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April 20, 2011
L-11-115

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 Number NPF-3
Reply to Request for Additional Information 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. 4

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 21, 2011 (ADAMS Accession No. ML11068A000), the Nuclear Regulatory Commission (NRC) requested additional information 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 Enclosure provides Amendment No. 4 to the DBNPS LRA.

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 20, 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, Sections 4.7 and A.2.7

Enclosure:
Amendment No. 4 to the DBNPS License Renewal Application

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 4.7.1.1 – Fatigue Flaw Growth

Question RAI 4.7.1.1-1

Section 4.7.1.1 states that " ... [t]he LBB analysis postulated surface flaws at the piping system locations with the highest stress coincident with the lower bound of the material properties for base metal and welds. The fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions ... "

Provide the following information regarding the fatigue flaw growth analysis:

(1) discuss the number of postulated surface flaws that were analyzed for fatigue flaw growth,

(2) discuss the initial and final flaw sizes due to fatigue,

(3) identify the plant design transients and associated cycle numbers that were used in performing the fatigue flaw growth analysis,

(4) discuss how the fatigue flaw growth was calculated,

(5) identify the piping system that contains the postulated fatigue flaws,

(6) clarify whether the number of the transient cycles used in the fatigue flaw growth calculation is based on 40 years or 60 years.

RESPONSE RAI 4.7.1.1-1

(1) The Leak-Before-Break (LBB) evaluation, BAW-1847 Rev. 1 [Reference 1], included fatigue flaw growth analyses to demonstrate that postulated surface flaws are likely to propagate in the through-wall direction and develop leakage before they will propagate circumferentially around the pipe (and produce pipe failure). BAW-1847, Rev. 1 evaluated large, high energy reactor coolant system piping for Babcock & Wilcox (B&W) designed plants and postulated flaws in piping with inner diameters of 28 to 38 inches; only the 36-inch and 28-inch diameters are applicable to Davis-Besse. The smallest and largest pipe straight sections were analyzed for fatigue. A longitudinal flaw and a circumferential flaw were evaluated for each diameter.

(2) The results of the fatigue flaw growth analysis are presented in Table 4-2 of BAW-1847, Rev. 1 (also provided in the response to item 4) and show the minimum postulated flaws that could grow to through-wall. These initial flaws are

all at least 30% through-wall. From the fatigue flaw growth analysis, the (postulated) flaw sizes that grow through-wall are clearly many times those allowed by the Code and represent a very conservative estimate of any existing flaws. Note, however, that whether or not the flaws grow through-wall does not affect the conclusion of the LBB evaluation. The important factor is that postulated flaws would propagate radially, and thus go through-wall and produce leakage before they propagate circumferentially and produce pipe failure.

(3) The design transients and associated design cycles used in the LBB fatigue flaw growth evaluation are listed in BAW-1847, Rev. 1, Tables 4-3, 4-4 and 4-5. Fatigue flaw growth is based on six categories of Nuclear Steam Supply System (NSSS) design transients. Category 1 includes deadweight, thermal expansion and operating pressure associated with 240 heatup (HU)/cooldown (CD) cycles and corresponds to Table 4-3 in BAW-1847, Rev. 1. Categories 2 through 5 include thermal stresses due to four groupings of NSSS design transients (i.e., Groups I through IV) in BAW-1847, Rev. 1, Tables 4-4 and 4-5. Category 6 includes 22 safe shutdown earthquake events. The generic NSSS design transients used in the BAW-1847, Rev. 1, LBB evaluation were selected to bound the participating B&W plants for a 28-inch cold leg straight section (Table 4-4 of BAW-1847, Rev. 1) and a 38-inch hot leg section (Table 4-5 of BAW-1847, Rev. 1).

A comparison of NSSS design cycles used in the BAW-1847, Rev.1 fatigue flaw growth evaluation to the Davis-Besse NSSS design cycles is provided in the following tables. As shown in these tables, the analyzed cycles used in the LBB fatigue flaw growth analysis will bound 60-year projected cycles.

**Comparison of Davis-Besse Design Transients
to the Transient Grouping in BAW-1847, Rev 1, Table 4-4**

Davis-Besse		BAW-1847, Rev 1, Table 4-4, 28-inch pipe		
Transient #	**60-year projected cycles (from LRA Table 4.3-1)**	**Group**	**Transient**	**# analyzed**
1A	128	I	HU/CD	240
1B	128	I	HU/CD	240
9A-D (1)	4	II	Rapid Depressurization	40
8A-C	77	III	Reactor Trips (A, B, and C)	
11	40	III	Rod Withdrawal	
14	18	III	Control Rod Drop	
15	6	III	Loss of Station Power	
17A	6	III	Loss of FW to 1 SG	
Total of Group:	147	III	Total cycles analyzed:	428
2A	205	IV	Power change 0-15%	
2B	94	IV	Power change 15-0%	
3 (2)	1800	IV	Power loading, 8-100%	
4 (2)	1800	IV	Power unloading, 100-8%	
5	67	IV	Step load increase of 10%	
6	140	IV	Step load decrease of 10%	
7A	8	IV	Step load reduction 100-8%	
7B	4	IV	Step load reduction 100-8%	
10	10	IV	Flow change of 20%	
Total of Group:	4128	IV	Total cycles analyzed:	57770

