FENOC

FirstEnergy Nuclear Operating Company

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July 11, 2008
L-08-209

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
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

Reference 1 provided the FirstEnergy Nuclear Operating Company (FENOC) License Renewal Application (LRA) for the Beaver Valley Power Station (BVPS). Reference 2 requested additional information from FENOC regarding the BVPS license renewal integrated plant assessment in Sections B.2.27, 4.3, and 4.7.4 of the BVPS LRA.

Attachment 1 provides the FENOC reply to the U.S. Nuclear Regulatory Commission request for additional information. Attachment 2 provides the Regulatory Commitment List. Enclosure A provides Amendment No. 15 to the BVPS License Renewal Application. Enclosure B provides a copy of Westinghouse letter FENOC-08-109, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue," Revision 1, dated June 25, 2008.

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.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 11, 2008.

Sincerely,

Mark A. Manoleras

A108
NRR

References:
1. FENOC Letter L-07-113, "License Renewal Application," August 27, 2007.
2. NRC Letter, "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)," May 28, 2008.

Attachments:
1. Reply to Request for Additional Information Regarding Beaver Valley Power Station, Units 1 and 2, License Renewal Application, Sections B.2.27, 4.3, and 4.7.4
2. Regulatory Commitment List

Enclosures:
A. Amendment No. 15 to the BVPS License Renewal Application
B. Westinghouse letter FENOC-08-109, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue," Revision 1, dated June 25, 2008.

cc: Mr. K. L. Howard, NRC DLR Project Manager
Mr. S. J. Collins, NRC Region I Administrator

cc: w/o Attachments or Enclosures
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

Section B.2.27

Question RAI B.2.27-1

License renewal application (LRA) B.2.27 states "Critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the sixty year operating life of the units." Please provide the list of the critical transients, and explain the basis for the selection of these transients, including the selection criteria.

RESPONSE RAI B.2.27-1

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is a time-limited aging analysis (TLAA) program that uses preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The preventive measures consist of monitoring and tracking critical thermal and pressure transients for Reactor Coolant System (RCS) components to prevent the fatigue design limit from being exceeded. Critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the 60-year operating life of the units. These critical transients include plant heatup, plant cooldown, reactor trip from full power (Unit 1 only), inadvertent auxiliary spray, safety injection activation (Unit 1 only), and RCS cold overpressurization. Supplemental transients were also identified by the program for monitoring. These supplemental transients include the pressurizer insurge transient, selected Chemical and Volume Control System (CVCS) transients, auxiliary feedwater injections and Residual Heat Removal System (RHR) actuation. For each critical transient and supplemental transient, the basis for monitoring is provided in the Unit 1 (Unit 2) Critical and Supplemental Transients tables (shown below).

Unit 1 Critical and Supplemental Transients

Critical/Supplemental Transient	Basis for Monitoring
Plant Heatup	The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with Improved Technical Specifications (ITS) 5.5.3. (Reference 1)
Plant Cooldown	
Reactor Trip from Full Power	
Inadvertent Auxiliary Spray	
RCS Cold Overpressurization	
Safety Injection Activation	The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. Evaluation has shown that detailed analysis could qualify approximately 110 cycles. This transient is monitored to ensure the administrative limit (60 cycles) is not exceeded without additional evaluation (Reference 1).
Pressurizer Insurge	The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 1). The analytical basis includes assumed future occurrences of insurges with a range of delta-T values (Pressurizer water space temperature - Hot Leg temperature). The plant has a thermocouple located in the transition elbow just outboard of the Pressurizer. That computer point (recorded to the plant computer data archiving system) is monitored during startup and shutdown operations to identify whether an insurge has occurred; if so, the delta-T is determined from the plant computer data archiving system. Delta-T values are compared to the analytical basis, and the future allocations are adjusted.

Unit 1 Critical and Supplemental Transients, cont.

Critical/Supplemental Transient	Basis for Monitoring
Auxiliary Feedwater Injections	The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 1). This transient was not part of the original BVPS design basis. The Extended Power Uprate project identified that this was a transient incorporated in the Westinghouse analysis for the Replacement Steam Generators.
Selected CVCS Transients	The selected transients requiring monitoring are: 1) Isolation of charging during Mode 4 operation, 2) Isolation of letdown during Mode 4 operation, and 3) Isolation of charging during Modes 1, 2 and 3. The projected cycles for these events may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 1).
RHR Activation	The projected cycles may approach the design cycles during the period of extended operation. The Unit 1 RHR system tee is a NUREG/CR-6260 location and was evaluated for environmentally-assisted fatigue. The Unit 1 RHR system tee is designed to ANSI B31.1 and was, therefore, re-evaluated in accordance with ASME Section III to determine a cumulative usage factor. The fatigue analysis assumed 600 cycles for RHR activation. As provided in the response to NRC request for additional information (RAI) RAI-B.2.27-9, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (License Renewal Application (LRA) Section B.2.27) is revised to show an enhancement to require monitoring of the Unit 1 RHR activation transient and to establish an administration limit of 600 cycles for the transient.

Unit 2 Critical and Supplemental Transients

<table>
<tr><th>Critical/Supplemental Transient</th><th>Basis for Monitoring</th></tr>
<tr><td>Plant Heatup</td><td rowspan="4">The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 2)</td></tr>
<tr><td>Plant Cooldown</td></tr>
<tr><td>Inadvertent Auxiliary Spray</td></tr>
<tr><td>RCS Cold Overpressurization</td></tr>
<tr><td>Pressurizer Insurge</td><td>The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 2).

The analytical basis includes assumed future occurrences of insurges with a range of delta-T values (Pressurizer water space temperature – Hot Leg temperature). The plant has a thermocouple located in the transition elbow just outboard of the Pressurizer. That computer point (recorded to the plant computer data archiving system) is monitored during startup and shutdown operations to identify whether an insurge has occurred; if so, the delta-T is determined from the plant computer data archiving system. Delta-T values are compared to the analytical basis, and the future allocations are adjusted.</td></tr>
<tr><td>Auxiliary Feedwater Injections</td><td>The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 2). The Extended Power Uprate project identified that this was a transient incorporated in the Westinghouse analysis for the Steam Generators.</td></tr>
</table>

Unit 2 Critical and Supplemental Transients, cont.

Critical/Supplemental Transient	Basis for Monitoring
RHR Activation	The projected cycles may approach the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 2).
Selected CVCS Transients	The selected transients requiring monitoring are: 1) Isolation of charging during Mode 4 operation, 2) Isolation of letdown during Mode 4 operation, and 3) Isolation of charging during Modes 1, 2 and 3. The projected cycles for these events may approach or exceed the design cycles during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3. (Reference 2). The design analysis has been revised to incorporate 60-year projected cycles for these transients as described in the response to RAI-4.3-3.

References:

1. BVPS Updated Final Safety Analysis Report (UFSAR) Unit 1, Rev. 24 (Table-4.1-10).
2. BVPS UFSAR Unit 2, Rev. 16 (Table 3.9N-1).

Question RAI B.2.27-2

Please provide the alert value (or triggering point) for transient cycle monitored of each component under the "Metal Fatigue of Reactor Coolant Pressure Boundary Program" and describe the follow-up actions or corrective actions when a triggering point is approached. Please explain how the process is incorporated into the current plant procedure.

RESPONSE RAI B.2.27-2

The current plant procedure that implements the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.27) addresses critical transients. Specifically, the procedure states that critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the 60-year operating life of the units. These critical transients are required to be monitored. The number of critical transient occurrences is periodically reviewed (annual basis) to determine if there are any adverse trends, adverse conditions, or deficient conditions with the primary objective of initiating evaluation of adverse trends and adverse conditions early to prevent the possibility of a deficient condition.

- "Adverse Trend" is an observed increase in the rate of critical transient occurrences that, if it continued, would result in exceeding the fatigue cycle design limit number of transients prior to the end of the Unit's 60-year operating life.
- "Adverse Condition" is a condition in which the number of actual transient occurrences exceeds 80% of the fatigue cycle design limit number of occurrences.
- "Deficient Condition" is a condition in which the current number of actual transient occurrences exceeds the fatigue cycle design limit number of occurrences.

Adverse trends and adverse conditions are evaluated by Engineering to determine if and when more rigorous analysis or alternate resolutions are required. Deficient conditions are addressed under the FENOC Corrective Action Program.

Question RAI B.2.27-3

During the on-site audit, the applicant stated "the surge line to hot leg nozzle, for [Beaver Valley Power Station (BVPS)] units 1 and 2, is included in a stress and fatigue model to be used in an on-line monitoring system (WESTEMS)"

a. Please explain the purpose of the WESTEMS in the management of components subject to metal fatigue including NUREG/CR-6260 components for the period of extended operation.

b. Please provide the benchmarking results for the WESTEMS software using relevant transient data, and proper 3-D model. Please justify the use of WESTEMS to update the cumulative usage factors (CUF) calculation by using the monitored or projected transient data (cycles) and discuss the conservatisms in the calculation on a plant specific basis.

RESPONSE RAI B.2.27-3

a. The WESTEMS software is used in the analysis of two specific locations for each Unit, the pressurizer lower shell and related components, and the surge line to hot leg nozzle. In addition, WESTEMS software is used in the analysis of the Unit 1 pressurizer spray nozzle. The nature of the analysis is different between the locations.

The Unit 1 and Unit 2 pressurizer lower shell analyses, and the Unit 1 pressurizer spray nozzle analysis used actual plant data for past operational transients, and a conservative set of analytical assumptions for future transients.

For the lower shell analyses, the basic method used by Westinghouse consisted of identifying the past instances of either a heatup or cooldown event and the system delta-T at the time an insurge might have occurred. Then ranges of delta-T were established and the past occurrences were counted within those bands. Future heatups and cooldowns were assigned to the delta-T band range based on past distribution. The analyses also assumed there were two insurges during each heatup/cooldown at the assigned delta-T. Delta-T is the difference between the pressurizer steam space temperature and the Hot Leg temperature. Therefore, all heatup and cooldown transients beyond those used in the analyses need to be evaluated to determine whether an insurge occurred, and if so, a delta-T band range must be assigned to be counted against the analytical assumptions in the analyses.

The analysis of the Unit 1 pressurizer spray nozzle assumed initial flow of stagnant water at ambient conditions, causing a delta-T of 380 °F, for each of the design 200 heatup cycles

Provided the analytical assumptions continue to be met, there is no need to revise the analyses for the pressurizer lower shell at either Unit or for the Unit 1 pressurizer spray nozzle.

The WESTEMS analysis for the surge line to hot leg nozzles is also based on past occurrences of various transients along with what are believed to be conservative assumptions of future transients. In this case, the input assumptions are to be verified by periodic reanalysis using updated plant history files.

b. The FENOC response to Part b of this RAI is provided in Westinghouse letter FENOC-08-109, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue," Revision 1, dated June 25, 2008, included as Enclosure B to this letter. Refer to pages 3 thru 16 of the Westinghouse letter.

Question RAI B.2.27-4

During the audit, the staff noted element 10 of Table 6.0-1 of license renewal basis document (FMP Program Document LRBV-PED-X.M1) states that, "The design transient assumed by original design analysis will be sufficient for 60-year operation." LRA B.2.27 states that, "Fatigue monitoring to date indicates that the number of design transient events assumed in the original design analysis will be sufficient for a 60-year operating period." Please provide a basis for statements

given that project 60-year cycles of residual heat removal system piping that are expected to exceed the design cycles by 50 percent, which is provided in the annotation (a) of Table 4.3-1.

RESPONSE RAI B.2.27-4

The thermal transient of interest for the Unit 2 RHR piping is "Placing RHR in Service" at approximately 350 °F during plant shutdown operations. Westinghouse initially performed counting of this event and documented their results in WCAP-16173-P (Reference 1). The method employed assumed that RHR was placed in service each time the plant transitioned from Mode 3 to Mode 4. While this is a valid assumption, it depends on an accurate accounting of plant Modes.

To accurately count this event, FENOC has recreated the plant mode history from Power Ascension Testing to the present. The initial Westinghouse count of the "Placing RHR in Service" transient during the period 1987 to October 15, 2003, was 85 events in approximately 17 years (Reference 1). The 60-year projection was:

$$\left(\frac{85}{17}\right)(60) = 300 \text{ cycles}$$

The corrected FENOC count of "Placing RHR in Service" from 1987 to October 15, 2003, was 31 events. The 60-year revised projection is:

$$\left(\frac{31}{17}\right)(60) \cong 110 \text{ cycles}$$

From the revised projection, it is clear that use of the design cycles of 200 bounds the projected cycles. This transient will be retained in the cycle counting program.

Since the design cycles of 200 bounds the projected cycles, LRA Sections 4.3.1.1 and A.3.3.1.1 are revised to remove the discussion pertaining to the Unit 2 RHR piping.

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

Reference:

1. WCAP-16173-P, "Beaver Valley Units 1 and 2 Design Basis Transient Evaluation for License Renewal," March 2004, including Errata dated August 11, 2004.

Question RAI B.2.27-5

Provide a comparison of the design transients in the LRA Table 4.3-2 with basis document (ADM 2115) and the transients in the latest associated piping design specification for BVPS Unit 2. Please justify any discrepancy between the LRA Table and plant documents (ADM 2115 and design specification).

RESPONSE RAI B.2.27-5

There are no discrepancies between the LRA Table 4.3-2, "Design and 60-Year Projected Operational Cycles," and plant documents.

Table 4.3-2 provides a listing of the original set of nuclear steam supply system (NSSS) transients, their design cycles, the operational cycles (as of October 15, 2003) and the 60-year projected cycles. Attachment A of procedure 1/2-ADM-2115, "Fatigue Monitoring Program," lists these same transients, identifies whether future monitoring is required, and provides a basis for that determination. Attachment A of the procedure also lists other transients that had previously been monitored for various reasons, and a basis for either continuing or discontinuing monitoring.

A BVPS Engineering Standard provides the design transients to be used in the fatigue analysis of Unit 2 Class 1 piping. Figures are included showing the parameters at the start and end of each transient.

Question RAI B.2.27-6

The LRA does not provide sufficient detail for the staff to determine whether "Metal Fatigue of Reactor Coolant Pressure Boundary Program" is adequate for the period of extended operation. Please provide sufficient detail in LRA Section B.2.27 (such as the scope of the program, method how to monitor the critical and thermal transients, periodic updates of fatigue usage calculation, and how to address environmentally assisted fatigue (EAF)).

RESPONSE RAI B.2.27-6

LRA Section B.2.27, "Metal Fatigue of Reactor Coolant Pressure Boundary Program," is revised to include the results of an evaluation of each of the 10 aging management program elements described in NUREG-1801, Section X.M1. In addition, enhancements were added to address fatigue management of the Unit 1 and Unit 2 NUREG/CR-6260 locations, the Unit 2 steam generator secondary manway bolts and the Unit 2 steam generator tubes. See Enclosure A to this letter for the revision to the BVPS LRA.

Question RAI B.2.27-7

LRA B.2.27 states, "Supplemental transients were also identified by the program for monitoring" (Unit 2 only)

a. Specify the major components affected by these transients. Confirm that the fatigue analysis of these components has been updated to include these transients.

b. Please justify the consistency between those supplemental transients and design transients specified in the design specification.

c. Explain how these supplemental transients will be monitored for the period of extended operation. Please provide the number of design cycles for the supplemental transients and indicate whether these design cycles will remain valid for the period of extended operation.

RESPONSE RAI B.2.27-7

LRA B.2.27 reads as follows:

> *"Supplemental transients were also identified by the program for monitoring. These supplemental transients include pressurizer insurge transient, selected Chemical and Volume Control System transients, Auxiliary Feedwater injections and RHR actuation (Unit 2 only)."*

Therefore, the RAI response is applicable to both Unit 1 and Unit 2 supplemental transients.

a. Major components affected by the transients:

Pressurizer Insurge/Outsurge	The pressurizer lower shell and related components for both Units are affected by pressurizer insurges and outsurges. Unit specific analyses have been performed to qualify these transients.
Selected CVCS Transients	For both Units, there are three specific transients grouped under this heading. They are: 1) Isolation of Letdown Flow during Mode 4 Operation, 2) Isolation of Charging Flow during Mode 4 Operation, and

	3) Isolation of Charging Flow during Modes 1, 2 and 3. The selected CVCS transients only affect the ANSI B31.1 Quality Class 1 portion of the Unit 1 charging piping, and the ASME Class 1 portion of the Unit 2 charging piping.
Auxiliary Feedwater Injections	The Unit 1 steam generators, Unit 2 pressurizer, Unit 2 reactor coolant pumps, Unit 2 loop stop valves and Unit 2 steam generators are affected by Auxiliary Feedwater Injections (also called Feedwater Cycling). The applicable analyses include this transient. No revision was required.
RHR Actuation	The ASME Class 1 portion of the Unit 2 RHR piping is affected by RHR actuation. The applicable piping analysis includes this transient. No revision was required.

b. Consistency with design transients:

Pressurizer Insurge/Outsurge	These transients were not part of the original design basis. Circa 1990, Westinghouse determined that their analysis of the pressurizer lower shell and related components was not consistent with the way the industry operated their plants. Westinghouse Owners Group (WOG) efforts initiated the analysis to resolve this issue.
Selected CVCS Transients	Isolation of letdown and charging flow during RHR Operation were not part of the original design basis, but were identified during Operator interviews into the way the Charging Systems for both Units were operated. Isolation of charging flow during Modes 1, 2 and 3 was part of the original Unit 2 design basis, but the Operator interviews identified that a test had been performed weekly during the early years of operation, causing the projected occurrences of this transient to exceed the original design number of occurrences.

	For Unit 1, the projected cycles remain below 7000 cycles, and reanalysis of the ANSI B31.1 piping was not necessary. For Unit 2, the analysis is being refined as described in the FENOC response to RAI-4.3-3. The reanalysis has incorporated the revised design cycles.
Auxiliary Feedwater Injections	This transient was included in the original analysis for the Unit 2 pressurizer, the Unit 2 reactor coolant pumps and the Unit 2 loop stop valves. However, the Westinghouse identified NSSS Transients did not identify this transient, so it was not previously considered to be part of the design basis. For the steam generators of both Units, this transient was specifically added to the design basis as part of the Extended Power Uprate Project.
RHR Actuation	The Unit 2 RHR actuation transient was part of the original design basis. It was identified as a "supplemental transient" because Westinghouse had indicated that the design cycles may be exceeded during the period of extended operation. That turned out to be incorrect as noted in the FENOC response to RAI-B.2.27-4.

c. Monitoring, design cycles and projection:

Pressurizer Insurge/Outsurge	The Plant Computer data archiving system can be used to identify if an insurge to the Pressurizer has occurred during plant startup or shutdown operations. If an insurge is detected via the surge line thermocouple, the system delta-T during the insurge will be allocated to one of the pre-existing band of delta-T's that were used as input assumptions in the analysis. Tables are included (below) to show the revised delta-T bands for each unit.

