UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 19, 2010

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 – AUDIT OF STEAM GENERATOR PROGRAM FOCUSING ON STEAM GENERATOR TUBE INTEGRITY DURING A LARGE-BREAK LOSS-OF-COOLANT ACCIDENT (TAC NO. ME2875)

Dear Mr. Allen:

On February 23, 2010, U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The purpose of the audit was for the NRC to gain a better understanding of the approach taken by DBNPS for ensuring steam generator tube integrity following a large-break loss-of-coolant accident. The audit report is enclosed.

The audit team members did not identify any plant-specific safety issues as a result of the audit. Two potential areas of inconsistency between various licensee documents were identified. These inconsistencies are discussed in detail in the enclosed audit report. The first inconsistency deals with the technical specifications and a license condition 2.C(7)c. The second inconsistency deals with the source term used in the Final Safety Analysis Report for the maximum hypothetical accident and a calculation used to satisfy a license condition. These inconsistencies were discussed with representatives of DBNPS at the conclusion of the audit.

Please provide a response within 60 days of the date of this letter on how the two inconsistencies indentified in the enclosed audit report will be addressed.

If you have any questions, please contact me at 301-415-3867 or by e-mail at Michael.Mahoney@nrc.gov.

Sincerely,

Michael Mahoney, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
Audit Report

cc w/encl: Distribution via Listserv

AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

STEAM GENERATOR PROGRAM: STEAM GENERATOR TUBE INTEGRITY FOLLOWING A

LARGE BREAK LOSS OF COOLANT ACCIDENT

On February 23, 2010, an audit by the U.S Nuclear Regulatory Commission (NRC) was conducted at the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) in Oak Harbor, Ohio. The purpose of the audit was to gain a better understanding of FirstEnergy Nuclear Operating Company's (FENOC, the licensee) approach for ensuring steam generator (SG) tube integrity following a large break loss of coolant accident (LBLOCA). The audit team members included Ken Karwoski of the Division of Component Integrity and John Parillo of the Accident Dose Branch in the Division of Risk Assessment, both of the Office of Nuclear Reactor Regulation.

As a result of the audit, the team members gained a better understanding of the licensee's approach for ensuring tube integrity following a LBLOCA. The audit team members identified two areas of potential inconsistency between various licensee documents. These are discussed further below. The team members did not identify any plant-specific safety issues as a result of the audit.

Background

Prior to July 2009, the NRC staff generically interacted with licensees regarding once through SGs to address tube integrity issues associated with a LBLOCA. This generic interaction was a result, in part, of the staff's review of Topical Report BAW-2374, "Risk Informed Assessment of Once Through Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping."

Since the NRC staff was generically addressing this issue, the staff's review of licensees' SG reports simply acknowledged this fact (i.e., the licensees' plant-specific assessments were not reviewed in detail). The last review of DBNPS SG reports was documented in a letter dated July 6, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091671135).

In May and June 2009, the NRC staff met with industry representatives to discuss the path forward on this issue. Summaries of these meetings are in memorandums dated July 13, 2009 (ADAMS Package Accession No. ML091811290) and July 27, 2009 (ADAMS Package Accession No. ML091820002).

In response to these meetings, on July 15, 2009 (ADAMS Accession No. ML092010129, not publically available), the Pressurized Water Reactor Owners Group withdrew Topical Report BAW-2374, "Risk Informed Assessment of Once Through Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," from review. The withdrawal of this topical report was acknowledged in a letter dated September 3, 2009 (ADAMS Accession No. ML092450533). In addition, the NRC staff began interacting with each licensee to resolve this issue on a plant-specific basis.

In response to these meetings, FENOC, provided information concerning their actions to address tube integrity following a LBLOCA in a letter dated August 31, 2009 (ADAMS Accession

No. ML092450685). Subsequently, by letter dated November 6, 2009 (ADAMS Accession No. ML093010483), the NRC staff requested more information on how DBNPS performed their tube integrity analyses for a LBLOCA. On February 10, 2010 (ADAMS Accession No. ML100491091), FENOC provided additional information concerning their tube integrity analysis and indicated that it would support an on-site review of their tube integrity documentation. An audit was subsequently performed on February 23, 2010 at the DBNPS.

Key FENOC Staff Interviewed

Kathy Nesser, Fleet Licensing
Dave Gerren, Steam Generator Engineer
Luke Twarek, Steam Generator Engineer
Al Wise, Manager, Technical Services Engineering
Jessica Kemp, Supervisor, Nuclear Engineering Programs
Jim Begley, AREVA

Documents Audited

1. Davis-Besse Degradation Assessment for 15th Refueling Outage (January 2008), AREVA 51-9064198-001

2. Procedure for Selection of In-Situ Pressure Testing Candidates, AREVA 51-5051884-005

3. A CMOA [Condition Monitoring Operational Assessment] Evaluation of Steam Generator Tubing at Davis-Besse, 15 RFO [Refueling Outage], AREVA 51-9071130-000

4. Davis-Besse Degradation Assessment for 16th Refueling Outage, AREVA 51-9114644-000

5. Procedure for Selection of In-Situ Pressure Testing Candidates, AREVA 51-5051884-006

6. Calculation number C-NSA-060.00-015, "Evaluation of Potential for Best Estimate Primary to Secondary Leakage (using BAW2374) Following a Large Break LOCA"

Other References

1. Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7

2. Steam Generator Integrity Assessment Guidelines, Revision 2

Audit Activities

The audit consisted of two distinct activities: (1) gaining an understanding of the tube integrity assessment performed for a LBLOCA and: (2) gaining an understanding of the licensee's basis for their accident induced primary-to-secondary leakage limit of 1 gallon per minute (gpm) for a LBLOCA. Each of these activities is discussed below.

