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September 26, 2016
L-16-279

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 Related to License Renewal Commitment 42 (CAC MF7626)

By letter dated April 21, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16112A079), FirstEnergy Nuclear Operating Company (FENOC) submitted a Fatigue Monitoring Program evaluation for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), to address License Renewal Commitment 42.

By email dated August 26, 2016, the NRC staff requested additional information on the Fatigue Monitoring Program evaluation to complete its review.

The Attachment provides the FENOC reply to the NRC request for additional information.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Patrick McCloskey, Manager – Regulatory Compliance, at (419) 321-7274.

Sincerely,

Brian D. Boles

Attachment:

Reply to Request for Additional Information Related to Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), License Renewal Commitment 42

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

By letter dated April 21, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16112A079), FirstEnergy Nuclear Operating Company (FENOC) submitted a Fatigue Monitoring Program evaluation for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), to address License Renewal Commitment 42.

By an email dated August 26, 2016, the NRC staff requested additional information on the Fatigue Monitoring Program evaluation to complete its review. The requested information is provided below. The NRC staff request is shown in bold text, followed by the FENOC reply.

NRC STAFF RAI

Background

Section 2.C(11), "License Renewal Conditions," of Renewed Facility Operating License No. NPF-3 specifies that the Commitments in Appendix A of NUREG-2193, Supplement 1, "Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station," published April 2016 (ADAMS Accession No. ML16104A350), are part of the DBNPS Updated Final Safety Analysis Report. License renewal Commitment No. 42 in Appendix A of NUREG 2193, Supplement 1, states the following:

Enhance the Fatigue Monitoring Program to:

- ***Evaluate additional plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than those considered in NUREG/CR-6260[1]. This evaluation will include identification of the most limiting fatigue location exposed to reactor coolant for each material type (i.e., [carbon steel] CS, [low-alloy steel] LAS, [stainless steel] SS, and [nickel-based alloy] NBA) and that each bounding material/location will be evaluated for the effects of the reactor coolant environment on fatigue usage. Nickel-based alloy items will be evaluated using NUREG/CR-6909[2]. Submit the evaluation to the NRC 1 year prior to the period of extended operation.***

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[1] **NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, dated February 1995, ADAMS Accession No. ML031480219.**
[2] **NUREG/CR-6909 "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," dated February 2007.**

Enclosure B of the licensee's letter dated June 17, 2011 (ADAMS Accession No. ML11172A389), provides AREVA Report No. 51-9157140-001. Table 3-9 of the AREVA Report contains the environmentally-assisted fatigue (EAF) values for the NUREG/CR-6260 locations. Table 3-8 of the AREVA Report contains a summary of the reactor coolant system pressure boundary locations with environmentally-adjusted cumulative usage factor (CUF_{en}) values that exceed the limit of 1.0.

In its April 21, 2016, letter, the licensee submitted the results of its evaluations associated with Commitment No. 42. The letter stated that locations were screened in accordance with the methodology of Electric Power Research Institute (EPRI) Technical Report 1024995, "Environmentally Assisted Fatigue Screening[:] Process and Technical Basis for Identifying EAF Limiting Locations," dated 2012. The letter also identified the most limiting locations for each of the four material types (i.e., CS, LAS, SS, and NBA). The CUF_{en} values are provided for two of the four limiting locations. The CUF_{en} values are not provided for the LAS and NBA locations. The LAS and NBA locations reference EPRI Technical Report 1024995.

<u>Issue</u>

The April 21, 2016, letter, describes the results of the licensee's evaluation and references the generic methodologies (e.g., EPRI Technical Report 1024995) used, but it does not provide sufficient details about the actual evaluation. The plant-specific methodology and criteria used to select the most limiting locations for EAF is not clearly described. The letter does not explain how the plant-specific screening methodology conservatively evaluates EAF effects, with the same degree of analytical rigor for all locations, to identify the bounding locations. Additionally, the licensee uses different material types to bound limiting locations for LAS and NBA without justification.

EPRI Technical Report 1024995 has not been submitted to the NRC for approval and has not been endorsed by the NRC. The licensee does not explain how the plant-specific implementation of the generic procedures in EPRI Technical Report 1024995 will identify the most limiting plant-specific locations. The NRC staff lacks sufficient information to evaluate the plant-specific methodology and criteria used to select the most limiting locations for EAF.

Request

(1) Describe the plant-specific methodology and criteria used to rank locations and select the most limiting locations for EAF. Describe relevant factors for each step of the process, such as thermal zones, material types, transient complexity, temperature effects, and complexity of the systems (as applicable). Justify the use of different material types to bound locations. Justify that the process is appropriately conservative.

