FENOC

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October 2, 2008
L-08-287

10 CFR 54

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Supplement to Reply to Request for Additional Information for the Review of the Beaver Valley Power Station, Units 1 and 2, License Renewal Application (TAC Nos. MD6593 and MD6594) and License Renewal Application Amendment No. 25

Reference 1 provided the FirstEnergy Nuclear Operating Company (FENOC) License Renewal Application (LRA) for the Beaver Valley Power Station (BVPS). Reference 2 provided the FENOC reply to a U.S. Nuclear Regulatory Commission (NRC) request for additional information regarding BVPS license renewal time-limited aging analyses (TLAA) related to metal fatigue in Sections 4.3 and B.2.27 of the BVPS LRA.

During conference calls between FENOC and the NRC on August 28, 2008 and September 4, 2008, related to the FENOC reply in Reference 2, the NRC staff asked for supplements to the responses to the NRC requests for additional information (RAIs) for RAIs 4.3-1, 4.3-3, 4.3-11 and 4.3-16 to clarify the information submitted. This letter provides the FENOC supplemented response to NRC RAIs 4.3-1, 4.3-3, 4.3-11 and 4.3-16. This letter also provides Amendment No. 25 to the BVPS LRA, including revised license renewal future commitments, based on changes resulting from the FENOC supplemental responses to the NRC RAIs.

The Attachment provides the FENOC replies to the request for supplemental information. The Enclosure provides Amendment No. 25 to the BVPS 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.

A108
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 2, 2008.

Sincerely,

Roy K. Brosi

References:
1. FENOC Letter L-07-113, "License Renewal Application," August 27, 2007.
2. FENOC Letter L-08-209, "Reply to Request for Additional Information for the Review of the Beaver Valley Power Station, Units 1 and 2, License Renewal Application (TAC Nos. MD6593 and MD6594), and License Renewal Application Amendment No. 15," July 11, 2008.

Attachment:
Supplement to the Response to Request for Additional Information Regarding Beaver Valley Power Station, Units 1 and 2, License Renewal Application, Section 4.3

Enclosure:
Amendment No. 25 to the BVPS License Renewal Application

cc: Mr. K. L. Howard, NRC DLR Project Manager
Mr. S. J. Collins, NRC Region I Administrator
Mr. J. E. Richmond, NRC Region I DRS/EB1

cc: w/o Attachment or Enclosure
Mr. B. E. Holian, NRC DLR Director
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRC DORL Project Manager
Mr. D. J. Allard, PA BRP/DEP Director
Mr. L. E. Ryan, PA BRP/DEP

RAI-4.3-01 NRC Follow-up Questions (Conference Call August 28, 2008):

New Question 1

Line 1 of the initial response (page 20 of FENOC Letter L-08-209) lists dates that seem to conflict with statements regarding monitoring start dates that appear in the referenced letters. The letters state that data collected began before the dates cited in the RAI response. FENOC stated during the conference call that data collection did start before the dates listed in the response, but that data collection was for the establishment of a baseline, and once the baseline was established FENOC deemed the monitoring began on the dates listed in the response. Please provide clarification.

RESPONSE for New Question 1

Thermocouple data collection for the establishment of a baseline was commenced on the monitoring start dates that appear in the referenced letters (References 1 and 2). Therefore, the 1st paragraph is revised as follows:

Collection of thermocouple monitoring data commenced in June 1989 (startup from the first refueling) for Unit 2 and in December 1989 (startup from the seventh refueling) for Unit 1, this data collection was suspended in 2002.

New Question 2

The initial response states that "....renewed thermocouple monitoring may be required for some lines." The phrase "may be required" is too vague. FENOC stated during the conference call that the requirement for monitoring will be determined and tracked via MRP-146.

Please provide a commitment added to Appendix A stating that monitoring will be done in accordance with MRP-146.

RESPONSE for New Question 2

FENOC provides a future License Renewal commitment to implement "needed actions" of MRP-146 (Reference 4) as follows:

FENOC will implement "needed actions" of MRP-146. These actions include screening, detailed analysis, inspections and temperature monitoring in accordance with the guidelines of MRP 146. FENOC has completed screening of the BVPS RCS branch lines.