The revised delta-T bands for Unit 1 are:

Delta-T Range	Future Allocation Cooldown	Future Allocation Heatup
>320	0	0
320	4	9
310	9	9
300	18	9
290	9	9
280	18	13
270	13	15
260	7	6
250	3	6
240	3	6
230	3	5
Total	87 events	87 events

The revised delta-T bands for Unit 2 are:

Delta-T Range	Future Allocation Cooldown	Future Allocation Heatup
>320	0	0
320	26	34
310	52	0
290	43	43
280	0	52
270	34	43
250	17	0
240	0	0
Total	172 events	172 events

Selected CVCS Transients

Isolation of the letdown flow and isolation of charging flow can be detected at any time by noting valve positions with the Plant Computer data archiving system. Only the charging piping is affected.

For Unit 1, the projected cycles remain below 7000 cycles, and reanalysis of the ANSI B31.1 piping was not necessary.

For Unit 2, the analysis is being refined as described in the FENOC response to RAI-4.3-3. The reanalysis has incorporated the revised design cycles as shown in the following transient count table:

Transient	Revised Design Cycles	Current Count (Oct. 2003)	60-Year Cycle Projection
Isolation of Letdown Flow during Mode 4 Operation	400	61	215
Isolation of Charging Flow during Mode 4 Operation	2000	305	1076
Isolation of Charging Flow during Modes 1, 2 and 3	435	423 [a]	433

a The current count is based on 419 occurrences of a test that will not be performed in the future, along with 4 inadvertent occurrences. The inadvertent cycles are projected to 60 years, giving a total of 433 cycles when added to the 419 test cycles.

Auxiliary Feedwater Injections

This transient can be identified, via the Plant Computer data archiving system, by noting the operation of the auxiliary feedwater pumps and the system flow rates during plant modes 1, 2 or 3. Past operation is based on a prorating of the design cycles in the Westinghouse equipment specifications, or 50 events per year. Transient counts are:

Transient	Design Cycles	Current Count	60-Year Cycle Projection
Unit 1 Auxiliary Feedwater Injections (Feedwater Cycling)	18,300	150	2000 [a]
Unit 2 Auxiliary Feedwater Injections (Feedwater Cycling)	2000	900	1975 [b]

a The projection of 2000 Unit 1 Auxiliary Feedwater Injections applies to the Steam Generators only, which were replaced in 2006. The current count (Year 2008) is based on the assumption of 50 events per year. The projection is based on 50 events per year. This transient will be monitored and it is anticipated that the actual cycles will never approach the design cycles.

b As noted, the current count (Oct. 2003) is based on the assumption of 50 events per year. It is highly unlikely that there have been 50 events per year. The projection is based on 25 events per year. This transient will be monitored and it is anticipated that the actual cycles will never approach the design cycles.

RHR Actuation

The Unit 2 RHR actuation transient can be counted by noting transitions from Mode 3 into Mode 4. The Plant Mode status will be tracked. Transient counts are:

Transient	Design Cycles	Current Count (Oct. 2003)	60-Year Cycle Projection
Placing RHR in Service	200	31	110

Question RAI B.2.27-8

LRA Section B.2.27 provides the operating experience for BVPS Unit 2 letdown, charging, and excess letdown piping. Please explain why no further evaluation was required for letdown or excess letdown piping. Please provide the results of the re-analysis of the charging piping, including the evaluation of EAF for the period of extended operation.

RESPONSE RAI B.2.27-8

The Class 1 portion of the Unit 2 Charging, Letdown and Excess Letdown Systems can be affected by three specific events:

1. Isolation of Letdown Flow,
2. Isolation of Charging Flow, and
3. Placing Excess Letdown in Service.

Occurrences of these charging system transients were not monitored prior to the generation of WCAP-16173-P, "Beaver Valley Units 1 and 2 Design Basis Transient Evaluation for License Renewal," (Reference 1). In addition, they are not transients that are readily detectable based on a review of historical plant documentation. To compile a count of these transients for the WCAP, Westinghouse developed a questionnaire for plant operators, asking them their recollections of number of times these events happened. In the 60 year projections of WCAP-16173-P, Westinghouse identified that, by their count, the Unit 2 charging, letdown and excess letdown transients would exceed the design limit prior to the end of plant life. The Westinghouse estimation of thermal cycles, as of October 15, 2003, was 1,076.

FENOC addressed this concern using the Corrective Action Program. The subsequent investigation determined that the Westinghouse count of CVCS transients was overly conservative, and combines three events (isolation of letdown flow, isolation of charging flow and placing excess letdown in service) as if they affect the same components. However, they do not affect the same components; the excess letdown piping is not affected by either charging isolation or letdown isolation events.

For the letdown piping:

The isolation of letdown flow event would allow the temperature of the letdown piping, outside of the zone heated by RCS loop flow, to slowly decay towards ambient. The applicable design calculations identified that the slow temperature decline (560°F to 100°F over 4.6 hours) in insulated piping is an insignificant event from a fatigue perspective.

Isolation of charging flow, by itself, has no affect on the letdown piping. If charging flow is not restored, then letdown flow must also be isolated. Isolation of both charging and letdown flow would lead to a plant shutdown, which was previously addressed in the piping analysis as a plant shutdown event with a bounding design limit of 200 events.

Placing excess letdown in service has no effect on the primary letdown piping.

For the excess letdown piping:

Charging isolation and letdown isolation have no impact on the excess letdown piping. The excess letdown piping is a completely independent system.

The 25 projected cycles of the "placing excess letdown in service" transient are bounded by the analyzed 100 cycles. Future monitoring of this transient is not required.

Regarding the re-analysis of the charging piping:

The analysis of record for the Unit 2 charging piping will be revised to incorporate new and revised thermal transients reflecting the operating experience at BVPS Unit 2. In addition, analytical conservatism will be reduced so that the effects of environmentally-assisted fatigue (EAF) may be addressed. A regulatory commitment is provided in Attachment 2 to this letter for completion of the reanalysis. See also the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

Reference:

1. WCAP-16173-P, "Beaver Valley Units 1 and 2 Design Basis Transient Evaluation for License Renewal," March 2004, including Errata dated August 11, 2004

Question RAI B.2.27-9

During the audit, the staff noted that FMP Basis Document (ADM 2115) indicates the design transient, RHR actuation, of Unit 1 is not required to be monitored. For the period of extended operation, please provide the basis considering the

ASME Section III analysis and EAF analysis for nozzle are affected by this transient.

RESPONSE RAI B.2.27-9

The Unit 1 RHR System tee is a NUREG/CR-6260 location that was evaluated for environmentally-assisted fatigue. The Unit 1 RHR System tee is designed to ANSI B31.1, and was, therefore, re-evaluated in accordance with ASME Section III to determine a cumulative usage factor. The fatigue analysis assumed 600 cycles for RHR activation. The projected cycles may approach the analyzed cycles during the period of extended operation and therefore, this transient will require monitoring.

The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) is revised to provide an enhancement to require monitoring of the Unit 1 RHR activation transient and to establish an administration limit of 600 cycles for the transient. In addition, the program will require monitoring of Unit 1 or Unit 2 transients where the 60-year projected cycles are used in the environmental fatigue evaluations, and will require establishing an administration limit that is equal to or less than the 60-year projected cycles number. The program enhancements are included as a new BVPS License Renewal Future Commitment in LRA Tables A.4-1 and A.5-1, "Unit 1 (Unit 2) License Renewal Commitments."

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

Question RAI B.2.27-10

LRA Section 4.3.2.2 indicates the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors the transients associated with non-regenerative (letdown) heat exchanger, regenerative heat-exchanger, and RHR heat exchangers. However, LRA Section B.2.27 did not indicate that monitoring for the relevant transients will be provided by this aging management program (AMP).

a. Please the list transients associated with these heat exchangers, and indicate which transients are monitored by the program.

b. Please explain the corrective action when current fatigue analyses of these heat exchangers are not bounding for 60 years of operation.

RESPONSE RAI B.2.27-10

All of the auxiliary system heat exchangers at BVPS Units 1 and 2 are installed on the Class 2 portions of their respective systems. The primary side of these Heat

Exchangers are designed to ASME III, Class 2 rules. For BVPS, detailed fatigue analysis (per ASME Class 1 rules) was not required.

In addition, the total number of thermal cycles expected for each heat exchanger is less than 7000 as required for ASME Class 2 thermal analysis. Therefore, per ASME Class 2 rules, monitoring and/or reanalysis is not required.

LRA Section 4.3.2.2, 3rd paragraph, and Section A.3.3.2.2 (LRA Section A.3 provides the Appendix A TLAA Summaries for BVPS Unit 2), 3rd paragraph, are revised to read:

> *"For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the original design (usually 7,000 cycles, as described in Section 4.3.2.1), the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required. FENOC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i)."*

Also, the LRA Section 4.3.2.2 paragraphs addressing the Unit 2 non-regenerative (letdown) heat exchanger, regenerative heat exchanger, and RHR heat exchangers are deleted.

In addition, the subsections of Section A.3.3.2.2 (A.3.3.2.2.1, A.3.3.2.2.2 and A.3.3.2.2.3) are deleted.

LRA Section A.2.3.2 (LRA Section A.2 provides the Appendix A TLAA Summaries for Unit 1) is revised to add a new subsection (A.2.3.2.2) to read:

> *A.2.3.2.2 Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings*
>
> *Non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings are typically designed in accordance with ASME Section VIII or ASME Section III, Subsection NC or ND (e.g., Class 2 or 3). Some tanks and pumps are designed to other industry codes and standards (such as American Water Works Association and Manufacturer's Standardization Society), reactor designer specifications, and architect engineer specifications. Only ASME Section VIII, Division 2, and ASME*

Section III, Subsection NC-3200, design codes include fatigue design requirements. Due to the conservatism in ASME Section VIII Division 1 and ASME Section III NC-3100/ND-3000 detailed fatigue analyses are not required. If cyclic loading and fatigue usage could be significant, the component designer is expected to specify ASME Section VIII Division 2 or NC- 3200. For components where there is no required fatigue analysis, cumulative fatigue damage is not an aging effect requiring management.

Fatigue analysis is not required for ASME Section VIII Division I, Section III NC- 3100 or ND vessels. It is also not required for NC/ND pumps and storage tanks (<15 psig). The design specification identifies the applicable design code for each component.

For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the original design (usually 7,000 cycles, as described in Section 4.3.2.1), the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required. FENOC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

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

Section 4.3

Question RAI 4.3-1

In responses to NRC Bulletin 88-08 (Letter to NRC, Beaver Valley Power Station, Unit No. 1, BV-1 Docket No. 50-334, License No. DPR-66, NRC Bulletin 88-08, 2/7/1990) and Letter to NRC, Beaver Valley Power Station, Unit No. 2, BV-2 Docket No. 50-412, License No. NPF-73, NRC Bulletin 88-08, 7/14/1989), the applicant stated "temperature monitoring will be continue until a long term solution is implemented" to address the thermal stress in piping connected to reactor coolant system. Please explain the follow-up actions in the above response letters to the NRC Bulletin 88-08 and indicate whether the temperature monitoring

program will be maintained to address the thermal stratification issue for the period of extended operation.

RESPONSE RAI 4.3-1

Collection of thermocouple monitoring data commenced in November 1989 for Unit 2 and in February 1990 for Unit 1. As noted below, this data collection was suspended in 2002.

The initial data collected was used to form the baseline temperature profiles for each line. Review of the baseline data identified that some lines experience steady state top-to-bottom temperature differences (References 1 and 2); this was expected, and consistent with stagnant branch lines attached to turbulent lines, such as the RCS. This temperature stratification causes a global bending moment, stress in the piping, and loading in the pipe supports. Detailed analyses were performed for the lines experiencing global bending due to stratification. The analyses superimposed the stratification moments on the existing stress analysis with acceptable results.

Snapshots of the thermocouple data were reviewed at roughly six month intervals. The Unit-specific data following startup from refueling outages were also reviewed. As evaluations of the monitored data continued, several lines were removed from the monitoring program, due to either:

(1) the analysis performed, or

(2) the presence of redundant isolation capabilities of the lines.

Subsequent data collected was consistent with the baseline stratification profiles. Minor changes in temperature at a single thermocouple were mirrored by changes at adjacent thermocouples. The top and bottom temperatures at a piping location were shown to be in-phase without large changes in the top-to-bottom delta-T. Examination of the data for individual thermocouples supported the position that temperature cycling, consistent with cold water intrusion, was not occurring. Over time, the temporary instrumentation installed to collect this data became degraded and unreliable; it was judged to be a fire hazard. Therefore, data collection was suspended in 2002.

Initial weld inspections required by NRC Bulletin 88-08 were performed during the Unit 1 Cycle 7 Refueling Outage (Fall 1989), and in the Unit 2 Cycle 1 Refueling Outage

(Spring 1989), with subsequent inspections of applicable welds performed during following refueling outages:

Refueling Outage Cycle No.	Timeframe	Number of Welds Inspected
Unit 1		
8	Spring 1991	3
10	Winter 1995	7
11	Spring 1996	4
12	Fall 1997	26
15	Spring 2003	3
17	Winter 2006	4
Unit 2		
3	Spring 1992	2
5	Spring 1995	3
6	Fall 1996	1
7	Spring 1999	16
10	Fall 2003	2
12	Fall 2006	2

No repairs were necessary as a result of these inspections.

FENOC has also participated in Electric Power Research Institute (EPRI) initiatives on this topic, including the Thermal Stratification, Cycling and Striping (TASCS) project, MRP-24 (Reference 3) and MRP-146 (Reference 4). All branch lines within the scope of NRC Bulletin 88-08 have been screened per the guidance of MRP-146. In addition, some lines, previously exempt from 88-08 since in-leakage was not possible, are included in the scope of the MRP-146 evaluations, and have been subsequently screened.

Additional actions associated with MRP-146 include detailed analysis of those lines not screened out and augmented non-destructive examination (NDE) inspections. These required inspections are scheduled for the next refueling outage at each Unit (Spring 2009 for Unit 1, and Fall 2009 for Unit 2). In addition, the MRP-146 process requires detailed analysis of those lines not initially screened out, and, depending on the results of that analysis, renewed thermocouple monitoring may be required for some lines.

The history of BVPS actions taken to address thermal stresses in piping connected to the RCS along with the current initiatives contained within the EPRI MRP-146 process provide reasonable assurance that this issue will be addressed throughout the period of extended operation.

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," July 14, 1989 (NRC PDR Accession 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," February 7, 1990 (NRC PDR Accession 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.

Question RAI 4.3-2

LRA Section 4.3.1.2 indicates that BVPS will perform a reanalysis, repair, or replacement of the affected unit 2 steam generator manway bolts and tubes components as part of an AMP. However, the AMP description in Appendix B did not indicate this. Explain the discrepancy.

RESPONSE RAI 4.3-2

LRA Section 4.3.1.2 states:

> *"As part of the Steam Generator Tube Integrity Program (Section B.2.38), BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design basis of these components is not exceeded for the period of extended operation."*

This statement in LRA Section 4.3.1.2 and the associated statement in LRA Section A.3.3.1.2 are revised to read:

> *"The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be enhanced to include a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation."*

In addition, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) is revised to include an enhancement, as follows:

> *"Add a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation."*

Based on this Metal Fatigue of Reactor Coolant Pressure Boundary Program enhancement, the existing License Renewal Future Commitment related to the Unit 2 manway bolts and steam generator tubes, LRA Table A.5-1 (Unit 2), Item Number 26, is deleted.

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

Question RAI 4.3-3

LRA Section 4.3.3.3 discusses the effects of primary coolant environment on fatigue life. During the audit, the applicant indicated that it will refine the analysis for NUREG/CR-6260 components in the near future. To assist the staff it its review:

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

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.**

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.**

RESPONSE RAI 4.3-3

a. Schedule for refining the NUREG/CR-6260 locations EAF analyses:

   FENOC hereby commits that the results of the refined analyses for EAF of the NUREG/CR-6260 locations for which the cumulative usage factor, including environmental effects (U_{env}), exceeded the design code allowable value of 1.0, and a summary of how the calculations were performed, will be submitted to the NRC no later than October 15, 2008.

The applicable NUREG/CR-6260 locations are listed as follows:

Unit 1 Surge Line to Hot Leg Nozzle

Unit 1 Charging System Nozzle

Unit 2 Surge Line to Hot Leg Nozzle

Unit 2 Charging System Nozzle

Unit 2 Safety Injection System Nozzle

Unit 2 Residual Heat Removal System Piping

It is anticipated that the Unit 1 Surge Line to Hot Leg Nozzle refined analysis will be successful. However, should the refined analysis not be successful, as an alternative, a fracture mechanics analysis of the Unit 1 Surge Line to Hot Leg Nozzle will be performed using the general methodology as described in NUREG/CR-6934. The results of the alternative analysis, if needed, will be submitted to the NRC no later than October 15, 2008.

See Attachment 2 to this letter for the Regulatory Commitment List.

b. Calculations for fatigue life correction factors (F_{en}):

The proposed actions discussed in the following paragraphs of Part "b" of this response represent intended or planned actions by FENOC. They are described only as information and are not Regulatory Commitments. Upon completion of the refined analyses for EAF of the NUREG/CR-6260 locations identified in Part "a" of this response, a summary of how the calculations for fatigue life correction factors were performed will be provided to the NRC. See Attachment 2 to this letter for the Regulatory Commitment List.