(2) Describe and justify any engineering judgement, plant-specific assumptions, and plant-specific criteria used in the EAF analysis or screening process. This should include the systematic process used to eliminate locations as limiting and examples showing how the process was implemented.

(3) Describe how the screening process was applied to the locations in Table 3-8 of the AREVA Report No. 51-9157140-001.

(4) State the locations being managed by the fatigue monitoring program to maintain the CUF_{en} values below the limit of 1.0 through the period of extended operation. Provide the CUF_{en} values for the locations being managed by the fatigue monitoring program.

FENOC REPLY TO RAI

(1) Describe the plant-specific methodology and criteria used to rank locations and select the most limiting locations for EAF. Describe relevant factors for each step of the process, such as thermal zones, material types, transient complexity, temperature effects, and complexity of the systems (as applicable). Justify the use of different material types to bound locations. Justify that the process is appropriately conservative.

To reduce the size of the set, a location was eliminated if it met one of the following prescreening criteria:

- The location has a usage factor that is so low that, even if the maximum possible environmental fatigue correction factor (F_{en}) were applied, the resulting EAF usage factor (U_{en}) would be less than 0.8, which is the example EAF screening limit provided in EPRI Technical Report 1024995.
- The location is not exposed to reactor water, and therefore EAF is not applicable.
- The location is not part of the primary pressure boundary and is therefore excluded from the EPRI screening process.

Sentinel locations are determined based on the locations remaining after prescreening. Sentinel locations are those locations chosen for more detailed analysis or monitoring. These locations are chosen to have bounding U_{en}* (estimated U_{en}) compared with other locations. Estimated U_{en} is determined using an estimated F_{en}. Estimated F_{en} (F_{en}*) is calculated as the average of the value based on a qualitative estimate of strain rate, and the value based on the worst possible strain rate, using the same values of dissolved oxygen and estimated upper bound temperature for design transients in both cases.

Based on the locations that pass prescreening, each location is conservatively assumed to exist in its own thermal zone. Therefore, the rules in the EPRI screening process for reducing the number of sentinel locations across thermal zones are used as follows:

- One thermal zone (that is, location) can bound another thermal zone (location) in a system if the cumulative usage factor (CUF) and F_{en} values for one sentinel location in one thermal zone are each higher than those for the sentinel locations in other thermal zones, and the U_{en} is more than twice those in the other zones.

- One material in a thermal zone (location) can bound other materials in the same thermal zone (location) if the CUF and F_{en} values for one sentinel location composed of one material are each higher than the CUF and F_{en} values for the sentinel locations composed of all other materials, and the U_{en} is more than twice those for the other materials.

- One material in a thermal zone can bound other materials in another thermal zone if both of the preceding two criteria are true.

The limiting locations were then evaluated per NUREG/CR-6260, Section 4.3, "Potential Adjustments to Licensees' Calculations that Might Reduce the CUF," to determine sources of conservatism in the existing fatigue usage calculations.

As described in FENOC letter L-11-203 dated June 17, 2011 (ADAMS Accession No. ML11172A389), the Reactor Coolant Pump Bearing Cavity, Pressurizer Heater Bundle Diaphragm Plate and Diaphragm Plate Seal Weld each had EAF values greater than 1.0, requiring a detailed analysis. Detailed analyses were completed for those components with EAF values greater than 1.0, and the results were provided in FENOC letter L-16-148 dated April 21, 2016 (ADAMS Accession No. ML16112A079).

(2) Describe and justify any engineering judgement, plant-specific assumptions, and plant-specific criteria used in the EAF analysis or screening process. This should include the systematic process used to eliminate locations as limiting and examples showing how the process was implemented.

Assumptions used in the evaluation:

- The initial screening applied a conservative F_{en} multiplier to each location. This approach is conservative since the F_{en} factor considered the worst strain rate of 0.0004%/sec and dissolved oxygen level of 0.05 ppm, which results in higher (conservative) F_{en} values.

- Each location that initially screens in is conservatively assumed to exist in a separate thermal zone.

The screening process is conservative since the original usage factors are determined based on the design transient cycles. The design transient cycles are bounded by the 60-year projected cycles (reference FENOC letter L-11-166 (ADAMS Accession No. ML11159A132), Enclosure Table 4.3-1, "60-Year Projected Cycles"), which are currently tracked under the Davis-Besse Allowable Operating Transient Cycles (AOTC) Program.

No credit is taken for components that were recently replaced to screen out limiting locations. Recently replaced components include those associated with the reactor vessel closure head and control rod drive mechanisms (2011), steam generators (2014) and reactor coolant system hot leg piping between the flow meter and the steam generators (2014).