References:

1. Sieber, John D. (BVPS), Letter to NRC, "Beaver Valley Power Station Unit No. 2, Docket No. 50-412, License No. NFP-73, NRC Bulletin 88-08," 7/14/1989 (NRC PDR Ascension Number 8907240226)

2. Sieber, John D. (BVPS), Letter to NRC, "Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, NRC Bulletin 88-08," 2/7/1990 (NRC PDR Ascension Number 9002150239)

3. EPRI Technical Report 1000701, Interim Thermal Fatigue Management Guideline (MRP-24)," January 2001

4. EPRI Technical Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)," June 2005

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

RAI 4.3-03 NRC Follow-up Questions (Conference Call September 4, 2008):

NUREG/CR-6934 is not endorsed by the NRC. If FENOC selects the analysis option that uses general methodology as described in NUREG/CR-6934 the following statement is required: "This option will require NRC review and approval." The previously provided commitment will need to be changed. Noting FENOC's present course for data reduction on the number of transients (Unit 1 Pressurizer Surge line Hot Leg Nozzle), that FENOC has reasonable expectation to be successful (60-year cumulative usage factor (CUF) with EAF considerations < 1.0), that FENOC has two paths they are pursuing (data reduction and fracture mechanics), and that when FENOC is done they will submit those results, please state that regarding (c)(1)(iii), FENOC will do one of the three options (with the above-mentioned NRC review and approval) and add it to the Metal Fatigue program as an enhancement. Then it will not be an open item.

RESPONSE RAI 4.3-03 NRC Follow-up Questions

The response to the original RAI 4.3-03 from Letter L-08-209 is replaced in its entirety with the following:

a. **Please provide the schedule for refining the analysis for the environmental assisted fatigue (EAF) of the NUREG/CR-6260 (Reference 1) locations in which the cumulative usage factor includes environmental effects (U_{env}) exceed the design code allowable value of 1.0.**

   The refined analyses for Unit 1 Charging System Nozzle, Unit 2 Charging System Nozzle, Unit 2 Safety Injection System Nozzle, and Unit 2 Residual Heat Removal System Piping are completed. For these NUREG/CR-6260 locations, the refined analyses resulted in cumulative usage factors including environmental effects (U_{env}) of less than the design code allowable (i.e., $U_{env} \leq 1.0$). A summary of how the calculations were performed is provided in item b of this response.

   LRA Table 4.3-1 and Section 4.3.3.3.3 (including the associated statements in LRA Appendix A, Sections A.2.3.3.2 and A.3.3.3.3) are revised to address the results of the refined analyses. See the Enclosure to this letter for the revision to the BVPS LRA.

   At two locations (Unit 1 pressurizer surge line to hot leg nozzle and Unit 2 pressurizer surge line to hot leg nozzle), U_{env} exceeded the design code allowable limit of 1.0. The refined analyses including other actions to manage the environmental-assisted fatigue for the Unit 1 pressurizer surge line to hot leg nozzle and the Unit 2 pressurizer surge line to hot leg nozzle will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.27). Previously, in FENOC Letter L-08-209, an enhancement was added to the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.27) and associated commitments provided in LRA Appendix A (Table A.4-1, Item Number 25 and Table A.5-1, Item Number 26 as follows:

"Add a requirement that fatigue will be managed for the NUREG/CR-6260 locations. This requirement will provide that management is accomplished by one or more of the following.

1. Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;

2. Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,

3. Repair or replacement of the affected locations."

In addition, the response to RAI-4.3-11 is revised in this letter to provide that a minimum of 50 cycles of OBE (5 events of 10 cycles each) are considered in each design analysis calculating CUF for NUREG/CR-6260 locations.

Therefore, the Regulatory Commitment for the refined analyses including the alternative analysis (fracture mechanics analysis) and use of a minimum of 50 cycles of OBE previously provided in Attachment 2 to FENOC Letter L-08-209 is withdrawn.

b. Please explain how the calculations for the fatigue life correction factor (F_{en}), used to express the effects of the reactor coolant environment, will be performed. Specifically, how the transient pairs will be considered in the calculations.

Unit 1 Surge Line to Hot Leg Nozzle:	For the surge line hot leg nozzle, reactor water environmental effects were evaluated by calculating F_{en} factors on fatigue usage using the general methodology in NUREG/CR-5704 (Reference 3) for stainless steel. According to this method, fatigue usage is calculated with environmental fatigue correction factors on each load pair incremental usage. See Enclosure B, page 18 of FENOC Letter L-08-209 that provided the Westinghouse input to this RAI response.