(1) Transient 9A-D, as recorded in LRA Table 4.3-1, includes both rapid depressurizations and HPI nozzle cycles. There have been only 2 actual rapid depressurizations to date at Davis-Besse, and are projected to 4 rapid depressurizations for 60 years.

(2) As provided in LRA Table 4.3-1: "Transients 3 and 4 are not monitored. Davis-Besse is not a load following plant and therefore; transients 3 and 4 could not credibly approach the number of design cycles during the period of extended operation." Therefore, the 60-year projected cycles are set to the design cycles.

Comparison of Davis-Besse Design Transients to the Transient Grouping in BAW-1847, Rev 1, Table 4-5

Davis-Besse		BAW-1847, Rev 1, Table 4-5, 38-inch pipe		
Transient #	**60-year projected cycles (from LRA Table 4.3-1)**	**Group**	**Transient**	**# analyzed**
1A/1B	128	I	HU/CD	240
9A-D (1)	4	II	Rapid Depressurization	
8A-D	187	II	Reactor Trips (all)	
Total of Group:	191	II	Total cycles analyzed:	572
7A	8	III	Step load reduction 100-8%	
7B	4	III	Step load reduction 100-8%	
14	18	III	Control Rod Drop	
2A	204	III	Power change 0-15%	
Total of Group:	234	III	Total cycles analyzed:	430
3(2)	1800	IV	Power loading, 8-100%	
4(2)	1800	IV	Power unloading, 100-8%	
5	67	IV	Step load increase of 10%	
6	140	IV	Step load decrease of 10%	
Total of Group:	3807	IV	Total cycles analyzed:	58000

(1) Transient 9A-D, as recorded in LRA Table 4.3-1, includes both rapid depressurizations and HPI nozzle cycles. There have been only 2 actual rapid depressurizations to date at Davis-Besse, and are projected to 4 rapid depressurizations for 60 years.

(2) As provided in LRA Table 4.3-1: "Transients 3 and 4 are not monitored. Davis-Besse is not a load following plant and therefore; transients 3 and 4 could not credibly approach the number of design cycles during the period of extended operation." Therefore, the 60-year projected cycles are set to the design cycles.

(4) The fatigue flaw growth analysis is discussed in Sections 3.1.4, 3.3.2 and 4.3 of BAW-1847, Rev. 1.

The fatigue flaw growth analysis was performed based on linear elastic fracture mechanics and uses the Paris equation provided in Appendix A to Section XI of the ASME code. This method is applicable to surface flaws that have not fully penetrated the wall.

The fatigue flaw growth analysis was based on the input stresses and number of cycles, which were based on ASME Code transients and Safe Shutdown Earthquake loads.

Results of the fatigue flaw growth analysis (given in Table 4-2 of BAW-1847, Rev. 1) show the minimum surface flaw depths (semi-elliptical shape) that will grow through the pipe wall thus creating a through-wall flaw. For the 28-inch ID pipe, any initial circumferential flaw depth less than six-tenths of the wall thickness (0.6 t) will not extend through-wall during the design life. Likewise, any initial longitudinal flaw depth less than four-tenths of the wall thickness (0.4 t) will not extend through-wall during the design life.

For LBB, it was important to determine if an initial surface flaw will grow in the radial direction to become a through-wall flaw instead of growing in the circumferential (longitudinal) direction. It was shown using the BIGIF code with the variable aspect ratio option, that the preferential flaw growth is in the through-wall direction of the pipe, thereby causing a detectable leak to occur prior to significant circumferential extension.

(5) The hot leg (36") and cold leg (28") piping of the reactor coolant system contains the postulated fatigue flaws.

(6) In the Leak-Before-Break (LBB) evaluation [Reference 1], the design cycles analyzed for LBB fatigue flaw growth analysis were based on 40 years of operation for all B&W designed plants. The design cycles were originally intended to conservatively bound 40 years of plant life. As shown in the tables provided in item (3) to this response, the analyzed cycles used in the LBB fatigue flaw growth analysis bound the 60 year projected cycles.

Reference for this response:

1. BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985 (ADAMS Accession Number ML8511180489)

Question RAI 4.7.1.1-2

Section 4.7.1.1 states that " ... [t]he updated analysis used 1.5 times the design cycles for the reactor coolant pump suction and discharge weld overlays ... "

(1) Submit the updated analysis.