The screening of initial sentinel locations is shown in Table 1, "Screening of Initial Sentinel Locations."

Table 1: Screening of Initial Sentinel Locations

System	Location	Material Type	F_{en}*	U	U_{en}*	F_{en} smaller?	U smaller?	U_{en} < half?	Not 6260?	Bounded?
RC	RV continuous vent nozzle, J-groove weld	Ni-Cr-Fe	4.784	0.90	4.31	TRUE	TRUE	TRUE	TRUE	TRUE
	RC pump cover, cooling hole ligament (Note 1)	SS								
	RC pump cover, bearing cavity	SS	13.117	0.964	12.64	FALSE	FALSE	FALSE	TRUE	FALSE
	PZR spray nozzle, internal pipe	SS	9.013	0.33	2.97	TRUE	TRUE	TRUE	TRUE	TRUE
	PZR heater bundle closure, diaphragm plate	SS	11.486	0.6	6.89	TRUE	TRUE	FALSE	TRUE	FALSE
	PZR heater bundle closure, seal weld	SS	11.486	0.86	9.88	TRUE	TRUE	FALSE	TRUE	FALSE
	PZR manway closure, studs	LAS	2.455	0.35	0.86	TRUE	TRUE	TRUE	TRUE	TRUE
	Spray piping (Node 73)	SS	9.013	0.5184	4.67	TRUE	TRUE	TRUE	TRUE	TRUE
	Spray piping (Node 74)	SS	9.013	0.0951	0.86	TRUE	TRUE	TRUE	TRUE	TRUE
	Spray piping (Node 80)	SS	9.013	0.4124	3.72	TRUE	TRUE	TRUE	TRUE	TRUE
	Letdown piping (Node 859)	SS	13.117	0.604	7.92	FALSE	TRUE	FALSE	TRUE	FALSE
	RC drain nozzles	SS	15.348	0.132	2.03	FALSE	TRUE	TRUE	TRUE	FALSE
	Hot leg, new upper 180° elbow	CS	1.740	0.827	1.44	TRUE	TRUE	TRUE	TRUE	TRUE
	SG primary manways	LAS	2.455	0.34	0.83	TRUE	TRUE	TRUE	TRUE	TRUE
	SG primary manway seal weld (Note 2)									
	SG primary handhole cover	LAS	2.455	0.37	0.91	TRUE	TRUE	TRUE	TRUE	TRUE
	SG primary handhole seal weld (Note 2)									
	SG inlet nozzle primary head juncture	LAS	2.455	0.79	1.94	TRUE	TRUE	TRUE	TRUE	TRUE
	SG tubesheet, postulated thin ligament, primary side	LAS	2.455	0.39	0.96	TRUE	TRUE	TRUE	TRUE	TRUE
	SG lower spherical/flat head juncture	LAS	2.455	0.33	0.81	TRUE	TRUE	TRUE	TRUE	TRUE

Table 1: Screening of Initial Sentinel Locations (cont.)

System	Location	Material Type	F_{en}*	U	U_{en}*	F_{en} smaller?	U smaller?	U_{en} < half?	Not 6260?	Bounded?
RC	SG tubes	Ni-Cr-Fe	4.784	0.36	1.72	TRUE	TRUE	TRUE	TRUE	TRUE
	SG tube seal welds	Ni-Cr-Fe	4.784	0.42	2.01	TRUE	TRUE	TRUE	TRUE	TRUE
	SG tube plug seal welds	Ni-Cr-Fe	4.784	0.23	1.10	TRUE	TRUE	TRUE	TRUE	TRUE
DHR	Decay heat nozzle	CS	1.740	0.89	1.55	TRUE	FALSE	FALSE	TRUE	FALSE
	Decay heat containment isol. valves	SS	9.568	0.14594	1.40	TRUE	TRUE	FALSE	TRUE	FALSE
	Low pressure injection check valves	SS	9.568	0.14099	1.35	TRUE	TRUE	FALSE	TRUE	FALSE
	Low pressure injection containment isol. valves	SS	13.117	0.18261	2.40	FALSE	FALSE	FALSE	TRUE	FALSE
CF	Core flood piping (Node 490)	SS	9.013	0.582	5.25	FALSE	FALSE	FALSE	TRUE	FALSE

Note 1 - This location is exposed to component cooling water and not Reactor Coolant, therefore is not evaluated for EAF.
Note 2 - This cover is sealed using gaskets, and the seal weld was never made. Therefore, fatigue calculations are not needed.

RC = Reactor Coolant
DHR = Decay Heat Removal
CF = Core Flooding

(3) Describe how the screening process was applied to the locations in Table 3-8 of the AREVA Report No. 51-9157140-001.