Unit 1 Charging System Nozzle: The B31.1 analysis for the Unit 1 charging system was modified to meet the requirements of ASME III, Class 1. The design transients for the corresponding Unit 2 piping are judged to be representative of the transients experienced by Unit 1. Design numbers for the CVCS transients were modified in accordance with operating experience at Unit 1. An appropriate F_{en} was calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF was multiplied by the calculated F_{en}.

Unit 2 Surge Line to Hot Leg Nozzle: For the surge line hot leg nozzle, reactor water environmental effects were evaluated by calculating F_{en} factors on fatigue usage using the general methodology in NUREG/CR-5704 for stainless steel. According to this method, fatigue usage is calculated with environmental fatigue correction factors on each load pair incremental usage. See Enclosure B, page 18 of FENOC Letter L-08-209 that provided the Westinghouse input to this RAI response.

Unit 2 Charging System Nozzle: The analysis of record for the Unit 2 charging piping was revised to incorporate new and revised thermal transients reflecting the operating experience at BVPS Unit 2. In addition, analytical conservatism was reduced to address the effects of environmentally assisted fatigue (EAF). All original design transients continue to be used without reduction for projected cycles. A design CUF was calculated. An appropriate F_{en} was calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF was multiplied by the calculated F_{en}.

Unit 2 Safety Injection System Nozzle: A supplemental design analysis was performed for the SI nozzle location as defined by NUREG/CR-6260. The original design transients were used; however, the cycles for some transients were reduced to a bounding number. A design CUF was calculated. An appropriate F_{en} was calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF was multiplied by the calculated F_{en}.

Unit 2 Residual Heat Removal System Piping:	A supplemental design analysis was performed for the RHR system piping location as defined by NUREG/CR-6260. The original design transients were used without reduction for projected cycles. A design CUF was calculated. An appropriate F_{en} was calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF was multiplied by the calculated F_{en}.

c. Please describe the criteria and methodology that will be performed for the additional analyses in calculating the CUF, including environmental effects, for the components where the CUF exceeds the design code allowable value of 1.0.

Unit 1 Surge Line to Hot Leg Nozzle:	The surge line hot leg nozzle fatigue analyses were performed according to the detailed methods of ASME Code Section III, NB-3200, as permitted by the NB-3600 piping design section. The NB-3200 evaluation was performed using program WESTEMS™. See Enclosure 1, page 19 of FENOC Letter L-08-209 that provided the Westinghouse input to this RAI response. The method used to evaluate the effects of reactor water environment on the ASME fatigue usage is addressed in part b of this response. Refined analysis is in progress as described in part a, above. While it is anticipated that the refined analysis will be successful, as an alternative a fracture mechanics analysis will be performed in accordance with the existing Metal Fatigue of Reactor Coolant Pressure Boundary Program.
Unit 1 Charging System Nozzle:	Using the method described in item b above, the reanalysis resulted in a cumulative usage factors including environmental effects (U_{env}) of less than the design code allowable (i.e., $U_{env} \leq 1.0$). No additional analysis is required.

Unit 2 Surge Line to Hot Leg Nozzle:	The surge line hot leg nozzle fatigue analyses were performed according to the detailed methods of ASME Code Section III, NB-3200, as permitted by the NB-3600 piping design section. The NB-3200 evaluation was performed using program WESTEMS™. See Enclosure 1, page 19 of FENOC Letter L-08-209 that provided the Westinghouse input to this RAI response. The method used to evaluate the effects of reactor water environment on the ASME fatigue usage is addressed in part b of this response. Refined analysis is in progress as described in part a, above. It is anticipated that the refined analysis will be successful.
Unit 2 Charging System Nozzle:	Using the method described in item b above, the reanalysis resulted in a cumulative usage factors including environmental effects (U_{env}) of less than the design code allowable (i.e., $U_{env} \leq 1.0$). No additional analysis is required.
Unit 2 Safety Injection System Nozzle:	Using the method described in item b above, the reanalysis resulted in a cumulative usage factors including environmental effects (U_{env}) of less than the design code allowable (i.e., $U_{env} \leq 1.0$). No additional analysis is required.
Unit 2 Residual Heat Removal System Piping:	Using the method described in item b above, the reanalysis resulted in a cumulative usage factors including environmental effects (U_{env}) of less than the design code allowable (i.e., $U_{env} \leq 1.0$). No additional analysis is required.

