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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Davis-Besse Nuclear Power Station, Unit No. 1 Response to Audit of Steam Generator Program Focusing on Steam Generator Tube Integrity During A Large-Break Loss-Of-Coolant Accident (TAC No. ME2875)

On February 23, 2010, the U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The purpose of the audit was for the NRC to gain a better understanding of the approach taken by the FirstEnergy Nuclear Operating Company (FENOC) for ensuring steam generator tube integrity following a large-break loss-of-coolant accident for DBNPS. On May 19, 2010, the NRC staff issued the audit report (Accession No. ML101170196) that identified two potential inconsistencies between various licensee documents. FENOC's response to the audit report is provided in Attachment 1.

The regulatory commitment contained in this submittal is listed in Attachment 2. If there are any questions or if additional information is requested, then please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry S. Allen

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NRR

Attachments:

1. Response to NRC Audit Report for Steam Generator Tube Integrity Following a Large-Break Loss-of-Coolant Accident for the Davis-Besse Nuclear Power Station, Unit No. 1
2. Regulatory Commitment List

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Utility Radiological Safety Board

On February 23, 2010, the U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The purpose of the audit was for the NRC to gain a better understanding of the approach taken by the FirstEnergy Nuclear Operating Company (FENOC) for ensuring steam generator tube integrity following a large-break loss-of-coolant accident for DBNPS. On May 19, 2010, the NRC staff issued the audit report (Accession No. ML101170196) that identified two potential areas of inconsistencies between various licensee documents. These items are provided below in bold type followed by the FENOC response for DBNPS.

Item 1 – Steam Generator Tube Integrity

The team interviewed licensee personnel and reviewed various documents to ascertain the scope of the licensee's [steam generator] SG tube inspections. The team confirmed that the licensee was inspecting all areas susceptible to circumferentially oriented degradation and that the licensee was analyzing the data. The staff noted that there is a potential inconsistency in the licensee's requirements contained in their Technical Specifications. Specifically, Technical Specification 6.8.4.g.4 [currently 5.5.8.d because of implementation of Improved Technical Specifications] indicates:

> ***For tubes that have undergone repair rolling, the tube and tube roll, outboard of the new roll area in the tube sheet, can be excluded from inspections because it is no longer part of the pressure boundary once the repair roll is installed.***

However, license condition 2.C(7)c requires the licensee to report the following prior to returning the SG to service:

> ***Determination of the best-estimate total leakage that would result from an analysis of the limiting [Large-Break Loss-of-Coolant Accident] LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.***

In order to determine the best-estimate leakage following a LBLOCA as required by License Condition 2.C(7)c, the licensee must inspect the portion of tube outboard of the repair rolls. These inspections are necessary since flaws outboard of the repair rolls may result in primary-to-secondary leakage following a LBLOCA. This License Condition, therefore, is potentially inconsistent with the requirement in Technical Specification 6.8.4.g.4 [currently 5.5.8.d because of implementation of Improved Technical Specifications] which indicates that portions of the tube outboard of the new roll area can be excluded from inspections. Since the

licensee is performing the inspections necessary to meet license condition 2.C(7)c, there is no safety concern; however, this potential inconsistency could result in misapplication in the future.

Response

The DBNPS Operating License and Technical Specification sections indicated above were added by Amendment No. 252 to Facility Operating License No. NPF-3 (Accession No. ML020450025). This amendment approved the use of repair rolls in accordance with Framatome Technologies Topical Report BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report." Based on the results of the qualification testing of the repair roll process detailed in BAW-2303P, Revision 4, the repair roll is sufficient to ensure adequate margins of structural and leakage integrity without crediting the original rolls or tube-to-tubesheet seal welds.

The tubing outboard of the repair roll was excluded from inspection as it was outside of the new credited pressure boundary created by the repair roll. This allows flaws to remain in the non-pressure boundary portion of the in-service tubing without violating the first repair criteria of Technical Specification 5.5.8.c.1, "tubes found by inservice inspection to contain flaws, in a region of the tube that contains no repair, with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged or repaired." This region of the tube contains no repair but is outboard of the installed repair roll, and thus is not pressure boundary. License Condition 2.C(7)c, however, requires the portion of the tubing excluded from the standard inspection scope to be included in the determination of best estimate total leakage, which must be communicated to the NRC prior to returning the steam generators to service.

DBNPS steam generators are scheduled for replacement during the 2014 refueling outage and FENOC will implement any necessary license amendments at that time. Accordingly, no licensing actions will be pursued to eliminate the identified inconsistency. In the interim, FENOC will update "Davis-Besse Steam Generator Management Program Manual," to explicitly state that the portion of the tubing outboard of a repair roll, which is not pressure boundary, is excluded from the inspection and repair requirements of Technical Specification 5.5.8, but is inspected to support the requirements of License Condition 2.C(7)c.