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} 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 F_{en} factors on each load pair incremental usage. See Enclosure B, page 17, which provided the Westinghouse input to this RAI response.
Unit 1 Charging System Nozzle:	The B31.1 analysis for the Unit 1 charging system will be 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 are

	modified in accordance with operating experience at Unit 1. An appropriate F_{en} will be calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design cumulative usage factor (CUF) will be 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 F_{en} factors on each load pair incremental usage. See Enclosure B, page 17, which provided the Westinghouse input to this RAI response.
Unit 2 Charging System Nozzle:	The analysis of record for the Unit 2 charging piping will be revised to incorporate new and revised thermal transients reflecting the operating experience at BVPS Unit 2. In addition, analytical conservatism will be reduced so that the effects of EAF may be addressed. All original design transients continue to be used without reduction for projected cycles. A design CUF will be calculated. An appropriate F_{en} will be calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF will be multiplied by the calculated F_{en}.
Unit 2 Safety Injection System Nozzle:	A supplemental design analysis will be performed for the safety injection nozzle location as defined by NUREG/CR-6260. The original design transients will continue to be used; however, the cycles for some transients may be reduced to a bounding number. A design CUF will be calculated. An appropriate F_{en} will be calculated in accordance with the guidance of NUREG/CR-5704 for stainless steel. The design CUF will be multiplied by the calculated F_{en}.
Unit 2 Residual Heat Removal System Piping:	A supplemental design analysis will be performed for the RHR system piping location as defined by NUREG/CR-6260. The original design transients will continue to be used; however, the cycles for some transients may be reduced to a bounding number. A design CUF will be calculated. An appropriate F_{en} will be calculated in accordance with the guidance of

NUREG/CR-5704 for stainless steel. The design CUF will be multiplied by the calculated F_{en}.

c. Criteria and methodology that will be performed for the additional CUF analyses, including environmental effects:

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™ (Reference 4). See Enclosure B, page 19, which 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" of this response. While it is anticipated that the refined analysis will be successful, as an alternative, a fracture mechanics analysis will be performed using the general methodology as described in NUREG/CR-6934.
Unit 1 Charging System Nozzle:	Preliminary evaluations show that the method described in part "b" of this response will be successful. No alternate methodologies will be 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 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" of this response. It is anticipated that the refined analysis will be successful.
Unit 2 Charging System Nozzle:	Preliminary evaluations show that the method described in part "b" of this response will be successful. No alternate methodologies will

	be required.
Unit 2 Safety Injection System Nozzle:	Preliminary evaluations show that the method described in part "b" of this response will be successful. No alternate methodologies will be required.
Unit 2 RHR System Piping:	Preliminary evaluations show that the method described in part "b" of this response will be successful. No alternate methodologies will be 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.

Question RAI 4.3-4

LRA Section 4.3.1.3 describes the Pressurizer Weld Overlay Project on BVPS Unit 2 only. Indicate whether the Pressurizer Weld Overlay Project will be performed for BVPS Unit 1. Explain the impact of the weld overlay on the fatigue usage for the period of extended operation for both units.

RESPONSE RAI 4.3-4

LRA Section 4.3.1.3 states:

> *"In addition, the pressurizer spray nozzle, the safety valve nozzles, the pressure operated relief valve nozzle and the surge line nozzle were potentially impacted by the Pressurizer Weld Overlay Project. Weld overlay was performed during the Unit 2 Cycle 12 Refueling Outage (October – November 2006)."*

During the BVPS Unit 1 Cycle 18 Refueling Outage (Fall 2007), FENOC completed the planned pressurizer structural weld overlay workscope, which included the application of structural weld overlays to the pressurizer spray nozzle, relief nozzle, and three safety nozzles (Reference 1). The Unit 1 pressurizer surge line nozzle is stainless steel and, as such, was not within the structural weld overlay workscope.

Applicable to both Unit 1 and Unit 2 pressurizer nozzles, fatigue crack growth analyses using ASME Code Section XI methodology were performed to demonstrate the fatigue qualification at the structural weld overlay regions. The impact of the addition of structural weld overlay material on the existing primary stress qualifications, considering both deadweight and dynamic loadings, were determined to be insignificant. Reconciliation of the existing fatigue evaluations were performed for the limiting locations outside the structural weld overlay and it was demonstrated that the pressurizer nozzles would still meet the applicable ASME Code Section III requirements. The transient assumptions for these analyses were consistent with the existing stress analyses.

Reference:

1. Sena III, Peter P. (FENOC), FENOC Letter L-07-143, "Beaver Valley Power Station, Unit No. 1 Docket No. 50-334, License No. DPR-66, Pressurizer Weld Overlay Examination Report," October 29, 2007 (ADAMS Accession Number ML073050334).

Question RAI 4.3-5

Please clarify the sentence at the end of first paragraph in LRA Section 4.3.2.2, "For components where there is no required fatigue analysis, cracking due to fatigue is not an aging effect requiring management."

RESPONSE RAI 4.3-5

The sentence in LRA Section 4.3.2.2:

> *"For components where there is no required fatigue analysis, cracking due to fatigue is not an aging effect requiring management,"*

and the identical sentence in LRA Appendix A, Section A.3.3.2.2 are revised to read:

> *"For components where there is no required fatigue analysis, cumulative fatigue damage is not an aging effect requiring management."*

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

Question RAI 4.3-6

LRA Section 4.3.3.2 states, "Therefore, the Unit 1 and Unit 2 pressurizer surge line fatigue time-limited aging analyses (TLAAs) have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i)." Please provide the basis for this statement and for Unit 2 indicate whether the design specification was updated and certified to incorporate changes to the design transients.

RESPONSE RAI 4.3-6

LRA Section 4.3.3.2.3 and the associated statements in LRA Appendix A, Sections A.2.3.3.1 and A.3.3.3.2 are revised to show that the pressurizer surge line fatigue TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The LRA is revised to read:

> *Section 4.3.3.2.3*
>
> *Both WCAP-12727 and WCAP-12093 determine the effect of thermal stratification through the imposition of defined thermal stratification cycles upon the stress and fatigue evaluations. The stratification cycles which are incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients. Therefore, these NRC Bulletin 88-11 analyses are TLAAs in accordance with 10 CFR 54.3. The BVPS original design basis transients including design cycles for the RCS are identified in Table 4.3-2 along with the projected operational cycles that BVPS anticipates will occur for 60 years of plant life. Table 4.3-2 demonstrates that the 200 heatup and cooldown cycles are bounding for 60 years of operation. Since 60-year projected operational cycles were used in determining that the 200 heatup and cooldown transient assumption remains bounding for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate this assumption. Therefore, the Unit 1 and Unit 2 pressurizer surge line thermal stratification TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*

Unit 2 Design Specification

The stratification cycles which are incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients. The Unit 2 design specification previously included 200 heatup and cooldown design transients and therefore, the design specification does not require a change.

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

Question RAI 4.3-7

LRA Section 4.3.3.1 states, "Typical cycle periods for thermal stratification events on the Unit 2 RHR lines were 6 to 8 days, which equated to approximately 2000 cycles for a 40-year plant life (assuming the stratification occurred continuously)". Discuss the technical basis or analyses supporting this statement and provide the supporting documentation.

RESPONSE RAI 4.3-7

In response to NRC Bulletin 88-08 Supplement 3, strategic locations on the Unit 2 RHR suction branch line were instrumented with thermocouples to monitor pipe temperatures for indications of thermal stratifications. Evaluation of the resulting temperature data determined that the typical cycle period for the thermal stratification events were 6 to 8 days, which equated to approximately 2000 cycles for a 40-year plant life (assuming the stratification occurred continuously). A bounding thermal stratification load assuming 7000 cycles was incorporated into the fatigue analysis as an additional load.

Projecting the 2000 cycles assumption for a 60-year plant life, results in 3000 stratifications cycles. Therefore, the 7000 stratification cycles assumption in the Unit 2 RHR line fatigue analysis remains acceptable for the period of extended operation.

The thermal stratification analysis of the Unit 2 RHR line was documented in an attachment to the stress calculation of record. This stress calculation is available for NRC review at BVPS.

Question RAI 4.3-8

The U60 for the Unit 2 RHR system piping in the LRA Table 4.3-1 is higher than Unit 1. In addition, Unit 2 RHR system piping is dispositioned in accordance with 10CFR54.21(c)(1)(iii). The applicant indicated that U60 analysis will be refined during the audit. Please explain in detail how the RHR system piping will be managed for aging effects.

RESPONSE RAI 4.3-8

As discussed in LRA Section 4.3.3.3.1, the NUREG/CR-6260 location for the RHR system tee in the Unit 1 older-vintage plant is different from the RHR system piping (inlet piping transition) location in the Unit 2 newer-vintage plant. Therefore, the CUF results are not directly comparable.

The Unit 1 U_{env} results are acceptable as shown in Table 4.3-1 of the LRA.

The Unit 2 U_{env} results identified in Table 4.3-1 are artificially high for two reasons. First, an inaccurate counting method was used to determine the current cycles for the "Placing RHR in Service" transient, as described in the FENOC response to RAI-B.2.27-4. The corrected cycle count shows that the design assumption of 200 cycles is bounding for 60 years. Second, the effects of thermal stratification were conservatively superimposed on the previous piping analysis. Reanalysis of this location is expected to reduce the design CUF to a value where the resulting U_{env} will be less than 1.0.

A regulatory commitment is provided for completion of the reanalysis. See the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

See Attachment 2 to this letter for the Regulatory Commitment List.

Question RAI 4.3-9

LRA Table 4.3-1 and LRA Section 4.3.3.3 provides the TLAA disposition for BVPS Units 1 and 2 to address EAF. The staff noted the TLAAs for some of the locations appeared to be dispositioned in accordance with 10 CFR 54.21(c)(1)(i), but LRA Section 4.3.3.3 indicated that these components were to be dispositioned under 10 CFR 54.21(c)(1)(ii). Please clarify the TLAA dispositions for the each location.

RESPONSE RAI 4.3-9

Clarification of the TLAA disposition for the each NUREG/CR-6260 location is provided in the revised LRA text that follows. The associated statements in LRA Appendix A, Sections A.2.3.3.2 and A.3.3.3.3 are similarly revised:

> *Section 4.3.3.3.3*
>
> *At several NUREG/CR-6260 locations (Unit 1 pressurizer surge line hot leg nozzle and charging system nozzle; Unit 2 pressurizer surge line hot leg nozzle, charging system nozzle, safety injection system nozzle, and RHR system piping), U_{env} exceeded the design code allowable limit of 1.0.*
>
> *Further refinement of the fatigue analyses for these NUREG/CR-6260 locations will be performed to lower the U_{env} to less than 1.0. While it is anticipated that the Unit 1 surge line to hot leg nozzle refined analysis will be successful, as an alternative, a fracture mechanics analysis will be performed using the general methodology as described in 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.

The U_{env} at the other NUREG/CR-6260 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), 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).

The License Renewal Future Commitments related to this issue provided in LRA Table A.4-1, Item Number 25, and in LRA Table A.5-1, Item Number 28, are deleted.

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

A regulatory commitment is provided for completion of the reanalysis. See the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

See Attachment 2 to this letter for the Regulatory Commitment List.

Question RAI 4.3-10

Various line items in Table 2 of the LRA Chapter 3 indicate TLAA as the AMP for its component type. Please indicate the AMP for these items.

RESPONSE RAI 4.3-10

No change is required to the LRA.

For the line items in the LRA Table 2s that indicate TLAA as the Aging Management Program, a link was provided to the applicable Table 1 item. In the Table 1 item, a link was provided to the applicable subsection of LRA Section 4, "Time-Limited Aging Analyses." The TLAA dispositions were provided in the subsections of Section 4, and, when dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), the applicable aging management program (AMP) was provided.

Question RAI 4.3-11

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.

RESPONSE RAI 4.3-11

As noted in the FENOC response to RAI-4.3-3, EAF analysis is continuing. For those analyses that are complete, a minimum of 50 cycles of operational basis earthquakes (OBE) have been included in each analysis. For those EAF analyses that are not complete, FENOC plans to include a minimum of 50 cycles of the OBE transient for the subject EAF analyses; if it is necessary to use less than 50 cycles of the OBE transient, this will be reported with the results of the analyses.

See Attachment 2 to this letter for the Regulatory Commitment List.

Question RAI 4.3-12

LRA Section 4.3.2.1 states that for the Unit 2 emergency diesel generator air start system, "BVPS will perform an assessment to determine whether the full-temperature cycles limit will be exceeded for 60 years of operation." Provide the estimated temperature cycles expected for 60 years of operation and please explain how these temperature cycles will be monitored during the period of extended operation.

RESPONSE RAI 4.3-12

LRA Section 4.3.2.1, 4th paragraph, reads:

> *"The Unit 2 EDG Air Start System contains components potentially subject to fatigue. As part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27), BVPS will perform an assessment to determine whether the full-temperature cycles limit will be exceeded for 60 years of operation. Corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the full-temperature cycles of the Unit 2 EDG Air Start System are not exceeded for the period of extended operation. Therefore, the Unit 2 EDG Air Start*

> *System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)."*

LRA Section 4.3.2.1 and the associated text in Section A.3.3.2.1 are revised to read:

> *"The Unit 2 EDG Air Start system is designed as a stand-by system. The compressor runs to fill the air start tank and does not run again until the tank needs to be either topped off or refilled after discharge. The act of compressing the air is the only source of heat for this piping. Measurement of discharge temperature during compressor operation indicates that the compressors, bolting, discharge piping and valves may exceed the threshold temperature for thermal fatigue during compressor operation. Confirmation of the full-temperature cycles of the Unit 2 EDG Air Start System would be a time consuming process and would not account for variables such as the leak tightness of the air system. Therefore, the design analysis for the piping has been revised to incorporate the observed temperatures as a new load case. The stress levels for this thermal load case are below the endurance limit for the piping material. In other words, the revised analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, the Unit 2 EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii)."*

In addition, the License Renewal Future Commitment related to this issue provided in LRA Appendix A, Table A.5-1, Item Number 27, is deleted.

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

Question RAI 4.3-13

During the audit, it was found that U60 value, as well as, the environmental CUF value in Table 4.3-1 of the LRA for the Unit 2 safety injection system, are not correct. These values in the LRA do not represent the result for injection nozzle to the cold leg. Please provide the U60 and EAF results for this location.

RESPONSE RAI 4.3-13

During the audit, it was discovered that the calculation of environmental CUF (U_{env}) for the Unit 2 safety injection system nozzle had used an incorrect location, and thus, provided incorrect results. This issue is being addressed under the FENOC Corrective Action Program.

Use of the current design CUF from the correct location with a F_{en} of 15.35 leads to a U_{env} in excess of 1.0. Reanalysis is required, and, therefore, a regulatory commitment is provided for completion of the reanalysis. See the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

See Attachment 2 to this letter for the Regulatory Commitment List.

Question RAI 4.3-14

Note (d) of the LRA Table 4.3-2 states the number of the design cycles for OBE are 400 for nuclear steam supply system (NSSS) equipment and 50 for the piping. During the audit, the staff noted that Table 6-1 of the basis document (WCAP-16173-P, "Beaver Valley Units 1 and 2 Design Basis Transient Evaluation for License Renewal," March 2004, including Errata dated 8/11/2004) shows the design cycles of OBE are 50 for several NSSS equipments of BVPS Unit 1, including the reactor vessel and pressurizer. Please explain the discrepancy between LRA Table 4.3-2 and Table 6-1 from WCAP-16173-P and how the design cycles for OBE will be considered in the CUF evaluation.

RESPONSE RAI 4.3-14

Part 1: Explanation of discrepancy between LRA Table 4.3-2 and Table 6-1 of WCAP-16173-P.

The design cycles for the Unit 1 OBE provided in Table 4.3-2 of the LRA were obtained from Table 4.1-10 of the Unit 1 UFSAR (Reference 1), and both documents indicate the OBE design cycles are 400 for nuclear steam supply system equipment, and 50 for piping. However, Table 6-1 of WCAP-16173-P (Reference 2) shows that the Unit 1 Reactor Vessel, pressurizer and steam generators (original steam generators) were designed for 50 cycles of OBE. It has been confirmed that the replacement Steam Generators are analyzed for 400 cycles of OBE. WCAP-16173-P Table 6-1 correctly shows that the Unit 1 Reactor Vessel and pressurizer are designed to 50 cycles of OBE.

LRA Table 4.3-2 footnote "d" states:

> *"Operating Basis Earthquake design cycles are 400 for nuclear steam supply system equipment and 50 for piping."*

This table footnote is revised to read:

> *"Operating Basis Earthquake design cycles are 400 for nuclear steam supply system equipment (except OBE cycles are 50 for both the Unit 1 reactor vessel and Unit 1 pressurizer) and 50 for piping."*

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

The error in Table 4.1-10 of the Unit 1 UFSAR is being addressed under the FENOC Corrective Action Program.

Part 2: Explanation of how the design cycles for OBE will be considered in the cumulative fatigue usage (CUF) evaluation.

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.

References:

1. Unit 1 UFSAR, Rev. 24.
2. WCAP-16173-P, "Beaver Valley Units 1 and 2 Design Basis Transient Evaluation for License Renewal," March 2004, including Errata dated August 11, 2004.

Question RAI 4.3-15

LRA Section 4.3.3.3.2 states that three of the NUREG/CR-6260 locations of BVPS Unit 1 were re-evaluated in accordance with ASME Section III, 1989 Edition with 1989 addenda to determine the U60. Please provide the design basis transients and the associated cycles used to calculate the U60 in LRA Table 4.3-1.

RESPONSE RAI 4.3-15

Charging System Nozzle:

As noted in LRA Table 4.3-1, the U_{env} for the charging nozzle exceeds 1.0. Reanalysis is required, and, therefore, a regulatory commitment is provided for completion of the reanalysis. See the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

See Attachment 2 to this letter for the Regulatory Commitment List.

Safety Injection Nozzle:

The analysis for the Unit 1 safety injection nozzle used the applicable design transients from the corresponding Unit 2 analysis. Use of the Unit 2 transients is representative due to the similarity of design and operation between the two units. The design cycles for these transients were used in the analysis and envelope the 60-year projected cycles.

Residual Heat Removal System Tee

The analysis for the Unit 1 RHR system tee used the applicable design transients from the corresponding Unit 2 analysis. Use of the Unit 2 transients is representative due to the similarity of design and operation between the two units. The design cycles for these transients were used in the analysis, with one exception. The design cycles for the "RHR operation" transient were increased to account for the projected cycles. With that change, the analyzed cycles envelope the 60-year projected cycles.