(2) Identify the design cycles and associated cycle numbers.

(3) Discuss why a multiple of 1.5 times the design cycles is adequate for the period of extended operation.

(4) Discuss how many calendar years the cycles used in the updated analysis (ie.,1.5 times the design cycles) will cover.

RESPONSE RAI 4.7.1.1-2

(1) Application of weld overlays on the reactor coolant pumps (RCPs) suction and discharge nozzle dissimilar metal welds required an update to the Davis-Besse LBB evaluation. In Section 4.7.1.1, this evaluation is referred to as the updated analysis. The updated LBB evaluation was submitted by FirstEnergy Nuclear Operating Company (FENOC) under letter L-09-227 [Reference 1]. NRC approved the updated LBB analysis on March 24, 2010, via issuance of Amendment No. 281 (TAC No. ME2310) [Reference 2].

(2) As part of the updated LBB evaluation, specific fatigue crack growth (FCG) analyses were performed for the Alloy 82/182 RCP nozzles dissimilar metal welds to demonstrate that the post weld overlay crack growth is very minimal for balance of plant life. The transients and associated cycles for the FCG analyses are provided in the table as follows:

From LRA Table 4.3-1				FCG Analyses
Transient Number	Design Cycles	60-Year Projected Cycles	Transient	Analyzed Cycles (Suction/Discharge)
1A	240	128	HU/CD	360/360
1B	240	128	HU/CD	360/360
8A	40	4	Rx Trip from Low RC Flow	885/192
8C	88	26	Rx Trip from High Pressure due to loss of Feedwater	225/282
10	20	10	Loss of RCP without Rx Trip	31590/31590
12A	15 + 5 shop test	4	Hydro-test	30/30

(3) In the fatigue crack growth analyses, design cycles multiplied by a factor of 1.5 were used to conservatively define cycles for 60 years of operation. As shown in the above table, the number of analyzed cycles bound the 60-year projected cycles.

(4) As shown in the above table, the design cycles bound the 60-year projected cycles. Therefore, design cycles multiplied by a factor of 1.5 would also bound the 60-year projected cycles. The analyzed cycles used in the fatigue crack growth analyses would remain valid for at least 90 years of operation.

References for this response:

1. FENOC letter L-09-227, Barry S. Allen to USNRC Document Control Desk, "License Amendment Request to Update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds," 28 September 2009
(ADAMS Accession Number ML092790438)

2. NRC letter from Michael Mahoney to Barry S. Allen, "Davis-Besse Nuclear Power Station, Unit 1 - Issuance of Amendment RE: Application to update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds (TAC No. ME2310)," 24 March 2010
(ADAMS Accession Number ML100640506)

Question RAI 4.7.1.1-3

Section 4.7.1.1 mentions three analyses: the updated analysis, the fatigue flaw growth analysis, and the LBB analysis.

It is not clear how these three analyses are related to each other and to the flaw growth calculation due to fatigue.

Explain the differences among the three analyses in terms of calculating the flaw growth due to fatigue.

RESPONSE RAI 4.7.1.1-3

The Leak-Before-Break (LBB) evaluation for the Davis-Besse RCS primary piping is contained in topical report BAW-1847, Revision 1 [Reference 1]. The LBB evaluation included fatigue flaw growth analyses, flaw stability analyses, and limit load analyses. The fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe.

Application of weld overlays on the reactor coolant pumps (RCPs) suction and discharge nozzle dissimilar metal welds required an update to the Davis-Besse LBB evaluation. In Section 4.7.1.1, this evaluation is referred to as the updated analysis. The updated LBB evaluation was submitted by FirstEnergy Nuclear Operating Company (FENOC) under letter L-09-227 [Reference 2]. The NRC approved the updated LBB evaluation on March 24, 2010, via issuance of Amendment No. 281 (TAC No. ME2310) [Reference 3]. As part of the updated LBB evaluation, specific fatigue crack growth analyses were performed for the Alloy 82/182 RCP nozzles DMW welds to demonstrate that the post weld overlay crack growth is very minimal for balance of plant life. In the fatigue crack growth analyses, plant design transients multiplied by a factor of 1.5 were used to conservatively define cycles for 60 years of operation.

References for this response:

1. BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985 (ADAMS Accession Number ML8511180489)

2. FENOC letter L-09-227, Barry S. Allen to USNRC Document Control Desk, "License Amendment Request to Update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds," 28 September 2009 (ADAMS Accession Number ML092790438)

3. NRC letter from Michael Mahoney to Barry S. Allen, "Davis-Besse Nuclear Power Station, Unit 1 - Issuance of Amendment RE: Application to update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds (TAC No. ME2310)," 24 March 2010 (ADAMS Accession Number ML100640506)

Question RAI 4.7.1.1-4

Section 4.7.1.1 states that "... [t]he effects of fatigue flaw growth on piping approved for LBB will be managed by the Fatigue Monitoring Program for the period of extended operation ... " Based on the fatigue monitoring program described in Appendix B to the license renewal application (LRA), the applicant would monitor the actual transient cycles.