Item 2 – Accident-Induced Leakage Limit

The NRC staff reviewed the dose consequence analysis provided by the licensee as documented in Calculation number C-NSA-060.00-015 entitled, "Evaluation of Potential for Best Estimate Primary to Secondary Leakage (using BAW-2374) following a Large Break LOCA." This calculation was initiated to support License Condition 2.C(7) which states in part that:

> ***"FENOC shall demonstrate by evaluation that the primary-to-secondary leakage following a LBLOCA, if any, as described in Appendix A to Topical***

> ***Report BAW-2374, July 2000, continues to be acceptable, based on the as-found condition of the steam generators. For the purpose of this evaluation, acceptable means that a best estimate of the leakage expected in the event of a LBLOCA would not result in a significant increase of radionuclide release (e.g., in excess of Title 10 of the Code of Federal Regulations (10 CFR) Part 100 limits). This is required to demonstrate that adequate margin and defense-in-depth continue to be maintained. A written summary of this evaluation shall be provided to the NRC within three months following completion of the steam generator tube inservice inspection."***

The results of the calculation indicate a two-hour site boundary thyroid dose of 3.21 rem and a 30-day dose of 1.52 rem at the outer boundary of the low population zone (LPZ). The licensee stated that the site boundary at DBNPS is the same as the exclusion area boundary (EAB). The calculation is based on the leakage of 1 [gallon per minute] gpm of post-LOCA reactor coolant system (RCS) liquid leaking into the secondary side of the affected steam generator under the driving force of the safety injection system. The source term considered in the analysis consists of 10 percent of the core inventory of iodine representing a gap release. The calculation states that for the break location which causes the maximum stress to the steam generator tubes, no cladding damage is predicted. Therefore the calculation states that the use of the 10 percent core inventory is conservative for this evaluation.

The licensee evaluated the transport of the leaking RCS using assumptions consistent with the regulatory guidance for the evaluation of a main steamline break accident. Consistent with regulatory guidance, the calculation did not consider the release of noble gases from this pathway since they would not be expected to be present in the liquid source term. Consistent with regulatory guidance, the calculation uses an iodine partition factor of 100 for releases from the steam generator. The calculation assumes that the main steam isolation valve fails to close and that the release continues for the 30-day accident analysis period with no credit for operator action to isolate the secondary side.

The licensee added the contribution from the primary to secondary leakage to the total thyroid dose shown in Table 15.4.6-1, "Resultant Doses From Maximum Break Size LOCA", of the FSAR to show that the resulting dose were well within the limits of 10 CFR 100.11. The staff noted that C-NSA-060.00-015 and the results shown in Table 15.4.6-1 are based on a gap release source term. The NRC staff noted that the gap release source term is not consistent with the source term description stated in 10 CFR 100.11.

<u>Response</u>

Consistent with Updated Final Safety Analysis Report (UFSAR) section 15.4.6.3, the source term used for this analysis is the activity contained in the fuel cladding gap. This activity is smaller than the maximum hypothetical accident (MHA), (UFSAR section

15.4.6.4) which uses the Technical Information Document (TID)-14844 source term. The MHA is a non-mechanistic worst case accident, thus applying specific failure mechanisms to this accident is not considered to be required, consistent with UFSAR section 15.4.6.4. Additionally, while License Condition 2.C(7) refers to 10 CFR 100 limits, which are expressed only in terms of total radiation dose, it does not specify the methodology used in determining the acceptability of as-found leakage.

For this particular analysis, the concern related to steam generator tube degradation is related to maximum emergency core cooling system injection with the lowest temperature in the borated water storage tank. Analysis documented in Areva Document 51-5054227-00, "BWOG Cladding Rupture Study for BAW-2374," shows that cladding failure will not occur during this transient. Maximum emergency core cooling systems (ECCS) injection coupled with low ECCS water temperature results in rapid heat removal of the decay heat and stored heat. Although fuel cladding failure does not occur, to maintain defense-in depth it is assumed that all the activity contained in the gap is released to the coolant injected ECCS fluid and is uniformly mixed. Therefore, for this analysis the gap activity is assumed to be 10 percent of the total iodine core activity, which is consistent with Regulatory Guide 1.25 gap activity source term assumptions.

As noted in the NRC audit report, the calculation contains additional conservative assumptions such as no credit for operator actions to isolate the secondary side, instantaneous release of source term, and conservative dose conversion factors. Also noted is if the MHA source term were utilized in the calculation, 10 CFR 100 limits would not be exceeded.

In summary, for the maximum primary-to-secondary leak rate of 1 gpm, the dose limits of 10 CFR 100 are not exceeded when analyzed consistent with the conservative methodology of UFSAR section 15.4.6.3. Therefore, this accident evaluation demonstrates that a significant increase of radionuclide release will not occur, which satisfies the requirements of License Condition 2.C(7). No additional actions are necessary to address the inconsistency identified in the audit report.

Regulatory Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC. They are described only as information and are not Regulatory Commitments. Please notify Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071 of any questions regarding this document or associated Regulatory Commitments.

Regulatory Commitment	Due Date
1. FENOC will update "Davis-Besse Steam Generator Management Program Manual," to explicitly state that the portion of the tubing outboard of a repair roll, which is not a pressure boundary, is excluded from the inspection and repair requirements of Technical Specification 5.5.8, but is inspected to support the requirements of License Condition 2.C(7)c.	1. Prior to the next required steam generator inspection in accordance with Technical Specification 5.5.8, "Steam Generator (SG) Program."