Question RAI 4.3-16

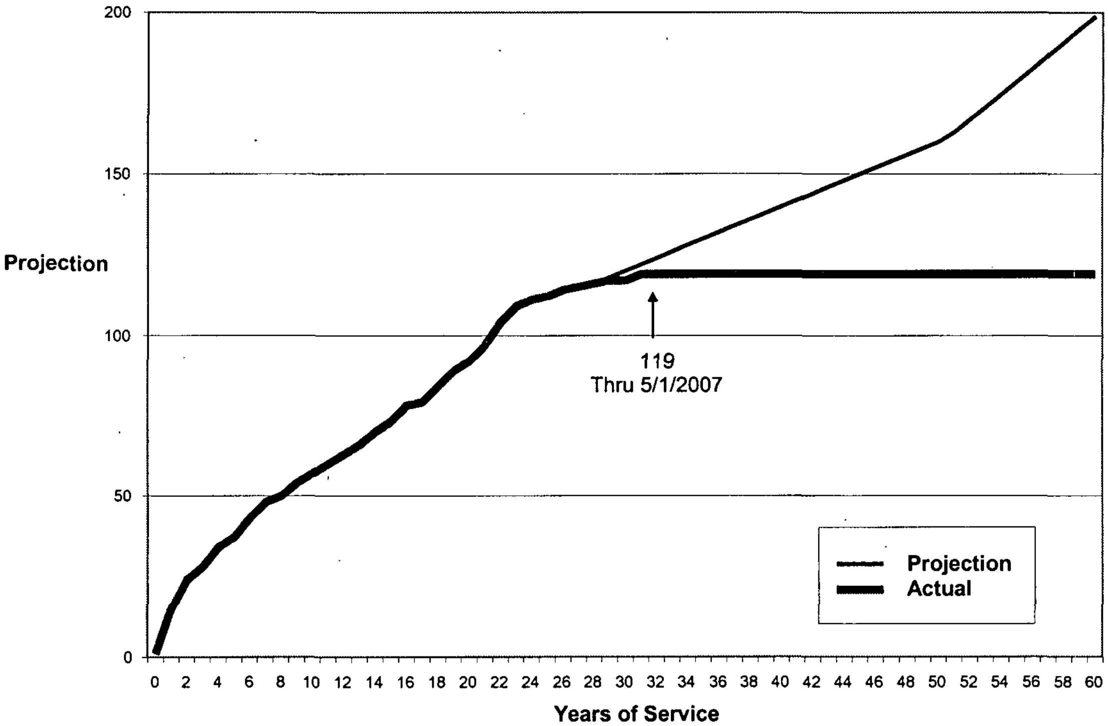
LRA Section 4.3.4 states that histograms were developed based on the last ten years in order to perform an extrapolation for the number of transients that will be accumulated in 60 years of operation for plant heatup and cooldown, and pressurizer cooldown. Please provide the histograms that were developed and describe the method used to extrapolate these cycles to 60 years of operation.

RESPONSE RAI 4.3-16

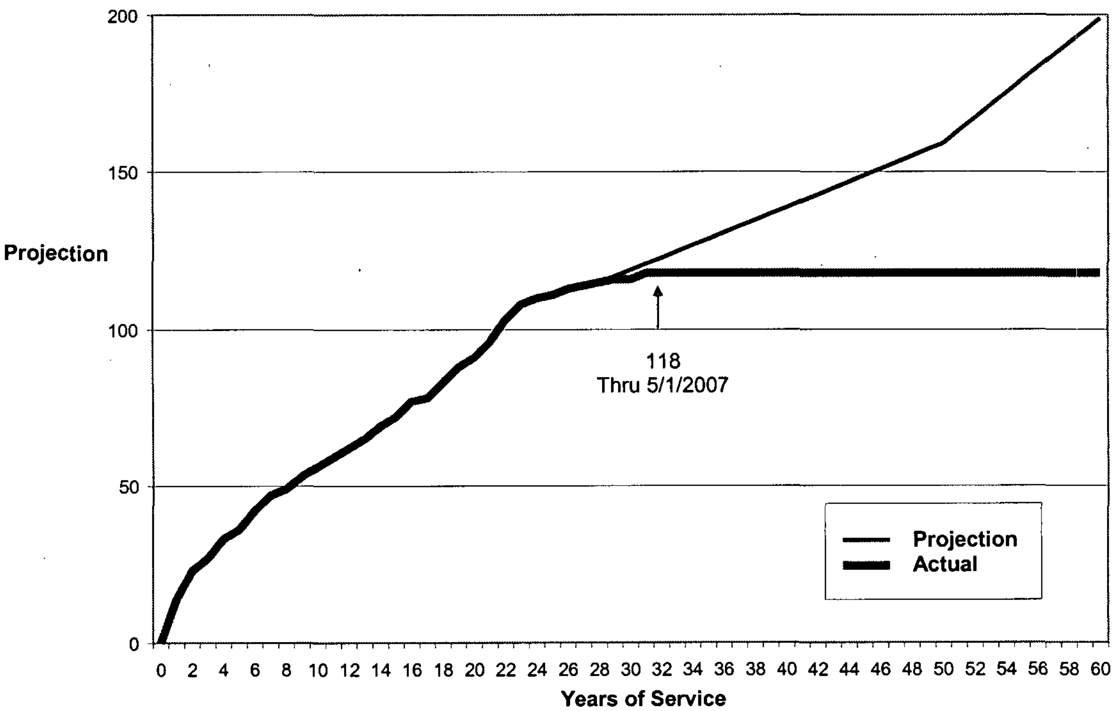
Histograms were developed using accrued cycles through October 15, 2003, for the Unit 1 plant heatup, plant cooldown, and reactor trip transients. For these transients, the 60-year projected operational cycles were based on recent operating experience (i.e., the last ten years). In addition, the Unit 1 reactor trip transient was biased with additional reactor trips as Unit 1 approaches end-of-life (60-years). For all other transients, the accrued cycles were linearly extrapolated to obtain the projected 60-year projected operational cycles. Since the Unit 1 plant heatup, cooldown, and reactor trip transients are expected to approach or exceed the number of design cycles during the sixty year operating life of the unit, the Metal Fatigue of Reactor Coolant Pressure Boundary Program requires monitoring of these transients and provides for preventive and/or corrective action prior to exceeding the fatigue design limit.

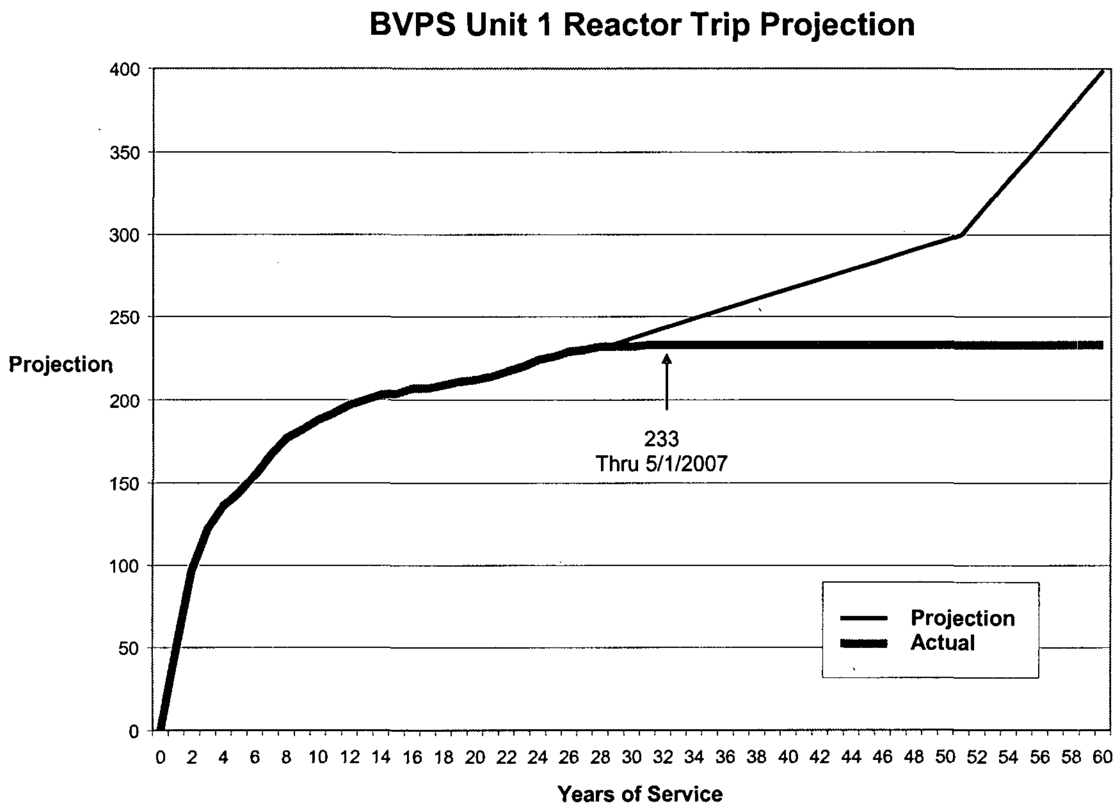
As part of an annual review of the monitored transients, the histograms for the Unit 1 plant heatup, plant cooldown, and reactor trip transients were updated with accrued cycles through May 1, 2007. In addition, all three of the subject transients were biased with additional cycles as Unit 1 approaches end-of-life (60-years). These histograms are provided as follows:

BVPS Unit 1 Heatup Projection



BVPS Unit 1 Cooldown Projection





Section 4.7.4

Question RAI 4.7.4-1

LRA section 4.7.4 states, "The cycle assumptions used in the fatigue crack growth analyses are conservative compared to the BVPS original design cycles", and "Since the 60-year projected operational cycles were used in determining that fatigue crack growth analyses remains valid for 60 years, the Metal Fatigue of Reactor Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations." Please explain why fatigue crack growth analyses are evaluated for the High Energy Line Break Postulation TLAA Evaluation.

RESPONSE RAI 4.7.4-1

Fatigue crack growth analyses are not evaluated for the High Energy Line Break Postulation TLAA Evaluation. The use of "fatigue crack growth analyses" in LRA Section 4.7.4 (4th paragraph) was a typographical error. The correct phrase is "fatigue analyses." It should be noted that LRA Section A.3.6.2, which provides the summary for the Unit 2 High Energy Line Break Postulation TLAA, correctly used the term "fatigue analyses." Corrections to LRA Section 4.7.4 will be made as follows:

LRA Section 4.7.4, "High Energy Line Break Postulation," is revised to replace the phrase *"fatigue crack growth analyses"* (two places) with *"fatigue analyses."*

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

Question RAI 4.7.4-2

Please clarify whether there any Class 1 high energy piping locations with a CUF less than 0.1 by the current design basis where the CUF may exceed 0.1 during the period of extended operation.

RESPONSE RAI 4.7.4-2

LRA Section 4.3.1 identifies that, for most cases, the original design cycles of transients are bounding for the period of extended operation. Since the original design cycles bound the period of extended operation, no reanalysis is required, and CUF values do not change or increase.

The exceptions to bounding transients were identified in LRA Sections 4.3.1.1, 4.3.1.2, and 4.3.1.3. Only Section 4.3.1.1 discusses piping locations. LRA Section 4.3.1.1 states:

> *"The Unit 2 RHR piping and the Unit 2 charging line cycles of operation are projected to exceed their respective design cycles during the period of extended operation."*

FENOC evaluation of the plant Mode history has identified that the Unit 2 RHR operational cycles (60-year projected cycles) will be bounded by the design assumption of 200 occurrences. Details of this evaluation are described in the response to RAI-B.2.27-4. Design CUF values for the RHR system are unchanged for the period of extended operation.

The applicable transients for the Unit 2 charging piping are discussed in the response to RAI-B.2.27-8. The "Isolation of Charging Flow (at power)" transient is predicted to exceed the design cycles prior to the end of plant life. In addition, two lesser transients have been identified for this piping, "Isolation of Charging Flow (during Mode 4)" and "Isolation of Letdown Flow (during Mode 4)". The applicable piping analysis will be revised, as described in the FENOC response to RAI-4.3-3, to incorporate the increased cycles for the first transient and to include the other two transients. A regulatory commitment is provided for completion of the Unit 2 charging piping reanalysis. See the FENOC response to RAI-4.3-3 that addresses the regulatory commitment.

See Attachment 2 to this letter for the Regulatory Commitment List.

In summary, there is currently no Class 1 high energy piping location where the design CUF has been increased. By extension, there are no locations where the CUF had been less than 0.1 and are now greater than 0.1. Therefore, no new pipe rupture locations are required to be postulated.

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC. They are described only as information and are not Regulatory Commitments. Please notify Mr. Clifford I. Custer, Project Manager, Fleet License Renewal, at (724) 682-7139, of any questions regarding this document or associated Regulatory Commitments.

Regulatory Commitment	Due Date
1. The results of the refined analyses for environmentally-assisted fatigue (EAF) of the NUREG/CR-6260 locations for which the cumulative usage factor, including environmental effects (U_{env}), exceeded the design code allowable value of 1.0, and a summary of how the calculations were performed, will be submitted to the NRC no later than October 15, 2008. In addition, FENOC plans to include a minimum of 50 cycles of the operational basis earthquake (OBE) transient for these EAF analyses; if it is necessary to use less than 50 cycles of the OBE transient, this will be reported with the results of the analyses. The applicable NUREG/CR-6260 locations are listed as follows: • Unit 1 Surge Line to Hot Leg Nozzle • Unit 1 Charging System Nozzle • Unit 2 Surge Line to Hot Leg Nozzle • Unit 2 Charging System Nozzle • Unit 2 Safety Injection System Nozzle • Unit 2 Residual Heat Removal System Piping It is anticipated that the Unit 1 Surge Line to Hot Leg Nozzle refined analysis will be successful. However, should the refined analysis not be successful, as an alternative, a fracture mechanics	October 15, 2008

analysis of the Unit 1 Surge Line to Hot Leg Nozzle will be performed using the general methodology as described in NUREG/CR-6934. The results of the alternative analysis, if needed, will be submitted to the NRC no later than October 15, 2008.

ENCLOSURE A

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

Letter L-08-209

Amendment No. 15 to the
BVPS License Renewal Application

Sections Affected

Table 4.1-1	Section A.2.3.3.2
Section 4.3.1.1	Section A.2.7
Section 4.3.1.2	Section A.3.3.1.1
Section 4.3.2.1	Section A.3.3.1.2
Section 4.3.2.2	Section A.3.3.2.1
Section 4.3.3.2.3	Section A.3.3.2.2
Section 4.3.3.3.3	Section A.3.3.3.2
Table 4.3-2	Section A.3.3.3.3
Section 4.7.4	Table A.4-1
Section A.1.27	Table A.5-1
Section A.2.3.2.2	Section B.2.27
Section A.2.3.3.1	

Enclosure A 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 listing, 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 4.1-1	**Page 4.1-4**	**8th, 9th, 12th and 13th Rows, Disposition**

The "Disposition" column in LRA Table 4.1-1, "List of BVPS Time-Limited Aging Analyses and Resolution," requires changes based on the FENOC responses to RAIs B.2.27-10, 4.3-6, 4.3-9, and 4.3-12. Table 4.1-1 "Disposition" column is revised to read:

Time-Limited Aging Analysis Description	*Disposition* [a]	LRA Section
Metal Fatigue	--	4.3
Class 1 fatigue	*(i) (iii)*	4.3.1
Non-Class 1 fatigue	--	4.3.2
Piping and In-Line Components	*(i) (ii) ~~(iii)~~*	4.3.2.1
Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings	*(i) ~~(iii)~~*	4.3.2.2
Generic Industry Issues on Fatigue	--	4.3.3
Thermal Stresses in Piping Connected to Reactor Coolant Systems (NRC Bulletin 88-08)	*(i)*	4.3.3.1
Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)	*(iii) ~~(i)~~*	4.3.3.2
Effects of Primary Coolant Environment on Fatigue Life	*~~(ii)~~ (iii)*	4.3.3.3

a. The disposition of the time-limited aging analysis for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii):
(i) = The analysis remains valid for the period of extended operation;
(ii) = The analysis has been projected to the end of the period of extended operation; or
(iii) = The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.1.1	**Page 4.3-3**	**1st Paragraph, 4th Sentence**

LRA Section 4.3.1.1, "Unit 2 RHR Piping and Unit 2 Charging Line," requires revision to remove the Unit 2 Residual Heat Removal (RHR) piping from the discussion because revised projections show that the design cycles of 200 bounds the revised projected cycles. LRA Section 4.3.1.1 is revised to read:

*"**4.3.1.1 ~~Unit 2 RHR Piping and~~ Unit 2 Charging Line***

The ~~Unit 2 RHR piping and the~~ Unit 2 charging line cycles of operation are projected to exceed their respective design cycles during the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be used to monitor the transient cycles for the ~~Unit 2 RHR piping and the~~ Unit 2 charging line. As required by the program, corrective actions will be taken (including reanalysis, repair or replacement) such that the design basis of the Unit 2 charging line is ~~these components are~~ not exceeded for the period of extended operation. Therefore, ~~the Unit 2 RHR piping and~~ the Unit 2 charging line fatigue TLAA~~s have~~ has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)."

