FENOC

FirstEnergy Nuclear Operating Company

Peter P. Sena III
Site Vice President

724-682-5234
Fax: 724-643-8069

June 2, 2008
L-08-147

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)

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 BVPS license renewal time-limited aging analyses in Sections 4.7.1 and 4.7.3 of the BVPS LRA.

The Attachment provides the FENOC reply to the U.S. Nuclear Regulatory Commission request for additional information.

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

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

Sincerely,

Peter P. Sena III

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)," April 1, 2008.

Attachment:
Reply to Request for Additional Information Regarding Beaver Valley Power Station, Units 1 and 2, License Renewal Application, Sections 4.7.1 and 4.7.3

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

cc: w/o Attachment
Dr. S. S. Lee, NRC DLR Acting 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 4.7.1, Piping Subsurface Indications (Unit 1 only)

Question RAI 4.7.1-1

Section 4.7.1 of the license renewal application (LRA) does not provide information regarding the inspection history and results of the subject weld on the reactor coolant system (RCS) loop C cold leg nor degradation mechanism of the indication. The staff needs this information to complete its evaluation.

(a) Discuss the inspection history and results (characterization) of the indication on the RCS loop C cold leg between the elbow and a section of straight pipe since the discovery in 1996.

(b) Discuss future inspection frequency of the subject indication.

(c) Clarify the exact location of the subject indication, e.g., in the weld that joins the elbow and pipe, on the elbow, or on the pipe.

(d) Discuss the degradation mechanism of the indication.

RESPONSE RAI 4.7.1-1

(a) Weld DLW-LOOP3-7-S-02 inspection history and results since the discovery in 1996:

A flaw indication was identified during a Unit 1 Inservice Inspection (ISI) examination performed in the Unit 1 Cycle 11 Refueling Outage (March-May, 1996) for weld DLW-LOOP3-7-S-02. Examination revealed the presence of four ID indications grouped in a band ranging from 4 inches to 14 inches below top dead center in the 9 o'clock to 12 o'clock quadrant looking toward the vessel. Individual depths were not reported. However, the four indications were considered bounded by a single composite flaw with the following dimensions (Reference 1):

Flaw Depth (a) = 0.68 inches

Flaw Length (l) = 10 inches

Wall Thickness (t) = 2.66 inches

Flaw Depth Parameter (a/t) = 25.6%

Flaw Shape Parameter (a/l) = 0.07

Since the subject flaw indication exceeded the ASME Code, Section XI, subsection IWB-3500 acceptance criteria, an analysis (Reference 1) was performed to ensure that this indication would remain within ASME Code, Section XI, Appendix C evaluation acceptance standards. The analysis concluded that the postulated flaw met the applicable requirements with significant margins of safety to the end of the service lifetime. This analysis was reviewed and approved by the NRC (Reference 2). The NRC concluded that the reported flaw was acceptable for continued service until the end of the service lifetime provided the weld was reexamined during each of the next three 40-month ISI periods.

Examinations were conducted for the first two 40-month periods (1st and 2nd periods of the 3rd ISI Interval) and the results indicated that no measurable growth was observed since the initial examination. The 3rd period exam should have been completed during the Unit 1 Cycle 17 (February 13 to April 19, 2006) or Cycle 18 (September 24 to October 24, 2007) refueling outages. However, this required ISI examination was inadvertently removed from the Unit 1 ISI 3rd Interval 10-year ISI Plan. This issue is being addressed under the FENOC Correction Action Program.

(b) Future inspection frequency of the subject indication:

A relief request to use an alternative Risk-Informed Inservice Inspection (RI-ISI) Program for the Beaver Valley Power Station (BVPS), Unit 1 (3rd ISI Interval) and Unit 2 (2nd ISI Interval), ASME Code Class 1 and 2 piping welds was approved by the NRC in a Safety Evaluation Report (SER) dated April 9, 2004 (Reference 3). Under this approved RI-ISI Program, the subject weld is not scheduled for future examinations.

As discussed in (a) above, the subject weld was not reexamined as required during the 3rd period of the 3rd ISI Interval, and this issue is being addressed under the FENOC Correction Action Program.

(c) Weld location:

Weld DLW-LOOP3-7-S-02 is the first circumferential weld after the Reactor Vessel nozzle-to-safe-end weld on the RCS C loop cold leg (the other end of the elbow is welded to the Reactor Vessel safe-end) (Reference 1).

(d) Degradation mechanism of the indication:

The piping is not susceptible to stress corrosion cracking (SCC) based on the conclusions of the flaw analysis (Reference 1) for weld DLW-LOOP3-7-S-02. The potential for SCC is minimized by assuring that materials selections are compatible with the plant operating parameters and a corrosive environment is not present. Since high residual stresses, materials susceptibility and a corrosive environment all have to be present in order to experience SCC, the materials specifications coupled with the absence of a corrosive environment assures that SCC is not a factor for this piping.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23, 1996 (NRC Public Document Room (PDR) Accession Number 9604300327).
2. Brinkman, Donald S. (NRC), Letter to J. E. Cross (Duquesne Light Company), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996 (NRC PDR Accession Number 9605060144).
3. Laufer, R.J. (NRC), Letter to L. W. Pearce (FENOC), *Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) - Risk-Informed Inservice Inspection (RI-ISI) Program (TAC Nos. MB5687 and MB5688)*, April 9, 2004.

Question RAI 4.7.1-2

The applicant stated that the subject pipe is made of cast austenitic stainless steel (CASS) piping. However, it is not clear to the staff the material specification of the elbow and subject weld. It is also not clear the indication characterization.

(a) Confirm that the elbow is made of CASS also.

(b) Discuss the material specification of the weld that joins the elbow and the pipe.

(c) Provide the indication size and characterization.

(d) The staff notes that ultrasonic testing of CASS material cannot be performed to meet the requirements of Appendix VIII to the ASME Code, Section XI. Therefore, discuss the reliability and accuracy of the detection and characterization of the subject indication.

RESPONSE RAI 4.7.1-2

(a) The elbow was fabricated of Grade CF8M cast austenitic stainless steel (CASS) (Reference 1).

(b) Weld DLW-LOOP3-7-S-02 (subject weld that joins the elbow and the pipe) material specification (Reference 1):

Weld Filler material is TP 308 stainless steel. The weld was made using a tungsten inert gas (tig) weld process for the root passes. It is believed that the remaining weld passes were made using a submerged arc welding process based on the weld appearance and observation of other circumferential welds on the sister train spool pieces which were radiographed in the as-welded condition. (A review of the fabrication records confirmed the use of a tig root pass with the balance of the weld being done by the sub-merged arc process.)

(c) Weld DLW-LOOP3-7-S-02 indication size and characterization (Reference 1):

A flaw indication was identified during a Unit 1 Inservice Inspection (ISI) examination performed in the Cycle 11 Refueling Outage (March-May, 1996) for weld DLW-LOOP3-7-S-02. Examination revealed the presence of four ID indications grouped in a band ranging from 4 inches to 14 inches below top dead center in the 9 o'clock to 12 o'clock quadrant looking toward the vessel. Individual depths were not reported. However, the four indications were considered bounded by a single composite flaw with the following dimensions:

Flaw Depth (a) = 0.68 inches

Flaw Length (l) = 10 inches

Wall Thickness (t) = 2.66 inches

Flaw Depth Parameter (a/t) = 25.6%

Flaw Shape Parameter (a/l) = 0.07

(d) Reliability and accuracy of the detection and characterization of the subject indication (Reference 1):

Initial ultrasonic testing (UT) examination was conducted on March 26, 1996, using a 45 degree dual element refracted longitudinal search unit set. The UT instrument used was a Panametric Epoch with modified bypass filter. A second examination was conducted on March 27, 1996, and used the same instrumentation. A third UT examination was conducted on March 29, 1996, using a Panametrics Epoch II instrument with a newly developed KBA ceramic search unit provided by Wesdyne. This search unit uses 30 x 40 mm dual elements in an integral case providing a 37 degree refracted longitudinal wave. The primary purpose of this scan was to confirm the reflector using another angle and try to observe any additional signal dynamics or tip diffracted peaks that may be generated. In addition, the 45 degree search units were once again used to observe the response using this instrument. A follow-up examination was conducted on the inside surface of the pipe using a driver pickup eddy current probe. The eddy current probe was used to do a complete surface examination of the elbow to pipe weld region of the cold leg for all three loops. The results of these circumferential scans were identical for all three loops, and showed no surface breaking indications. Even though this examination confirmed the lack of any surface indications, the fracture mechanics analysis used an umbrella of all the UT characterized indications, for conservatism, assuming the indication to be surface breaking. The indications were evaluated as a single composite size which bounds all four of these indications. The single composite flaw dimensions are provided in (c) above.

The NRC in an SER (Reference 2) documented that the NRC staff confirmed the presence of this flaw indication during an independent inspection on April 25, 1996. The SER also states that the details of this inspection are documented in NRC Integrated Inspection Report 50-334/96-04 and 50-412/96-04.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23,1996 (NRC PDR Accession Number 9604300327).
2. Brinkman, Donald S. (NRC), Letter to J. E. Cross (Duquesne Light Company), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996 (NRC PDR Accession Number 9605060144).

Question RAI 4.7.1-3

In Section 4.7.1, the applicant stated that the fully aged fracture toughness properties of the CASS straight pipe were used in the flaw evaluation. However, the applicant did not state the CASS piping has limiting material properties as compared to the subject weld. Confirm that the fracture toughness properties of the CASS piping are more limiting than the fracture toughness properties of the weld at the end of 60 years.

RESPONSE RAI 4.7.1-3

The fracture toughness values for full service life are listed as follows (Reference 1):

1. elbow: J_{Ic} = 750 in-lb/in^2, T_{mat} = 60
2. weld: J_{Ic} = 650 in-lb/in^2, T_{mat} = 100
3. piping: J_{Ic} = 410 in-lb/in^2, T_{mat} = 13

Therefore, the limiting material properties are those of the piping and these properties were used in the flaw evaluation.

The NRC in an SER (Reference 2) considered the piping material properties used in the flaw evaluation to be conservative.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23, 1996 (NRC PDR Accession Number 9604300327).
2. Brinkman, Donald S. (NRC), Letter to J. E. Cross (Duquesne Light Company), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996 (NRC PDR Accession Number 9605060144).

Question RAI 4.7.1-4

In Section 4.7.1, third paragraph, the applicant stated that it postulated an initial flaw and considered the flaw growth based on imposed loading transients in the flaw evaluation without providing detailed information.

(a) Discuss the initial flaw size assumed.

(b) Discuss the flaw growth rate used and associated references.

(c) The applicant stated that "...The cycle assumptions used in the analysis are conservative compared to the BVPS [Beaver Valley Power Station] original design cycles..." Clarify whether the operating cycles used in flaw evaluation that was performed in 1996 exceed the projected operational cycles at the end of 60 years.

RESPONSE RAI 4.7.1-4

(a) Initial flaw size assumed in the flaw analysis (Reference 1):

Flaw Depth (a) = 0.68 inches

Flaw Length (l) = 10 inches

Wall Thickness (t) = 2.66 inches

Flaw Depth Parameter (a/t) = 25.6%

Flaw Shape Parameter (a/l) = 0.07

(b) The crack growth rate used in the flaw analysis and associated references (Reference 1):

A compilation of data for austenitic stainless steels in a PWR water environment was made by Bamford (Reference 2), and it was found that the effect of the environment on the crack growth rate was very small. From this information it was estimated that the environment factor should be conservatively set at 2.0 in the crack growth rate equation provided in James, et al. (Reference 3).

(c) The transients considered in the flaw analysis (Reference 1) are the design transients contained in the equipment specification. It was noted in the analysis that the Beaver Valley plants were designed to a slightly different set of transients than those used in the flaw analysis. The transients used in the flaw analysis were considered to be conservative, relative to the original design transients.

As demonstrated in Table 4.3-2 of the BVPS License Renewal Application (LRA), the Unit 1 original design transients bound the 60-year projected cycles.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23, 1996 (NRC PDR Accession Number 9604300327).
2. Bamford, W. H., *Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment*, Trans ASME, Journal of Pressure Vessel Technology, February 1979.
3. James, L. A., and Jones, D. P., *Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air, in Predictive Capabilities in Environmentally Assisted Cracking*, ASME publication PVP-99, December 1985.

Question RAI 4.7.1-5

Section 4.7.1 of the LRA does not contain sufficient information regarding the stress evaluations of the subject flaw for the staff to make a conclusion on the validity of the evaluations.

(a) Besides the indication on the cold leg, identify all Class 1 components that contain indications or flaws that have remained in service at BVPS, Units 1 and 2.

(b) Discuss briefly the flaw evaluations (such as procedures and assumptions) performed for the affected components in accordance with the ASME Code, Section XI.

(c) Discuss how the indications or flaws were accepted and reference the appropriate ASME Code requirements.

(d) Provide indication / flaw characterization.

(e) Discuss the analyses performed to accept the degraded components (other than the subject weld) for the extended period of operation.

RESPONSE RAI 4.7.1-5

Flaws detected during examination are evaluated by comparing the examination results to acceptance standards established in ASME Section XI. Unacceptable indications require detailed analyses (e.g., ASME Section XI, IWB-3640), repair, or replacement. The acceptance standards ensure that all service conditions are accommodated by maintaining the safety margin of the component throughout its service life. Analytical evaluation in accordance with IWB-3640 requires that flaw growth be considered for an evaluation period equal to the time until the next inspection following discovery of the flaw or until the end of the service life of the component. Flaw growth evaluations performed for the remaining service life of the component are TLAA that must be addressed for license renewal.

Only reportable flaws were researched.. A reportable flaw is one that exceeds acceptance standards provided in Subsections IWB, IWC, or IWD of ASME Section XI and applicable addenda. There were three possible courses of action for any reportable flaw. Repairs may be made, affected portions of the component may be replaced, or the flaw may be shown to be acceptable through analytical evaluation. The BVPS ISI inspection summary reports for Unit 1 and Unit 2 were reviewed to identify analytical evaluations of flaws discovered during ISI that were analyzed to the end of the service life of the component. Other flaws were found, but were not mentioned because they were repaired, replaced, or not reportable according to the Section XI acceptance criteria.

Therefore, the contents of LRA Section 4.7.1 contain only reportable flaws left in-service with acceptance provided through analytical evaluations.

Question RAI 4.7.1-6

In Section 4.7.1, the applicant referenced an NRC safety evaluation of the subject indication on the RCS loop C cold leg. However it is unclear to the staff which safety evaluation is being referenced. Confirm that May 1, 1996, is the date of the NRC's safety evaluation, If the date is correct, please provide a copy of the safety evaluation because the staff cannot find the reference (Reference 4.7-1) in the NRC database (ADAMS).

RESPONSE RAI 4.7.1-6

May 1, 1996, is the correct date of the subject NRC safety evaluation. The submittal (Reference 1) and subsequent SER (Reference 2) pertaining to the flaw evaluation on the Unit 1 RCS loop C cold leg are accessible through the Public Document Room, PDR Accession Numbers 9604300327 and 9605060144, respectively.

During the research to provide responses to the NRC request for additional information (RAI) for Sections 4.7.1 and 4.7.3 of the BVPS LRA, it was discovered that the subject flaw evaluation was not updated using revised loading conditions derived from the Extended Power Uprate (EPU) and Steam Generator Replacement Projects. This issue is being addressed under the FENOC Corrective Action Program. At this time, Westinghouse has completed a review of the flaw evaluation previously performed for the indication at weld DLW-LOOP3-7-S-02 (Reference 1) in accordance with the flaw evaluation procedure and acceptance criteria in the 1989 Edition of the ASME Section XI Code by assessing the impact of the following items on the previous flaw evaluation results:

- Applicable Thermal Transients Reflecting EPU Conditions.
- Latest Piping Reaction Loads Reflecting EPU Conditions, including the replacement Steam Generators.
- Thermal Aging/Fracture Toughness per NUREG/CR-4513, Rev. 1.

The results of the assessment indicated that the indication at weld DLW-LOOP3-7-S-02 would remain acceptable for the duration of plant life including the license renewal period.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23, 1996 (NRC PDR Accession Number 9604300327).
2. Brinkman, Donald S. (NRC), Letter to J. E. Cross (Duquesne Light Company), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996 (NRC PDR Accession Number 9605060144).

<u>Section 4.7.3, Leak Before Break</u>

Question RAI 4.7.3-1

Nickel-based Alloy 600/82/182 material in the pressurized-water reactor environment has been shown to be susceptible to primary stress corrosion cracking (PWSCC). This is an emerging issue. The applicant did not address this issue in Section 4.7.3 of the LRA.

(a) Identify all Alloy 82/182 weld metal and Alloy 600 components used in the piping that have been approved for LBB.

(b) If LBB piping contains Alloy 600/82/182 material, discuss any plans to mitigate these components (such as weld overlays or mechanical stress improvement) to reduce their susceptibility to PWSCC.

(c) Discuss the inspection history of these components.

(d) Discuss future inspection frequency of the Alloy 600/82/182 components and whether the inspection frequency will provide reasonable assurance that PWSCC will be detected early, should it occur in the LBB piping.

RESPONSE RAI 4.7.3-1

(a) Alloy 82/182 weld metal and Alloy 600 components used in the primary loop piping that have been approved for Leak Before Break (LBB):

- Unit 1 Main Coolant Loop Piping – There is no Alloy 82/182 weld metal or Alloy 600 components in the Unit 1 primary loop piping.
- Unit 1 Pressurizer Surge Line Piping – There is no Alloy 82/182 weld metal or Alloy 600 components in the Unit 1 pressurizer surge line piping.
- Unit 2 Main Coolant Loop Piping – Reactor Vessel (RV) Inlet/Outlet Nozzles Safe-End Welds.
- Unit 2 Pressurizer Surge Line Piping – Pressurizer Surge Nozzle Safe-End Weld.
- Unit 2 Branch Line Piping (Residual Heat Removal (RHR), Safety Injection System (SIS) and Reactor Coolant System (RCS) loop bypass lines) – There is no Alloy 82/182 weld metal or Alloy 600 components in the Unit 2 Branch Line Piping (RHR, SIS and RCS loop bypass lines).

(b) Weld Overlay / Mechanical Stress Improvement

As shown in the response to (a), above, LBB piping that contains Alloy 600/82/182 material is the Unit 2 Main Coolant Loop Piping (RV Inlet/Outlet Nozzles Safe-End Welds) and Unit 2 Pressurizer Surge Line Piping (Pressurizer Surge Nozzle Safe-End Weld). At this time there are no plans to perform full structural weld overlays or mechanical stress improvements of the Unit 2 RV Inlet/Outlet Nozzles Safe-End Welds to reduce their susceptibility to primary water stress corrosion cracking (PWSCC). Application of structural weld overlays on the Unit 2 Pressurizer Nozzle Welds was implemented due to increased inspection requirements and difficulties associated with application of ultrasonic inspection technology to the current weld geometry. During the Unit 2 Cycle 12 refueling outage (October 2 to November 12, 2006) FENOC completed application of full structural weld overlays to the pressurizer nozzle welds which included the Unit 2 Pressurizer Surge Nozzle Safe-End Weld (Reference 1).

(c) Inspection History

The Unit 2 RV Inlet/Outlet Nozzles Safe-End Welds were examined using Performance Demonstration Initiative (PDI)-qualified ultrasonic examination techniques in the Spring of 2008. Greater than 90 percent ultrasonic examination coverage was obtained at all six dissimilar metal (DM) weld locations with no recordable indications identified.

The Unit 2 Pressurizer Surge Nozzle Safe-End Weld was volumetrically examined following structural weld overlay in the Fall of 2006. The examination was performed on the required inspection volume in the outer 25 percent of the original DM weld using PDI-qualified ultrasonic examination techniques. Greater than 90 percent ultrasonic examination coverage was obtained and no recordable indications were identified.

(d) Future Inspection Frequency

Future inspection frequency of the Alloy 600/82/182 components will be determined by the BVPS Nickel-Alloy Nozzles and Penetrations Program. Implementation of this program is a commitment in the LRA, Appendix A, Table A.4-1 (item 15) and Table A.5-1 (item 17), for Unit 1 and Unit 2 respectively.

Reference:

1. Lash, James H. (FENOC), Letter (L-06-163) to NRC, *Beaver Valley Power Station Unit No. 2, Docket No. 50-412, License No. NPF-73, Pressurizer Weld Overlay Examination Report*, November 20, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML063310045).

Question RAI 4.7.3-2

In Section 4.7.1, the applicant stated that an indication on the RCS loop C cold leg between the elbow and a section of straight pipe was detected in 1996. The staff needs to clarify the location of this indication to make a proper evaluation. Discuss whether this indication is located on a segment of the pipe that has been approved for LBB. If it is, discuss whether the assumptions in the LBB analyses are still valid in light of the indication. Discuss whether the indication was fabrication or service induced. The NRC-approved LBB approach excludes piping with active degradation mechanism for LBB.

RESPONSE RAI 4.7.3-2

A flaw indication was identified during a Unit 1 in-service inspection (ISI) performed in the Unit 1 Cycle 11 Refueling Outage (March-May, 1996) for weld DLW-LOOP3-7-S-02.

Examination revealed the presence of four ID indications grouped in a band ranging from 4 inches to 14 inches below top dead center in the 9 o'clock to 12 o'clock quadrant looking toward the vessel. Since the subject flaw indication exceeded the ASME Code, Section XI, subsection IWB-3500 acceptance criteria, an analysis (Reference 1) was performed to ensure that this indication would remain within ASME Code, Section XI, Appendix C evaluation acceptance standards. The analysis concluded that the postulated flaw met the applicable requirements with significant margins of safety to the end of the service lifetime. This analysis was reviewed and approved by the NRC (Reference 2).

Weld DLW-LOOP3-7-S-02 is the first circumferential weld after the Reactor Vessel nozzle-to-safe-end weld on the RCS C loop cold leg (the other end of the elbow is welded to the Reactor Vessel safe-end) and is located on a segment of the pipe that has been approved for LBB. The LBB evaluation for the Unit 1 main coolant loop piping is documented in WCAP- 11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1* (Reference 3).

The NRC-approved LBB approach excludes piping with active degradation mechanism for LBB. As stated in the *Leak-Before-Break Evaluation Procedures* published in the Federal Register (Reference 4):

> *"Piping susceptible to IGSCC* [inter-granular stress corrosion cracking] *with any planar flaws in excess of the standards in IWB 3514.3 of section XI of the ASME Code, would not be permitted to use leak-before-break analysis."*

The piping in this case is not susceptible to stress corrosion cracking (SCC) based on the conclusions of the original flaw analysis (Reference 1) for weld DLW-LOOP3-7-S-02. The potential for SCC is minimized by assuring that materials selections are compatible with the plant operating parameters and a corrosive environment is not present. Since high residual stresses, materials susceptibility and a corrosive environment all have to be present in order to experience SCC, the materials specifications coupled with the absence of a corrosive environment assures that SCC is not a factor for this piping. Therefore, the assumptions of the LBB analysis remain valid in light of the indication.

As documented in a follow-up submittal dated May 1, 1996 (Reference 5), the results of the video and eddy current examinations of the inner diameter (ID) surface were reviewed and evaluated. The results of these examinations verified that there was no surface breaking indications or geometric irregularities on the ID surface. The lack of any ID surface breaking indication provides assurance that there is no in-service failure mechanism to be addressed.

The weld was reexamined during the first two 40-month periods (1st and 2nd periods of the 3rd ISI Interval) following discovery, and the results indicated that no measurable growth was observed in the flaw since the initial examination. This provides additional confirmation that the material is not susceptible to SCC and confirms that the LBB analysis remains valid.

References:

1. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: 1989 Edition of ASME XI, Article IWB-3640*, April 23, 1996 (NRC PDR Accession Number 9604300327).
2. Brinkman, Donald S. (NRC), Letter to J. E. Cross (Duquesne Light Company), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996 (NRC PDR Accession Number 9605060144).
3. WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1*, March 1987 (including Supplements 1 and 2).
4. *Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures*; Federal Register Vol. 52, No. 167, Friday, August 28, 1987, Notices, pp. 32626-32627.
5. Jain, Sushil C. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 1, Docket No. 50-334, License No. DPR-66, Analysis of Flaw Indications: Follow-up Submittal*, May 1, 1996.

Question RAI 4.7.3-3

For the pressurizer surge line in Section 4.7.3.2, the applicant stated that a system temperature difference of about 360°F was experienced in the plant during heatup. The staff needs further information on this temperature difference to validate whether the safety margins are still met.

(a) Clarify whether the original LBB analysis has been evaluated with a 360°F temperature difference.

(b) Clarify whether the cycle counts in Table 4.3-2 include this temperature transient in the LBB analysis.

RESPONSE RAI 4.7.3-3

(a) As presented in WCAP-12093-P, Supplement 3 (Reference 1), the characteristics of the thermal stratification transients were discussed in WCAP-12093 (Reference 2). Of significance to WCAP-12093 is that, based on historical data associated with a high system temperature difference, is a series of less severe transients. Supplement 3 evaluated the original LBB analysis with the addition of the observed 360°F temperature difference that occurred during a Unit 2 plant heatup. Specifically, the 360°F transient scenario observed was postulated to consist of one transient of 360°F, one transient of 342°F and three transients of 320°F; this is in addition to all the design cycles developed in the original LBB analysis. Supplement 3 concludes that the maximum stress intensity, fatigue usage factor, and growth of postulated cracks are not significantly affected by the observed transient of 360°F and that the design life is not impacted by this larger temperature difference.

(b) WCAP-12093-P, Supplement 3, incorporated the postulated transients of 360°F (1 ea.), 342°F (1 ea.) and 320°F (3 ea.) into the integrity evaluation of the Unit 2 pressurizer surge line. These postulated transients were a subset of the observed 360°F temperature difference that occurred during the Unit 2 plant heatup following the first refueling outage. This Unit 2 plant heatup is included in the operational cycle count (operational cycles as of October 15, 2003) presented in LRA Table 4.3-2.

References:

1. Sieber, J. D. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No. 2, Docket No. 50-412, License No. NPF-73, Primary Component Support Snubber Elimination*, August 10, 1990.
2. WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line*, December 1988.

Question RAI 4.7.3-4

In Section 4.7.3.2, the applicant stated that operating procedures have been revised to add precautions and limitations to prevent exceeding a 320°F system temperature difference. The revised LBB analysis for the surge line is based on a temperature difference of 320°F. The LBB analysis would be invalid if the temperature difference is greater than 320°F. Discuss whether the 320°F limitation has been exceeded since the implementation of the limit. This information is needed for the staff to assess the adequacy of the precautions and limitations implemented by the applicant.

RESPONSE RAI 4.7.3-4

A system temperature differential between the pressurizer and the hot leg of approximately 360°F was experienced in Unit 2 during heatup from the Cycle 1 refueling outage (March-June 1989). Subsequently, the Unit 1 and Unit 2 operating procedures were revised to add precautions and limitations to prevent exceeding the 320°F system temperature differential design limit. Included in the operating procedures is a requirement to verify that the pressurizer to C loop hot leg differential temperature is less than 320°F when the conditions to start a reactor coolant pump (RCP) are established during heatup. Otherwise, the RCS temperature shall be raised to reduce the differential temperature to less than 320°F prior to start of an RCP. Therefore, this operating restriction is sufficient to preclude exceeding the 320°F differential temperature design limit between the pressurizer and the hot leg during heatup.

Question RAI 4.7.3-5

The applicant has an Aging Management Program B.2.41, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*, to manage CASS components, However, Section 4.7.3 of the LRA did not mention this aging management program (AMP) manage the LBB piping that are made of CASS. Discuss whether AMP B.2.41 will be used to monitor the CASS components in the LBB piping systems for thermal aging embrittlement.

RESPONSE RAI 4.7.3-5

The Unit 1 and Unit 2 main coolant loop piping (approved for LBB) are fabricated from cast austenitic stainless steel (CASS). For this piping, the BVPS Aging Management Program B.2.41, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*, will be used to manage loss of fracture toughness due to thermal aging embrittlement.

As presented in Appendix B of the LRA, Program B.2.41 monitors the effects of loss of fracture toughness on the intended function of the component by identifying CASS materials that are susceptible to thermal aging embrittlement. For potentially susceptible materials that are part of the reactor coolant pressure boundary, the program will consists of either volumetric examination of the base metal or a component-specific flaw tolerance evaluation.

Question RAI 4.7.3-6

By letter dated May 19, 2000, Christopher I. Grimes of the NRC forwarded Douglas J. Walters of Nuclear Energy Institute an evaluation of thermal aging embrittlement of CASS components (ADAMS Accession ML003717179). In the NRC's evaluation, the staff provided its positions on how to manage CASS components. For clarity of record, please discuss whether the CASS components in the LBB piping satisfy the staff positions in its evaluation dated May 19, 2000.

RESPONSE RAI 4.7.3-6

NUREG-1801, Rev. 1 (Reference 1), Section XI.M12 incorporated the NRC Staff positions in its evaluation dated May 19, 2000 (Reference 2). As presented in LRA Section B.2.41, "The *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program* is a new aging management program that will be consistent with NUREG-1801, Section XI.M12, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*."

The Unit 1 and Unit 2 main coolant loop piping (approved for LBB) are fabricated from cast austenitic stainless steel (CASS). For this piping, the Aging Management Program B.2.41, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*, will be used to manage loss of fracture toughness due to thermal aging embrittlement.

References:

1. NUREG-1801, Vol. 2, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1.

2. Christopher I. Grimes, NRC, License Renewal and Standardization Branch, Letter to Douglas J. Walters, Nuclear Energy Institute, *License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000, (ADAMS Accession No. ML003717179).

Question RAI 4.7.3-7

In Sections 4.7.3.1 and 4.7.3.2, the applicant stated that the LBB evaluation for the main coolant loop piping and pressurizer surge line continues to be justified at power uprate operating conditions for both units without providing details.

(a) Discuss in detail how the LBB evaluations of these two piping systems are validated for uprate power conditions.

(b) Clarify whether the LBB evaluation for the Unit 2 branch lines continues to be valid at power uprate operating conditions. Discuss the technical basis for the conclusion.

RESPONSE RAI 4.7.3-7

(a) The LBB evaluations performed for the main coolant loop piping and pressurizer surge line piping remain valid for the EPU operating conditions (Reference 1, Enclosure 2, Sections 4.5.2.1 and 4.5.2.2).

Main Coolant Loop Piping

The current LBB analyses (References 2 and 3) were updated to address EPU conditions. The loadings, operating pressure and temperature parameters for the EPU were used in the evaluation. The parameters, which are important in the evaluation, are the piping forces, moments, normal operating temperature and normal operating pressure. These parameters were used in the evaluation. The evaluation results show that the LBB conclusions provided in the current LBB analyses for Unit 1 and Unit 2 remain unchanged for the EPU conditions.

Pressurizer Surge Line Piping

The current LBB analyses (References 4 and 5) were updated to address EPU conditions. An evaluation was performed to determine the impact of the loadings and other parameters on the LBB analysis due to the EPU conditions. The results of the evaluation show that all the LBB acceptance criteria and recommended margins are satisfied at the EPU conditions.

(b) The LBB evaluations performed for the Unit 2 RCS branch lines remain valid for the EPU operating conditions (Reference 1, Enclosure 2, Section 4.5.2.3).

Branch Line Piping (Unit 2 only)

The RCL branch line piping forces and moments, operating pressure, temperature, and material properties were the important input parameters for the previous WHIPJET evaluation (Reference 6). The impact of EPU conditions on these input parameters was evaluated and shown to be one percent or less. Therefore, the Unit 2 RCL branch line WHIPJET evaluations for LBB are insignificantly affected and remain acceptable for EPU conditions.

References:

1. Pearce, L. W. (FENOC), Letter L-04-125 to NRC, *Beaver Valley Power Station Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, License Amendment Request Nos. 302 and 173*, October 4, 2004 (ADAMS Accession Number ML042920300).
2. WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1*, March 1987 (including Supplements 1 and 2).

3. WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers*, September 1988.
4. WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, November 1990.
5. WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line*, December 1988 (including Supplements 1 and 3).
6. *WHIPJET Program Final Report, Beaver Valley Unit 2*, January 30, 1987.

Question RAI 4.7.3-8

In Sections 4.7.3.1, 4.7.3.2, and 4.7.3.3, the applicant discussed fatigue crack growth analyses of the main coolant loop piping, surge line, and branch lines without details. Provide the bounding final fatigue crack size for each of the piping system and the corresponding pipe wall thickness. The information will assist the staff with an independent assessment of the structural integrity of the subject piping.

RESPONSE RAI 4.7.3-8

<u>LRA Section 4.7.3.1—Main Coolant Loop Piping Leak Before Break</u>

The current LBB evaluation for the Unit 1 main coolant loop piping is documented in WCAP-11317 (Reference 1) and includes the fatigue crack growth analysis. This evaluation (including Supplements 1 and 2) was approved by the NRC in an SER (Reference 2) in 1987.

The current LBB evaluation for the Unit 2 main coolant loop piping is documented in WCAP-11923 (Reference 3) and includes the fatigue crack growth analysis. This evaluation was approved by the NRC in an SER (Reference 4) in 1991.

As documented in the LBB evaluations for Unit 1 and Unit 2, a finite element stress analysis was carried out for the inlet nozzle safe end region of a plant typical in geometry and operational characteristics to any Westinghouse PWR System. The specific system was a plant with piping outside diameter 33 inches, and wall thickness of 2.85 inches. The corresponding dimensions for BVPS Unit 1 are 34.0 inches in diameter and 3.27 inches wall thickness. The corresponding dimensions for BVPS Unit 2 are 32.46 inches in diameter and 2.5 inches wall thickness. The difference in dimensions (between the typical Westinghouse plant and the Beaver Valley plants) are insignificant as far as fatigue crack growth analysis is concerned. The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths are summarized in Table 4.7.3-8A, shown below, which shows that crack growth is very small, regardless of which material and flaw size are assumed.

TABLE 4.7.3-8A Main Coolant Loop Piping Fatigue Crack Growth Results (Unit 1 and Unit 2)		
Initial Flaw (in.)	**Final Flaw for Ferritic steel (in.)**	**Final Flaw for Stainless (in.)**
0.292	0.31097	0.30107
0.300	0.31949	0.30953
0.375	0.39940	0.38948
0.425	0.45271	0.44350

<u>LRA Section 4.7.3.2—Pressurizer Surge Line Piping Leak Before Break</u>

The current LBB evaluation for the Unit 1 pressurizer surge line piping is documented in WCAP-12727 (Reference 5) and includes the fatigue crack growth analysis. This evaluation was approved by the NRC in an SER (Reference 6) in 1991. Unit 1 transients exceeding a 320°F temperature differential between the pressurizer and hot leg were included in the analyses of WCAP-12727.

The current LBB evaluation for the Unit 2 pressurizer surge line piping is documented in WCAP-12093 (Reference 7) and includes the fatigue crack growth analysis. This evaluation (including Supplements 1 and 2) was approved by the NRC in an SER (Reference 8) in 1990. These analyses were based on a maximum temperature difference of 315°F between the pressurizer and the hot leg. Subsequent to the 1990 SER (Reference 8), a system temperature difference of approximately 360°F was experienced in the plant during heatup. To address this issue, WCAP-12093-P, Supplement 3 (Reference 9) was prepared and submitted to the NRC. This evaluation was approved by the NRC in an SER (Reference 4) in 1991.

Presented here is a brief summary of the fatigue crack growth analyses. Five locations representing various cross sections of the surge line where thermal stratification could occur were evaluated for fatigue crack growth. Figure 4-6 of WCAP-12727 (WCAP-12093) identifies the five locations. Figure 4-7 of WCAP-12727 (WCAP-12093) identifies the positions at each location where fatigue crack growth analyses were performed. These positions are controlling positions because the global structural bending stress is maximum at two of the positions while the local axial stress on the inside surface is maximum at two of the other positions. The largest initial flaw assumed to exist was one with a depth equal to ten percent of the wall thickness, the maximum flaw size that could be found acceptable by Section XI of the ASME Code. The results of the fatigue crack growth analysis are presented in Table 4.7.3-8B, shown below, for the ten percent nominal wall initial flaw (initial size 0.141 inches). The maximum depth for full service life was less than 25% of the nominal wall thickness. In Table 4.7.3-8B, Unit 2 values for fatigue crack growth results from WCAP-12093 Supplement 3 are shown in parentheses.

TABLE 4.7.3-8B
Surge Line Piping Fatigue Crack Growth Results

Location	Position	Unit 1 Final Size (in.)	Unit 2 Final Size (in.)
1	A	0.149	0.149
	B	0.149	0.149
	C	0.163	0.162 (0.163)
	D	0.141	0.141
2	A	0.143	0.143
	B	0.142	0.142
	C	0.306	0.305 (0.306)
	D	0.141	0.141
3	A	0.142	0.142
	B	0.141	0.141
	C	0.348	0.347 (0.348)
	D	0.141	0.141
4	A	0.141	0.141
	B	0.142	0.142
	C	0.280	0.280 (0.281)
	D	0.144	0.144
5	A	0.142	0.142
	B	0.168	0.168

TABLE 4.7.3-8B Surge Line Piping Fatigue Crack Growth Results			
Location	**Position**	**Unit 1 Final Size (in.)**	**Unit 2 Final Size (in.)**
	C	0.170	0.170 (0.170)
	D	0.145	0.145

<u>LRA Section 4.7.3.3—Branch Line Piping Leak Before Break (Unit 2 only)</u>

The LBB evaluation for the Unit 2 Reactor Coolant Loop branch line piping is documented in the *WHIPJET Program Final Report* (Reference 10) and includes the fatigue crack growth analysis. This evaluation was approved by the NRC in an SER (Reference 11) in 1987.

The analyses were performed to apply LBB to all Reactor Coolant Loop branch line piping greater than or equal to 6 inches as identified in Section 3 of the *WHIPJET Program Final Report* and the BVPS Unit 2 *Updated Final Safety Analysis Report* (UFSAR) Table 3.6-4. Fatigue crack growth calculations were performed at the piping limiting locations; namely the piping locations with the highest stress based on normal and safe shutdown earthquake loads. An assumed crack of a size which exceeds the ASME, Section XI acceptance criteria was analytically subjected to the internal piping loads occurring at these limiting locations. The calculated fatigue crack growth results for the limiting locations are summarized in Table 4.7.3-8C, shown below.

TABLE 4.7.3-8C Branch Line Piping Fatigue Crack Growth Results		
System/Line Size	**Initial Wall Thickness (in.)**	**Final Surface Crack Length (in.)**
SIS / 6 inch line	0.718	0.56
RCS / 8 inch line	0.906	0.69
RHR (RHS) / 10 inch line	1.125	0.76
RHR (RHS) / 12 inch line	1.312	0.87
SIS / 12 inch line	1.312	0.96
RCS / 14 inch line	1.406	1.26

References:

1. WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1*, March 1987 (NRC PDR Accession Number 8706120011).
2. Tam, Peter S. (NRC), Letter to J.D. Sieber (Duquesne Light Company), *Beaver Valley Unit 1 - Removal of Large-Bore Snubbers from Primary Coolant Loops*, December 9, 1987 (NRC PDR Accession Number 8712160053).
3. WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers*, September 1988 (NRC PDR Accession Number 8901250444)
4. De Agazio, Albert W. (NRC), Letter to J. D. Sieber (Duquesne Light Company), *Elimination of Dynamic Effects of Postulated Pressurizer Surge Line Rupture and Elimination of Reactor Coolant System Component Support Snubbers*, April 8, 1991(NRC PDR Accession Number 9104150077).
5. WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, Rev. 0. (NRC PDR Accession Number 9101290161).
6. De Agazio, Albert W. (NRC), Letter to J. D. Sieber (Duquesne Light Company), *Approval of Leak-Before-Break Analysis*, May 2, 1991. (NRC PDR Accession Number 9105080014).
7. WCAP-12093, Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line, December 1988. (NRC PDR Accession Number 8901090282)..
8. Tam, Peter S. (NRC), Letter to J. D. Sieber (Duquesne Light Company), *Beaver Valley Unit 2 – Completion of Review on Pressurizer Surge Line Thermal Stratification*, January 18, 1990. (NRC PDR Accession Number 9001250295).
9. Sieber, J. D. (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station, Unit No. 2, Docket No. 50-412, License No. NPF-73, Primary Component Support Snubber Elimination*, August 10, 1990. (NRC PDR Accession Number 9002110013).
10. *WHIPJET Program Final Report, Beaver Valley Unit* 2, January 30, 1987 (NRC PDR Accession Numbers 8702170628 and 8702170635).
11. NUREG-1057, Supplement No. 4, *Safety Evaluation Report Related to the Operation of Beaver Valley Power Station Unit 2; Docket No. 50-412 Duquesne Light Company*, March 1987.

Question RAI 4.7.3-9

As part of the technical basis for the LBB, the applicant cited Reference 4.7-8, Westinghouse report WCAP-11923, "Technical Justification for Elimination Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers," September 1988, The staff needs to verify information in this document for its evaluation. Please provide a copy of Westinghouse report WCAP-11923.

RESPONSE RAI 4.7.3-9

WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers*, (Reference 1) and it's cover letter dated October 27, 1988 (Reference 2), are accessible through the Public Document Room, PDR Accession Number 8901250444.

WCAP-11923 is a Westinghouse proprietary document that was withheld from public disclosure as described in the cover letter of the submittal (Reference 2). The evaluation provided by WCAP-11923 was approved by the NRC in an SER (Reference 3) in 1991.

References:

1. WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers*, September 1988.
2. J.B. Sieber (Duquesne Light Company), Letter to NRC, *Beaver Valley Power Station Unit No.21, Docket No. 50-412, License No. NPF, Primary Component Support Snubber Eimination*, October 27, 1988 (NRC PDR Accession Number 8901250444).
3. De Agazio, Albert W. (NRC), Letter to J. D. Sieber (Duquesne Light Company), *Elimination of Dynamic Effects of Postulated Pressurizer Surge Line Rupture and Elimination of Reactor Coolant System Component Support Snubbers*, April 8, 1991(NRC PDR Accession Number 9104150077).