(1) Describe the processes and/or procedures of how the actual transient cycles are monitored and how they are compared to the cycles used in the updated LBB analysis.

(2) Describe how and when the corrective actions will be implemented if the actual transient cycles exceed the transient cycles used in the updated LBB analysis.

RESPONSE RAI 4.7.1.1-4

FENOC has elected to disposition the fatigue flaw growth analysis in accordance with 10 CFR 54.21(c)(1)(i) and therefore, will not credit the Fatigue Monitoring Program for managing the effects of fatigue flaw growth on piping approved for leak-before-break (LBB).

As provided in the response to RAI 4.7.1.1-1, FENOC performed a comparison of the design cycles that were used in the LBB fatigue flaw growth evaluation provided in BAW-1847, Rev.1 to the Davis-Besse 60-year projected cycles and determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the LBB fatigue flaw growth evaluation remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

As provided in the response to RAI 4.7.1.1-2, FENOC performed a comparison of the design cycles (multiplied by a factor of 1.5) that were used in the fatigue crack growth analyses for the Alloy 82/182 RCP nozzles dissimilar metal welds to the 60-year projected cycles and determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the flaw crack growth analyses, associated with the RCP nozzles dissimilar metal welds, remain valid for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(i).

Based on the responses provided in RAI 4.7.1.1-1 and RAI 4.7.1.1-2, the LBB evaluations will be dispositioned using 10 CFR 54.21(c)(1)(i).

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

Section 4.7.1.2 – Thermal Aging

Question RAI 4.7.1.2-1

Under the Disposition heading, Section 4.7.1.2 states that the effects of thermal aging on cast austenitic stainless steel (CASS) components in the approved LBB piping is not a time limited aging analysis (TLAA) because the effects of thermal aging will be managed by the inservice inspection program for the period of extended operation. The staff believes that if an aging effect is monitored by an inspection program, 10 CFR 54.21 (c)(1)(iii) may be applicable.

Explain why thermal aging of CASS component is not a TLAA if it is monitored by the inservice inspection program. The outcome of this issue may affect the conclusion in Section A.2.7.1 of the LRA.

RESPONSE RAI 4.7.1.2-1

Thermal aging of cast austenitic stainless steel (CASS) components (reactor coolant pump casings including the pump suction and discharge nozzles) in the approved leak-before-break (LBB) piping is not a time-limited aging analysis (TLAA) because saturated embrittlement (lowest and worst-case fracture toughness) was used in the updated analysis, and 10 CFR 54.21(c)(1)(iii) is not applicable.

As stated in LRA Section 4.7.1.2, "The updated LBB analysis was based on saturated embrittlement of the CASS casings such that there is no embrittlement TLAA." As further detailed in response to RAI 4.7.1.2-4, this saturated embrittlement represents the lowest and worst-case fracture toughness for the CASS reactor coolant pump (RCP) casing. Therefore, thermal aging of CASS components is not a TLAA, but is an aging effect that will be managed by the Inservice Inspection Program.

As provided in LRA Table 3.1.1, item 3.1.1-55, Inservice Inspection program is credited with management of thermal aging for cast austenitic stainless steel (CASS) components. This is consistent with NUREG-1800 Table 3.1-1 that states, "Inservice inspection (IWB, IWC, and IWD). Thermal aging susceptibility screening is not necessary, inservice inspection requirements are sufficient for managing these aging effects. ASME Code Case N-481 also provides an alternative for pump casings."

In preparing the response to RAI 4.7.1.2-1, it was noted that the two sentences related to weld overlays should be deleted from Section 4.7.1.2 and the associated TLAA summary in Section A.2.7.1.

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

Question RAI 4.7.1.2-2

The ultrasonic examination technique has not yet been qualified by the American Society of Mechanical Engineers (ASME) to detect flaws in CASS material.

Discuss the ASME-qualified inservice inspection method(s) that will be used to inspect the CASS components to monitor their thermal aging effects and discuss the associated inspection frequency (intervals).

RESPONSE RAI 4.7.1.2-2

The primary inspection of cast austenitic stainless steel (CASS) components (i.e. valve bodies and pump casings) is external visual examination. Internal visual inspections and volumetric inspections are performed only when a valve/pump is disassembled for maintenance. Details of the inspections and their frequencies are provided below.

As stated in NUREG-1801, Section XI.M12,

> "For pump casings and valve bodies, based on the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI), screening for susceptibility to thermal aging is not required. The existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies."