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Section 4.3.1.2	**Page 4.3-4**	**1st Paragraph, 4th Sentence**

LRA Section 4.3.1.2, "Unit 2 Steam Generator Manway Bolts and Tubes," requires revision to identify that the program that will be used to manage the Unit 2 steam generator manway bolts and steam generator tubes (U-bend fatigue) fatigue TLAAs is changed from the Steam Generator Tube Integrity Program to the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section 4.3.1.2, 1st paragraph on LRA page 4.3-4, is revised to read:

"The Unit 2 steam generator secondary manway bolts and the steam generator tubes fatigue analyses are based on a 40-year life (current operating license expires in 2027). In the Extended Power Uprate T_{AVG} coastdown analysis for the secondary manway bolts, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. In the Uprate analysis for the U-bends, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. ~~As part of the Steam Generator Tube Integrity~~ The <u>Metal Fatigue of Reactor Coolant Pressure Boundary</u> Program (Section <u>B.2.27</u> ~~B.2.38~~) <u>will be enhanced to include a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation.</u> ~~BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design basis of these components is not exceeded for the period of extended operation~~. Therefore, the Unit 2 steam generator secondary manway bolts and the Unit 2 steam generator tubes fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.2.1	**Pages 4.3-5 & 6**	**4th Paragraph**

LRA Section 4.3.2.1, "Piping and In-Line Components," requires changes to address the fact that the Unit 2 Emergency Diesel Generator Air Start System piping design analysis was revised to incorporate observed temperatures as a new load case. The revised design analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, Section 4.3.2.1, 4th paragraph, is revised to read:

"The Unit 2 EDG Air Start system is designed as a stand-by system. The compressor runs to fill the air start tank and does not run again until the tank needs to be either topped off or refilled after discharge. The act of compressing the air is the only source of heat for this piping. Measurement of discharge temperature during compressor operation indicates that the compressors, bolting, discharge piping and valves may exceed the threshold temperature for thermal fatigue during compressor operation. Confirmation of the full-temperature cycles of the Unit 2 EDG Air Start System would be a time consuming process and would not account for variables such as the leak tightness of the air system. Therefore, the design analysis for the piping has been revised to incorporate the observed temperatures as a new load case. The stress levels for this thermal load case are below the endurance limit for the piping material. In other words, the revised analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, the Unit 2 EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii). ~~*The Unit 2 EDG Air Start System contains components potentially subject to fatigue. As part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27), BVPS will perform an assessment to determine whether the full temperature cycles limit will be exceeded for 60 years of operation. Corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the full-temperature cycles of the Unit 2 EDG Air Start System are not exceeded for the period of extended operation. Therefore, the Unit 2 EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~"

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.2.2	**Pages 4.3-6 & 7**	**Entire Section**

LRA Section 4.3.2.2, "Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings," 1st paragraph, last sentence, is revised to discuss "cumulative fatigue damage" instead of "cracking due to fatigue." The remainder of Section 4.3.2.2 is revised to address the fact that auxiliary system heat exchangers at BVPS Unit 1 and Unit 2 are installed on the Class 2 portions of their respective systems, and the primary sides of these heat exchangers are designed to American Society of Mechanical Engineers (ASME) III, Class 2 rules. Therefore, detailed fatigue analyses (per ASME Class 1 rules) of these heat exchangers are not required. Section 4.3.2.2, is revised to read:

"Non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings are typically designed in accordance with ASME, Section VIII, or ASME, Section III, Subsection NC or ND (i.e., Class 2 or 3). Some tanks and pumps are designed to other industry codes and standards (such as American Water Works Association and Manufacturer's Standardization Society), reactor designer specifications, and architect engineer specifications. Only ASME, Section VIII, Division 2, and ASME, Section III, Subsection NC-3200 design codes include fatigue design requirements. Due to the conservatism in ASME, Section VIII, Division 1, and ASME, Section III, NC-3100/ND-3000, detailed fatigue analyses are not required. If cyclic loading and fatigue usage could be significant, the component designer is expected to specify ASME, Section VIII, Division 2 or NC-3200. For components where there is no required fatigue analysis, <u>cumulative fatigue damage</u> ~~cracking due to fatigue~~ is not an aging effect requiring management.

Fatigue analysis is not required for ASME, Section VIII, Division I, Section III, NC-3100, or ND vessels. It is also not required for NC/ND pumps and storage tanks (<15 psig). The design specification identifies the applicable design code for each component.

<u>For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the original design (usually 7,000 cycles, as described in Section 4.3.2.1), the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required.</u>

FENOC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

~~Only the Unit 2 non-regenerative (letdown), regenerative, and RHR heat exchangers were identified as having a fatigue TLAA, and are dispositioned as described in the following text.~~

~~Unit 2 Non-regenerative (Letdown) Heat Exchanger~~

~~The Unit 2 non-regenerative (letdown) heat exchanger is designed to ASME, Section III, Class C (tubes) and ASME, Section VIII, Division 1 (shell). The transients for the Unit 2 non-regenerative (letdown) heat exchanger are defined in Westinghouse Equipment Specification G-679150 [Reference 4.3-1]. The fatigue analysis associated with the Unit 2 non-regenerative (letdown) heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be used to monitor the Unit 2 non-regenerative (letdown) heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 non-regenerative (letdown) heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 non-regenerative (letdown) heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~

~~Unit 2 Regenerative Heat Exchanger~~

~~The Unit 2 regenerative heat exchanger was built to ASME, Section III, Class 2. The transients for the Unit 2 regenerative heat exchanger are defined in Westinghouse Equipment Specification G-679150. The fatigue analysis associated with the Unit 2 regenerative heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be used to monitor the Unit 2 regenerative heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 regenerative heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 regenerative heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~

~~***Unit 2 Residual Heat Removal (RHR) Heat Exchangers***~~

~~*The tube side of the Unit 2 RHR heat exchangers were designed in accordance with ASME Section III, Class 2. The shell side of these heat exchangers were designed in accordance with ASME Section III, Class 3. The transients for the Unit 2 RHR heat exchangers are defined in Westinghouse Equipment Specification G-679150. The fatigue analyses associated with the Unit 2 RHR heat exchangers are not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be used to monitor the Unit 2 RHR heat exchangers transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 RHR heat exchangers are not exceeded for the period of extended operation. Therefore, the Unit 2 RHR heat exchangers fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~"

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.3.3.2.3	**Page 4.3-9**	**Entire Section**

LRA Section 4.3.3.2.3, "Unit 1 and Unit 2 Disposition for License Renewal," requires revision to identify the FENOC strategy for ongoing validation of the assumptions used in the pressurizer surge line thermal stratification TLAAs using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section 4.3.3.2.3 is revised to read:

"Both WCAP-12727 and WCAP-12093 determine the effect of thermal stratification through the imposition of defined thermal stratification cycles upon the stress and fatigue evaluations. The stratification cycles which are incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients. Therefore, these NRC Bulletin 88-11 analyses are TLAAs in accordance with 10 CFR 54.3. <u>The BVPS original design basis transients including design cycles for the RCS are identified in Table 4.3-2 along with the projected operational cycles that BVPS anticipates will occur for 60 years of plant life. Table 4.3-2 demonstrates that the 200 heatup and cooldown cycles are bounding for 60 years of operation. Since 60-year projected operational cycles were used in determining that the 200 heatup and cooldown transient assumption remains bounding for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate this assumption. Therefore, the Unit 1 and Unit 2 pressurizer surge line thermal stratification TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).</u> ~~Section 4.3.4 demonstrates that the 200 heatup and cooldown cycles are bounding for 60 years of operation. Therefore, the Unit 1 and Unit 2 pressurizer surge line fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).~~"

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

LRA Section 4.3.3.3.3, "Unit 1 and Unit 2 Disposition for License Renewal," requires revision to address the details for ongoing validation of the assumptions used in the environmentally-assisted fatigue evaluations for the NUREG/CR-6260 locations using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section 4.3.3.3.3 is revised in its entirety to read:

"At several NUREG/CR-6260 locations (Unit 1 pressurizer surge line hot leg nozzle and charging system nozzle; Unit 2 pressurizer surge line hot leg nozzle, charging system nozzle, safety injection system nozzle, and RHR system piping), U_{env} *exceeded the design code allowable limit of 1.0.*

Further refinement of the fatigue analyses for these NUREG/CR-6260 locations will be performed to lower the U_{env} *to less than 1.0. While it is anticipated that the Unit 1 surge line to hot leg nozzle refined analysis will be successful, as an alternative, a fracture mechanics analysis will be performed using the general methodology as described in 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.*

The U_{env} *at the other NUREG/CR-6260 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), 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), Uenv 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
Table 4.3-2	**Page 4.3-17**	**Table Footnote "d"**

LRA Table 4.3-2, "Design and 60-Year Projected Operational Cycles," requires a revision to Table footnote "d" to clarify the number of operating basis earthquake (OBE) design cycles for specific nuclear steam supply system equipment and piping. Table 4.3-2, footnote "d", is revised to read:

"d. Operating Basis Earthquake design cycles are 400 for nuclear steam supply system equipment <u>(except Operating Basis Earthquake design cycles are 50 for the Unit 1 Reactor Vessel and Unit 1 pressurizer)</u> and 50 for piping."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 4.7.4	**Page 4.7-7**	**2nd Paragraph, 3rd & 6th Sentences**

LRA Section 4.7.4, "High Energy Line Break Postulation," requires revision to delete the term "crack growth" when discussing fatigue analyses related to high energy line break of Unit 2 Class 1 systems. Section 4.7.4, 2nd paragraph on LRA page 4.7-7, is revised to read:

"For the Unit 2 Class 1 systems, Regulatory Guide 1.46 states that postulated break locations be determined, in part, using any intermediate locations between terminal ends where the cumulative usage factor derived from the piping fatigue analysis under the loadings associated with specified seismic events and operational plant conditions exceeded 0.1. These fatigue evaluations are TLAAs since they are based on a set of fatigue transients that are based on the life of the plant. The cycle assumptions used in the fatigue ~~crack growth~~ analyses are conservative compared to the BVPS original design cycles [Reference 4.7-18]. The BVPS original design basis transients including design cycles for the RCS are identified in Table 4.3-2 along with the projected operational cycles that BVPS anticipates will occur for 60 years of plant life. BVPS has reviewed the design cycles against the 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the fatigue ~~crack growth~~ analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate the assumptions used in the evaluations. Therefore, the piping fatigue analyses used for determining the postulation of break locations in Class 1 lines remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.1.27	**Pages A.1-12**	**Entire Section**

LRA Section A.1.27, "Metal Fatigue of Reactor Coolant Pressure Boundary Program," requires revision to include the results of an evaluation of each of the 10 aging management program elements described in NUREG-1801, Section X.M1, including enhancements. LRA Section A.1.27 is replaced in its entirety, and is revised to read:

"A.1.27 METAL FATIGUE OF THE REACTOR COOLANT PRESSURE BOUNDARY PROGRAM

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is a time-limited aging analysis (TLAA) program that uses preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The preventive measures consist of monitoring and tracking critical thermal and pressure transients for RCS components to prevent the fatigue design limit from being exceeded. Prior to exceeding the fatigue design limit, preventive and/or corrective actions are triggered by the program.

In addition, environmental effects are evaluated in accordance with NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components *[Reference B.3-18], and the guidance of EPRI Technical Report MRP-47,* Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application *[Reference B.3-19]. Selected components are evaluated using material specific guidance presented in NUREG/CR-6583,* Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels *[Reference B.3-20], and in NUREG/CR-5704,* Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels *[Reference B.3- 21].*

Aging Management Program Elements

The results of an evaluation of each of the 10 aging management program elements described in NUREG-1801, Section X.M1, are provided as follows:

- *Scope of Program*

  The program tracks critical transient cycles to ensure RCS components remain within their design fatigue usage limits. This program utilizes the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded.

  Fatigue analyses validated by this program include nuclear steam supply system (NSSS) equipment for BVPS Unit 1 and Unit 2 as well as the ASME Class 1 portions of the primary piping for Unit 2. Also, included is BVPS Unit 1 Surge Line.

  The program addresses the effects of the reactor coolant environment on component fatigue life by including, within the program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260.

- *Preventive Actions*

  The program provides for monitoring and tracking critical thermal and pressure transients for RCS components to prevent the fatigue design limit from being exceeded. Critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the sixty year operating life of the units. These critical transients include plant heatup, plant cooldown, reactor trip from full power (Unit 1 only), inadvertent auxiliary spray, safety injection activation (Unit 1 only), and RCS cold overpressurization. Supplemental transients were also identified by the program for monitoring. These supplemental transients include pressurizer insurge transient, selected Chemical and Volume Control System transients, Auxiliary Feedwater injections and RHR actuation.

  The number of critical transient occurrences is periodically reviewed (annual basis) to determine if there are any adverse trends; adverse conditions or deficient conditions with the primary objective of initiating evaluation of adverse trends and adverse conditions early to prevent the possibility of a deficient condition. Adverse Trend is an observed increase in the rate of critical transient occurrences that, if it continued, would result in exceeding the fatigue cycle design limit number of transients prior to the end of the Unit's 60-year operating life. Adverse Condition is a condition in which the number of actual transient occurrences exceeds 80% of the fatigue cycle design limit number of occurrences. Deficient Condition is a condition in which the current number of actual transient occurrences exceeds the fatigue cycle design limit number of occurrences. Adverse trends and adverse conditions are evaluated by Engineering to determine if and when more rigorous analysis or alternate resolutions are required. Deficient conditions are addressed under the BVPS Corrective Action Program.

- Parameters Monitored / Inspected

  The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.

  The program, for the most part, is a transient cycle counting program and does not require analysis of operational data (local monitoring) to obtain an effective number of transients.

  The WESTEMS™ Integrated Diagnostics and Monitoring System analysis for the surge line to hot leg nozzles is based on past occurrences of various transients along with what are believed to be conservative assumptions of future transients. In this case, the input assumptions will be verified by periodic reanalysis using updated plant history files.

- Detection of Aging Effects

  The program requires the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded. When the accrued operational cycles approach the component design cycles, corrective action is required by the program to ensure the design cycle limit is not exceeded. If the corrective action has an impact on the cumulative fatigue usage factor (CUF), an updated CUF will be generated.

- Monitoring and Trending

  The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks *fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.*

- *Acceptance Criteria*

  The program verifies that the fatigue usage remains below the design code limit considering environmental fatigue effects as described under the program description.

- *Corrective Actions*

  This element is discussed in LRA Section B.1.3.

- *Confirmation Process*

  This element is discussed in LRA Section B.1.3.

- *Administrative Controls*

  This element is discussed in LRA Section B.1.3.

- *Operating Experience*

  Concerns for the overall health of the transient/cycle counting program were documented using the FENOC Corrective Action Program. Corrective actions included identifying a program owner, developing an administration program document and updating it to incorporate responsibilities, improving cycle counting, and establishing a process for engineering to evaluate plant data. Fatigue monitoring to date indicates that the number of design transient events assumed in the original design analysis will be sufficient for a 60-year operating period. The program has remained responsive to emerging issues and concerns, particularly the pressurizer surge and spray nozzle, hot leg surge nozzle, and surge line transients.

  For example, in 2002, a Westinghouse evaluation identified that the BVPS Unit 2 letdown, charging, and excess letdown piping could potentially exceed their design allowable cycle counts for several design transients. However, further evaluation of existing plant operations and the physical separation distance of the letdown and excess letdown piping demonstrated that no further evaluation of the letdown or excess letdown piping was required for current operation or for the period of extended operation. A re-analysis of the charging piping was required to account for the appropriate transients for a 60-year plant life.

> *This responsiveness to emerging issues and continued program improvements provide evidence that the program will remain effective for managing cumulative fatigue damage for passive components.*

Conclusion

Continued implementation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program provides reasonable assurance that the aging effects will be managed so that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.2.3.2.2	**Page A.2-7**	**New Subsection**

LRA Section A.2.3.2, "Non-Class 1 Fatigue Evaluations," requires the addition of a new sub-Section A.2.3.2.2, "Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings," to address the fact that auxiliary system heat exchangers at BVPS Unit 1 are installed on the Class 2 portions of their respective systems, and the primary sides of these heat exchangers are designed to American Society of Mechanical Engineers (ASME) III, Class 2 rules. Therefore, detailed fatigue analyses (per ASME Class 1 rules) of these heat exchangers are not required. New Section A.2.3.2.2, is added, which reads:

"Non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings are typically designed in accordance with ASME, Section VIII, or ASME, Section III, Subsection NC or ND (i.e., Class 2 or 3). Some tanks and pumps are designed to other industry codes and standards (such as American Water Works Association and Manufacturer's Standardization Society), reactor designer specifications, and architect engineer specifications. Only ASME, Section VIII, Division 2, and ASME, Section III, Subsection NC-3200 design codes include fatigue design requirements. Due to the conservatism in ASME, Section VIII, Division 1, and ASME, Section III, NC-3100/ND-3000, detailed fatigue analyses are not required. If cyclic loading and fatigue usage could be significant, the component designer is expected to specify ASME, Section VIII, Division 2 or NC-3200. For components where there is no required fatigue analysis, cumulative fatigue damage is not an aging effect requiring management.

Fatigue analysis is not required for ASME, Section VIII, Division I, Section III, NC-3100, or ND vessels. It is also not required for NC/ND pumps and storage tanks (<15 psig). The design specification identifies the applicable design code for each component.

For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the original design (usually 7,000 cycles, as described in Section A.2.3.2.1), the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required. FENOC evaluated the validity of this assumption for 60 years of plant _operation. The results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs 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.2.3.3.1	**Page A.2-8**	**4th Paragraph**

LRA Section A.2.3.3.1, "Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)," requires revision to address the details for ongoing validation of the heatup and cooldown assumption in the pressurizer surge line thermal stratification evaluation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section A.2.3.3.1, 4th paragraph, is revised to read:

"The 200 heatup and cooldown transients were determined to remain bounding for the period of extended operation. Since 60-year projected operational cycles were used in determining that the 200 heatup and cooldown transient assumption remains bounding for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate this assumption. Therefore, the Unit 1 pressurizer surge line fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) ~~10 CFR 54.21(c)(1)(i)~~."

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

LRA Section A.2.3.3.2, "Effects of Primary Coolant Environment on Fatigue Life," requires revision to address the details for ongoing validation of the assumptions used in the environmentally-assisted fatigue evaluations for the NUREG/CR-6260 locations using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The last 3 paragraphs on LRA page A.2-10 of Section A.2.3.3.2 are revised to read:

"At two locations (pressurizer surge line and charging system nozzle), U_{env} exceeded the design code allowable limit of 1.0. Further refinement of the fatigue analyses for these NUREG/CR-6260 locations will be performed to lower the U_{env} to less than 1.0. While it is anticipated that the surge line to hot leg nozzle refined analysis will be successful, as an alternative, a fracture mechanics analysis will be performed using the general methodology as described in NUREG/CR-6934, "Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping – A Basis for Improvements to ASME Code Section XI Appendix L" [Reference A.2-28].

The U_{env} at the other NUREG/CR-6260 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.

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).

~~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.2.7	**Page A.2-20**	**New Reference A.2-28**

LRA Section A.2.7, "Appendix A.2 References," requires revision to include a new reference in LRA Section A.2.3.3.2. Section A.2.7 is revised to add new reference A.2-28, which reads:

"A.2-28 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."