References:

1. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," February 1995

2. NUREG/CR-6934, "Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping – A Basis for Improvements to ASME Code Section XI Appendix L," May 2007

3. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," March 1999

4. WESTEMS™ Integrated Diagnostics and Monitoring System

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

RAI-4.3-11 NRC Follow-up Question (Conference Call September 4, 2008):

Regarding operating basis earthquake (OBE) cycles, FENOC is using 50 OBE cycles (which is the current licensing basis), but also says that, if necessary, FENOC will use less than 50 cycles. Once you have an OBE you're going to have a certain number of cycles. You can't parse that. Do you need to reduce the OBE cycles? If not, please restate your response.

RESPONSE RAI-4.3-11 NRC Follow-up Question

The Regulatory Commitment referenced in the original response to RAI 4.3-11 has been withdrawn. In order to remove the reference to that commitment and reply to the follow-up question above, the response to the original RAI 4.3-11 from Letter L-08-209 is replaced in its entirety with the following:

The 60-year projected operational cycles for operational basis earthquakes (OBE) is not provided in LRA Table 4.3-2. Please explain how many OBE occurrences or stress cycles will be included in the 60-year EAF.

A minimum of 50 cycles of OBE (5 events of 10 cycles each) is utilized in each design analysis calculating CUF for NUREG/CR-6260 locations.

RAI-4.3-16 NRC Follow-up Question (Conference Call August 28, 2008):

The LRA discussed the heat-up and cooldown pressurizer transients. RAI 4.3-16 requested the associated histograms. No pressurizer cooldown histogram was provided. Why?

RESPONSE RAI-4.3-16 NRC Follow-up Question

There is no independent transient that is tracked for pressurizer cooldown.

The LRA Section 4.3.4, 3rd paragraph, is revised in its entirety as follows:

Because plant performance has improved with time, the first option typically results in a more accurate projection, while the second option provides a more conservative number of thermal cycles. With the exception of the Unit 1 plant heatup and cooldown, and Unit 1 reactor trip transients, the extrapolation for all transients was completed using the second option. For the Unit 1 plant heatup and cooldown, the projected cycles were determined using the first option. For the Unit 1 reactor trip transient, the first option was also chosen, but then biased with additional reactor trips as the unit approaches end-of-life. Accrued operational cycles are based on initial operations for Unit 1 of 1975 and Unit 2 of 1986, and use a current plant life as of October 2003. Therefore, the operating lifetimes used for the evaluations were 28 and 17 years for Unit 1 and Unit 2, respectively. The results of the transient cycle extrapolation are presented in Table 4.3-2.

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

ENCLOSURE

Beaver Valley Power Station (BVPS), Unit Nos. 1 and 2

Letter L-08-287

<u>Amendment No. 25 to the</u>
<u>BVPS License Renewal Application</u>

License Renewal Application
Sections Affected

Table A.4-1

Table A.5-1

Table 4.3-1

Section 4.3.3.3.3

Section A.2.3.3.2

Section A.3.3.3.3

Section 4.3.4

The Enclosure identifies the correction by Affected License Renewal Application (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**
Table A.4-1	**Page A.4-9**	**New Item Number 31**

LRA Table A.4-1, "Unit 1 License Renewal Commitments," requires a new license renewal future commitment to implement "needed actions" of MRP-146. New Item Number 31 is created, and LRA Table A.4-1 is revised to read as follows:

Table A.4-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
31	*Implement "needed actions" of MRP-146. These actions include screening, detailed analysis, inspections and temperature monitoring in accordance with the guidelines of MRP-146. FENOC has completed screening of the BVPS RCS branch lines.*	*FENOC will perform detailed evaluations (analysis, inspections and/or monitoring) in accordance with MRP-146 schedule requirements, or as established by the MRP committee.*	*FENOC Letter L-08-287*	*None*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A.5-1	**Page A.5-10**	**New Item Number 32**

LRA Table A.5-1, "Unit 2 License Renewal Commitments," requires a new license renewal future commitment to implement "needed actions" of MRP-146. New Item Number 32 is created, and LRA Table A.5-1 is revised to read as follows:

Table A.5-1, cont.

