7.18. b, c, and d were developed by Davis-Besse using manual calculations; no computer codes were used. The calculations used conservative inputs and assumptions in developing the original TS requirements.

No changes are proposed for TS LCO 3.7.18.b. The restrictions imposed by TS LCO 3.7.18. c and d need to be revised for the ROTSGs to address the same considerations as discussed in the response to question 2, above. Specifically, some of the ROTSG component dimensions, such as the shroud and the lower tubesheet, are different from those of the same components in the original OTSG, resulting in more mass inventory in the ROTSG than the original OTSG at the same water level.

New calculations were performed for the ROTSGs, using the same techniques and conservative assumptions that were used in the original calculations. These calculations determined the maximum initial mass inventory that could exist in the ROTSGs, combined with the release from a period of continued feedwater flow to the failed steam generator, that would ensure the current Mode 1 Main Steam Line Break mass and energy would bound the results. One of the most conservative inputs was that the average enthalpy of the steam exiting the break was calculated assuming it was the same as for the Mode 1 break, even though the actual steam temperature and enthalpy would be significantly less in Mode 3. Additionally, the calculations for LCO

3.7.18.b and c were performed at 100 percent and 75.5 percent operating range level, respectively. However, for additional conservatism, the TS LCOs would impose more restrictive level limits of 96 percent and 74 percent operating range level, respectively.

Using the same techniques and conservatisms that were used to develop the original Technical Specification limits for LCO 3.7.18.b, c, and d results in revised limitations for LCO 3.7.18.c and d that are appropriate and conservative.

4. **Chapter 15.5 of the USAR reference 56 is BAW-10193PA, "RELAP5/MOD2-B&W for Safety Analysis of B&W Designed Pressurized Water Reactors," Rev 0, January 2000. BAW-10193(P)(A), Appendix A, Table A1, lists those accidents and transients that are modeled using the modeling scheme illustrated in Figure A.1 of the report. Appendix A of the report, Section A.1.1, also provides a general statement regarding the dominant characteristics, which could be affected by a SG replacement, that are essential in determining how to use the Figure A1 model.**

   Provide a disposition for each Appendix A, Table A.1 of BAW-10193PA, of these events that summarizes the impact that the new SG inventory would have on the consequences postulated for each event.

FENOC Response

As stated above, BAW-10193P-A, Rev 0, January 2000, is identified as reference 56 in the DBNPS USAR; however, the methodology outlined in Appendix A is referenced only in the analysis of the Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident) and Loss of Normal Feedwater, as indicated in USAR subsections 15.2.1.2.2 and 15.2.8.4.2, respectively.

As discussed with members of the NRC staff on August 19, 2013, the final analysis of the effects of steam generator replacement and new steam generator inventory on the consequences postulated for each of the events listed in the USAR is in progress; however, preliminary evaluations of the Chapter 15 analyses and a complete review of the DBNPS TS indicate that all necessary TS modifications have been addressed in the application for license amendment.