Pump Casings:

ASME Boiler and Pressure Vessel Code, Section XI, inservice inspection requirements of pressure retaining welds of pump casings (Category B-L-l) are delineated in Table IWB-2500-1 of the Code. Examination category B-L-1 requires volumetric examination on pump casing welds. Davis-Besse uses Code Case N-481 in lieu of the examination requirements of this Code Category. These alternate requirements consist of visual inspections and an analytical evaluation to demonstrate the safety and serviceability of the pump casings in the presence of an assumed flaw. Each of the four RCP casings is visually examined every 10-year inservice inspection (ISI) interval. The Davis-Besse evaluation for implementation of Code Case N-481 was submitted to the NRC in 2001 [Reference 1].

Valve bodies:

There are no Code Category B-M-1 welds in valves installed at Davis-Besse.

There are a total of ten Code Category B-M-2 valves in four groups. One valve per group will be examined over the ISI interval when disassembled for routine maintenance, repair, or volumetric examination. Per ASME Section XI, Table IWB-2500, B.12.30, the inspection of Code Category B-M-2 valves less than 4-inch NPS is limited to surface examination. Per ASME Section XI, Table IWB-2500, B.12.40 and B.12.50, the inspection of valves greater than 4-inch NPS includes volumetric and visual (VT-3) examination. However, the inspection is limited to an external visual inspection (VT-3) unless the valve is opened for maintenance and repair.

Reference for this response:

1. Letter Serial 2700, G.G. Campbell to USNRC Document Control Desk, "Submittal of ASME Code Case N-481 Evaluation for the Davis-Besse Nuclear Power Station Reactor Coolant Pumps," 23 April 2001
(ADAMS Accession Number ML011200090)

Question RAI 4.7.1.2-3

It appears that Davis Besse LRA does not include an aging management program to monitor thermal aging embrittlement of CASS. GALL AMP XI.M12 provides guidance on such an aging management program and many license renewal applications have implemented such an aging management program.

Explain why this program has not been proposed to be implemented at Davis Besse for the purpose of license renewal in light of the fact that the reactor coolant pump casing is made of CASS which is susceptible to thermal aging degradation.

RESPONSE RAI 4.7.1.2-3

The LRA does not include a Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) program similar to XI.M12. Davis-Besse has no CASS components other than pump casings and valve bodies subject to thermal embrittlement. As reduction of fracture toughness of these component types is managed by the Inservice Inspection Program (See LRA Section B.2.24), a program similar to XI.M12 is not required.

NUREG-1801 Aging Management Program XI.M12 applies to CASS pressure boundary components that are susceptible to thermal embrittlement. The second paragraph of the Program Description in NUREG-1801, Section XI.M12 specifically exempts pump casings and valve bodies from this program and states "For pump casings and valve bodies, based on the assessment documented in the letter dated May 19, 2000 from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI), screening for susceptibility to thermal aging embrittlement is not required. The existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies." This position is re-iterated in NUREG-1801 line item IV.C2-6, "For pump casings and valve bodies, screening for susceptibility to thermal aging is not necessary. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings and valve bodies. Alternatively, the requirements of ASME Code Case N-481 for pump casings are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings."

A review of LRA Table 3.1.2-3 for the reactor coolant pressure boundary finds only three CASS component types managed for reduction of fracture toughness: reactor coolant pump casings (row 196), small bore valve bodies (row 234), and large bore valve bodies (row 255). As there are no piping, piping components or piping elements, other than pump casings and valve bodies, GALL AMP XI.M12 is not required.

Question RAI 4.7.1.2-4

Section 4.7.1.2 states that "... The updated LBB analysis was based on saturated embrittlement of the cast austenitic stainless steel (CASS) casings such that there is no embrittlement TLAA .... "

(1) Submit the updated LBB analysis. If the updated LBB analysis was previously submitted, reference the date of the submittal and identify the pages that discuss the saturated embrittlement.

(2) Demonstrate that the value of fracture toughness used in the updated LBB analysis represents the lowest and worst-case fracture toughness value for the reactor coolant pump casing.

RESPONSE RAI 4.7.1.2-4

The updated Leak-Before-Break (LBB) analysis was based on saturated embrittlement of the cast austenitic stainless steel (CASS) casings such that there is no embrittlement time-limited aging analyses (TLAA).

(1) The updated LBB analysis was submitted by First Energy Nuclear Operating Company (FENOC) under Letter L-09-227 [Reference 1]. Included as Enclosure B to that letter was Structural Integrity Associates (SIA) Report 0800368.404, Revision 1, "Leak-Before-Break Evaluation of Reactor Coolant Pump Suction and Discharge Nozzle Weld Overlays for Davis-Besse Nuclear Power Station."