Section A.3.3.1.1 **Page A.3-6** **1st Paragraph, 4th Sentence**

LRA Section A.3.3.1.1, "Unit 2 RHR Piping and Unit 2 Charging Line," requires revision to remove the Unit 2 Residual Heat Removal (RHR) piping from the discussion because revised projections show that the design cycles of 200 bounds the revised projected cycles. LRA Section A.3.3.1.1 is revised to read:

"A.3.3.1.1 ~~Unit 2 RHR Piping and~~ Unit 2 Charging Line

The ~~RHR piping and the~~ charging line cycles of operation are projected to exceed their respective design cycles during the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the transient cycles for the ~~Unit 2 RHR piping and the~~ Unit 2 charging line. As required by the program, corrective actions will be taken (including reanalysis, repair or replacement) such that the design basis of the Unit 2 charging line is ~~these components are~~ not exceeded for the period of extended operation. Therefore, ~~the Unit 2 RHR piping and~~ the charging line fatigue TLAA~~s have~~ has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.3.3.1.2	**Page A.3-7**	**2nd Paragraph, 4th Sentence**

LRA Section A.3.3.1.2, "Unit 2 Steam Generator Manway Bolts and Tubes," requires revision to identify that the program that will be used to manage the Unit 2 steam generator manway bolts and steam generator tubes (U-bend fatigue) fatigue TLAAs is changed from the Steam Generator Tube Integrity Program to the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section A.3.3.1.2, 2nd paragraph, 4th sentence, is revised to read:

"The Unit 2 steam generator secondary manway bolts and the steam generator tubes fatigue analyses are based on a 40-year life (i.e., to 2027). In the Extended Power Uprate T_{AVG} coastdown analysis for the secondary manway bolts, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. In the Uprate analysis for the U-bends, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. The Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) will be enhanced to include a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation. ~~As part of the Steam Generator Tube Integrity Program, BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design bases of the these components are not exceeded for the period of extended operation.~~ Therefore, the steam generator secondary manway bolts and the steam generator tubes fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.3.3.2.1	**Page A.3-8**	**4th Paragraph**

LRA Section A.3.3.2.1, "Piping and In-Line Components," requires changes to address the fact that the Unit 2 Emergency Diesel Generator Air Start System piping design analysis was revised to incorporate observed temperatures as a new load case. The revised design analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, Section A.3.3.2.1, 4th paragraph, is revised to read:

"The Unit 2 EDG Air Start system is designed as a stand-by system. The compressor runs to fill the air start tank and does not run again until the tank needs to be either topped off or refilled after discharge. The act of compressing the air is the only source of heat for this piping. Measurement of discharge temperature during compressor operation indicates that the compressors, bolting, discharge piping and valves may exceed the threshold temperature for thermal fatigue during compressor operation. Confirmation of the full-temperature cycles of the Unit 2 EDG Air Start System would be a time consuming process and would not account for variables such as the leak tightness of the air system. Therefore, the design analysis for the piping has been revised to incorporate the observed temperatures as a new load case. The stress levels for this thermal load case are below the endurance limit for the piping material. In other words, the revised analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, the Unit 2 EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii). ~~The EDG Air Start System contains components potentially subject to fatigue. As part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program, BVPS will perform an assessment to determine whether the full-temperature cycles limit will be exceeded for 60 years of operation. Corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the full-temperature cycles of the EDG Air Start System are not exceeded for the period of extended operation. Therefore, the EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~"

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.3.3.2.2	**Page A.3-9**	**Entire Section**

LRA Section A.3.3.2.2, "Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings," 1st paragraph, last sentence, is revised to discuss "cumulative fatigue damage" instead of "cracking due to fatigue." The remainder of Section A.3.3.2.2 is revised to address the fact that auxiliary system heat exchangers at BVPS Unit 2 are installed on the Class 2 portions of their respective systems, and the primary sides of these heat exchangers are designed to American Society of Mechanical Engineers (ASME) III, Class 2 rules. Therefore, detailed fatigue analyses (per ASME Class 1 rules) of these heat exchangers are not required. Section A.3.3.2.2, including Subsections A.3.3.2.2.1, A.3.3.2.2.2 and A.3.3.2.2.3, is revised to read:

"Non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings are typically designed in accordance with ASME Section VIII or ASME Section III, Subsection NC or ND (e.g., Class 2 or 3). Some tanks and pumps are designed to other industry codes and standards (such as American Water Works Association and Manufacturer's Standardization Society), reactor designer specifications, and architect engineer specifications. Only ASME Section VIII, Division 2, and ASME Section III, Subsection NC-3200, design codes include fatigue design requirements. Due to the conservatism in ASME Section VIII Division 1 and ASME Section III NC-3100/ND-3000 detailed fatigue analyses are not required. If cyclic loading and fatigue usage could be significant, the component designer is expected to specify ASME Section VIII Division 2 or NC-3200. For components where there is no required fatigue analysis, <u>cumulative fatigue damage</u> ~~cracking due to fatigue~~ is not an aging effect requiring management.

Fatigue analysis is not required for ASME Section VIII Division I, Section III NC-3100 or ND vessels. It is also not required for NC/ND pumps and storage tanks (<15 psig). The design specification identifies the applicable design code for each component.

<u>For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the original design (usually 7,000 cycles, as described in Section A.3.3.2.1), the component is suitable for extended operation. If the number of equivalent full-temperature</u>

cycles exceeds the limit, evaluation of the individual stress calculations will be required. FENOC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i). ~~*Only the Unit 2 non-regenerative (letdown), regenerative, and RHR heat exchangers were identified as having fatigue TLAAs, and are dispositioned as described in the following text.*~~

~~*A.3.3.2.2.1 Non-regenerative (Letdown) Heat Exchanger*~~

~~*The Unit 2 non-regenerative (letdown) heat exchanger is designed to ASME, Section III, Class C (tubes) and ASME, Section VIII, Division 1 (shell). The transients for the non-regenerative (letdown) heat exchanger are defined in Westinghouse Equipment Specification G-679150 [Reference A.3-9]. The fatigue analysis associated with the Unit 2 non-regenerative (letdown) heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the Unit 2 non-regenerative (letdown) heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 non-regenerative (letdown) heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 non-regenerative (letdown) heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~

~~*A.3.3.2.2.2 Regenerative Heat Exchanger*~~

~~*The Unit 2 regenerative heat exchanger was built to ASME, Section III, Class 2. The transients for the Unit 2 regenerative heat exchanger are defined in Westinghouse Equipment Specification G-679150. The fatigue analysis associated with the Unit 2 regenerative heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the regenerative heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the regenerative heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 regenerative heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*~~

~~*A.3.3.2.2.3 Residual Heat Removal (RHR) Heat Exchangers*~~

~~*The tube side of the Unit 2 RHR heat exchangers were designed in accordance with ASME Section III, Class 2. The shell side of these heat exchangers were*~~

~~*designed in accordance with ASME Section III, Class 3. The transients for the RHR heat exchangers are defined in Westinghouse Equipment Specification G-679150. The fatigue analyses associated with the RHR heat exchangers are not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the Unit 2 RHR heat exchangers transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 RHR heat exchangers are not exceeded for the period of extended operation. Therefore, the Unit 2 RHR heat exchangers fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).*"~~

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Section A.3.3.3.2	**Page A.3-12**	**3rd Paragraph**

LRA Section A.3.3.3.2, "Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)," requires revision to address the details for ongoing validation of the heatup and cooldown assumption in the pressurizer surge line thermal stratification evaluation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Section A.3.3.3.2, 3rd paragraph on LRA page A.3-12, is revised to read:

"The 200 heatup and cooldown transients were determined to remain bounding for the period of extended operation. Since 60-year projected operational cycles were used in determining that the 200 heatup and cooldown transient assumption remains bounding for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate this assumption. Therefore, the Unit 2 pressurizer surge line thermal stratification ~~fatigue~~ TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) ~~10 CFR 54.21(c)(1)(i)~~."

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section A.3.3.3.3	**Pages A.3-13 & 14**	**Last 3 Paragraphs of Section**

LRA Section A.3.3.3.3, "Effects of Primary Coolant Environment on Fatigue Life," requires revision to address the details for ongoing validation of the assumptions used in the environmentally-assisted fatigue evaluations for the NUREG/CR-6260 locations using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The last 3 paragraphs of Section A.3.3.3.3 are revised to read:

"At four ~~three~~ locations (pressurizer surge line hot leg nozzle, charging system nozzle, safety injection system nozzle and RHR system piping), U_{env} exceeded the design code allowable limit of 1.0. Further refinement of the fatigue analyses for these NUREG/CR-6260 locations will be performed to lower the U_{env} to less than 1.0.

The U_{env} at the other NUREG/CR-6260 locations (reactor vessel shell and lower head, and reactor vessel inlet and outlet nozzles), 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).

~~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 fatigue 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
Table A.4-1	**Page A.4-9**	**Item Number 25**

LRA Table A.4-1, "Unit 1 License Renewal Commitments," Item Number 25 regarding the Unit 1 NUREG/CR-6260 locations, is no longer necessary because of a new FENOC Regulatory Commitment (see Attachment 2 to this letter). Therefore, Item Number 25 in LRA Table A.4-1 is deleted. Item Number 25 is revised to read:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
25	*[Deleted]* ~~*Of the NUREG/CR-6260 locations, the U_{env} (60 years) of the Unit 1 surge line hot leg nozzle and charging nozzle exceeded the design code allowable of 1.0. For these two locations, BVPS will implement one or more of the following:*~~ • ~~*Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*~~ • ~~*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*~~ • ~~*Repair or replacement of the affected locations.*~~	~~*January 29, 2016*~~	~~*LRA*~~	~~*A.2.3.3.2*~~ ~~*4.3.3.3.3*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A.4-1	**Page A.4-9**	**New Item Number 25**

In LRA Table A.4-1, "Unit 1 License Renewal Commitments," a new Item Number 25, previously deleted (this Enclosure), is added to identify a new FENOC License Renewal Future Commitment regarding the enhancements to the Metal Fatigue of the Reactor Coolant Pressure Boundary Program. Therefore, new Item Number 25 in LRA Table A.4-1 is created, and reads:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
25	*Enhance the Metal Fatigue of the Reactor Coolant Pressure Boundary Program to:* • *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.* • *Add a requirement that provides for monitoring of the Unit 1 RHR activation transient and establishes an administration limit of 600 cycles for the transient.*	*January 29, 2016*	*LRA*	*B.2.27*

	• *Add a requirement to monitor Unit 1 transients where the 60 year projected cycles are used in the environmental fatigue evaluations, and establish an administration limit that is equal to or less than the 60-year projected cycles number.* *~~[Deleted]~~*			

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A.5-1	**Page A.5-9**	**Item Number 26**

LRA Table A.5-1, "Unit 2 License Renewal Commitments," Item Number 26, is no longer necessary due to the fact that the Metal Fatigue of Reactor Coolant Pressure Boundary Program is revised (this Enclosure, Section B.2.27) to include the information from this commitment (#26) as an enhancement. A new License Renewal Future Commitment (this Enclosure) will be created to capture the Metal Fatigue of Reactor Coolant Pressure Boundary Program enhancements. Therefore, Item Number 26 of LRA Table A.5-1 is deleted. Item Number 26 is revised to read:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
26	*[Deleted]* ~~*The Unit 2 steam generator secondary manway bolts and the steam generator tubes fatigue analyses are based on a 40-year life (current operating license expires in 2027). As part of the Steam Generator Tube Integrity Program, BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design basis of these components is not exceeded for the period of extended operation.*~~	~~*May 27, 2027*~~	~~*LRA*~~	~~*A.3.3.1*~~ ~~*4.3.1*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A.5-1	**Page A.5-9**	**New Item Number 26**

In LRA Table A.5-1, "Unit 2 License Renewal Commitments," a new Item Number 26, previously deleted (this Enclosure), is added to identify a new FENOC License Renewal Future Commitment regarding the enhancements to the Metal Fatigue of the Reactor Coolant Pressure Boundary Program. Therefore, new Item Number 26 in LRA Table A.5-1 is created, and reads:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
26	*Enhance the Metal Fatigue of the Reactor Coolant Pressure Boundary Program to:* • *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.* • *Add a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of*	*May 27, 2027*	*LRA*	*B.2.27*

	these components are not exceeded for the period of extended operation. • *Add a requirement to monitor Unit 2 transients where the 60 year projected cycles are used in the environmental fatigue evaluations, and establish an administration limit that is equal to or less than the 60-year projected cycles number.* ~~[Deleted]~~			

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A.5-1	**Page A.5-9**	**Item Number 27**

LRA Table A.5-1, "Unit 2 License Renewal Commitments," Item Number 27 is no longer necessary due to the fact that the Unit 2 Emergency Diesel Generator Air Start System piping design analysis was revised to incorporate observed temperatures as a new load case. The revised design analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, Item Number 27 of LRA Table A.5-1 is deleted. Item Number 27 is revised to read:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
27	*[Deleted]* *~~BVPS will perform an assessment of the Unit 2 Emergency Diesel Generator Air Start System to determine whether the full temperature cycles limit would be exceeded for 60 years of operation. This assessment will be performed prior to the period of extended operation.~~*	*~~May 27, 2027~~*	*~~LRA~~*	*~~A.3.3.2.1~~* *~~4.3.2.1~~*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table A.5-1	**Page A.5-10**	**Item Number 28**

LRA Table A.5-1, "Unit 2 License Renewal Commitments," Item Number 28 regarding the Unit 2 NUREG/CR-6260 locations, is no longer necessary because of a new FENOC Regulatory Commitment (see Attachment 2 to this letter). Therefore, Item Number 28 in LRA Table A.4-1 is deleted. Item Number 28 is revised to read:

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
28	*[Deleted]* ~~*Of the NUREG/CR-6260 locations, the U_{env} (60 years) of the Unit 2 surge line nozzle, charging nozzle, and RHR line exceeded the design code allowable of 1.0. For these three locations, BVPS will implement one or more of the following:*~~ • ~~*Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;*~~ • ~~*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,*~~ • ~~*Repair or replacement of the affected locations.*~~	~~*May 27, 2027*~~	~~*LRA*~~	~~*A.3.3.3.3*~~ ~~*4.3.3.3.3*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section B.2.27	**Page B.2-75**	**Entire Section**

LRA Section B.2.27, "Metal Fatigue of Reactor Coolant Pressure Boundary Program," requires revision to include the results of an evaluation of each of the 10 aging management program elements described in NUREG-1801, Section X.M1, including enhancements. LRA Section B.2.27 is replaced in its entirety, and is revised to read:

"B.2.27 METAL FATIGUE OF THE REACTOR COOLANT PRESSURE BOUNDARY

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is a time-limited aging analysis (TLAA) program that uses preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The preventive measures consist of monitoring and tracking critical thermal and pressure transients for RCS components to prevent the fatigue design limit from being exceeded. Prior to exceeding the fatigue design limit, preventive and/or corrective actions are triggered by the program.

In addition, environmental effects are evaluated in accordance with NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components *[Reference B.3-18], and the guidance of EPRI Technical Report MRP-47,* Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application *[Reference B.3-19]. Selected components are evaluated using material specific guidance presented in NUREG/CR-6583,* Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels *[Reference B.3-20], and in NUREG/CR-5704,* Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels *[Reference B.3- 21].*

NUREG-1801 Consistency

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary.

Exceptions to NUREG-1801

None

Enhancements

The following enhancements will be implemented prior to the period of extended operation.

Program Element Affected:

- *Preventive Actions*

  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.*

  Add a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation.

- *Parameters Monitored/Inspected*

  Add a requirement that provides for monitoring of the Unit 1 RHR activation transient and establishes an administration limit of 600 cycles for the transient.

  Add a requirement to monitor Unit 1 and Unit 2 transients where the 60-year projected cycles are used in the environmental fatigue evaluations, and establish an administration limit that is equal to or less than the 60-year projected cycles number.

Aging Management Program Elements

The results of an evaluation of each of the 10 aging management program elements described in NUREG-1801, Section X.M1, are provided as follows:

- *Scope of Program*

  The program tracks critical transient cycles to ensure RCS components remain within their design fatigue usage limits. This program utilizes the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded.

  Fatigue analyses validated by this program include nuclear steam supply system (NSSS) equipment for BVPS Unit 1 and Unit 2 as well as the ASME Class 1 portions of the primary piping for Unit 2. Also, included is BVPS Unit 1 Surge Line.

  The program addresses the effects of the reactor coolant environment on component fatigue life by including, within the program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260.

- *Preventive Actions*

  The program provides for monitoring and tracking critical thermal and pressure transients for RCS components to prevent the fatigue design limit from being exceeded. Critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the sixty year operating life of the units. These critical transients include plant heatup, plant cooldown, reactor trip from full power (Unit 1 only), inadvertent auxiliary spray, safety injection activation (Unit 1 only), and RCS cold overpressurization. Supplemental transients were also identified by the program for monitoring. These supplemental transients include pressurizer insurge transient, selected Chemical and Volume Control System transients, Auxiliary Feedwater injections and RHR actuation.

  The number of critical transient occurrences is periodically reviewed (annual basis) to determine if there are any adverse trends; adverse conditions or deficient conditions with the primary objective of initiating evaluation of adverse trends and adverse conditions early to prevent the possibility of a deficient condition. Adverse Trend is an observed increase in the rate of critical transient occurrences that, if it continued, would result in exceeding the fatigue cycle design limit number of transients prior to the *end of the Unit's 60-year operating life. Adverse Condition is a condition in which the number of actual transient occurrences exceeds 80% of the fatigue cycle design limit number of occurrences. Deficient Condition is a condition in which the current number of actual transient occurrences exceeds the fatigue cycle design limit number of occurrences. Adverse trends and adverse conditions are evaluated by Engineering to determine if and when more rigorous analysis or alternate resolutions are required. Deficient conditions are addressed under the BVPS Corrective Action Program.*

- *Parameters Monitored / Inspected*

  The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.

  The program, for the most part, is a transient cycle counting program and does not require analysis of operational data (local monitoring) to obtain an effective number of transients.