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
32	*Implement "needed actions" of MRP-146. These actions include screening, detailed analysis, inspections and temperature monitoring in accordance with the guidelines of MRP-146. FENOC has completed screening of the BVPS RCS branch lines.*	*FENOC will perform detailed evaluations (analysis, inspections and/or monitoring) in accordance with MRP-146 schedule requirements, or as established by the MRP committee.*	*FENOC Letter L-08-287*	*None*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 4.3-1	**Page 4.3-12 & 13**	**Entire table**

As described in the amended response to RAI 4.3-03, Table 4.3-1 is revised to address the results of the refined analyses and to round the U_{env} values to three decimal places. Table 4.3-1 is revised to read as follows:

Location	**Material**	**Design CUF (U_{60})**	**NUREG/CR Multiplier**	**Environmental CUF (U_{env})**
		UNIT 1		
Reactor Vessel Shell and Lower Head	Low Alloy Steel	0.0102	2.53	~~0.0258~~ 0.026
Reactor Vessel Inlet Nozzle	Low Alloy Steel	0.0663	2.53	~~0.1679~~ 0.168
Reactor Vessel Outlet Nozzle	Low Alloy Steel	0.0508	2.53	~~0.1286~~ 0.129
Surge Line Hot Leg Nozzle	Stainless Steel	~~0.8600~~ 0.4155	~~15.35~~ 10.18	~~13.201~~ 4.231
Charging System Nozzle	Stainless Steel	~~0.1271~~ 0.0998	~~15.35~~ 2.86	~~1.95~~ 0.285
Safety Injection System Nozzle	Stainless Steel	0.0121	15.35	~~0.1857~~ 0.186
Residual Heat Removal System Tee	Stainless Steel	0.0087	15.35	~~0.1335~~ 0.134

		UNIT 2		
Reactor Vessel Shell and Lower Head	Low Alloy Steel	0.0016	2.53	~~0.0041~~ 0.004
Reactor Vessel Inlet Nozzle	Low Alloy Steel	0.0891	2.53	~~0.2256~~ 0.226
Reactor Vessel Outlet Nozzle	Low Alloy Steel	0.0601	2.53	~~0.1522~~ 0.152
Surge Line Hot Leg Nozzle	Stainless Steel	~~0.93~~ 0.4995	~~15.35~~ 9.7	~~14.276~~ 4.844
Charging System Nozzle	Stainless Steel	~~0.75~~ 0.0301	15.35	~~11.513~~ 0.462
Safety Injection System Nozzle	Stainless Steel	~~0.0149~~ 0.3586	~~15.35~~ 2.715	~~0.229~~ 0.974
Residual Heat Removal System Piping	Stainless Steel	~~1.0305[a]~~ 0.3889	~~15.35~~ 2.55	~~15.818~~ 0.992

~~a. Projected 60-year cycles are expected to exceed the design cycles by 50 percent. To account for the increased cycles, the design fatigue usage (0.687) was increased by 50 percent.~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.3.3.3	**Page 4.3-13 & 14**	**Entire Section**

The following supersedes the FENOC letter L-08-209 LRA changes shown for Section 4.3.3.3.3 (Enclosure A, pages 10 and 11). Section 4.3.3.3 is replaced in its entirety to read as follows:

At two locations (Unit 1 pressurizer surge line to hot leg nozzle and Unit 2 pressurizer surge line to hot leg nozzle), U_{env} exceeded the design code allowable limit of 1.0. For these two locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27):

1. *Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*

2. *Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,*

3. *Repair or replacement of the affected locations.*

The U_{env} at the other NUREG/CR-6260 locations (Unit 1 reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, charging system nozzle, safety injection nozzle and RHR system tee; Unit 2 reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, charging system nozzle, safety injection system nozzle and RHR system piping), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation.