(2) Saturated embrittlement is discussed in Section 4.3.3 of the SIA Report on pages 4-3 and 4-4. As provided in the report, a previous Code Case N-481 evaluation [Reference 2] for the Davis-Besse reactor coolant pumps used actual material properties to develop lower bound J-R curves and J_{Ic}/K_{Ic} for the RCP pump CASS material heats with consideration of thermal embrittlement. The NRC has approved the updated LBB analysis through Amendment No. 281 to Davis-Besse's Operating License by NRC letter dated March 24, 2010 [Reference 3]. In Section 3.3.3 (page 8) of the safety evaluation, the NRC concluded that the licensee used the lower bound fracture toughness of the worst pump casing material heats considering thermal embrittlement.

References for this response:

1. FENOC letter L-09-227, Barry S. Allen to USNRC Document Control Desk, "License Amendment Request to Update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds," 28 September 2009
(ADAMS Accession Number ML092790438)

2. FENOC letter Serial Number 2700, Guy G. Campbell to USNRC Document Control Desk, "Submittal of ASME Code Case N-481 Evaluation for the Davis-Besse Nuclear Power Station Reactor Coolant Pumps," 23 April 2001
(ADAMS Accession Number ML011200090)

3. NRC letter from Michael Mahoney to Barry S. Allen, "Davis-Besse Nuclear Power Station, Unit 1 - Issuance of Amendment RE: Application to update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds (TAC No. ME2310)," 24 March 2010 (ADAMS Accession Number ML100640506)

<u>Section 4.7.1.3 – Primary Water Stress Corrosion Cracking (PWSCC)</u>

Question RAI 4.7.1.3-1

Section 4.7.1.3 discusses the weld overlays to mitigate PWSCC in piping approved for LBB. Section 4.7.1.3 also states that PWSCC (or the weld overlay design) is not a TLAA. The staff disagrees with this assessment.

The staff believes that the weld overlay design is a TLAA. The design of weld overlays requires a fatigue flaw growth calculation based on a postulated or an actual detected flaw. The fatigue flaw growth calculation uses transient cycles which are time dependent. Therefore, the staff believes that the weld overlay is a TLAA.

Clarify why the subject matter in Section 4.7.1.3 is not a TLAA.

RESPONSE RAI 4.7.1.3-1

FENOC concurs with the staff that the design of the subject weld overlays is a TLAA.

In the response to RAI 4.7.1.1-4, FENOC addressed the fatigue flaw growth TLAA for the Alloy 82/182 RCP nozzles dissimilar metal welds and provided a revision to the LRA to include the subject TLAA in Section 4.7.1.1.

Therefore, Section 4.7.1.3 is deleted.

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

Section 4.7.2 – Metal Corrosion Allowance for Pressurizer Instrument Nozzles

Question RAI 4.7.2-1

Section 4.7.2 states that " ... [t]his resulted in an increase of the general corrosion rate of the pressurizer shell base metal in the nozzle bores from zero to 1.42 thousandths of an inch (mils) per year ... The allowable radial corrosion limit, calculated per ASME Section III, is 293 mils for the level instrument nozzles, 493 mils for the sample nozzle and 495 mils for the vent and thermowell nozzles ... "

(1) Discuss in detail how 1.42 thousandths of an inch, 293 mils, 493 mils, and 495 mils were obtained or submit the calculations.

(2) Discuss whether the general corrosion rate of 1.42 thousandths of an inch per year has been verified to be adequate for use for the period of extended operation. Discuss whether this corrosion rate increases as the component ages. As it is applied, this corrosion rate is assumed to be constant throughout the remaining life of the plant.

RESPONSE RAI 4.7.2-1

1. General corrosion rate of 1.42 thousandths of an inch (mils) per year was developed in Structural Integrity Associates, Inc. Report SIR-07-188-NPS, "Evaluation of the Corrosion of Carbon Steel and Low Alloy Steel in Portions of Pressurizer Vessels Exposed to Primary Water Following Repair of Small Bore Instrument Nozzles," dated November 2007.

   All of the repairs of small bore instrument nozzles or other penetrations that expose carbon steel or low alloy steel to primary coolant by "uncovering" some carbon steel or low alloy steel vessel material will expose that steel to borated water of nominal boron concentrations under immersion conditions rather than conditions of dripping, evaporation, etc. that occur for borated water that leaks from the pressure boundary. The corrosion rate methodology described in Report SIR-07-188-NPS used the corrosion rates reported from the literature for such worst case full immersion conditions compared to steam at high temperature, low temperature, and very low oxygen environments (e.g., normal operation) or higher oxygen environments that may occur during refueling. The overall metal loss is the sum of the products of the time at given conditions and the corrosion rates for each of those environments. Assuming that Davis-Besse operates 85% of the year at high temperatures, spends 10% of the year under shutdown conditions and 5% of the year at intermediate temperatures, the total general corrosion rate (GCR), actually an average annualized metal loss rate, would be the following:

   GCR = 0.85 x 0.6 (500°F) + 0.1 x 8 (100°F) + 0.05 x 2.2 (300°F) = 1.42 mils per year

   The allowable radial corrosion limits provided in LRA Section 4.7.2 were developed in Structural Integrity Associates, Inc. Calculation Package DB-09Q-303, "Determination of Allowable Corrosion of Pressurizer Vessel Shell," Rev. 1, dated

November 17, 2007. In this calculation, the maximum allowable corrosion of the pressurizer material in the penetration bore was quantified by determining the corroded radius such that resulting stresses due to primary loads in the repair pad still meet ASME Code, Section III Design Conditions allowable values. General primary membrane and primary membrane-plus-bending stress intensity values due to pressure and mechanical loads (where applicable) were determined and compared to ASME Code allowable values using the maximum corroded bore radius that is possible.

2. For carbon steel and low alloy steel, oxidation involves the oxidation of metallic iron to ferrous ion (Fe^{+2}, a soluble ionic species) or ferric ion (Fe^{+3}, an insoluble ion). Corrosion products provide a level of protection against continuing corrosion. Therefore, the general corrosion rate of 1.42 mils per year is an average annualized metal loss rate and is assumed to be constant throughout the remaining life of the plant.

Enclosure

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

Letter L-11-115

Amendment No. 4 to the DBNPS License Renewal Application

License Renewal Application Sections Affected

Section 4.7.1

Section 4.7.1.1

Section 4.7.1.2

Section 4.7.1.3

Section A.2.7.1

Table 4.1-1

Table of Contents

The Enclosure identifies the changes 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 *underlined*.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
4.7.1.1	**Page 4.7-1**	**Entire section**

Section 4.7.1.1 is replaced in its entirety to read:

The LBB analysis postulated surface flaws at the piping system locations with the highest stress coincident with the lower bound of the material properties for base metal and welds. The fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions. The fatigue flaw growth analysis used plant design transients.

Application of weld overlays on the reactor coolant pumps (RCPs) suction and discharge nozzle dissimilar metal welds (DMW) required an update to the Davis-Besse LBB evaluation. As part of the updated LBB evaluation, specific fatigue crack growth analyses were performed for the Alloy 82/182 RCP nozzles DMW to demonstrate that the post weld overlay crack growth is very minimal for balance of plant life. In the fatigue crack growth analyses, plant design transients multiplied by a factor of 1.5 were used to conservatively define cycles for 60 years of operation.

FENOC performed a comparison of the design cycles that were used in the LBB fatigue flaw growth evaluation provided in BAW-1847, Rev.1 to the 60-year projected cycles provided in LRA Table 4.3-1 and determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the LBB fatigue flaw growth evaluation remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

FENOC performed a comparison of the design cycles (multiplied by a factor of 1.5) that were used in the fatigue crack growth analyses for the Alloy 82/182 RCP nozzles dissimilar metal welds to the 60-year projected cycles provided in LRA Table 4.3-1 and determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the flaw crack growth analyses, associated with the RCP nozzles dissimilar metal welds, remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Disposition: 10 CFR 54.21(c)(1)(i)	*The LBB fatigue flaw growth analyses remain valid for the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.7.1	**Page A-47**	**"Fatigue Flaw Growth" section**

The Fatigue Flaw Growth subsection of Section A.2.7-1 is replaced in its entirety to read:

The LBB analysis postulated surface flaws at the piping system locations with the highest stress coincident with the lower bound of material properties for the base metal and welds. The fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions. The fatigue flaw growth analysis used plant design transients.

Application of weld overlays on the reactor coolant pumps (RCPs) suction and discharge nozzle dissimilar metal welds (DMW) required an update to the Davis-Besse LBB evaluation. As part of the updated LBB evaluation, specific fatigue crack growth analyses were performed for the Alloy 82/182 RCP nozzles DMW to demonstrate that the post weld overlay crack growth is very minimal for balance of plant life. In the fatigue crack growth analyses, plant design transients multiplied by a factor of 1.5 were used to conservatively define cycles for 60 years of operation.

The design cycle assumption used in these fatigue flaw growth analyses bound 60-year projected cycles.

