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April 15, 2011
L-11-107

10 CFR 54

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
Reply to Request for Additional Information on the Reactor Vessel Surveillance Aging Management Program and Time-Limited Aging Analyses for Neutron Embrittlement for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 2

By letter dated August 27, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1. By letter dated March 17, 2011 (ADAMS Accession No. ML110680172), the Nuclear Regulatory Commission (NRC) requested additional information (RAI) to complete its review of the License Renewal Application (LRA).

The Attachment provides the FENOC reply to the NRC request for additional information. The NRC request is shown in bold text followed by the FENOC response. The Enclosures provide Amendment No. 2 to the DBNPS LRA and several documents that support RAI responses.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at (724) 682-7139.

A145
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 15, 2011.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:

Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application

Enclosures:

A. Amendment No. 2 to the DBNPS License Renewal Application
B. AREVA NP Document 86-910440-000, "Fracture Mechanics Analysis of Postulated Underclad Cracks in the DB-1 Reactor Vessel for 60 Years"
C. Centerior Energy Letter, "High Pressure Injection/Makeup Nozzle and Thermal Sleeve Program Davis-Besse Nuclear Power Station Unit 1 (Serial No. 1968)," (ADAMS Accession Number ML9109030090), August 23, 1991
D. Babcock and Wilcox document 32-1172294-00, "Davis Besse 1 SG Flaw Evaluation," dated 6/9/1988
   Babcock and Wilcox document 32-1172294-01, "Davis Besse 1 SG Flaw Evaluation," dated 7/18/1988
   Babcock and Wilcox document 32-1172523-00, "DB-1 SG Flaw Evaluation," dated 7/18/1988

cc: NRC DLR Project Manager
NRC Region III Administrator

cc: w/o Attachment or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Section B.2.35 – Reactor Vessel Surveillance Program

RAI B.2.35-1

License Renewal Application (LRA) Section B.2.35 states that "[c]apsule TE1-C contains the Davis Besse limiting material and has been exposed to fluence slightly above the 60-year projected fluence for the Davis-Besse plant."

Please state whether the limiting material referred to in this statement is Upper Shell to Lower Shell Circumferential Weld WF-182-1. Please state the fluence value for Capsule TE1-C.

RESPONSE RAI B.2.35-1

The limiting material referenced in LRA Section B.2.35 is the Upper Shell Forging to Lower Shell Forging Circumferential Weld WF-182-1.

Capsule TE1-C contains weld metal WF-182-1. The estimated fluence recorded by the Master Integrated Reactor Vessel Surveillance Program for Capsule TE1-C is 1.81E+19 n/cm^2. The 52 EFPY peak fluence reported in Table 4.2-1 of the LRA is 1.70E+19 n/cm^2.

RAI B.2.35-2

LRA Section B.2.35 states that Capsule TE1-E was removed from the reactor vessel and has been discarded.

Please explain why this surveillance capsule was discarded.

RESPONSE RAI B.2.35-2

Capsule TE1-E was discarded in accordance with the NRC approved Master Integrated Reactor Vessel Surveillance Program because the specimens in this capsule would not contribute significant data to the Babcock and Wilcox Owners Group (B&WOG) Master Integrated Reactor Vessel Surveillance Program.

The NRC staff was notified on March 17, 2000 by the B&WOG Reactor Vessel Working Group of plans to dispose of irradiated plant-specific materials, including Davis-Besse plant-specific Capsules TE1-C and TE1-E (ADAMS Accession Number ML003693967).

The disposal schedule was incorporated into BAW-1543, Revision 4, Supplement 4, which was reviewed and approved by the NRC (SER is included in BAW-1543A).

BAW-1543A, Table IV, Page 12, identifies both TE1-C and TE1-E as capsules that will be discarded. BAW-1543A, Revision 4, Supplement 6, Page 2, amended the disposal schedule by stating that Capsule TE1-C is slated for disposal; however, it will likely be tested since it contains the Davis-Besse limiting material and has a fluence between 1 and 2 times Davis-Besse's 60 year projection.