  The WESTEMS™ Integrated Diagnostics and Monitoring System analysis for the surge line to hot leg nozzles is based on past occurrences of various transients along with what are believed to be conservative assumptions of future transients. In this case, the input assumptions will be verified by periodic reanalysis using updated plant history files.

- *Detection of Aging Effects*

  The program requires the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded. When the accrued operational cycles approach the component design cycles, corrective action is required by the program to ensure the design cycle limit is not exceeded. If the corrective action has an impact on the cumulative fatigue usage factor (CUF), an updated CUF will be generated.

- *Monitoring and Trending*

  The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks *fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.*

- *Acceptance Criteria*

  The program verifies that the fatigue usage remains below the design code limit considering environmental fatigue effects as described under the program description.

- *Corrective Actions*

  This element is discussed in LRA Section B.1.3.

- *Confirmation Process*

  This element is discussed in LRA Section B.1.3.

- *Administrative Controls*

  This element is discussed in LRA Section B.1.3.

- *Operating Experience*

  Concerns for the overall health of the transient/cycle counting program were documented using the FENOC Corrective Action Program. Corrective actions included identifying a program owner, developing an administration program document and updating it to incorporate responsibilities, improving cycle counting, and establishing a process for engineering to evaluate plant data. Fatigue monitoring to date indicates that the number of design transient events assumed in the original design analysis will be sufficient for a 60-year operating period. The program has remained responsive to emerging issues and concerns, particularly the pressurizer surge and spray nozzle, hot leg surge nozzle, and surge line transients.

  For example, in 2002, a Westinghouse evaluation identified that the BVPS Unit 2 letdown, charging, and excess letdown piping could potentially exceed their design allowable cycle counts for several design transients. However, further evaluation of existing plant operations and the physical separation distance of the letdown and excess letdown piping demonstrated that no further evaluation of the letdown or excess letdown piping was required for current operation or for the period of extended operation. A re-analysis of the charging piping was required to account for the appropriate transients for a 60-year plant life.

This responsiveness to emerging issues and continued program improvements provide evidence that the program will remain effective for managing cumulative fatigue damage for passive components.

Conclusion

Continued implementation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program provides reasonable assurance that the aging effects will be managed so that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation."

ENCLOSURE B

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

Letter L-08-209

**Westinghouse letter FENOC-08-109,
"FirstEnergy Nuclear Operating Company,
Beaver Valley Unit 1 and 2,
Responses to NRC RAIs Regarding
Pressurizer Surge Line Environmental Fatigue,"
Revision 1, dated June 25, 2008**

Page 1 of 22

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Our ref: FENOC-08-109, Revision 1

June 25, 2008

Note: Revision 1 is being issued to change the document to Westinghouse Non-Proprietary Class 3

FirstEnergy Nuclear Operating Company
Beaver Valley Unit 1 and 2
Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue

Dear Mr. Custer:

Attached are the Westinghouse inputs to the BVPS responses for the following NRC RAIs concerning pressurizer surge line environmental fatigue evaluations: RAI B.2.27-3, RAI 4.3-3 (b) and (c).

Should you have any questions, please feel free to contact Mr. Charlie Meyer at (724) 722-6017, or me at (412) 374-5216.

Regards,
WESTINGHOUSE ELECTRIC COMPANY

Kenneth B Blanchard

K. Blanchard
Customer Project Manager

with attachment

cc: BVRC Central File, SEB-1
Larry Hinkle – (FENOC)
Steve Buffington - (FENOC)

bcc: K. Blanchard – Energy Center
N. B. Closky – Energy Center
R. R. Jewell – Energy Center
C. Meyer – Waltz Mill
M. A. Gray – Waltz Mill

Responses to NRC Requests for Additional Information Regarding Pressurizer Surge Line *Environmental Fatigue*

RAI-B.2.27-3:

During the on-site audit, the applicant stated "the surge line to hot leg nozzle, for BVPS units 1 and 2, is included in a stress and fatigue model to be used in an on-line monitoring system (WESTEMS)" ...

b. Please provide the benchmarking results for the WESTEMS software using relevant transient data, and proper 3-D model. Please justify the use of WESTEMS™ to update the CUF calculation by using the monitored or projected transient data (cycles) and discuss the conservatisms in the calculation on a plant specific basis.

Westinghouse Input to Response:

The following provides the benchmarking results for WESTEMS™.

WESTEMS™ uses the transfer function method (TFM) [reference 1] to calculate six (6) components of stresses due to time varying mechanical and thermal loads. Time varying component stresses are calculated through wall as a function of the time varying mechanical and thermal boundary conditions. The resulting through wall stress components are processed and categorized according ASME Section III, Division 1, Subsection NB criteria. The processing first involves the calculation and categorization of the membrane, bending, and peak categories of mechanical and thermal stresses. These calculations are performed at the stress component levels, for each time step and for each applied loading type. The resulting stresses are then added to form the total stress and primary plus secondary stress according to ASME rules. Stress peak selection for fatigue evaluation purposes is based on analysis of the total stress time history and of the primary plus secondary stress time history. Both total stress and primary plus secondary stress are retained for future consideration in online fatigue evaluations. The discussion below will help to clarify the transfer function methodology, the transfer function database role, and provide an example of the current benchmarking process.

The transfer function method is a mathematical device that is capable of quantifying the effects experienced by a system due to an external disturbance, or excitation, with the aid of a characteristics function known as transfer function. In essence, the transfer function method is a means that correlates time-dependent behavior, in terms of input and output, of a system as seen in the thermal and dynamic problems. Examples of "disturbance" are mechanical forces, thermal transients, etc. Examples of "effects" include stresses, strains, displacements, temperature, etc. For typical structural applications, the "disturbance" can be surface temperature changes T(t), pressure P variation, forces (Fx, Fy, Fz), and moments (Mx, My, Mz) in a structural body (in vector notations: $\vec{F}$, and $\vec{M}$), whereas typical "effects" refer mostly to the stresses, displacements and metal interior temperatures.

In WESTEMS™, the transfer function methodology uses 2 or more unit load databases that have 4 or 6 components of stress depending on the nature of the original finite element model method that was used. If a two dimensional finite element model was used to create the transfer function database, then 4 components of stress are applicable (Sxx, Syy, Szz, Sxy). If a three dimensional model was used, then there are 6 components of stress in the transfer function databases (Sxx, Syy, Szz, Sxy, Syz, and Szx). The total number of stress states in the transfer function databases is dependent on the complexity of the thermal and mechanical boundary conditions being simulated.

For thermal applications, the transfer function is a characteristics function of a thermal-mechanical system. The characteristics include geometry, boundary conditions, insulation conditions, material properties, and thermal zones. These characteristics are all built into the transfer function for a predefined thermal-mechanical system. Therefore, a transfer function database is fixed for a particular type of thermal-mechanical problem. However, a single set of transfer function databases can be used to evaluate the system responses caused by any kind of transients. This means that a transfer function database is created only once but can be used to obtain solutions for unlimited numbers of transient cases.

It is important to realize that thermal stresses in materials or any structural systems arising from temperature transients are evolving because heat transfer is an energy transport process that will continue until thermal equilibrium is established. This means that it requires an appreciable amount of time for a thermally disturbed material or structural system to come to a steady state, even if the disturbance is as brief as an impulse. In short, a thermal transient is a time-dependent problem. On the contrary, all mechanical loads, pressure, direct forces, and moments, encountered in the general structural applications are treated as static problems, unless the loading rates are so high that the dynamic effects cannot be ignored. To appropriately reflect the types of loads being dealt with, the databases are split into two types:

- Thermal transfer Function DataBase (TFDB)
- MEchanical transfer function DataBase (MEDB).

Westinghouse has validated the thermal stress capability of the WESTEMS™ transfer function method by performing identical analyses using the Westinghouse transfer function method and an independent finite element program like ANSYS or WECAN. Examples of the predicted stress component results for benchmarking the transfer function models are shown below. The benchmarking process is generally performed for every transfer function database created. The following example was taken directly from the appendix of a Westinghouse Transfer function database calculation note. The verification of WESTEMS™ thermal and mechanical stress calculations have been performed in the program's verification and validation documentation. However, each application verification of the finite element models and of the final thermal transfer function databases should be performed to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results, Westinghouse performs thermal stress analyses using both the finite element program and WESTEMS™. The example below shows these comparisons and results. Certain information has been removed and text has been modified in order to clarify the example, which is taken from reference 2.

Verification of the Surge Line Hot Leg Nozzle TFDB and MEDB Databases

Verification of the databases being used for the WESTEMS™ analyses is a required step to ensure good analysis results. All databases are herein examined through suitable benchmarking problems.

The database files, TFDB and MEDB, generated in the unit load finite element analyses, represent the thermal and mechanical characteristics of the structural component considered. By using these databases, the stresses at the specified analysis sections (ASN or cut) can be evaluated for any combination of load conditions. To correctly produce the results, each load type requires an appropriate scaling factor, which is being developed in the following subsections. The scaling factor provides a means to correct the effects arising from differences in the stress units used in ANSYS and in WESTEMS™. It also is a means that permits non-standard unit loads to be used to generate the database.

Verifying the Bending Moment Database – M_x

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment about the x-axis portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_x represents bending about the global x-axis. The analysis for this bending case was performed using an ANSYS model and documented. The applied moment is 1000 in-kips.

Consider the well known bending stress equation

$$\sigma = Mr/I$$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-1, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

Ro = 7.0 in.

Ri = 5.594 in.

$I = \pi (Ro^4 - Ri^4) / 4 = 1116.6 \text{ in}^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-1. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips, whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $fb = f_1 * f_2 = 10^{-6}$.

Table B-1: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28106	5010	-5010	0.0
Outside node	26787	6269	-6269	0.0

Note: The sign of stress produced by ANSYS was negative since "CUT4" is in compression due to the direction of moment loading in ANSYS.

Verifying the Torsion Database – My

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the torsion moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The moment M_y represents the moment (or twist) about the global y-axis. The analysis for this torsion case was performed in ANSYS and documented. The applied moment is 1000 in-kips.

Consider the well known torsion shearing stress equation

$$\tau = Mr/J$$

where M is the applied torque, r is distance from the neutral axis, and J is the polar moment of inertia in torsion of the cross sectional area.

Two nodes, as listed in Table B-2, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above equation can be applied.

At this location, the following data apply:

$R_o = 7.0$ in.

$R_i = 5.594$ in.

$J = \pi (D_o^4 - D_i^4) / 32 = 2233.3 \text{ in}^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-2. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied torque is in-kips, whereas 1000 in-kips of torque was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the torsion load to be used for WESTEMS™ analyses is found to be $f_t = f_1 * f_2 = 10^{-6}$.

Table B-2: Comparison of ANSYS and Analytical Results

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28106	2504.82	2531.80	-1.08
Outside node	26787	3134.39	3131.90	0.08

Verifying the Bending Moment Database – Mz

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment about the z axis portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_z represents bending about the global z-axis. The analysis for this bending case was performed in ANSYS and documented. The applied moment is 1000 in-kips.

Consider the well known bending stress equation

$$\sigma = Mr/I$$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-3, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

$R_o = 7.0$ in.

$R_i = 5.594$ in.

$I = \pi (R_o^4 - R_i^4) / 4 = 1116.6 \text{ in}^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-3. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips, whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $f_b = f_1 * f_2 = 10^{-6}$.

Table B-3: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28093	5010	-5010	0.0
Outside node	26754	6269	-6268	0.0

Verifying the Pressure Database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the pressure portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the pressure loading case was performed in ANSYS and documented. The applied pressure is 1000 psi.

Consider the well known hoop stress equation for a pressurized pipe:

$$\sigma_\theta = \frac{p R_i^2}{R_o^2 - R_i^2}\left(1 + \frac{R_o^2}{r^2}\right)$$

where p is the internal pressure, R_o is the outside radius, R_i is the inside radius, and r is the radius at any point.

Two nodes, as listed in Table B-4, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above equation can be applied. At this location, the following data apply:

R_o = 7.0 in.

R_i = 5.594 in.

Comparisons of the ANSYS FE and analytical results are shown in Table B-4. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied pressure is psi, whereas 1000 psi of pressure was used in the database creation, a second factor, f_2=0.001, is required. Combining the two factors, the scaling factor for the pressure load to be used for WESTEMS™ analyses is found to be $f_p = f_1 * f_2 = 10^{-6}$.

Table B-4: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28093	4534.48	4756.50	-4.9
Outside node	26754	3534.48	3427.00	3.0

Verifying the Thermal Stress Database

Two benchmarking problems are considered here, which serve two purposes: (1) to determine the scaling factor corresponding to the transfer function thermal stress database, and (2) to verify the database created.

To benchmark and verify this portion of the database and determine the appropriate scaling factor for the thermal loads, an arbitrary shock transient and a stratification transient were used. The transients used for this benchmarking problem are defined in the data shown in Table B-5 and Table B-6.

Table B-5: The Transient Used for the Thermal Shock Benchmarking Case

Time (seconds)	Temperature -Zone1 through Zone10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone10 (Btu/hr-ft^2-°F)
0	100	8000
1	100	8000
10	100	8000
20	100	8000
21	500	8000
22	500	8000
23	500	8000
24	500	8000
25	500	8000
26	500	8000
27	500	8000
28	500	8000
30	500	8000
31	500	8000
40	500	8000
50	500	8000
60	500	8000
75	500	8000
85	500	8000
95	500	8000
110	500	8000
125	500	8000
160	500	8000

Time (seconds)	Temperature -Zone1 through Zone10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone10 (Btu/hr-ft^2-°F)
210	500	8000
410	500	8000
710	500	8000
1000	500	8000
2000	500	8000
4000	500	8000

Table B-6: The Transient Used for the Thermal Stratification Benchmarking Case

Time (seconds)	"Nozzle Top" Temperature - Zones 1, 3, 5, 7, 8, and 9 (°F)	"Nozzle Bottom" Temperature - Zones 2, 4, 6, and 10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone 10 (Btu/hr-ft^2-°F)
0.001	110	110	275
19	110	110	275
21	430	110	275
22	430	110	275
23	430	110	275
24	430	110	275
25	430	110	275
26	430	110	275
27	430	110	275
28	430	110	275
29	430	110	275
30	430	110	275
31	430	110	275
32	430	110	275
35	430	110	275
40	430	110	275
45	430	110	275
50	430	110	275
55	430	110	275
60	430	110	275
65	430	110	275
70	430	110	275
75	430	110	275
80	430	110	275
85	430	110	275
90	430	110	275
95	430	110	275
100	430	110	275

Time (seconds)	"Nozzle Top" Temperature - Zones 1, 3, 5, 7, 8, and 9 (°F)	"Nozzle Bottom" Temperature - Zones 2, 4, 6, and 10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone 10 (Btu/hr-ft^2-°F)
105	430	110	275
110	430	110	275
115	430	110	275
120	430	110	275
140	430	110	275
150	430	110	275
160	430	110	275
210	430	110	275
410	430	110	275
610	430	110	275
810	430	110	275
1210	430	110	275
1610	430	110	275
2500	430	110	275
5000	430	110	275

In the shock benchmarking transient, all zones undergo a severe thermal shock. In the stratification benchmarking transient, the surge line piping undergoes stratification with a temperature change of 320 °F. The transients are intentionally made severe on the temperature rate so as to allow a vigorous examination of the integrity of the transfer function database. This set of transients was analyzed by both WESTEMS™ and ANSYS. Note that the ANSYS results represent full finite element analyses whereas the WESTEMS™ results are produced by the transfer function method, which utilizes the transfer function databases produced by ANSYS.

The results, as shown in Figures B-1 through B-4 for the controlling location of the nozzle (units: ksi for stress, seconds for time), are then graphically compared on both the shapes and the magnitudes. It can be seen from these figures that the WESTEMS™ results compare very well with those calculated by ANSYS, both in magnitudes and curve shapes for both the shock transient loading and the stratification transient loading.

The shapes of the curves of the stresses from the WESTEMS™ analysis are visually compared with those from the ANSYS full finite element analysis. In general, good comparisons are observed for all cases. The stratification case shows slight differences in stress magnitude in the steady state stratification condition, which is expected. This is caused by the inside surface film coefficients changing values between zones, which is accounted for in ANSYS by two-dimensional heat transfer, but is not fully accounted for in the WESTEMS™ benchmark run. The results from WESTEMS™ predict slightly higher stresses at the stratified steady state condition, which therefore leads to conservative answers and is not considered a concern.

Overall, very good benchmarking results have been achieved, which assures good results can be produced through the TFDB created. Since WESTEMS™ results are either close in magnitude or slightly higher than the ANSYS benchmark results, the factor, f_l=1.00, is applied. Since the stress unit in the ANSYS FEA results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a factor, f_2=0.001, is required. Combining the two factors, the scaling factor for the thermal load to be used for WESTEMS™ analyses is $f_T = f_l * f_2 = 0.001$, which is to be registered to the "TFDB_Factor" box in the WESTEMS™ ASN Analysis Models.

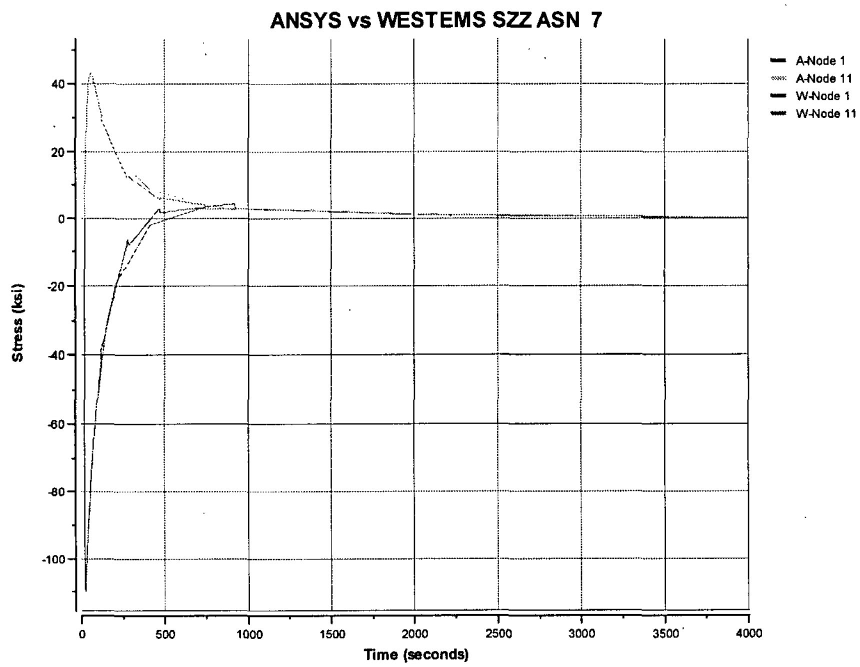


Figure B-1: ASN 7 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Shock Transient Loading

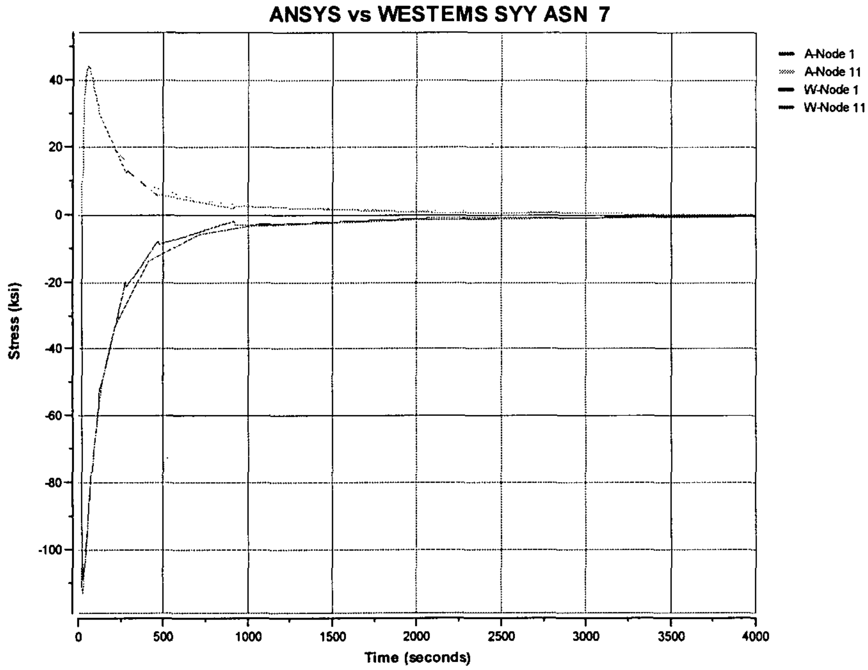


Figure B-2: ASN 7 Axial Stress Comparison (ANSYS vs. WESTEMS) for Shock Transient Loading

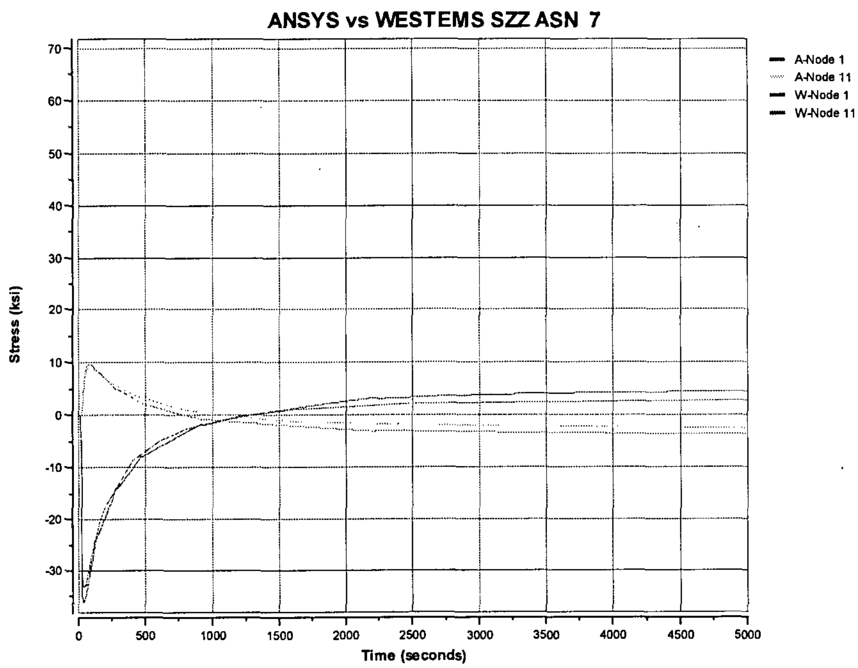


Figure B-3: ASN 7 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Stratification Transient Loading

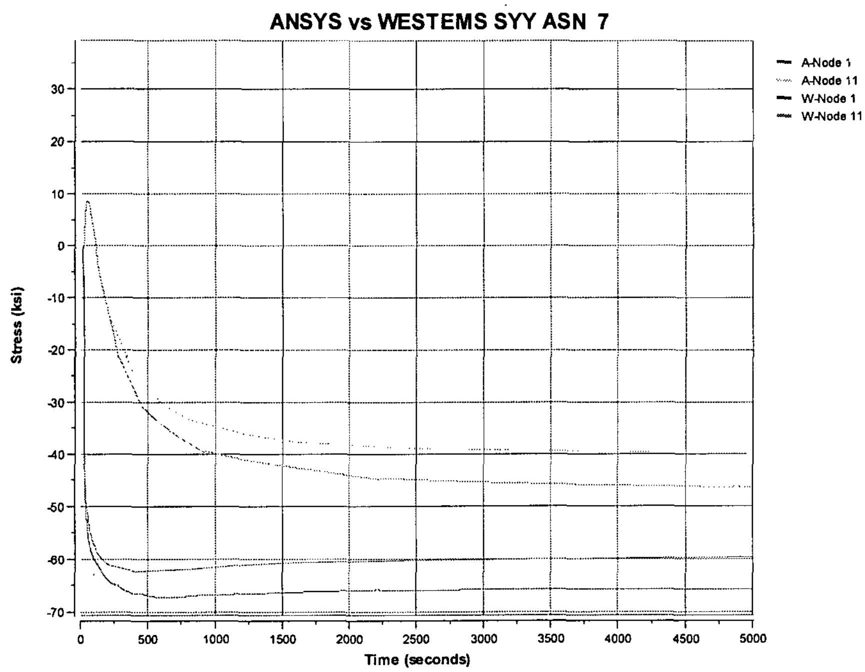
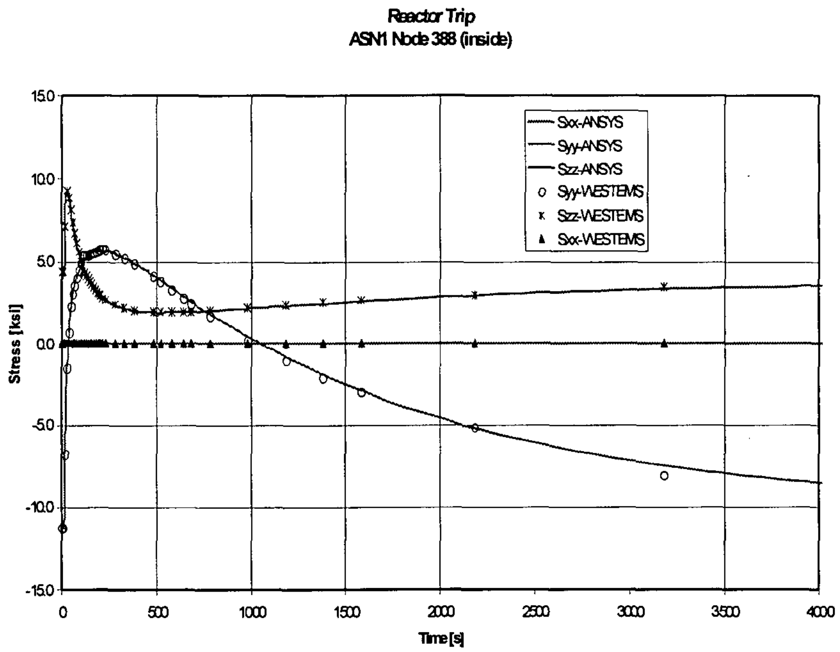
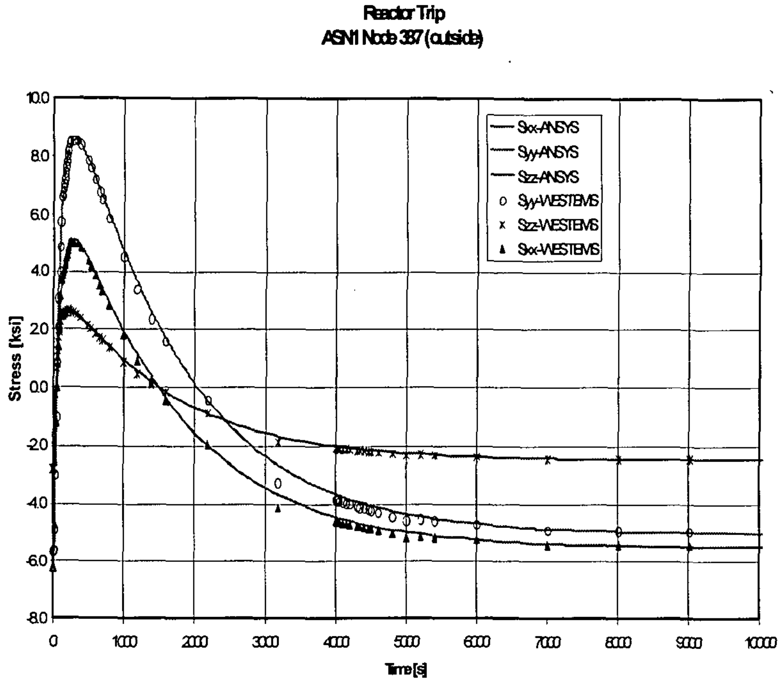


Figure B-4: ASN 7 Axial Stress Comparison (ANSYS vs. WESTEMS) for Stratification Transient Loading

The results shown below, obtained by a WESTEMS™ user in a Westinghouse European site, serve as additional verification of the transfer function methodology.



Additional Thermal Stress Benchmark Results, Sample 1.



Additional Thermal Stress Benchmark Results, Sample 2.

References:

1. "Transfer Function Method for thermal Stress and Fatigue Analysis: Technical Basis", WCAP-12315, Westinghouse Proprietary Class 2, C. Y. Yang, May 1990.
2. Westinghouse Calculation No. CN-PAFM-07-60, Rev. 0, "Beaver Valley Unit 2: Transfer Function Database Development for a 14-inch Hot Leg Surge Nozzle." S. F. Hankinson.

RAI-4.3-3 (b & c)

LRA Section 4.3.3.3 discusses the effects of primary coolant environment on fatigue life. During the audit, the applicant indicated that it will refine the analysis for NUREG/CR-6260 components in the near future. To assist the staff it its review:

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

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.

Westinghouse Input to Response for Part b

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 as:

$$U_{en} = U_1 * F_{en,1} + \ldots + U_i * F_{en,i} + \ldots + U_n * F_{en,n}$$

where i = 1, 2, ...n
U_i = incremental fatigue usage contribution, calculated according to NB-3222.4
$F_{en,i}$ = environmental fatigue penalty factor

For stainless steels, F_{en} is calculated as follows:

$$F_{en} = \exp[0.935 - T{*}O{*}\varepsilon'{*}]$$

where T* = transformed temperature

O* = transformed oxygen content

ε'* = transformed strain rate

The terms are explained below in detail.

Thresholds are defined where the following parameters for the pair are within the following ranges for stainless steel (per NUREG/CR-5704):

$T \leq 200°C$

$\varepsilon' > 0.4\%/sec$

When one of these is satisfied, the negative term in the F_{en} equation above is zero, and the minimum value of F_{en} is calculated as:

$$F_{en} = \exp(0.935) = 2.547$$

A strain amplitude threshold is also discussed in NUREG/CR-5704 and clarified in NUREG/CR-6717 ($\varepsilon_{amp} <= 0.10\%$), where the environmental effect is insignificant ($F_{en} = 1.0$) for the pair. This was not applied in the evaluation since pairs in this range did not have a significant effect on fatigue.

T* = transformed temperature

$T^* = 0$ (T<200°C)

$T^* = 1.0$ (T ≥ 200°C)

Where T = metal surface temperature of the component being considered

O* = transformed oxygen content

$O^* = 0.260$ (DO < 0.05 ppm)

$O^* = 0.172$ (DO ≥ 0.05 ppm)

Where DO = dissolved oxygen (DO) content (ppm).

For PWRs, it is easily assumed that: DO < 0.05 ppm.

Therefore, O* = 0.260 for all cases for stainless steels.

ε'* = transformed strain rate, for stainless steels is:

$\varepsilon'^* = 0$ ($\varepsilon' > 0.4\%/\text{sec}$)

$\varepsilon'^* = \ln(\varepsilon'/0.4)$ ($0.0004 \leq \varepsilon' \leq 0.4\%/\text{sec}$)

$\varepsilon'^* = \ln(0.0004/0.4)$ ($\varepsilon' < 0.0004\%/\text{sec}$)

This may be determined using various methods depending on the degree of conservatism retained for qualification.

A detailed integrated method was used to incorporate strain rate, called the modified rate approach, where the Fen is integrated over the strain range for the tensile strain producing cycle of the transient pair. The modified rate approach is represented below:

$$Fen = \frac{\sum Fen_i \Delta\varepsilon_i}{\sum \Delta\varepsilon_i}$$

where:

Fen_i = Fen computed for time interval i, based on ε_i'=$100\Delta\varepsilon_i/\Delta t_i$ and transformed parameters (T*), (ε*), and (O*) computed for the interval

$\Delta\varepsilon_i$ = change in strain for time interval i, $(\sigma_i-\sigma_{i-1})/E$

σ_i = stress intensity for time i

σ_{i-1} = stress intensity for time i-1

Δt_i = change in time for time interval i, $\Delta t = t_i-t_{i-1}$

E = Young's Modulus

For load pairs that include dynamic OBE loading, a minimum Fen = 2.55 was used for the dynamic portion of the strain included in the pair. This was considered to be conservative, since dynamic load cycling occurs at a frequency that is too high for environmental effects to be significant, and an Fen = 1.0 could be justified. Based on this, when OBE occurred in a pair with a thermal transient, it was conservative to use the Fen determined based on the thermal transient only.

The stress cycle pairs obtained from the fatigue analysis of the safe end to pipe weld of the surge line hot leg nozzle were used to calculate F_{en} factors. The most dominant stress cycle pairs in these evaluations came from the heatup and cooldown transients. F_{en} for all stress cycle pairs was calculated using the F_{en} modified rate approach discussed above. This approach, integrating the F_{en} over the positive strain rate portions of the pair's history, resulted in F_{en} values for each stress cycle pair. After calculation of the appropriate F_{en} values for the respective stress cycle pairs, the final cumulative usage factor for the surge line hot leg nozzle with environmental effects was calculated by summing the corrected usage for each pair.

Westinghouse Input to Response for Part c (for surge line hot leg nozzles)

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 method used to evaluate the effects of reactor water environment on the ASME fatigue usage is discussed in the response to RAI 4.3-3.b. The NB-3200 evaluation was performed using program WESTEMS™.

Inputs to the fatigue evaluation were provided or confirmed by Beaver Valley engineering in a Design Information Transmittal (DIT), "DIT-WEST-ENV-02". The information provided included the design mechanical loads for Units 1 and 2. It also confirmed the applicability of thermal loads related to stratified and non-stratified conditions. The inputs were consistent with those used in the evaluations of surge line stratification in WCAP-12727, Supplement 1 for Unit 1 and WCAP-12093, Supplement 5 for Unit 2. The DIT also provided primary stresses calculated separately from the fatigue evaluations performed by Westinghouse.

Transients used in the fatigue evaluations were developed based on the design transients used in the original evaluations of surge line stratification, WCAP-12727 and WCAP-12093, and updated information on stratification loading developed from plant operating data. Transient input information was supplied and/or confirmed in the Beaver Valley DIT.

The fatigue evaluation followed the procedures given in the ASME Code Section III, NB-3200. Transient loadings representing the transients defined for the surge line hot leg nozzle were input to WESTEMS™ using binary "history files." The history files contain all the local parameter tagnames needed to calculate stress at the controlling locations, using the WESTEMS™ Derived Value functions and transfer functions. The methodology used to develop and benchmark transfer functions was discussed in the response to RAI B.2.27-3.

The stress ranges, cycle pairing and fatigue usage factors were calculated using WESTEMS™, consistent with the ASME Code as outlined by the steps below.

1. The stress histories were calculated for stress cuts (ASNs) in structural components subjected to thermal, pressure, and piping loads from the defined transients using the unit load transfer function databases. WESTEMS™ model information was used to calculate stress and related inputs for the fatigue evaluation based on ASME Section III, NB-3200, methodology. Stress component histories and stress component ranges were determined and used in the fatigue evaluations.

2. The stress peak and valley times were determined for each transient stress history, and associated stress component values at each selected time were input to the fatigue usage calculation.

3. The Primary plus Secondary Stress Intensity Ranges were calculated. Since the ASME Code fatigue curves are based on elastic stress results, adjustments to the alternating stress intensity range were required if this stress range exceeded the elastic range. In the ASME Code evaluation, the linearized primary plus secondary stress ranges are compared to the $3S_m$ allowable to determine if the elastic range is exceeded. If the 3Sm allowable is exceeded, then a Simplified Elastic-plastic analysis per NB-3228.5 is performed to obtain the appropriate adjustment factor (K_e).

4. Appropriate correction factors were calculated for each possible load set range pair formed from the stress components at each peak and valley time.

    a. Primary plus Secondary Stress Intensity Range (S_n) was compared to the 3Sm allowable. If $S_n > 3S_m$, the elastic-plastic penalty factor, K_e, for that pair was applied, in addition to evaluation of other requirements of NB-3228.5.

b. If $S_n \leq 3S_m$, then Poisson's ratio correction factor for Local Thermal stress was calculated according to NB-3227.6.

c. The elastic modulus correction factor, $E_{curve}/E_{analysis}$, was calculated according to NB-3222.4(e)(4).

5. For each load set pair, correction factors were applied, and final adjusted alternating stress, S_a, was determined. Cumulative fatigue usage was calculated using the method of NB-3222.4(e)(5) and the appropriate material fatigue curve.

The surge line hot leg nozzle environmental fatigue evaluations are documented in WCAP-16830-P for Unit 1 and WCAP-16867-P for Unit 2