As discussed in Section 4.3.1, since 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the TLAAs associated with the NUREG/CR-6260 locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)

~~At several locations (Unit 1 pressurizer surge line and charging system nozzle; Unit 2 pressurizer surge line, charging system nozzle, and RHR system piping), U_{env} exceeded the design code allowable limit of 1.0. For these locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27):~~

1. *~~Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;~~*

2. *~~Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,~~*

3. *~~Repair or replacement of the affected locations.~~*

~~Should BVPS select the option to manage environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC prior to the period of extended operation. Therefore, the TLAAs associated with the Unit 1 pressurizer surge line and charging system nozzle, and the Unit 2 pressurizer surge line, charging system nozzle, and RHR system piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~

~~The CUFs, including environmental fatigue at the other limiting locations (Unit 1 reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, safety injection nozzle and RHR system tee; Unit 2 reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, and safety injection nozzle), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation. Therefore, the TLAAs associated with these other locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.2.3.3.2	**Page A.2-10**	**Last 3 paragraphs of section**

The following supersedes the FENOC letter L-08-209 LRA changes shown for Section A.2.3.3.2 (Enclosure A, pages 22 and 23). The last three paragraphs of Section A.2.3.3.2 are replaced in their entirety to read as follows:

At the pressurizer surge line to hot leg nozzle, U_{env} exceeded the design code allowable limit of 1.0. For this location, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:

1. *Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*

2. *Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,*

3. *Repair or replacement of the affected locations.*

The U_{env} at the other NUREG/CR-6260 locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, charging system nozzle, safety injection nozzle and RHR system tee), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation.

As discussed in Section A.2.3.1, since 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the TLAAs associated with the NUREG/CR-6260 locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

~~*At two locations (pressurizer surge line and charging system nozzle), U_{env} exceeded the design code allowable limit of 1.0. For these locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:*~~

1. ~~*Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*~~

2. ~~*Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,*~~

3. ~~*Repair or replacement of the affected locations. Should BVPS select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program, such as scope, qualification, method, and frequency, will be submitted to the NRC prior to the period of extended operation. Therefore, the pressurizer surge line and charging system nozzle TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~

~~*The cumulative usage factors including environmental fatigue at the other locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, safety injection nozzle and RHR system tee) have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation. Therefore, the fatigue TLAAs associated with these locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.3.3.3.3	**Page A.3-13**	**Last 3 paragraphs of section**

The following supersedes the FENOC letter L-08-209 LRA changes shown for Section A.3.3.3.3 (Enclosure A, pages 31 and 32). The last three paragraphs of Section A.3.3.3.3 are replaced in their entirety to read as follows:

At the pressurizer surge line to hot leg nozzle, U_{env} exceeded the design code allowable limit of 1.0. For this location, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:

1. *Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*
2. *Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,*
3. *Repair or replacement of the affected locations.*

The U_{env} at the other NUREG/CR-6260 locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, charging system nozzle, safety injection system nozzle and RHR system piping), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation.

As discussed in Section A.3.3.1, since 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the TLAAs associated with the NUREG/CR-6260 locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

~~*At three locations (pressurizer surge line, charging system nozzle, and RHR system piping),* U_{env} *exceeded the design code allowable limit of 1.0. For these locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:*~~

1. ~~*Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*~~

2. ~~*Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,*~~

3. ~~*Repair or replacement of the affected locations.*~~

~~*Should BVPS select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program, such as scope, qualification, method, and frequency, will be submitted to the NRC prior to the period of extended operation. Therefore, the TLAAs associated with the pressurizer surge line, charging system nozzle, and RHR system piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~

~~*The cumulative usage factors including environmental fatigue at the other locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, and safety injection nozzle), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation. Therefore, the TLAAs associated with these locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.4	**Page 4.3-14 &15**	**Third paragraph**

Section 4.3.4 requires revision because no independent transient is tracked for pressurizer cooldown. The third paragraph is modified to read as follows:

Because plant performance has improved with time, the first option typically results in a more accurate projection, while the second option provides a more conservative number of thermal cycles. With the exception of the Unit 1 plant heatup and cooldown, ~~pressurizer cooldown,~~ and Unit 1 reactor trip transients, the extrapolation for all transients was completed using the second option. For the Unit 1 plant heatup and cooldown ~~and for pressurizer cooldown~~, the projected cycles were determined using the first option. For the Unit 1 reactor trip transient, the first option was also chosen, but then biased with additional reactor trips as the unit approaches end-of-life. Accrued operational cycles are based on initial operations for Unit 1 of 1975 and Unit 2 of 1986, and use a current plant life as of October 2003. Therefore, the operating lifetimes used for the evaluations were 28 and 17 years for Unit 1 and Unit 2, respectively. The results of the transient cycle extrapolation are presented in Table 4.3-2.