Steam Generator Tube Integrity

In order to obtain a better understanding of the methodology used for assessing tube integrity following a LBLOCA, the audit team focused its review on the degradation assessment for refueling outages (RFOs) 15 (2008) and 16 (2010) and the portions of the RFO 15 condition monitoring and operational assessment addressing tube integrity for a LBLOCA.

The team interviewed licensee personnel and reviewed various documents to ascertain the scope of the licensee's 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 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 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 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.

The team also interviewed licensee personnel and reviewed various documents to gain a better understanding of the licensee's methods for assessing tube integrity following a LBLOCA. Based on these discussions and reviews, the team understands that:

> For circumferentially oriented degradation in the pressure boundary (e.g., for tubes with repair rolls, all circumferentially oriented degradation in the repair roll or inboard of the repair roll), the licensee's assessment of tube integrity is performed consistent with industry guidelines. These guidelines generally require tube integrity to be demonstrated at a 95 percent probability level with 50 percent confidence.
>
> For circumferentially oriented degradation outside the pressure boundary (i.e., for tubes with no repair rolls, the portion of tube outboard the tube-to-tubesheet weld; for tubes with repair rolls, the portion of tube outboard the repair roll), the licensee's assessment of tube integrity is performed consistent with an amendment dated February 20, 2002, "Davis-Besse Nuclear Power Station, Unit 1 – Issuance of Amendment (TAC No. MB2107)" (ADAMS Accession No. ML020450025). This amendment approved the use of repair rolls at DBNPS. This amendment resulted in the adoption of license

condition 2.C(7)c which permits the amount of leakage from the tubesheet region to be determined on a best-estimate basis (50 percent probability, 50 percent confidence level).

Beginning in RFO 16, assessments of tube integrity for circumferentially oriented degradation outside the pressure boundary may be performed more conservatively (i.e., consistent with industry guidelines with 95 percent probability and 50 percent confidence).

Structural limits were reported for LBLOCA loading in the February 10, 2010 letter. Structural limits were reported as the maximum circumferential extent for a 100 percent through-wall flaw and the percent degraded area. These two structural limits were determined from different correlations. As a result, one can not use the tube dimensions to convert from one structural limit (maximum circumferential extent for a 100 percent through-wall flaw) to the other (percent degraded area). The tubes have an outside diameter of 0.625 inches and a wall thickness of 0.037 inches.

The uncertainties associated with sizing flaws are based on a specific data set. This data set is comprised of mainly small flaws. If larger flaws are detected, the sizing uncertainties would need to be reassessed.

A correlation is used to convert the maximum depth from eddy current examination to the structurally equivalent depth, which is used in tube integrity analyses.

The team concluded that the above approach is consistent with standard industry practice.

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 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. The source term that should be used to show compliance with the dose limits in 10 CFR 100.11 is described in a footnote 1 to the regulation as show below:

> *The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.*

Further information on the specifics needed to perform the dose calculations required by 10 CFR 100 is provided in a note to the regulation.

> *Note: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document [TID] 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.*

TID-14844 defines substantial core melt with specific percentages of core fission products released into the containment; 100 percent of the noble gases, 50 percent of the halogens and 1 percent of the solids. A plateout factor of 50-percent is applied to the halogen activity resulting in 25 percent of the halogens available for release as a result of containment leakage. TID-14844 specifies that the full evolution of the accident source term is assumed to occur instantaneously. TID-14844 is silent on the apportionment of fission products to the containment sump. NRC guidance conservatively assumes that 50 percent of the core inventory of iodine is assumed to reside in the containment sump for the evaluation of the Emergency Core Cooling System (ECCS) leakage contribution to the LOCA dose consequence analysis.

Calculations performed to show compliance with the 10 CFR 100.11 dose criteria should use a substantial core melt source term, notwithstanding the fact that ECCS operation is designed to preclude this degree of core damage. The substantial core melt source term is used as a surrogate to represent a potential hazard not exceeded by those from any accident considered credible. This dose consequence analysis is often referred to as the maximum hypothetical accident (MHA) or as the maximum credible accident and is separate and distinct from all other dose consequence analyses in a plant's licensing basis.