The Reactor Vessel Surveillance Program will be enhanced to schedule testing of the TE1-C capsule as committed by FENOC in License Renewal Application Appendix A, Table A-1, Commitment Item Number 17.

Section 4.2.2 – USE Evaluation

RAI 4.2.2-1

LRA Section 4.2.2.1 states that, as no initial USE data is available for the reactor vessel (RV) beltline welds, plant operation through 32 EFPY (40 years) was justified based on an equivalent margins analysis (EMA). LRA Section 4.2.2.1 provides references for the subject EMA in LRA Section 4.8, References 4.8-2 and 4.8-3. LRA Section 4.2.2.3 states that the equivalent margins analysis (EMA) for the limiting beltline weld (WF-182-1) is projected to satisfy the acceptance criteria of the ASME Code, Section XI (the Code), Appendix K through the period of extended operation (PEO) (52 EFPY).

a. **Are the existing criteria for minimum acceptable USE developed in References 4.8-2 and 4.8-3 (using ASME Code, Section XI, Appendix K EMA procedures) valid for demonstrating RV beltline weld acceptability through 52 EFPY, based on the calculations of the projected percentage decrease in USE for 52 EFPY, as listed in LRA Table 4.2-2?**

b. **If the existing criteria for minimum acceptable USE developed in References 4.8-2 and 4.8-3 are not valid for demonstrating RV beltline weld acceptability through 52 EFPY, then please provide the reports documenting the EMA calculations for demonstrating that all RV beltline welds, including the limiting beltline weld (WF-182-1), will satisfy the requirements of 10 CFR Part 50, Appendix G for equivalent margins against ductile fracture through the PEO (52 EFPY).**

RESPONSE RAI 4.2.2-1

Response for a:

Yes, the existing criteria for minimum acceptable USE developed in LRA References 4.8-2 and 4.8-3 (using ASME Code, Section XI, Appendix K EMA procedures) are valid for demonstrating RV beltline weld acceptability through 52 EFPY.

The methodology and acceptance criteria reported in LRA References 4.8-2 and 4.8-3 were based on Code Case N-512, which later was incorporated into Appendix K to the ASME Code, Section XI. The ASME Section XI Appendix K acceptance criteria and evaluation procedure may be used to determine the acceptability for operation of a reactor vessel when the vessel metal temperature is in the upper shelf range.

The criteria and methods used to evaluate the limiting RV beltline weld, WF-182-1, for 32 EFPY and 52 EFPY are based on ASME Section XI, Appendix K, 1995 Edition through 1996 Addenda, for Level A through D Service Loadings. The criteria and methodology in Appendix K require that the applied J-integral be compared to the J-integral of the material. The applied J-integral and J-integral of the material are dependent on the change in material properties of weld WF-182-1 but are independent of the calculation of upper shelf energy at 52 EFPY for weld WF-182-1 reported in Table 4.2-2 of the LRA. As discussed in Section 3.1 and Appendix B of BAW-2192PA, J-integral is a function of weld copper content, fluence, metal temperature, and net specimen thickness. The material properties reported in Section 3.2 of BAW-2192PA are used in the calculation of the applied J integral as reported in Appendix A of BAW-2192PA.

Response for b:

While the methods and acceptance criteria remain valid, the analysis was extended to 52 EFPY for license renewal, using the methodology and acceptance criteria from ASME Section XI, Appendix K, 1995 Edition through 1996 Addenda, for Level A through D Service Loadings.

As presented in Section 4.2.2.3 and Table 4.2-2 of the LRA, the limiting reactor vessel beltline weld WF-182-1 is the only 60-year (52 EFPY) beltline location with a projected Charpy impact energy level below 50 ft-lbs and therefore, requires an equivalent margins analysis (EMA) for the period of extended operation. The EMA at 32 EFPY for weld WF-182-1 was updated for the measurement uncertainty recapture (MUR) uprate. The MUR EMA was extended to 52 EFPY to address the period of extended operation. The results of the equivalent margins analysis at 52 EFPY are included in Section 4.2.2.3 of the LRA.

LRA Section 4.2.2.1, 3rd paragraph contains an incorrect reference for the USE evaluation associated with the measurement uncertainty recapture. See Enclosure A to this letter for the revision to the LRA.

RAI 4.2.2-2

LRA Table 4.2-2 lists the initial USE for all RV beltline welds (Linde 80) as 70 ft-lbs. Please explain how the 70 ft-lbs initial USE value for these welds was obtained or derived (i.e., from a conservative estimate based on a statistically significant sample of existing Charpy USE data for this type of weld, or other method), given the statement in LRA Section 4.2.2.1 that no initial USE data is available for the RV beltline welds. If the 70 ft-lbs initial USE value is a conservative estimate based on a statistically significant sample of existing Charpy USE data for this type of weld, please state whether the EMA calculation reports (References 4.8-2 and 4.8-3) included the statistical analysis of the Charpy USE data for Linde 80 welds. Otherwise, please provide the statistical analysis of the Charpy USE data for this type of weld.

RESPONSE RAI 4.2.2-2

The statement in Section 4.2.2.1 that no initial USE is available for beltline welds (Linde 80 welds) requires clarification and a change to the LRA.

The reactor vessel was designed in accordance with ASME III 1968 Edition with Addenda through Summer 1968, and the only requirement relative to Charpy V-Notch testing was to test a set of three specimens at temperatures 60 °F below the lower of the hydrotest temperature or the lowest service metal temperature. Therefore, unirradiated upper shelf energy data was not available from the ASME III required Charpy V-Notch tests associated with the original construction of the reactor vessel. However, archived weld metal specimens (Weld WF-182-1) associated with the plant-specific capsules were tested and unirradiated upper shelf energy values for weld WF-182-1 are reported in the plant-specific capsule reports. The capsules contain the most limiting weld metal (WF-182-1), heat affected zone metal, and base metal; there are no measured unirradiated USE data available for the remaining beltline Linde 80 welds listed in Table 4.2-2 of the LRA since no plant-specific archived material exists for these welds.

In order to obtain unirradiated USE values for the beltline Linde 80 welds with no plant-specific capsule or archived data, the B&WOG Master Integrated Reactor Vessel Program (MIRVP) established a generic mean value for all Linde 80 welds using measured data from archived specimens designated with plant-specific capsules from each of the participating MIRVP plants. Specifically, the statistical analysis of upper shelf energy for Linde 80 welds is reported in topical report BAW-1803 [Reference 1]. A generic initial upper shelf energy (USE) of 69.7 ft-lbs, rounded to 70 ft-lb, was established for all Linde 80 welds in BAW-1803, Table 3-5. This value is a mean value with a standard deviation of 5.6, minimum value of 64 ft-lb and maximum value of 81 ft-lb. The population of Linde 80 welds considered in the statistical evaluation includes measured data and is reported in Table 3-2 of BAW-1803 [Reference 1]. The data set reported in BAW-1803, Table 3-2, includes 16 unirradiated measured upper shelf energy values from 7 heats of weld wire that cover the most limiting welds for the MIRVP plants.

A generic mean initial USE of 70 ft-lb was used for all Linde 80 welds in Table 4.2-2 of the LRA. As was the case for 32 EFPY, weld WF-182-1 is the limiting Linde 80 weld at 52 EFPY and the equivalent margins analysis reported in Section 4.2.2.3 bounds the remaining Linde 80 locations within the upper and lower shells.

Reference 1 – BAW-1803, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," January 1984 (ADAMS Accession Number ML8504240333)

See Enclosure A to this letter for the revision to the LRA, Section 4.2.2.1.

RAI 4.2.2-3

LRA Section 4.2.2.2 states that Regulatory Position (RP) 2.2 of Regulatory Guide 1.99, Rev. 2 was used to calculate 52 EFPY USE values for weld WF-182-1 and forging BCC 241 using surveillance data. Please state whether the 52 EFPY USE values (2nd line entry for 52 EFPY USE in Table 4.2-2 for each component) for these materials are based on two credible sets of USE surveillance data for these materials.

RESPONSE RAI 4.2.2-3

The 52 EFPY USE values for weld WF-182-1 and forging BCC 241 provided in LRA Table 4.2-2 citing footnote 2 are based on two or more credible sets of USE surveillance data for these materials.

The second line entry in LRA Table 4.2-2 for forging BCC 241 is based on four credible sets of surveillance data.

The second line entry in LRA Table 4.2-2 for weld WF-182-1 is based on five credible sets of surveillance data.

Section 4.2.6 – Intergranular Separation / Underclad Cracking

RAI 4.2.6-1

Please provide a reference for the report documenting the detailed analyses for demonstrating that the postulated underclad cracks in the Davis-Besse RV SA-508, Class 2 forging materials are acceptable for the period of extended operation.

RESPONSE RAI 4.2.6-1

The non-proprietary version of the fracture mechanics analysis for demonstrating that the postulated underclad cracks in the Davis-Besse RV SA-508, Class 2 forging materials are acceptable for the period of extended operation is AREVA NP Document 86-910440-000, "Fracture Mechanics Analysis of Postulated Underclad Cracks in the DB-1 Reactor Vessel for 60 Years," dated July, 2010.

A copy of AREVA NP Document 86-910440-000 is provided in the Enclosure B to this letter.

Section 4.2.7 – Reduction in Fracture Toughness of RV Internals

RAI 4.2.7-1

The staff notes that cast austenitic stainless steel (CASS) components are susceptible to reduction in fracture toughness due to the synergistic effects of both neutron embrittlement and thermal embrittlement. The LRA aging management review (AMR) results for the RV internals (LRA Table 3.1.2-2) lists a number of CASS RV internal components.

a. **Will these CASS RV internals components be screened for susceptibility to thermal embrittlement based on ferrite content, molybdenum content, and casting method under the Davis-Besse PWR Reactor Vessel Internals Aging Management Program (AMP)?**
b. **Will CASS RV internals components determined to be susceptible to thermal embrittlement (based on ferrite content, molybdenum content, and casting method) receive supplemental examinations and/or component specific evaluations of reduction in fracture toughness (due to the synergistic effects of both neutron embrittlement and thermal embrittlement) under the Davis-Besse PWR Reactor Vessel Internals AMP?**

RESPONSE RAI 4.2.7-1

Response for a:

The screening of the reactor vessel internals CASS components was performed as part of the development of MRP-227 [Reference 1] which forms the basis of the Davis-Besse PWR Reactor Vessel Internals Program. As an input document to MRP-227, MRP-189 [Reference 2] performed the screening of the CASS internals components and included ferrite and molybdenum content in the parameters screened. The detailed screening criteria are contained in MRP-175 [Reference 3].

LRA Table 3.1.2-2 lists four reactor vessel internals component types made from Cast Austenitic Stainless Steel (CASS) that are susceptible to reduction of fracture toughness; Core Support Assembly (CSA) Incore Guide Tube Assembly Spider, CSA Vent Valve Assembly Valve Body; Plenum Assembly (PA) control rod guide tube spacer casting, and PA reinforcing plate (line items are 130, 144, 182 and 204 from LRA Table 3.1.2-2). As shown in Table 3.1.2-2, reduction of fracture toughness for the subject component types is managed by the PWR Reactor Vessel Internals Program.

Response for b:

Component specific evaluations were performed as part of the development of MRP-227 [Reference 1] for the Core Support Assembly (CSA) Incore Guide Tube Assembly Spider, Plenum Assembly (PA) control rod guide tube spacer casting, and PA reinforcing plate. MRP-227 provided a functionality assessment of degradation for components and assemblies of components; then developed an aging management strategy combining results of the functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

In addition, MRP-276 [Reference 4] found that implementation of MRP-227 guidelines provides appropriate aging management for irradiated cast austenitic stainless steels.

As provided in the LRA Appendix A License Renewal Commitment List, Item Number 15 of Table A-1, FENOC will revise the PWR Reactor Vessel Internals Program, as necessary, to incorporate the final recommendations and requirements following NRC approval of MRP-227 and re-issuance of the guidelines as MRP-227-A.

References included with this response:

1. EPRI Report No. 1016596, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev.0),” December 2008 (ADAMS Accession Number ML090160204)
2. ERPI Report No.1018292, “Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev.1),” March 2009 (ADAMS Accession Number ML091671777)

3. ERPI Report No. 1012081, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," December 2005 (ADAMS Accession Number ML071500462)

4. ERPI Report No.1020959, "Materials Reliability Program: Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals (MRP-276)," May 2010 (ADAMS Accession Number ML102950165)

Section 4.7.4 – High Pressure Injection / Makeup Nozzle Thermal Sleeves

RAI 4.7.4-1

LRA Section 4.7.4 states that the high pressure injection (HPI) makeup flow path was re-routed from HPI/makeup nozzle A-1 to nozzle A-2 during the Cycle 6 refueling outage (1990) as one of the corrective actions for the subject failed thermal sleeve. LRA Section 4.7.4 then states that fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions predicted a thermal sleeve lifetime exceeding 20 eighteen-month operating cycles under current makeup flow control conditions. LRA Section 4.7.4 stated that, since that analysis, Davis-Besse had an extended (approximately two year) Cycle 13 refueling outage, converted to a 24-month fuel cycle, and performed a measurement uncertainty recapture power uprate. The corresponding predicted end-of-life for the HPI/makeup nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles.

The staff notes that Davis-Besse has committed to replacing all four makeup nozzle thermal sleeves prior to the beginning of the period of extended operation. Based on the above discussion, please state which specific HPI/makeup nozzle thermal sleeves (thermal sleeves for HPI/makeup nozzle A-2 or other nozzle thermal sleeves) were analyzed as discussed above.

RESPONSE RAI 4.7.4-1

The fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions was performed to predict the life of the thermal sleeve for the high pressure injection (HPI) nozzle that is used for both HPI and makeup duty. This analysis was applicable to HPI nozzle A-1 (also known as HPI nozzle 2-1) and A-2 (also known as HPI nozzle 2-2). The other two HPI nozzles, B-1 (also known as HPI nozzle 1-1) and B-2 (also known as HPI nozzle 1-2) were not addressed in the analysis.

Clarification and Change to LRA Table A-1, "Davis-Besse License Renewal Commitments," Item Number 23.

As provided in the LRA, the commitment reads as follows:

"In association with the TLAA for cracking of the high pressure injection / makeup nozzle thermal sleeves, FENOC commits to replace all four high pressure injection / makeup nozzle thermal sleeves and safe ends prior to the period of extended operation. In addition, FENOC commits to evaluate the environmental effects on the replacement HPI nozzle safe ends and associated welds in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." Any nickel-based alloy locations will be evaluated in accordance with NUREG/CR-6909."

However, as provided in LRA Section A.2.3.4.2, this commitment, Item Number 23, is also associated with the effects of environmentally assisted fatigue of the HPI nozzle safe end and associated Alloy 82/182 weld (weld that connects the safe end to the nozzle). Therefore, the commitment requires a change to include the association with the effects of environmentally assisted fatigue and addition of the associated Alloy 82/182 weld.

See the Enclosure A to this letter for the revision to the LRA.

RAI 4.7.4-2

LRA Section 4.7.4 states that the high pressure injection (HPI) makeup flow path was re-routed from HPI/makeup nozzle A-1 to nozzle A-2 during the Cycle 6 refueling outage (1990) as one of the corrective actions for the subject failed thermal sleeve. LRA Section 4.7.4 then states that fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions predicted a thermal sleeve lifetime exceeding 20 eighteen-month operating cycles under current makeup flow control conditions. LRA Section 4.7.4 stated that, since that analysis, Davis-Besse had an extended (approximately two year) Cycle 13 refueling outage, converted to a 24-month fuel cycle, and performed a measurement uncertainty recapture power uprate. The corresponding predicted end-of-life for the HPI/makeup nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles.

Please provide a reference for the subject thermal sleeve fracture mechanics analysis.

RESPONSE RAI 4.7.4-2

The fracture mechanics analysis to evaluate thermal sleeve reliability of the HPI/makeup nozzle (currently HPI nozzle A-2) was prepared by Structural Integrity Associates and is as follows:

SIR-91-047, Rev. 0, "Fracture Mechanics Evaluation of Davis-Besse HPI/Makeup Nozzle Thermal Sleeve," August 1991.

Centerior Energy described the methodology and the results of the subject analysis in a letter, Serial Number 1968, to the NRC dated August 21, 1990.

For convenience, a copy of Serial Number 1968, "High Pressure Injection/Makeup Nozzle and Thermal Sleeve Program Davis-Besse Nuclear Power Station Unit 1," dated August 23, 1991 (Agencywide Documents Access and Management System (ADAMS), Accession Number ML9109030090) is provided in Enclosure C to this letter.

Section 4.7.5.1 – Reactor Coolant System Loop 1 Cold Leg Drain Line Weld Overlay Repair

RAI 4.7.5.1-1

LRA Section 4.2.5.1 states that the applicant performed a full structural overlay repair for an axial indication found on the Reactor Coolant System Loop 1 cold leg drain line during the Cycle 14 refueling outage. The structural weld overlay of the cold leg drain nozzle was designed consistent with the requirements of the ASME Code, Section XI; ASME Code Case N-504-2; ASME Code, Section XI, non-mandatory Appendix Q; and was supplemented by additional design considerations specific to the unique nature of the geometry and materials of the cold leg drain nozzle-to-elbow weld. The overlay is designed as a full structural overlay that assumes the as-found flaw propagates to a 100% through-wall 360-degree crack, rather than performing a crack growth analysis of the as-found flaw. The fatigue analysis for the repaired configuration conservatively estimated cycles for 60 years at 1.5 times the original design cycles.

Please provide a reference for the fatigue analysis of the repaired configuration discussed above, if this fatigue analysis is not referenced elsewhere in the LRA.

RESPONSE RAI 4.7.5.1-1

The reference for the fatigue analysis addressed in LRA Section 4.7.5.1 is Calculation Number DB-06Q-304 Rev.1, "RCS Cold Leg Letdown Line Nozzle Weld Overlay Repair ASME Code Section III Evaluation."

By letter dated May 22, 2006 (Serial Number 3266), (Agencywide Documents Access and Management System (ADAMS) Accession Number ML061440282), FENOC submitted a summary of the calculation packages that were prepared to document the design and analysis of the DBNPS Reactor Coolant Pump 1-1 inlet cold leg drain line nozzle-to-elbow weld overlay. As provided in the summary, Calculation Number DB-06Q-304, Rev.1 performed an ASME Code, Section III, Class 1 evaluation for the repaired configuration.

Section 4.7.5.2 – Once-Through Steam Generator 1-2 Flaw Evaluations

RAI 4.7.5.2-1

LRA Section 4.7.5.2 discusses a number of flaws that were discovered in the steam generator 1-2 shell base material and shell welds during the Cycle 5 refueling outage (May 1988). The staff requests the following information concerning these flaws and the analyses performed for these flaws.

a. **How many flaw indications were found in total (in 1988) that did not pass the initial ASME Code, Section XI, IWB-3500 screening criteria?**
b. **Were these flaws determined to be the result of service-induced degradation or fabrication defects?**
c. **Have the components with the flaws received subsequent/supplemental examinations, in accordance with ASME Code, Section XI requirements since May 1988?**
d. **When is the next inservice examination scheduled for the components with the flaws?**
e. **Have the flaw dimensions increased since discovery in 1988? If so, were the flaws re-analyzed in accordance with ASME Code, Section XI, IWB-3600 requirements based on the new flaw dimensions?**
f. **Are the existing flaw growth analyses for the subject flaws bounded by the projected number of thermal cycles for the period of extended operation? If not, the staff requests that the applicant provide a specific license renewal commitment to re-evaluate the subject flaws (per IWB-3600) for the period of extended operation, based on the results of the next scheduled inservice examination of the components with the flaws.**
g. **No flaw evaluation reports are referenced. Please provide references to reports documenting IWB-3600 analytical evaluations of the subject flaws. Were these reports previously submitted to the NRC?**

RESPONSE RAI 4.7.5.2-1

a. A total of twelve indications where found in the Cycle 5 refueling outage (year 1988) for steam generator 1-2 that did not meet ASME Code, Section XI, IWB-3500 screening criteria. Note that Article IWC-3000, "Acceptance Standards For Flaw Indications," was in the course of preparation and allowed for usage of IWB-3000, "Acceptance Standards For Flaw Indications." Of the twelve indications, ten were associated with the shell welds near the lower tubesheet-to-shell juncture and two were associated with the shell to steam outlet nozzle welds. Applicable code year was ASME Boiler and Pressure Vessel Code, 1977 Edition with Addenda through Summer 1978.

b. The flaws were analyzed in accordance with IWB-3612, as required by the ASME Section XI Acceptance Standards, and found to be acceptable for continued operation.

c. Initial reexamination of the subject components was performed during the Cycle 6 refueling outage (year 1990). In addition, the subject components were reexamined in Cycle 7 (year 1991), with the exception of the W axis longitudinal seam weld intersection with the shell to lower tubesheet weld, to meet the ASME Section XI, IWC-2420(b) requirements. Although the W axis longitudinal seam weld was not identified as a required examination in ASME Section XI, 1977 Edition through Summer 1978 addenda, Table IWC-2500-1, Examination Category C-A, Pressure Retaining Welds in Pressure Vessels, the weld was analyzed and determined to be acceptable for continued service in accordance with IWB-3612 and reexamined during the Cycle 6 refueling outage (year 1990) with no flaw growth noted.

d. As this is the end of the Davis-Besse 3rd ISI interval (ends 9/20/2012), the only steam generator flaw location still scheduled for examination this interval is the SG 1-2 W/X axis outlet nozzle to shell weld currently scheduled for the 17 Midcycle Outage (year 2011). The SG 1-2 X/Y axis outlet nozzle to shell weld and the SG 1-2 lower tubesheet to shell weld were examined in the 2nd and 1st periods of the interval, respectively.

e. The subject components were reexamined during Cycle 6 (year 1990) and no flaw growth was noted. The subject components, with the exception of the W axis longitudinal seam weld intersection with the shell to lower tubesheet weld, were also reexamined during Cycle 7 (year 1991) and no flaw growth was noted.

f. As provided in LRA Section 4.7.5.2, the simplified evaluation of fatigue crack growth was based on 240 heatup and cooldown cycles. As shown in LRA Table 4.3-1, the 60-year projection of heat and cooldown cycles is 128 and is bounded by the 240 design cycles used in the subject flaw evaluations.

g. The flaw evaluations are documented in Babcock & Wilcox Reports listed as follows:

   1) 32-1172294-00, "Davis Besse 1 SG Flaw Evaluation," dated 6/9/1988

   2) 32-1172294-01, "Davis Besse 1 SG Flaw Evaluation," dated 7/18/1988

   3) 32-1172523-00, "DB-1 SG Flaw Evaluation," dated 7/18/1988

A copy of each report is provided in Enclosure D to this letter.

Enclosure A

Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1

Letter L-11-107

Amendment No. 2 to the
DBNPS License Renewal Application

License Renewal Application
Sections Affected

Section 4.2.2.1

Section 4.8

Table A-1

The Enclosure identifies the change to the License Renewal Application (LRA) by Affected LRA Section, LRA Page No., and Affected Paragraph and Sentence. The count for the affected paragraph, sentence, bullet, etc. starts at the beginning of the affected Section or at the top of the affected page, as appropriate. Below each section the reason for the change is identified, and the sentence affected is printed in *italics* with deleted text ~~*lined-out*~~ and added text <u>*underlined*</u>.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.2.2.1	**Page 4.2-5**	**Third paragraph, first sentence**

A reference cited in the LRA needs to be corrected. The first sentence of the third paragraph of LRA Section 4.2.2.1, "Background," is revised to read:

USE was re-evaluated for the measurement uncertainty recapture power uprate [Reference ~~4.8-3~~ 4.8-18 and 4.8-19].

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.8	**Page 4.8-2**	**New references**

The following two references are added to LRA Section 4.8, "References:"

4.8-18 Serial 3198, "Davis-Besse Nuclear Power Station License Amendment Application for Measurement Uncertainty Recapture Power Uprate (License Amendment Request No. 05-0007)," April 12, 2007 (Agencywide Documents Access and Management System (ADAMS), Accession Number ML071030396)

4.8-19 NRC Letter, "Davis-Besse Nuclear Power Station, Unit No.1 – Issuance of Amendment (278) RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD8326), June 30, 2008 (Agencywide Documents Access and Management System (ADAMS), Accession Number ML081410652)

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.2.2.1	**Page 4.2-5**	**Second paragraph, second sentence**

The statement in Section 4.2.2.1 that no initial USE is available for beltline welds (Linde 80 welds) requires clarification and a change to the LRA. The second sentence of the second paragraph of LRA Section 4.2.2.1, "Background," is revised to read:

~~As no initial USE is available for the beltline welds (Linde80 welds), operation for~~ Operation to 32 EFPY for Davis-Besse reactor vessel beltline Linde 80 welds was justified based on an equivalent margins analysis (fracture mechanics analysis) [References 4.8-2 and 4.8-3].

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A-1	**Page A-69**	**Commitment No. 23**

Commitment No. 23 requires a change to include the association with the effects of environmentally assisted fatigue and addition of the associated Alloy 82/182 weld. Table A-1, "Davis-Besse License Renewal Commitments," Item No. 23 is revised as follows:

"Commitment" column:

In association with the TLAA for effects of environmentally assisted fatigue of the high pressure injection (HPI) nozzle safe end including the associated Alloy 82/182 weld (weld that connects the safe end to the nozzle), and cracking of the ~~high pressure injection / makeup~~ HPI/makeup nozzle thermal sleeve~~s~~, FENOC commits to replace ~~all four high pressure injection / makeup nozzle~~ the HPI nozzle safe end including the associated Alloy 82/182 weld, and the thermal sleeve~~s~~ for all four HPI nozzles prior to the period of extended operation. In addition, FENOC commits to evaluate the environmental effects on the replacement HPI nozzle safe end~~s~~ and the weld~~s~~ that connects the safe end to the nozzle in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," Rev. 1. Any nickel-based alloy locations will be evaluated in accordance with NUREG/CR-6909.

"Related LRA Section No./Comments" column:

A.2.3.4.2
A.2.7.4