The screening process described in response to Request 1 of this RAI was applied to all non-NUREG/CR-6260 reactor coolant pressure boundary components exposed to reactor coolant water listed in Tables 3-1 through 3-7 of AREVA Report No. 51-9157140-001[3] that are not exempt from fatigue, and not just those components previously identified in Table 3-8 with EAF CUF greater than 1.0. The locations evaluated are based on current plant design and analysis. The EAF associated with the reactor vessel closure head and control rod mechanisms (2011), steam generators (2014) and hot leg portion between the flow meter and the steam generators (2014) have changed due to replacements.

In addition to the components listed in AREVA Report No. 51-9157140-001, the following valves were also evaluated for EAF:

- core flood tank discharge check valves
- decay heat containment isolation valves
- decay heat bypass containment isolation valves
- low pressure injection check valves
- low pressure injection containment isolation valves

(4) State the locations being managed by the fatigue monitoring program to maintain the CUF_{en} values below the limit of 1.0 through the period of extended operation. Provide the CUF_{en} values for the locations being managed by the fatigue monitoring program.

The Davis-Besse Allowable Operating Transient Cycles (AOTC) Program tracks the actual number of plant transients. The program prevents the fatigue time limited aging analysis (TLAA) from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. The procedure uses the systematic counting of transient cycles and the evaluation of operating data to ensure that the allowable cycle limits are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded. Transient documentation is updated at least once per plant operating cycle.

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[3] AREVA Report No. 51-9157140-001, "DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal," dated June 10, 2011 was included as an Enclosure to FENOC letter L-11-203 dated June 17, 2011 (ADAMS Accession No. ML11172A389).

When the sum of the transients in the AOTC Logs indicate that the total number of transients is within 75% of the designated number of design cycles or within 10% of the 60-year projected cycles for any transient classification, a condition report shall be initiated. Some of the EAF evaluations used 60-year projected cycles versus design cycles. The 60-year projected cycles are more limiting than the design cycles and therefore, become the allowable when considering environmental effects. When the accumulated cycles approach the allowable cycles, corrective action is taken that includes an engineering evaluation to ensure the Code design limit of 1.0 is not exceeded, including environmental effects where applicable. For transient cycles that are projected to exceed the allowable cycle limit by the end of the next plant operating cycle (Davis-Besse operating cycles are normally two years in duration), the corrective action shall include an update of the fatigue usage calculation for the affected component(s). Acceptance criterion is to maintain the cumulative fatigue usage below the Code design limit of 1.0 through the period of extended operation, including environmental effects where applicable.

All non-NUREG/CR-6260 locations evaluated for EAF are evaluated for at least the 60-year projected Cycles documented in FENOC letter L-11-166 (ADAMS Accession No. ML11159A132), Enclosure Table 4.3-1, "60-Year Projected Cycles," with the following exceptions. The pressurizer heater bundle diaphragm plate and seal weld were evaluated for the best estimate 60-year heatup and cooldowns of 114 (reference FENOC letter L-11-203 dated June 17, 2011 (ADAMS Accession No. ML11172A389)) instead of 128. Locations for the core flood piping, letdown piping, Low Pressure Injection (LPI) check valves, LPI containment isolation valves, and DHR containment isolation valves as listed in Table 2 are currently evaluated for one transient number 11 (rod withdrawal accident), 18 (loss of feedwater heater), and 28 (maximum probable earthquake (OBE)) since these transients have not occurred. The limiting number of transients evaluated in the EAF analysis will be tracked in the AOTC program as the transient cycle limit. Therefore, all locations are managed by the Fatigue Monitoring Program.

The CUF_{en} values for locations that screen out as described in the FENOC response to Request 1, above, were not determined. Table 2, below, lists the CUF_{en} value for the components evaluated.

Table 2: EAF Usage Factors (U_{en})

Location	Material Type	U_{en}
Reactor Coolant Drain Nozzle and Adjacent Piping*	SS	0.8866
Core Flood Piping Point 490	SS	0.61
Letdown Piping Point 859	SS	0.73
Low Pressure Injection Check Valves	SS	0.14
Low Pressure Injection Containment Isolation Valves	SS	0.20
Decay Heat Removal Containment Isolation Valves	SS	0.19
Decay Heat Removal Nozzle, CS Portion	CS	0.92
RCP Cover, Bearing Cavity	SS	0.49
Pressurizer Heater Closure, Plate	SS	0.92
Pressurizer Heater Closure, Seal Weld	SS	0.75

* U_{en} determined using NUREG/CR-6909 rules. The strain amplitude is less than the strain amplitude threshold of 0.10% in all cases, F_{en} does not need to be applied.