In order to show compliance with the conservative assumptions used in the MHA, the contribution from primary to secondary leakage resulting from degraded SG tubes can be assessed using the conservative assumption that 50 percent of the iodine core inventory resides in the leaking fluid with the resulting contribution added to the doses calculated for the licensing basis MHA. By simple ratio, the primary to secondary leakage dose contribution would be five times the values calculated in C-NSA-060.00-015. This equates to a two hour EAB thyroid dose of 16 rem, and a 30-day dose of 7.6 rem at the outer boundary of the LPZ. When these contributions are added to the results for the MHA as described in FSAR Table 15.4.6-2, the total MHA doses can be shown to be less than the limits of 10 CFR 100.11.

The NRC staff discussed using the conservative substantial core melt assumptions associated with the MHA in C-NSA-060.00-015 in order to show that a primary to secondary leak rate of 1.0 gpm results in doses that meet the limits of 10 CFR 100.11. This would make the source term used in C-NSA-060.00-015 inconsistent with the source term used in FSAR Table 15.4.6.2 for the MHA.

The NRC staff identified additional conservatisms in C-NSA-060.00-015 which could be used to add margin to the LBLOCA dose contribution from primary to secondary leakage. Principal among these is the assumption that the release will continue without credit for operator action to isolate the secondary side. Plant personnel confirmed that procedures are in place that would direct operators to secure the secondary side in the event of a LBLOCA. Although C-NSA-060.00-015 did not use the TID-14844 source term magnitude, the calculation does assume that the release from the fuel to the RCS occurs instantaneously. This assumption was always known to be very conservative. Subsequent research, as documented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," describes the updated timing of an unmitigated core melt source term. According to NUREG-1465, during the first 30 minutes of the accident, the release would be characterized as a gap release prior to the onset of the early in-vessel release. In other words, updated guidance suggests that a substantial meltdown of the core with subsequent release of appreciable quantities of fission products would not be expected during the first 30 minutes of a LBLOCA. Using an assumption of 30 minutes for operator action to isolate the secondary side, together with the insights on the timing of the source term as described in NUREG-1465, it is plausible that the release from the secondary side could be isolated before the onset of the early in-vessel release phase.

The NRC staff also noted that C-NSA-060.00-015 used dose conversion factors (DCFs) from TID-14844 which have been shown to be conservative relative to the updated DCFs by approximately 30 percent.

Conclusions

The team concluded that the approach for assessing tube integrity following a LBLOCA is consistent with NRC staff safety evaluations or industry guidance, as appropriate. Industry guidance was used by the licensee in instances where the NRC staff had not specifically approved or prescribed a methodology to be used (i.e., in the absence of specific/prescriptive regulatory guidance or requirements).

The team also concluded, based on the audit of dose calculation C-NSA-060.00-015, that the dose contribution from a 1.0 gpm primary to secondary leak resulting from a LBLOCA at the DBNPS, when added to the current licensing basis MHA dose, results in a total MHA dose below the limits of 10 CFR 100.11.

The audit team members identified two areas of potential inconsistency between various licensee documents. The first potential inconsistency is between the technical specifications and a license condition regarding the extent of tube inspections. The second potential inconsistency is between the source term used in the FSAR MHA and Calculation number C-NSA-060.00-015. These were discussed with the licensee. These inconsistencies are not safety significant for the reasons discussed above.

The team members did not identify any plant-specific safety issues as a result of the audit.

May 19, 2010

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 – AUDIT OF STEAM GENERATOR PROGRAM FOCUSING ON STEAM GENERATOR TUBE INTEGRITY DURING A LARGE BREAK LOSS OF COOLANT ACCIDENT (TAC NO. ME2875)

Dear Mr. Allen:

On February 23, 2010, U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The purpose of the audit was for the NRC to gain a better understanding of the approach taken by DBNPS for ensuring steam generator tube integrity following a large break of coolant accident. The audit report is enclosed.

The audit team members did not identify any plant-specific safety issues as a result of the audit. Two potential areas of inconsistency between various licensee documents were identified. These inconsistencies are discussed in detail in the enclosed audit report. The first inconsistency deals with the technical specifications and a license condition 2.C(7)c. The second inconsistency deals with the source term used in the Final Safety Analysis Report (FSAR) for the maximum hypothetical accident and a calculation used to satisfy a license condition. These inconsistencies were discussed with representatives of DBNPS at the conclusion of the audit.

Please provide a response within 60 days of the date of this letter on how the two inconsistencies indentified in the enclosed audit report will be addressed.

If you have any questions, please contact me at 301-415-3867 or by e-mail at Michael.Mahoney@nrc.gov.

Sincerely,
/RA/
Michael Mahoney, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346
Enclosure:
As stated
cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL3-2 R/F
RidsNrrDorlLpl3-2 Resource
RidsNrrPMMMahoney
RidsNrrLATHarris Resource
RidsRgn3MailCenter Resource
AJohnson. NRR
KKarwoski, NRR

ADAMS Accession No. ML *By Memo Dated

OFFICE	NRR/LPL3-2/PM	NRR/LPL3-2/LA	NRR/DRA/AADB	NRR/DCI/CSGB	NRR/LPL3-2/BC
NAME	MMahoney	THarris	JParillo	KKarwoshi	SCampbell
DATE	04/29 /10	04/29/10	04/02/10*	04/05/10*	05/19 /10

OFFICIAL RECORD COPY