Therefore, the LBB fatigue flaw growth analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

<u>**Affected LRA Section**</u>	<u>**LRA Page No.**</u>	<u>**Affected Paragraph and Sentence**</u>
Table 4.1-1	**Page 4.1-4**	**"Leak-Before-Break" row**

The "Leak-Before-Break" row, under the "Other Plant-Specific Time-Limited Aging Analyses" section of Table 4.1-1, "Time-Limited Aging Analyses" is revised as follows:

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Leak-Before-Break	~~*(iii)*~~ <u>*(i)*</u>	4.7.1

<u>**Affected LRA Section**</u>	<u>**LRA Page No.**</u>	<u>**Affected Paragraph and Sentence**</u>
Table of Contents	**Page xii**	**"Fatigue Flaw Growth" section**

The Table of Contents is revised as follows:

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.7.1.2	**Page 4.7-2**	**Third paragraph**

The third paragraph of Section 4.7.1.2 is revised as follows:

An aging management review of the Reactor Coolant System, including the reactor coolant pumps, has been performed for license renewal (see Section 3.1). Reduction of fracture toughness due to thermal embrittlement of CASS components is an aging effect requiring management for the reactor coolant pump casings and is managed by the Inservice Inspection Program. ~~*The acceptability of a 10-year inspection interval for these weld overlays was demonstrated in the updated LBB analysis. This analysis does not justify operation of the weld overlays for the life of the plant, but for the 10 years between inspections.*~~ *Therefore, the effects of thermal aging on CASS components in the approved LBB piping, will be managed by the Inservice Inspection Program for the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.7.1	**Page A-47**	**Thermal Aging subsection, third paragraph**

The third paragraph of the "Thermal Aging" subsection of Section A.2.7.1 is revised as follows:

Aging management review of the RCS determined reduction of fracture toughness due to thermal embrittlement of CASS components to be an aging effect requiring management for the reactor coolant pump casings. ~~*The acceptability of a 10-year inspection interval for these weld overlays was demonstrated in the updated LBB. This analysis does not justify operation of the weld overlays for the life of the plant, but for the 10 years between inspections.*~~ *Therefore, the effects of thermal aging on CASS components in the approved LBB piping will be managed by the Inservice Inspection Program for the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.7.1	**Page 4.7-1**	**Fifth paragraph, second sentence**

The second sentence of the fifth paragraph of the of Section 4.7.1 is revised as follows:

The time-limited aspects of fatigue flaw growth~~,~~ and thermal aging ~~and PWSCC~~ are addressed separately in the subsections below.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.7.1.3	**Page 4.7-3**	**Entire section**

Section 4.7.1.3 is deleted in its entirety:

~~*4.7.1.3 Primary Water Stress Corrosion Cracking*~~

~~*FENOC received relief to install weld overlays on certain Alloy 600 components and Alloy 82/182 dissimilar metal welds for mitigation of PWSCC. As presented in Section 4.7.1, this relief included Alloy 82/182 dissimilar metal welds that are located in piping approved for LBB. FENOC updated the original leak-before-break calculations for Davis-Besse with an evaluation demonstrating that the weld overlays resolve the concerns for original welds susceptibility to primary water stress corrosion cracking. Critical crack sizes and leakage rates with the weld overlay in place were evaluated to demonstrate that margins exist for detection of leakage, i.e., the conclusions of the existing leak-before-break analysis remain valid.*~~

~~*For license renewal, an aging management review of the Reactor Coolant System, including the nickel-alloy weld locations, has been performed (see Section 3.1). Cracking due to PWSCC is an aging effect requiring management for the period of extended operation and is managed by the Inservice Inspection Program and Nickel-Alloy Management Program.*~~

~~*Disposition:*~~	~~*Not a TLAA.*~~	~~*The effects of PWSCC on the Reactor Coolant System piping will be managed by the Inservice Inspection Program and Nickel-Alloy Management Program for the period of extended operation*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.7.1	**Page A-48**	**PWSCC subsection**

The PWSCC subsection of Section A.2.7.1 is deleted in its entirety:

~~*PWSCC*~~

~~*FENOC received relief to install weld overlays on certain Alloy 600 components and Alloy 82/182 dissimilar metal welds for mitigation of PWSCC, including Alloy 82/182 welds in piping approved for LBB. FENOC updated the original leak-before-break calculations for Davis-Besse with an evaluation demonstrating that the weld overlays resolve the concerns for original welds susceptibility to primary water stress corrosion cracking. Critical crack sizes and leakage rates with the weld overlay in place were evaluated to demonstrate that margins exist for detection of leakage, i.e., the conclusions of the existing leak-before-break analysis remain valid.*~~

~~*Aging management review of the RCS, including the nickel alloy weld locations, identified cracking due to PWSCC as an aging effect requiring management for the period of extended operation. Cracking due to PWSCC is managed by the Inservice Inspection Program and the Nickel-Alloy Management Program.*~~

~~*The analyses associated with the effects of PWSCC of Alloy 600/82/182 materials on the LBB analysis are not a TLAA.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table of Contents	**xii**	**4.7.1.3**

The Table of Contents entry for Section 4.7.1.3 is deleted in its entirety: