UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

February 13, 2008

Mr. Peter P. Sena III
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2007005 AND 05000412/2007005

Dear Mr. Sena:

On December 31, 2007, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Beaver Valley Power Station Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 10, 2008, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, this report documents a Severity Level IV violation of NRC requirements, a NRC-identified finding, and three (3) self-revealing findings evaluated using the significance determination process (SDP) as being of very low safety significance (Green). Of the findings evaluated using the SDP, three (3) were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues have been entered in your corrective action program, the NRC is treating the findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the findings in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Beaver Valley.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at 610-337-5200 if you have any questions regarding this letter.

Sincerely,

/RA/

Ronald R. Bellamy, Ph.D., Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-334, 50-412
License Nos: DPR-66, NPF-73

Enclosures: Inspection Report 05000334/2007005; 05000412/2007005
w/Attachment: Supplemental Information

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We appreciate your cooperation. Please contact me at 610-337-5200 if you have any questions regarding this letter.

Sincerely,
/RA/
Ronald R. Bellamy, Ph.D., Chief
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REGION I

Docket Nos.	50-334, 50-412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2007005 and 05000412/2007005
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	October 1, 2007 through December 31, 2007
Inspectors:	D. Werkheiser, Senior Resident Inspector D. Spindler, Resident Inspector A. Ziedonis, Resident Inspector T. Fish, Senior Operations Engineer P. Frechette, Physical Security Inspector P. Kaufman, Senior Reactor Inspector J. Lilliendahl, Reactor Inspector T. Moslak, Health Physicist S. Pindale, Senior Reactor Inspector G. Smith, Physical Security Inspector
Approved by:	R. Bellamy, Ph.D., Chief Reactor Projects Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2007005, IR 05000412/2007005; 10/01/2007 – 12/31/2007; Beaver Valley Power Station, Units 1 & 2; Heat Sink Performance, Maintenance Risk Assessments and Emergent Work Control, Post Maintenance Testing, Refueling and Other Outage Activities, Event Follow-Up.

The report covered a 3-month period of inspection by resident inspectors, regional reactor inspectors, and a regional health physics inspector. One Severity Level IV non-cited violation (NCV), three Green NCVs, and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3 dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A green self-revealing NCV of TS LCO 3.8.1, "Electrical Power Systems - AC Sources - Operating," was identified due to FENOC's failure to comply with the LCO actions for one required offsite power circuit inoperable within the specified allowed time requirements. The performance deficiency is that the power availability of the 138kV Bus 1 to the Unit-1 1A System Station Service Transformer (SSST) [TR-1A] was not effectively monitored such that an open circuit on the phase 'A' was not identified. This resulted in exceeding the TS 3.8.1 allowed outage time. This issue was entered into the corrective action program as CR 07-30614. The open circuit on phase 'A' was repaired and immediate compensatory measures were taken to augment monitoring of off-site power system availability. A root cause investigation was initiated. Long term corrective actions are under development.

  The finding is more than minor because it is associated with the Initiating Events Cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the finding is determined to be of very low risk significance (Green) because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

  The cause of this finding is related to the cross-cutting area of problem identification and resolution, in that FENOC did not completely and accurately identify this issue in a timely manner [P.1 (a)]. (Section 1R13)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a green finding because FENOC failed to meet a commitment made in their Generic Letter (GL) 89-13 program. Specifically, after a self-assessment identified the potential for postponing the cleaning of safety related heat exchangers, which was contrary to the Beaver Valley GL 89-13 commitments, corrective actions were developed to prevent postponement of the planned cleanings. These actions were insufficient to prevent the postponement of cleanings of both the Jacket Water and Intercooler heat exchangers for the 2A Emergency Diesel Generator (EDG) in October 2006. This issue was entered into the corrective action program as CR 07-29900. The licensee performed a prompt operability determination to show reasonable assurance of operability through the rest of the operating cycle.

  This finding is more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspector conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function or loss of a single train for greater than its allowed technical specification time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding did not involve a violation of NRC regulatory requirements.

  The cause of this finding is related to the cross-cutting area of problem identification and resolution, in that a 2004 self-assessment identified this potential vulnerability, but the resulting corrective actions were ineffective in preventing it [P.3 (c)]. (Section 1R07)

- Green. A green self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified in that the licensee failed to properly implement and control work activities associated with the alarm and status relays for Unit 1 ‘C’ steam generator water level (SGWL), which resulted in a degraded SGWL Hi-Hi and Lo-Lo alarm status for approximately 9 days. Safety functions for the ‘C’ SGWL were unaffected and one alarm status remained available during the degraded condition. This issue was entered into the corrective action program as CR 07-29487. FENOC performed an apparent cause evaluation, evaluated appropriate human performance and organizational contributors, and initiated corrective actions and procedure revisions to prevent recurrence.

  The finding is more than minor because it affected the equipment performance attribute of the associated Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspector conducted a Phase 1 SDP screening and the finding was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function or loss of a single train for greater than its allowed technical specification time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events.

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC failed to ensure appropriate coordination of work activities during work scope changes to activities affecting the use of ‘C’ SGWL instrumentation during outage periods, which resulted in a loss of configuration control that degraded a safety-related alarm and status indicator [H.3.(b)]. (Section 1R19)

- **Severity Level IV**. The inspectors identified that the licensee did not perform an adequate safety evaluation in accordance with 10 CFR 50.59 associated with changing the periodicity of IST testing of valves MOV-1SI-890 A&B in May 2006. The review did not identify that the change allowed operations of these valves in Operational Modes where operation was prohibited by TS. The change was approved and implemented and as a result, from May 2006 until July 2007, valves MOV-1SI-890 A&B were cycled nine times total. Upon discovery, the licensee entered this issue into their corrective action program as CR 07-23462, conducted a root cause analysis and an extent of condition review, and revised the LHSI surveillance procedures. The licensee also determined that this event was reportable and issued LER 05000334/2007-001.

  The performance deficiency and violation is that the licensee did not perform an adequate safety evaluation in accordance with 10 CRF 50.59, due to the fact that the evaluation failed to identify that a change would proceduralize an operation which was prohibited by TS. This change would have required prior approval from the NRC via Technical Specification Amendment, to allow this change. A 10 CFR 50.59 violation is considered to potentially impede or impact the regulatory process; therefore, Traditional Enforcement applies. Comparing this item to the examples in NUREG 1600 Supplement I, this finding is more than minor because NRC approval would have been required. The inspectors completed a Significance Determination Review using IMC 0609, Appendix A “Significance Determination of Reactor Inspection Findings for At Power Situations.” Using the Phase I Screening worksheet the finding was determined to be of very low safety significance (Green) since the finding did not represent an actual loss of safety function for greater than the Technical Specification allowed outage time. Therefore, the finding is similar to Item D.5 in NUREG 1600 Supplement I, “Violations of 10 CFR 50.59 that result in conditions evaluated as having very low safety significance (i.e., green) by the SDP.” This is an example of a Severity Level IV violation.

  There is no cross cutting aspect for this finding, because it was determined that this finding is not reflective of current licensee performance. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. A green self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified, in that FENOC failed to properly establish and implement adequate work instructions and acceptance criteria to inspect the fuel transfer system cables. This led to the failure of the cable associated with the Unit 1 spent fuel pool up-ender frame during refueling operations. A new fuel assembly (FA) and an irradiated rod cluster control assembly (RCCA) were contained in the frame during cable failure. The FA and RCCA were not visibly damaged. The affected FA was not used in the core reload. Additional FAs were purchased to satisfy the core design. The licensee affected repairs and performed an extent of condition on the containment side up-ender as well as the Unit 2 up-ender equipment. This issue was entered into the corrective action program as CR 07-28471.

  This finding was more than minor because it affected the procedure quality attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding barrier protects the public from radionuclide release. The inspectors determined the affected FA fuel clad barrier remained intact and that containment controls were unaffected. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance), in accordance with Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process."

  This finding has a cross-cutting aspect in the area of human performance in that safety-related maintenance decisions regarding the inspection and replacement of fuel transfer system cables were based on assumptions (adequate inspection personnel and program) that were not validated and did not consider all possible unintended consequences, [H.1.(b)]. (Section 4OA3)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period shutdown in a planned refueling outage, 1R18 (Section 1R20). On October 24, the unit commenced a reactor startup and reached full power on October 28. The unit remained essentially at full power for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power and essentially remained at full power for the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Seasonal Site Inspection

a. Inspection Scope (1 seasonal sample)

The inspectors reviewed the Beaver Valley Power Station (BVPS) design features and FENOC's implementation of procedures to protect risk significant mitigating systems from cold weather conditions and high winds. The inspectors walked down risk significant plant areas for several days in November and December 2007 and assessed FENOC's protection activities for cold weather conditions. Specifically, the inspectors evaluated to outside instrument line conditions and the potential for unheated ventilation. The walkdown included the safety-related heat tracing, Intake Structure cubicles, and ventilation heating for safety-related areas. The inspectors also reviewed 1OST-45.11, "Cold Weather Protection Verification," Rev. 17 and 2OST-45.11, "Cold Weather Protection Verification," Rev. 18. Other documents that were reviewed are listed in the attachment.

b. Findings
No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns (71111.04Q)

a. Inspection Scope (4 samples)

The inspectors performed a partial walkdown of the following four systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control system components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the attachment.

- On October 31, Unit 2's 2-2 Emergency Diesel Generator (EDG) during an unplanned outage of the 2-1 EDG;

- On November 27, Unit 1 'B' train offsite-to-onsite electrical power source line-up during an identified failure of the 'A'-phase to the 'A' train System Station Service Transformer (SSST) [TR-1A];

- On December 7, Unit 2 'B' train Service Water (SW) using the 'C' SW pump during planned cleaning of the 'A' intake bay; and

- On December 28, Unit 1 'A' train River Water (RW) and Auxiliary River Water (ARW) during planned 'B' train RW surveillance testing.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Sample Review (71111.05Q)

a. Inspection Scope (10 samples)

The inspectors reviewed the conditions of the fire areas listed below, to verify compliance with criteria delineated in Administrative Procedure 1/2-ADM-1900, "Fire Protection," Rev. 16. This review included FENOC's control of transient combustibles and ignition sources, material condition of fire protection equipment including fire detection systems, water-based fire suppression systems, gaseous fire suppression systems, manual firefighting equipment and capability, passive fire protection features, and the adequacy of compensatory measures for any fire protection impairments. Documents reviewed are listed in the Attachment:

- Unit 1, Reactor Containment Area (Fire Area RC-1);
- Unit 1, Steam Generator Blowdown Room (Fire Area SGPD-1);
- Unit 1, Primary Auxiliary Building elevation 768 (Fire Area PA-1A);
- Unit 1, Main Exhaust Filter Bank (Fire Area MF-1);
- Unit 1, Main Exhaust Filter Bank (Fire Area MF-2);
- Unit 2, Fuel Building (Fire Area FB-1);
- Unit 2, Pipe Tunnel (Fire Area PT-1);

- Unit 2, Battery Room 2-4 (Fire Area SB-9);
- Unit 2, Main Feed Reg Valve Room (Fire Area SB-5); and
- Unit 2, Battery Room 2-6 (Fire Area TB-2).

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Resident Sample Review (71111.07A)

a. Inspection Scope (1 sample)

The inspectors reviewed a thermal performance test for the Unit 2 ‘C’ Primary Plant Component Cooling heat exchanger [2CCP-E21C] conducted on March 20, 2007, in accordance with 1/2-ADM-2106, Rev. 2, “River/Service Water System Control and Monitoring Program.” The review included an assessment of the testing methodology and verified consistency with Electric Power Research Institute document NP-7552, “Heat Exchanger Performance Monitoring Guidelines,” December 1991, and Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.” The inspectors reviewed inspection results, related condition reports, and component cooling heat exchanger leak test results against applicable acceptance criteria.

b. Findings

No findings of significance were identified.

.2 Biennial Sample Review (71111.07B)

a. Inspection Scope (3 samples)

Based on a plant specific risk assessment, past inspection results, and resident inspector input, the inspector selected a sample of the following heat exchangers:

- 1C Recirculation Spray (1RS-E-1C)
- 2A Emergency Diesel Generator (2EGS-E21A/E22A)
- 2B Primary Plant Component Cooling (2CCP-E21B)

The inspector verified that common cause heat sink performance problems that had the potential to increase risk were identified and corrected by the licensee. The inspector also verified that potential macro fouling (silt, debris) issues and biotic fouling issues were closely examined. The inspector reviewed FENOC’s methods and frequency of inspection, cleaning, chemical control, and performance monitoring for the selected components to ensure agreement with FENOC’s response to Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment” (GL 89-13). The inspector compared surveillance and inspection results, including as found conditions, photographs, and eddy current examination reports, to the established acceptance criteria to verify that heat exchanger operation was acceptable and consistent with design. The inspector reviewed heat exchanger design basis values and assumptions, plugging limit calculations, and vendor information, to verify that they were incorporated into the heat exchanger inspection and maintenance procedures.

The inspector walked down portions of the Service Water System, River Water System, Emergency Diesel Generators, Primary Plant Component Cooling Water System, and the intake structures, to assess the material condition and operational functioning of these systems and components. The inspector reviewed a sample of condition reports related to the selected heat exchangers and the service water systems and interviewed responsible system engineers to ensure that FENOC was appropriately identifying, characterizing, and correcting problems related to these systems and components.

b. Findings

Introduction: The inspectors identified a green finding because FENOC failed to meet a commitment made in their GL 89-13 program. Specifically, after a self-assessment identified the potential for postponing the cleaning of safety related heat exchangers, which was contrary to the Beaver Valley GL 89-13 commitments, corrective actions were developed to prevent postponement of the planned cleanings. These actions were insufficient to prevent the postponement of cleanings of both the Jacket Water and Intercooler heat exchangers for the 2A Emergency Diesel Generator (EDG) in October 2006.

Description: In their response to GL 89-13, FENOC committed to either test or perform frequent regular maintenance to verify heat transfer capability for the unit 2 EDG Jacket Water and Intercooler heat exchangers. Since there is currently no testing program in place for the EDG heat exchangers, frequent regular maintenance was required. The GL 89-13 commitment for maintenance on the unit 2 EDG heat exchangers is “as required.” Supplement 1 to GL 89-13, part III.B.1 states that the licensee should determine the appropriate frequency of maintenance to ensure that the heat removal requirements for the service water system are satisfied, and in the absence of a routine test program, the frequency of maintenance may have to be a maximum value to provide proper assurance. The River/Service Water System Control and Monitoring Program states that the EDG heat exchanger validation method is mechanical cleaning and inspection once each operating cycle.

Beaver Valley completed a self-assessment of the Heat Exchanger Program in 2004 and created a corrective action in CR 04-05808 CA 12 to revise preventative maintenance tasks to ensure all tasks do not give an option to “clean if necessary.”

In October 2006, the 2A EDG Jacket Water and Intercooler heat exchangers were inspected by the system engineer. Based on the inspections, the system engineer informed the maintenance personnel that a cleaning was not needed. This changed the cleaning frequency for the 2A EDG heat exchangers from once each operating cycle to once in two operating cycles.

Without a testing program to provide a basis for extending the cleaning interval, there existed the potential that microfouling of the tubes could degrade the heat transfer capability of the heat exchangers such that the heat exchangers would not be capable of operating under design conditions. This issue was entered into the corrective action program under CR 07-29900, and a prompt operability determination was completed by FENOC showing reasonable assurance of operability for the 2A EDG Jacket Water and Intercooler heat exchangers.

The inspector determined that FENOC's inappropriate postponement of a cleaning of safety related heat exchangers without sufficient basis, was a performance deficiency that was reasonably within their ability to foresee and prevent. Specifically, these actions were insufficient to prevent the postponement of cleanings for both the Jacket Water and Intercooler heat exchangers for the 2A Emergency Diesel Generator (EDG) in October 2006.

Analysis: This issue was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and did not involve a willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspector conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function or loss of a single train for greater than its allowed technical specification time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events.

The cause of this finding is related to the cross-cutting area of problem identification and resolution, in that a 2004 self-assessment identified this potential vulnerability, but the resulting corrective actions were ineffective in preventing it [P.3 (c)].

Enforcement: No violation of NRC regulatory requirements occurred. However, FENOC's response to GL 89-13 contained commitments that were not met which constituted a performance deficiency and a finding. This finding was of very low safety significance and has been entered into FENOC's corrective action program (CR 07-29900). **(FIN 05000412/2007005-01, Inappropriate Postponement of Safety Related Heat Exchanger Cleaning)**

1R08 Inservice Inspection Activities (71111.08)

Beaver Valley Unit 1

a. Inspection Scope (3 Samples)

From October 1-18, the inspectors conducted a review of the implementation of FENOC's risk-informed Inservice Inspection Program for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries for Beaver Valley Unit 1 using the criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage and upon inservice inspection (ISI) activities available for review during the on-site inspection period. The inspectors reviewed documentation, observed in process non-destructive examinations (NDE), and interviewed technicians to verify that the ISI activities were performed in accordance with the ASME Boiler and Pressure Vessel Code Section XI requirements.

<u>Non-Destructive Examinations Activities and Welding Activities</u>

The inspectors observed the following NDE activities and reviewed completed NDE inspection data records to evaluate compliance with the ASME Code Section V and Section XI requirements and to verify that the indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements.

- Manual Ultrasonic Examinations (UT) of pressurizer relief valve nozzle weld overlay and pressurizer spray nozzle weld overlay and completed UT examination data records

- Manual UT examination inspection data report record review of UT-07-1011 Pressurizer longitudinal weld, location RC 740 PZR and Reactor in-vessel automated UT examination report and indication report record review of UT-07-1012 circumferential weld, location RC-062A-780

- Dye penetrant testing (PT) examination record PT-07-1006 review of reinforcement plate to nozzle weld RH-E-1A-N-5 (inlet) & RH-E-1A-N-6 (outlet)

- PT and magnetic particle testing (MT) examination record review of five pressurizer nozzles prior to weld overlays

- Bare Metal Inspection (BMI) visual examination records Report No. BOP-VT-07-031 and digital photographic review of reactor vessel lower head penetrations inspection, and

- Visual Testing (VT) examination records review of Reactor Vessel and Internals Report No. VT-07-1142, VT-07-1144, VT-07-1145, and VT-07-1146.

The inspectors reviewed pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous 1R17 refueling outage to determine if the welding acceptance and pre-service examinations (e.g., VT, PT, and weld procedure qualification tests) were performed in accordance with ASME Code Sections III, V, IX,

and XI requirements.
During the current 1R18 refueling outage, FENOC mitigated the pressurizer nozzle Alloy 82/182/600 welds to prevent Primary Water Stress Corrosion Cracking (PWSCC) induced through wall cracking in the Reactor Coolant System (RCS) pressure boundary. Mitigation activities included weld overlays on three safety valve nozzles, spray nozzle, and relief valve nozzle on the pressurizer. The inspectors remotely observed automated welding activities associated with the structural weld overlays of the pressurizer dissimilar metal welds on ASME Class 1 pressurizer piping nozzles.

The inspectors reviewed procedures and records associated with the welding activity and observed the weld overlay process and ensured that the correct welding variable settings were being employed. In addition, certifications of the NDE technicians performing the manual-driven, encoded phased array UT examinations, as well as ASME Welder Maintenance Logs of the individual contractors performing the weld overlay activities on the pressurizer nozzles were reviewed.

On October 2, during installation of weld overlays on pressurizer safety nozzles, it was discovered that welding was being performed with a procedure that had not been qualified for the application, and therefore, did not meet ASME Construction Codes (ASME Section III ’65 edition, Winter ’66 addenda, ASME Section IX, latest edition) requirements. The PCI Energy Services welding procedure WPS 3-8/52-TB MCGTAW-N638 used for the P-1 portion of layer 1 of the weld overlays on pressurizer safety nozzles “A”, “B”, and “C was not qualified to ASME Section III and IX requirements for P-1 materials (Condition Report 07-27664). The implemented welding procedure was qualified for "P-3" material; therefore, the contractor proceeded to qualify the welding procedure. FENOC made the decision to proceed with the weld overlays at risk while the procedure qualification testing was in progress. The weld procedure was subsequently qualified for “P-1" material and acceptable for use. This issue remains unresolved until NRC completes its final evaluation of FENOC’s assessment of the use of the referenced procedure for P-1 material. **(URI 05000334/2007005-02, Procedure used for Weld Overlays on Pressurizer Safety Nozzles was not Initially qualified for P-1 Material)**.

<u>Pressurized Water Reactor Vessel Upper Head Penetration (VUHP) Inspection Activities</u>

No inspections were performed of the reactor pressure vessel upper head penetration nozzles during this outage because the Beaver Valley Unit 1 reactor vessel closure head was replaced during the previous 1R17 outage.

<u>Boric Acid Corrosion Control (BACC) Inspection Activities</u>

The inspectors reviewed the BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05 “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.” The inspectors conducted a direct observation of BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements.

The inspectors reviewed the boric acid guidance and accompanied the boric acid walkdown team as part of the initial 1R18 Unit 1 containment entry on September 24, with Unit 1 in Mode 3 at full operating pressure and temperature. The inspectors observed visual inspections to determine if locations where boric acid leaks could cause degradation of safety significant components were emphasized. The inspectors monitored the licensee's inspection of the containment basement level, 692' and the 718' level. The BACC team also inspected the two upper levels. The BACC team had containment maps, lists of previous leakers, and camera equipment. The team members appeared to be knowledgeable of the plant layout and competent in their inspection and evaluation. A total of 52 minor leaks were identified.

The inspectors also sampled the photographic database of all examined areas to verify that visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components. The inspectors also reviewed the procedures being used for visual inspection for evidence of boric acid leakage. The inspectors confirmed that sampled condition reports were assigned corrective actions consistent with the requirements of the ASME Code and 10 CFR 50 Appendix B Criterion XVI. The documents reviewed during this inspection are listed in the Attachment to this report.

<u>Steam Generator (SG) Tube Inspection Activities</u>

The inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements. Beaver Valley Unit 1 replaced all three steam generators during the previous refueling outage 1R17. The inspectors observed acquisition of eddy current testing (ECT) data for tubes in all three steam generators to identify and quantify tube degradation mechanisms and to confirm tube integrity following the first cycle of operation. They also interviewed ECT data analysts and reviewed documents related to the steam generator inservice inspection program.

The inspectors reviewed plant specific steam generator information, tube inspection criteria, integrity assessments, and degradation modes. The inspectors observed a sample of tubes being examined from each generator to verify the entire length was examined and all testing was performed in accordance with EPRI technical report "Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6.

The inspectors reviewed examination data for selected tubes from each of the three steam generators, and evaluated the characterization and disposition of the identified flaws to assess the implementation of the steam generator inspection program. The SG tube ECT examination scope included tube areas which represent potential ECT challenges such as top-of-tube sheet, tube support plates, and U-bends.

The inspectors reviewed portions of the steam generator eddy current acquisition procedure, management plan, and the operational assessment to assess the steam generator inspection and management program. No primary-to-secondary leakage (e.g., SG tube leakage) was identified during the previous operating cycle; no detrimental affects of loose parts on the SG tubes were identified;, no new SG tube degradation mechanisms were identified; no SG tubes required in-situ pressure testing; and one SG tube (R11 C2), in the “C” steam generator, was preventively plugged as a result of an indication being sized at 29% max depth due to support plate wear. The inspectors assessed whether FENOC adequately performed steam generator testing activities and documented the results in accordance with EPRI guidelines and site procedures.

b. Findings

No findings of significance were identified; however, an unresolved item was opened regarding the procedures used for weld overlays on pressurizer safety nozzles were not initially qualified for P-1 material.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope (1 sample)

The inspectors observed Unit 2 licensed operator simulator-based testing on November 15, 2007 during the Green-Team drill. The inspectors evaluated licensed operator performance regarding command and control, implementation of normal, annunciator response, abnormal, and emergency operating procedures, communications, technical specification review and compliance, and emergency plan implementation. The inspectors evaluated the licensee staff training personnel to verify that deficiencies in operator performance were identified, and that conditions adverse to quality were entered into the licensee’s corrective action program for resolution. The inspectors reviewed simulator response to ensure the simulator appropriately modeled expected plant conditions and configurations. The inspectors verified that the training evaluators adequately addressed that the applicable training objectives had been achieved. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Biennial Review by Regional Specialist (71111.11B)

a. Inspection Scope

On December 18, a region-based inspector conducted an in-office review of results of Unit 1 licensee-administered annual operating tests for 2007. Unit 1 comprehensive written exams were administered in the fall of 2006, and therefore were not included in this review. (Results of Unit 2's requalification tests are documented in NRC Inspection

Report 05000334/2007004 and 05000412/2007004). The inspector assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspector verified that:

- Crew failure rate was less than 20%. (Crew failure rate was 0%);
- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Individual failure rate was 0%);
- Individual failure rate on the walk-through test was less than or equal to 20%. (Individual failure rate was 0%); and
- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Overall pass rate was 100%).

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

.1 Routine Maintenance Effectiveness Inspection (71111.12Q)

a. Inspection Scope (2 samples)

The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. The inspectors evaluated specific attributes, such as MR scoping, characterization of failed structures, systems, and components (SSCs), MR risk characterization of SSCs, SSC performance criteria and goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65 and the licensee's program for MR implementation. For the selected SSCs, the inspectors evaluated whether performance was properly dispositioned for MR category (a)(1) and (a)(2) performance monitoring. MR System Basis Documents were also reviewed, as appropriate. Documents reviewed are listed in the Attachment.

- CR 07-27188, "Breaker for 1CH-P-1C(DF) Did Not Stay Closed During 1OST-7.11B"; and
- CR 07-28237, "2-2 D/G Failure to Flash Field."

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13)

a. Inspection Scope (4 samples)

The inspectors reviewed the scheduling and control of four activities, and evaluated their effect on overall plant risk. This review was conducted to ensure compliance with applicable criteria contained in 10 CFR 50.65(a)(4). Documents reviewed during the inspection are listed in the Attachment. The inspectors reviewed the planned or emergent work for the following activities:

- Unit 2, #2 emergency diesel generator voltage regulator failure, repair, and retest impact on plant risk with Unit 1 in a planned outage, on October 10;
- Station risk assessment of work activities during warm weather conditions, October 31 to November 1;
- Unit 1 Yellow PRA risk due to System Station Service Transformer (SSST) [TR-1A] isolation for repairs of the A phase and removal of metering equipment on the 'B' and 'C' phases; and
- Unit 1 and 2 work-week risk which included Solid State testing on Unit 2 and a River Water alignment change due to intake structure work, week of December 3.

b. Findings

Introduction. A green self-revealing NCV of TS LCO 3.8.1, "Electrical Power Systems - AC Sources - Operating," was identified due to FENOC's failure to comply with the LCO actions for one required offsite power circuit inoperable within the specified allowed time requirements. This was due to an inadequate surveillance procedure such that a degraded phase on the 138kV bus 1 off-site line was not identified.

Description. The Beaver Valley Power Station offsite power system consists of two independent 138kV buses (bus 1 and 2), with each bus fed from multiple sources. Each 138kV bus supplies power to its own system station service transformer (SSST) through each phase's integrated revenue equipment (current and voltage potential transformers, CT/PT) and normally-closed bus disconnects. The SSST steps down voltage to 4kV and is the preferred source to the emergency train 4kV bus during shutdown operations. Load tap changers are provided on both secondary windings of transformers TR-1A and TR-1B which improves the voltage regulation of the 4kV buses. Manual capability for controlling the tap changers is available from the control room. The normal power to the emergency train bus during power operations is the unit station service transformer (USST), which is supplied by the associated Unit's main generator output. During power operation, the SSST is the standby power source to the emergency train 4kV bus. The emergency 4kV buses supply safety-related equipment under normal, shutdown, and design basis accident (LOCA) loads. Also, the capability exists to supply the emergency 4kV bus from the other unit.

On November 3, operations staff identified that 138kV Bus 1 and Bus 2 indicated voltages diverged more than expected and the 1A SSST tap changer positions were at significantly different positions than in the past. The operations department determined indications and breaker lineups were satisfactory based on surveillance test acceptance criteria and reports from the switchyard traveling operator (Duquesne Light Co.) that the 138kV buses were 'balanced' and 'solid'. The operations department requested support from engineering to explain the voltage and tap changer position differences.

On November 14, during an offsite power surveillance (1OST-36.7) the ‘A’ train SSST [TR-1A] load tap changer had to be placed in manual to return its phase voltages to within specification. This was entered into the corrective action process as CR 07-30165 to evaluate why the tap changer was not correctly controlling the SSST voltage in automatic.
On November 27, during a walkdown as part of the investigation to CR 07-30165, it was identified that the ‘A’ phase conductor on the Unit 1 three-phase 138kV power line had broken off from the switchyard side of integrated revenue metering equipment. The operations department declared the ‘A’ train power circuit inoperable and entered Technical Specification 3.8.1 Action A for one of the two required offsite circuits inoperable and established immediate compensatory actions to perform switchyard walkdowns as part of their offsite power availability operability check. Based on an extent of condition review, these actions were also implemented on Unit 2. The line was repaired and returned to service on November 28. Through review of local and remote computer data, FENOC determined that phase A to TR-1A SSST had failed at 12:26 pm on November 1. This issue was entered into the corrective action program as CR-07-30614.

A December 2007 report from BETA Laboratory (a FirstEnergy ISO 9001 laboratory) determined the primary terminal on the 138kV Kuhlman KA-145 CT/VT metering unit failed due to an improper braze of the two-piece terminal design. FENOC has communicated with the vendor and has issued operating experience to the industry. FENOC plans to remove all affected metering units of this terminal design. The vendor no longer manufactures a two-piece design and has changed to a one-piece terminal design.

The delay in identifying the degraded condition of the ‘A’ phase to the 1A SSST until November 27 is attributed to FENOC not having effective surveillance to monitor the line on a lightly-loaded transformer and did not physically identify the broken bus bar during traveling operator rounds in the switchyard. This is documented in condition report 07-30764. The ‘A’ train offsite power was inoperable for 641 hours, from November 1, 12:26 pm to November 28, 5:32 pm. The allowed TS 3.8.1 A LCO outage time is 72 hours. During the time ‘A” train of offsite power was inoperable, the associated ‘A’ emergency diesel generator (EDG) was out of service for planned maintenance for 7.2 hours. This is less than the TS 3.8.1 D LCO outage time for 1 source of offsite power and 1 EDG Inoperable of 12 hours. In addition, the affected emergency bus was powered from the unit service transformer throughout this period, and current plant procedures contain contingencies, and operators are trained, to recover an affected dead bus by powering it from the other unit’s EDG. The licensee has initiated a root cause investigation under CR 07-30614. Short term corrective actions include six of the twelve metering units have been removed, and interim changes to the offsite power availability surveillance test and switchyard walkdowns have been implemented. Long term corrective actions are under development.

The performance deficiency is that the power availability of the 138kV Bus 1 to the Unit-1 1A System Station Service Transformer (SSST) [TR-1A] was not effectively monitored such that an open circuit on the phase ‘A’ was not identified, and it was reasonable that

FENOC could have provided appropriate surveillance criteria to detect a phase degradation. This resulted in exceeding the TS 3.8.1 allowed outage time.

Analysis. The finding is more than minor because it is associated with the Initiating Events Cornerstone attribute of procedural quality and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements.

In accordance with inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the finding was determined to be of very low risk significance (Green) because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Because this finding is of very low safety significance and has been entered into FENOC's corrective action program, the violation is being treated as a non-cited violation.

The cause of this finding is related to the cross-cutting area of problem identification and resolution, in that FENOC did not completely and accurately identify this issue in a timely manner [P.1 (a)].

Enforcement. TS LCO 3.8.1 requires, in part, that two qualified offsite power circuits be operable. TS LCO 3.8.1 required action "A" states that an inoperable offsite power circuit must be restored to operable status within 72 hours. If the required action and associated completion time of condition "A" are not met, TS LCO 3.8.1 required action "G" states that the plant must be placed in hot shutdown in 6 hours and cold shutdown in 36 hours. Contrary to the above, from November 1 to November 28, 2007, one qualified offsite power circuit was inoperable for greater than 72 hours and the plant was not placed in the cold shutdown condition. Because the violation is of very low risk significance and FENOC entered the deficiency into its corrective action program as CR-07-30614, this finding is being treated as an NCV consistent with Section VI.A.1of the Enforcement Policy. **(NCV 05000334/2007005-03, Failure to Comply with TS 3.8.1 Required Actions for One Offsite Power Source Inoperable)**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (3 samples)

The inspectors evaluated the technical adequacy of selected immediate operability determinations (IOD), prompt operability determinations (POD), or operability assessments, to verify that determinations of operability were justified, as appropriate. In addition, the inspectors verified that TS LCO requirements and UFSAR design basis requirements were properly addressed. Documents reviewed are listed in the Attachment.

- On October 1, inspectors evaluated the licensee's review and assessment of a Part 21 Notification received by ABB, Inc., in reference to deficiencies identified in samples of L-2 auxiliary switches used in 4160 Volt circuit breaker assemblies, which the licensee documented in CR 07-27544 and CR 07-28465. Susceptible switch assemblies were inspected and no deficiencies were identified. The licensee incorporated additional inspection criteria into work orders for future switch assembly installations;

- On October 11, the inspectors reviewed the IOD and the extent of condition assessment documented in CR 07-28237 concerning emergency diesel generator (EDG) voltage regulator K-1 relay assemblies. A K-1 relay assembly had failed on the Unit 2 #2 EDG on October 10; and

- Inspectors reviewed the POD and follow-up information associated with CR 07-26849, "Flow Lower Than Required During 1OST-30.12A for Train A Recirculation Spray Heat Exchangers".

b. <u>Findings</u>

No findings of significance were identified.

1R19 <u>Post-Maintenance Testing</u> (71111.19)

a. <u>Inspection Scope</u> (4 samples)

The inspectors reviewed the following activities to determine whether the post-maintenance tests (PMT) adequately demonstrated that the safety-related function of the equipment was satisfied given the scope of the work specified, and that operability of the system was restored. In addition, the inspectors evaluated the applicable acceptance criteria to verify consistency with the associated design and licensing bases, as well as TS requirements. The inspectors also verified that conditions adverse to quality were entered into the corrective action program for resolution. Documents reviewed during the inspection are listed in the Attachment.

- On October 14, replacement of Unit 1 spent fuel pool side up-ender cable (WO 200285260);

- On November 1, installation of Unit 1 'C' steam generator water level alarm circuit relays (WO 2002238177, 200287197);

- On November 1, replacement of Unit 2 EDG voltage regulator relay (K1) into panel BV-PNL-2DIGEN-1A (WO 200287210); and

- On November 27, 1OST-7.4, "Centrifugal Charging Pump Test [1CH-P-1A]," Rev. 36, performed following preventative and corrective maintenance activities on Unit 1 Charging Pump 1CH-P-1A (WO 200240002).

b. Findings

Introduction. A green self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified in that the licensee failed to properly implement and control work activities associated with the alarm and status relays for Unit 1 'C' steam generator water level (SGWL), which resulted in a degraded SGWL Hi-Hi and Lo-Lo alarm status for approximately 9 days.

Description. On October 31, 2007, during a planned surveillance test of a Unit 1 'C' SGWL instrumentation loop (L-1FW-496), FENOC personnel identified that the expected status light and alarm annunciator did not activate. Initial investigation of the issue led to the determination that an additional instrumentation loop (L-1FW-495) had been similarly affected. A total of four status and alarm relays between the two instrument loops had not been reinstalled following instrument restoration during the past refueling outage (Section 1R20). Upon discovery of the issue, the four relays were reinstalled and the instruments were satisfactorily retested.

During the refueling outage, SGWL instruments L-1FW-475 and L-1FW-494 are normally reconfigured to provide two separate reactor coolant system (RCS) temporary level indication and alarms during reduced inventory timeframes. These signals are processed through the solid-state protection system (SSPS). However, due to work schedule activities affecting SSPS, an alternate pair of instrumentation loops were utilized (L-1FW-495, L-1FW-496), via an interposing signal path, to process the alarm and status signals. This was accomplished by a temporary alteration process and revisions to the affected maintenance procedures were made. In their normal configuration, L-1FW-495 and L-1FW-496 only provide alarm and indications in the control room. The automatic safety functions of the 'C' SGWL Instrument (Reactor Trip Signal and ESF actuation signals) were not impacted by this temporary alteration.

An apparent cause investigation revealed that during the Unit 1 outage, the 'A' loop was properly restored and retested since changes were made in the body of the procedure to restore and retest the relays. However, due to differences in the 'C' loop procedure, restoration and retest required the use of an appendix. Changes were made to the 'C' loop procedure and an addendum to the work order was developed. However, these changes did not include sufficient information, or reference, in the appendix to ensure reinstallation and retest of the relays for the 'C' loop (L-1FW-495, L-1FW-496). This resulted in a degraded 'C' SGWL Hi-Hi and Lo-Lo alarm status for approximately 9 days. The inspector verified that the safety functions for the 'C' SGWL were unaffected and that alternate alarm status remained available during the degraded condition. This issue was entered into the corrective action program as CR 07-29487. The inspectors reviewed the licensee's apparent cause evaluation. The following observations and conclusions were made:

- In order to provide temporary reactor coolant level alarm status and not impact solid state protection testing, the use of an alternate signal path to process the RCS temporary level alarm status was required;

- The temporary level ‘A’ loop maintenance procedure is written differently than the ‘C’ loop procedure, in that ‘C’ loop requires the use on an appendix to restore and retest the instrument. This difference in method contributed to the cause of the finding;
- There were missed opportunities to identify the procedure deficiency during the review process; and
- There were missed opportunities to identify the deficient condition based on as-left conditions (jumper tags, lifted leads, and stored relays).

The failure to properly implement and control work activities associated with the alarm and status relays for Unit 1 ‘C’ steam generator water level (SGWL) was considered a performance deficiency.

Analysis. The finding is more than minor because it affected the equipment performance attribute of the associated Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, in that alarms and indications which are used by operators to enter alarm response, abnormal, and emergency procedures were unavailable. Traditional enforcement does not apply because the issue did not have an actual safety consequence or potential for impacting the NRC’s regulatory function, and it was not the result of any willful violation of NRC requirements.

The significance of this finding was evaluated using Appendix A, of the NRC’s Significance Determination Process (Manual Chapter 0609). The inspectors determined that this finding was of very low safety significance (Green), The finding was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function or loss of a single train for greater than its allowed technical specification time, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. Because this finding is of very low safety significance and has been entered into FENOC’s corrective action program, the violation is being treated as a non-cited violation.

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC failed to ensure appropriate coordination of work activities during work scope changes to activities affecting the use of ‘C’ SGWL instrumentation during outage periods, which resulted in a loss of configuration control that degraded a safety-related alarm and status indicator [H.3.(b)].

Enforcement. 10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, and shall be accomplished in accordance with these instructions. Contrary to this requirement, in October 2007, FENOC failed to ensure work activities affecting the ‘C’ SGWL instrumentation were appropriately accomplished in accordance with approved procedures which resulted in a degraded ‘C’ SGWL Hi-Hi and Lo-Lo alarm status for approximately 9 days. Because this deficiency is considered to be of very low safety significance (Green), and was entered into the corrective action program (CR 07-29487), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000334/2007005-04, Failure to control Work Activity results in Degraded 'C' Steam Generator Water Level Alarm Function.**)

1R20 Refueling and Outage Activities (71111.20)

Unit 1 Refueling Outage (1R18)

a. Inspection Scope (1 sample)

Unit 1 began refueling outage 1R18 on September 24. This sample is a continuation of the partial sample documented in NRC Inspection Report 50-334 & 50-412/2007004. The inspectors observed selected Unit 1 outage activities to determine whether shutdown safety functions (e.g. reactor decay heat removal, spent fool pool cooling, and containment integrity) were properly maintained as required by TS and plant procedures. The inspectors evaluated specific performance attributes including operator performance, communications, and instrumentation accuracy. The inspectors reviewed procedures and/or observed selected activities associated with the Unit 1 refueling outage. The inspectors verified activities were performed in accordance with procedures and verified required acceptance criteria were met. The inspectors also verified that conditions adverse to quality identified during performance of selected outage activities were entered into the licensee's corrective action program. Documents reviewed are listed in the Attachment. The inspectors also evaluated the following activities:

- Maintenance of decay heat removal flow paths;
- Coordination of electrical bus work and minimization of shutdown risk;
- Emergency diesel generator auto-load tests;
- Low Head Safety Injection full flow test;
- Containment sump installation;
- Weld Overlays on safety nozzles on the upper pressurizer;
- Reactor vessel lower internal lift and minimization of radiation dose;
- Control rod drive split-pin replacements;
- Drain down of reactor coolant and detention of reactor vessel head bolts;
- Replacement of 'C' Reactor Coolant Pump Motor;
- Refueling operations;
- 1R18 Core Map / fuel assembly loading verification;
- Final containment walk down and closeout inspection;
- Low power physics testing;
- Control rod drop measurement and testing;
- Initial approach to criticality for Cycle 19;
- Plant startup, heat-up, and evaluation of heat-up rates; and
- Balance of plant walk down during power ascension.

The inspectors also observed selected management review activities associated with restart readiness of Unit 1. The restart readiness review meeting was accomplished as required by NOBP-OM-4010, "Restart Readiness for Plant Outages" Rev. 4 on October 22.

b. Findings

Introduction. A green self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified in that the licensee failed to properly establish and implement adequate work instructions and acceptance criteria to

inspect the fuel transfer system cables. This led to the failure of the cable associated with the Unit 1 spent fuel pool (SFP) up-ender frame during refueling operations.

Description. The fuel transfer system facilitates the movement of fuel assemblies (FAs) between the SFP and reactor vessel. Two components of this system are the up-ender subsystem and the transfer tube. The up-ender subsystem facilitates repositioning FAs from the vertical position to the horizontal position. This is accomplished with a frame, cable, and pulley system, with the frame tilting at its base. The FA is then moved through the containment wall via the transfer canal. There are two up-ender subsystems per unit (SFP-side and containment-side), which are underwater during refueling operations.

On October 12, 2007, during refueling operations and after the successful transfer of 117 FAs from the SFP to the reactor vessel, the cable associated with the Unit 1 spent fuel pool up-ender frame broke. A new FA and an irradiated rod cluster control assembly (RCCA) were contained in the up-ender frame when the cable failed. The up-ender frame was being lowered from vertical and estimated to be within 2 to 24 inches from the horizontal position when the 3/8-inch 18-8 stainless steel cable supporting the up-ender frame broke underwater. The cable had been in service since 1983. Fuel handling operations were promptly suspended. There were no adverse radiological consequences from the event. Camera inspection showed no visible damage to the FA or RCCA. The inspection also revealed the cable failed near a pulley. Subsequent analysis of the failed cable concluded that the failure was the result of fatiguing of individual strands during service.

The inspectors observed that procedure 1RP-3.2, “Fuel transfer System”, Issue 0 Rev. 3, is performed prior to outage to inspect the system and static visual inspection of the cables. The procedure does not address vendor criteria for replacement or qualification standards (ANSI B30.2-1976) to perform the inspection. The licensee formed an Event Review Team to evaluate the cause of the event, initiated CR 07-28471 to capture the issue in the corrective action program, and performed a root cause evaluation to determine the root and contributing causes of the event.

Corrective actions included an extent of condition on the containment side up-ender as well as the Unit 2 up-ender equipment, replacement of the failed cable, and an event briefing for all refueling personnel. The affected FA was returned to the SFP and not placed in service. Additional FAs were purchased to satisfy core design requirements. The RCCA was inspected satisfactorily to a vendor standard and returned to service. The licensee determined that a program inadequacy in implementing inspection/replacement criteria was a cause of the cable failure.

The inspector reviewed the licensee root cause evaluation. The following observations and conclusions were made:

- In 1983 the vendor (UE&C Nuclear) recommended cable replacement every 10th refueling outage and Repetitive Task 10001 was created;
- Repetitive Task 10001, was scheduled to replace the Unit 1 cable in 2000 (10th refueling outage since last cable replacement during 1R3), but was changed by the licensee to an ‘inspection only’ activity in May 2000.
- No consideration for damage to irradiated fuel was documented as a basis for the Task revision;
- Qualifications per ANSI B30.2-1976, “Overhead and Gantry Cranes” are needed to perform Repetitive task 10001, but not addressed during inspections; and
- Preventive Maintenance (PM) tasks for cable inspections do not reference standards for cable inspection.

The failure to properly establish and implement adequate work instructions and acceptance criteria to inspect the fuel transfer system cables is considered a performance deficiency.

Analysis. This finding was more than minor because it affected the procedure quality attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding barrier protects the public from radionuclide release. Traditional enforcement does not apply because the issue did not have an actual safety consequence or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements.

The finding was evaluated using Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1 – Checklist 4, Refueling Operation. The FA and RCCA were not visibly damaged, however the affected FA was not used in the core reload. The inspectors determined the affected FA fuel clad barrier remained intact and that containment controls were unaffected. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance).

This finding has a cross-cutting aspect in the area of human performance in that safety-related maintenance decisions regarding the inspection and replacement of fuel transfer system cables were based on assumptions (adequate inspection personnel and program) that were not validated and did not consider all possible unintended consequences, [H.1.(b)].

Enforcement. 10 CFR 50, Appendix B, Criterion V, requires in part, that activities affecting quality shall be accomplished in accordance with appropriate procedures, and contain sufficient criteria to ensure satisfactory accomplishment. Contrary to these requirements, from May 2000 until October 2007, FENOC failed to properly establish and implement adequate work instructions and acceptance criteria to inspect and/or replace the fuel transfer system cables, which resulted in the failure of the cable associated with the Unit 1 spent fuel pool up-ender frame during refueling operations. Because this violation is considered to be of very low safety significance (Green) and was entered into the corrective action program as CR 07-28471, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000334/2007005-05, Inadequate Inspection led to a subsequent failure of a Fuel Transfer Up-Ender Cable)**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (1 RCS Leak Detection sample, 1 Isolation Valve sample, and 1 routine surveillance sample)

The inspectors observed Pre-Job test briefings, observed selected test evolutions, and reviewed the following completed Operation Surveillance Test (OST) and Maintenance Surveillance Packages (MSP). The reviews verified that the equipment or systems were being tested as required by TS, the UFSAR, and procedural requirements. Documents reviewed are listed in the Attachment. The following three activities were reviewed:

- 2OST-6.7, “Accident Monitoring Instrumentation Channel Checks,” Rev. 17;
- 1OST-6.2A, “Computer Generated Reactor Coolant System Water Inventory Balance”, Rev.15; and
- 1OST-11.15, “Safety Injection Accumulator Check Valve Test.”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed the following temporary modifications (TMOD) based on risk significance. The associated 10 CFR 50.59 screening was reviewed against the system design basis documentation, including the UFSAR and the TS. Implementation was performed in accordance with Administrative (ADM) Procedure, NOP-CC-2003, “Engineering Changes,” Rev. 11. Documents reviewed are listed in the Attachment.

- Engineering Change Package 07-0315, associated with temporary modifications to terminal connections on the K-1 relay associated with the Unit 1 and Unit 2 Emergency Diesel Generators (EDG). Inspectors walked down the system to verify that the TMOD described was appropriately implemented, and that EDG operability would not be challenged or adversely impacted.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed a Unit 2 licensed-operator simulator evaluation conducted on November 15. Senior licensed-operator performance regarding event classifications and notifications were specifically evaluated. The inspector evaluated a simulator-based scenario that involved multiple, safety-related component failures and plant conditions that would have warranted emergency plan activation, emergency facility activation, and escalation to the event classification of Site Area Emergency. The licensee credited this evolution toward Emergency Preparedness Drill/Exercise Performance (DEP) Indicators, therefore, the inspectors reviewed the applicable event notifications and classifications to determine whether they were appropriately credited, and properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, Rev. 5, “Regulatory Assessment Performance Indicator Guideline.” The inspectors reviewed licensee evaluator worksheets regarding the performance indicator acceptability, and reviewed other crew and operator evaluations to ensure adverse conditions were appropriately entered into the Corrective Action Program. Other documents utilized in this inspection include the following:

- 1/2-ADM-1111, Rev. 2, “NRC EPP Performance Indicator Instructions;”
- 1/2-ADM-1111.F01, Rev. 1, “Emergency Preparedness Performance Indicators Classifications/Notifications/PARS;”
- EPP/I-1a/b, Rev. 11, “Recognition and Classification of Emergency Conditions;”
- 1/2-EPP-I-2, Rev. 29, “Unusual Event;”
- 1/2-EPP-I-3, Rev. 27, “Alert;” and
- 1/2-EPP-I-4, Rev. 27, “Site Area Emergency.”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (10 samples)

During the period October 15 - 19, the inspectors conducted the following activities to verify that the licensee was properly implementing physical, administrative, and engineering controls for access to locked high radiation areas, and other radiologically controlled areas during the Unit 1 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant TS, and the licensee's procedures. This inspection represents the completion of ten (10) samples.

Plant Walkdown and Radiation Work Permits (RWP) Reviews

- The inspectors toured accessible radiologically controlled areas in the Unit 1 reactor building containment, primary auxiliary building, radwaste building and safeguards building, and with the assistance of a radiation protection technician, performed independent radiation surveys of selected areas to confirm the accuracy of survey data, and the adequacy of postings. Radiation protection technicians were questioned regarding their knowledge of plant radiological conditions for selected jobs, and the associated controls.

- The inspectors identified radiologically significant jobs being performed in the Unit 1 reactor building containment. The inspector reviewed the applicable RWPs, ALARA Plans, and the electronic dosimeter dose/dose rate setpoints, for the associated tasks, to determine if the radiological controls were acceptable and if the setpoints were consistent with plant policy. Jobs reviewed included Containment Sump modification (RWP 107-4039, AP 07-01-19), Cavity Decontamination (RWP 107-4009, AP 07-01-16), Pressurizer Weld Overlay (RWP 107-4044, AP 07-01-36), and scaffolding installation/removal activities (RWP 107-4037, AP 07-01-31).

- For the jobs reviewed, the inspector determined that there were no significant dose gradients requiring relocation of dosimetry. The inspectors determined that tele-dosimetry was extensively used to monitor and control worker exposure for high risk jobs including reactor internals removal, diving operations, steam generator entries, and pressurizer weld overlay.

- There were no current radiation work permits for airborne radioactivity areas with the potential for individual worker internal exposures to exceed 50 mrem during the 1R18 outage.

  Additionally, the inspectors determined that during 2007, there were no actual internal exposures greater than 50 mrem Committed Effective Dose Equivalent (CEDE). The inspectors reviewed the CEDE dose assessments for the five highest internal exposures for 2007; no dose exceeded 10 mrem.

  The inspectors also reviewed Bioassays Evaluations for diving operations, steam generator entries, and various personnel contamination incidents, occurring during 1R18, and reviewed the methodology for assessing internal exposure for the subject individuals.

High Radiation Area and Very High Radiation Area Controls

- Changes made to high dose rate high radiation area and very high radiation area procedures, since the last inspection of this area, were reviewed and management of these changes was discussed with the Radiation Protection Supervisor.

- Keys to locked high radiation areas (LHRA), located in Units 1 and 2 were inventoried, and accessible LHRAs were verified to be properly secured and posted during plant tours in Unit 1.

- The inspectors reviewed the preparations made for various high dose rate jobs including the removal of the reactor vessel internals, to perform 10 year in-service inspections. This task required the use of large area water shields, tele-dosimetry, remote monitoring, and using temporary shielding for the crane operator. Additionally, access controls were applied during the lift to assure that no unplanned exposure occurred.

Radiation Worker and Radiation Protection Technician Performance

- During tours of radiologically controlled areas in the Unit 1 reactor building containment, the inspector questioned radiation workers and radiation protection technicians regarding the radiological conditions at the work site and the radiological controls that applied to their task. Additionally, radiologically-related condition reports, including dose/dose rate alarm reports, were reviewed to evaluate if the incidents were caused by repetitive radiation worker or technician errors and to determine if an observable pattern traceable to a similar cause was evident.

- The inspectors attended the pre-job RWP briefing for the reactor cavity decontamination to determine if access to this high radiation area was properly controlled, that workers were informed of past operating experiences, that electronic dosimetry alarm setpoints were appropriate, and that individual responsibilities were discussed.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (9 samples)

During the period October 15 - 19, the inspectors conducted the following activities to verify that the licensee was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for activities performed in the 1R18 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee's procedures. This inspection represents the completion of nine (9) samples.

Radiological Work Planning

- The inspectors reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities to assess past outage ALARA performance, current (2007) exposure trends, and the exposure challenges for the Unit 1 outage.

- The inspectors reviewed the exposure status for tasks performed during the Unit 1 outage and compared actual exposure with forecasted estimates contained in ALARA Plans. Outage jobs reviewed included the containment sump modification (ALARA Plan 07-01-33), the pressurizer weld overlay (ALARA Plan 07-01-56), reactor disassembly/reassembly (ALARA Plan 07-01-22), diving operations (07-01-59) and outage scaffolding construction (ALARA Plan 07-01-31).

- The inspectors evaluated the departmental interfaces between radiation protection, operations, maintenance crafts, and engineering to identify missing ALARA program elements and interface problems. The evaluation was accomplished by reviewing outage Work-in-Progress and Post-Job ALARA reviews, Station ALARA Committee meeting minutes, and interviewing the station Radiation Protection Manager.

Verification of Dose Estimates

- The inspectors reviewed the assumptions and basis for the annual (2007) site collective exposure projections for site operations and for the Unit 1 refueling outage. The inspectors also reviewed the revisions made to various outage project dose estimates due to elevated system source terms.

- The inspectors reviewed the licensee's procedures associated with monitoring and re-evaluating dose estimates when the forecasted cumulative exposure for tasks was approached and the implementation of these procedures during the outage. The inspectors reviewed the exposures for the ten (10) workers who received the highest doses for 2007 to confirm that no individual exceeded the regulatory annual limit.

Job Site Inspections

- The inspectors reviewed the ALARA controls contained in RWP 107-4009, Cavity Decontamination, and attended the pre-job ALARA briefing. During tours of the reactor building containment, the inspector observed workers performing containment sump modifications, valve repairs, and de-mobilization activities. Workers were questioned regarding their knowledge of job site radiological conditions and ALARA measures.

Source Term Reduction and Control

- The inspectors reviewed the status and historical trends for the Unit 1 source term. Through review of survey maps and interviews with the Senior Nuclear Specialist-ALARA, the inspector evaluated recent source term measurements and control strategies. Specific strategies being employed included zinc addition, increased filtration flow, enhanced chemistry controls, system flushes, and temporary shielding.

Declared Pregnant Workers

- The inspectors reviewed the procedural controls for managing declared pregnant workers (DPW) and determined that no DPW was employed during the Unit 1 outage.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (15 samples)

The inspectors sampled licensee data and submittals for fifteen (15) Performance Indicators (PI) listed below for Unit 1 and Unit 2. Inspectors discussed the methods for compiling and reporting the PIs with cognizant licensing and other station personnel. To verify the accuracy of the PI data reported during this period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 5, were used to verify the reporting basis for each data element. The inspectors compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report.

.1 Cornerstone: Mitigating Systems (10 samples)

The inspectors reviewed data from the third quarter 2006 through the fourth quarter 2007.

Mitigating Systems Performance Index (MSPI)
The inspectors reviewed portions of the operations logs and raw PI data developed from monthly operating reports, and train / system unavailability data. Inspectors reviewed the Consolidated Data Entry MSPI Derivation Reports for availability and reliability and MSPI component risk coefficients for the systems listed below:

- Emergency AC power systems (Emergency Diesel Generator)
- High pressure safety injection systems (High Head Safety Injection)

• Auxiliary feedwater systems
• Residual heat removal systems (Low Head Safety Injection & Recirculation Spray)
• Support cooling water systems (River Water [Unit 1] & Service Water [Unit 2])

.2 Cornerstone: Occupational Exposure Radiation Safety (1 sample)

Occupational Exposure Control Effectiveness

The inspectors reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed condition reports, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures.

.3 Cornerstone: Public Radiation Safety (1 sample)

The inspector reviewed relevant effluent release condition reports for the period from the fourth quarter 2006 through the third quarter 2007.

RETS/ODCM Radiological Effluent Occurrences

Reports were reviewed for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5mrad/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrad/qtr for organ dose for gaseous effluents. The inspector reviewed the following documents:

- Monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- Quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- Dose assessment procedures.

.4 Cornerstone: Physical Protection (3 samples)

Security PIs were inspected during the annual security baseline inspection and the documentation was inadvertently omitted from the security baseline inspection report issued on March 6, 2007. This inspection activity represents the completion of three (3) samples relative to this inspection area; completing the annual inspection requirement.

Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment

The review was conducted of the licensee's programs for gathering, processing, evaluating, and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators (PIs). The review included the licensee's tracking and trending reports, personnel interviews and security event reports for the PI data collected since the last security baseline inspection. The inspector noted from the licensee's submittal that there were no reported failures to properly implement the requirements of 10 CFR 73 and 10 CFR 26 during the reporting period.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Daily Review of Problem Identification and Resolution

a. Inspection Scope

In order to help identify repetitive equipment failures or specific human performance issues for follow up, the inspectors performed a daily screening of items entered into FENOC's corrective action program. This review was accomplished by reviewing summary lists of each CR, attending screening meetings, and accessing FENOC's computerized CR database.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

Focused Review of Unit 2 Charging Pump 21C Degraded Bearing

a. Inspection Scope (1 sample)

The inspectors performed a focused review of a degraded pump outboard bearing on Unit 2 charging pump 21C, which was identified by FENOC during a planned component inspection on April 7, 2006. The inspectors reviewed Condition Reports (CR) 06-02686 and 06-02954. The first CR determined the degradation was due to foreign material in the pump’s lubricating oil system and an associated oil flush was not proceduralized. The second CR identified missed opportunities by FENOC staff to manage the equipment degradation with additional rigor. The inspectors toured the applicable plant components, interviewed engineering and maintenance personnel, and reviewed additional related documentation to verify that FENOC appropriately evaluated and addressed the conditions that resulted in the bearing degradation. Documents reviewed are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The condition report specific to the charging pump 21C degraded bearing adequately evaluated the details surrounding the issues. In particular, the cause analysis, as documented in CR 06-02686, determined that the abnormal bearing wear was caused by foreign material left in the oil system following maintenance and a post-maintenance oil flush (Fall 2003). Detailed flush guidance was subsequently added to a charging pump overhaul procedure. However, in response to the inspectors' questions regarding the effectiveness of corrective actions, the system engineer identified an additional maintenance procedure that was not revised to include the detailed oil flush guidance. CR 07-030729 was written to address this item.

In addition, the inspectors noted that there were other weaknesses surrounding FENOC's overall response and coordination of charging pump issues. For example, during maintenance activities, technicians identified additional "hide-out" locations in the lubricating oil system where foreign material could remain in the system during a flush. This was addressed by the licensee by revising the recently developed detailed flush guidance. Also, though CR 06-02954 documented and evaluated the organization's fragmented response and missed opportunities in managing the equipment degradation with additional rigor, the inspectors identified that there were multiple CRs documenting several types of problems associated with both the Unit 1 and Unit 2 charging pumps, but a collective review was not conducted. This type of review could have provided a broader perspective of the number and nature of performance issues that were occurring on the charging pumps. In response to the inspectors' observations, FENOC initiated CRs 07-30720 and 07-30749 to further assess whether a potential adverse trend or common cause exists for the several instances of charging pumps issues.

Review of The Operator Work-Around Program

a. Inspection Scope (1 sample)

The inspectors reviewed the cumulative effects of the existing operator work-arounds (OWA), the list of operator burdens, existing operator aids and disabled alarms, and the list of open main control room deficiencies. This review was performed to identify any effect on emergency operating procedure operator actions, and any impact on possible initiating events and mitigating systems. The inspectors evaluated whether station personnel were identifying, assessing, and reviewing OWAs as specified in administrative procedure BVBP-OPS-0002, "Operator Work-Arounds, Operator Burdens, and Control Room Deficiencies" Rev. 11.

The inspectors reviewed FENOC's process to identify, prioritize and resolve main control room distractions to minimize operator burden. The inspectors reviewed the system used to track these operator work-arounds and burdens and recent licensee self assessments of the program. The inspectors toured the control room and discussed the open items with the operators to ensure the items were being addressed on a schedule consistent with their relative safety significance.

b. Findings and Assessment

No findings of significance were identified. At the time of the inspection, FENOC had no issues classified as operator work-arounds and relatively few operator burdens. These operator burdens were determined to have a minimal impact on the ability of the operator to promptly and appropriately respond to an event. The operators interviewed were aware of the status of the active operator burdens for their unit.

The tracking system in place (SAP) appeared to be effective at ensuring operators and management were aware of operator work-arounds and burdens and ensuring these items were addressed in a timely fashion. However, it was noted that two operator burdens appear to be long standing issues.

- 0600343638, Unit 1 PAB Sump requires manual pump down since 12/15/2000. At least seven CRs have been written for this issue and have been closed to this operator burden item. It is scheduled to be repaired in May 2008; and

- 0600055702, Unit 2 gland seal PCV operator 2GSS-PCV-205B, has leaked by since 2003. Multiple CRs have been written and closed to this operator burden. It is scheduled to be repaired in October 2009.

.3 Observation of Training required by Confirmatory Order EA-07-199

a. Inspection Scope (1 sample)

On November 13, 2007, the inspectors observed regulatory sensitivity training conducted by the FENOC Director of Fleet Regulatory Affairs to the Beaver Valley Power Station (BVPS) site leadership team at BVPS. The team consisted of the Site Vice President and five station directors. Also in attendance was the station Fleet Oversight Manager. The inspectors observed the training and reviewed the training material to verify it was conducted to the appropriate population and accomplished the objectives specified in the confirmatory order.

b. Findings and Observations

No findings of significance were identified. The appropriate station personnel were trained and the specified enabling objectives were covered in sufficient detail specified in the confirmatory order. The confirmatory order specifies four directors at BVPS, but due to the addition of a director-level Work and Outage Management position, five directors were present. Additional information can be found in Davis-Besse inspection report 05000346/2007005 and Perry inspection report 05000440/2007005.

.4 Problem Identification and Resolution (PI&R) Review

Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors performed a review of inservice inspection (ISI) and steam generator related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action documents to verify that FENOC was identifying ISI problems at an appropriate threshold, implementing appropriate corrective actions, and evaluating operating experience and industry generic issues related to inservice inspection activities and pressure boundary integrity. The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified

Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors evaluated the licensee’s program for assuring that access controls to radiologically significant areas were effective and properly implemented by reviewing six (6) departmental self assessments, a Quality Assurance Audit Report, seventeen (17) Quality Field Observation Reports, and thirty-two (32) relevant condition reports. The inspector evaluated if problems were identified in a timely manner, that an extent of condition and cause evaluation were performed, previous radiation surveys remained valid, and corrective actions were appropriate to preclude repetitive problems.

b. Findings

No findings of significance were identified

ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors reviewed elements of the licensee’s corrective action program related to implementing the ALARA program to determine if problems were being entered into the program for timely resolution. Seventeen (17) condition reports related to programmatic dose challenges, and the effectiveness in predicting and controlling worker exposure were reviewed.

b. Findings

No findings of significance were identified

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 Review of Licensee Event Reports (LERs) (1 sample)

a. (Closed) LER 05000334/2007-001, “Valve Testing Program Change Inadvertently Leads to Condition Beyond Design Basis During Test.”

On July 13, 2007, during a review to develop a clearance boundary in preparation to perform a routine quarterly surveillance procedure at Beaver Valley Power Station (BVPS) Unit 1, an operator questioned whether stroking valve MOV-1SI-890B, per the Low Head Safety Injection System (LHSI) surveillance procedure, was appropriate with the plant in Mode 1. BVPS Technical Specifications Surveillance Requirement 3.5.2.1 requires that valves MOV-1SI-890 A and B be verified closed with power to the valve operator control circuit removed, every 12 hours. If either valve were to be open, then discharge from both LHSI pumps would be aligned to simultaneously inject into the Reactor Coolant System (RCS) Hot Legs and Cold Legs. This condition was not analyzed for and it could not be confirmed that, while in this abnormal line up, all design basis accident safety analysis conclusions described in the BVPS Unit 1 UFSAR could be maintained if a Loss of Coolant Accident (LOCA) were postulated to occur.

Investigation discovered that in May 2006, the surveillance procedure was changed, increasing the periodicity from 18 months (the TS required action is not applicable during refueling modes) to quarterly. The surveillance procedure change, Inservice Testing (IST) program change, and 10 CFR 50.59 screenings performed in April and May of 2006 did not identify that stroking of MOV-1SI-890 A & B was prohibited by TS or the impact on the UFSAR safety analysis. From May 2006 to July 2007, these valves had been stroked quarterly with the plant in Mode 1. MOV-1SI-890A was stroked five times and MOV-1SI-890B was stroked four times. Each valve stroke was no longer than 30 minutes and was preformed in accordance with the surveillance procedure. The licensee entered this into the corrective action program as CR 07-23462, conducted a root cause analysis and an extent of condition review, and revised the LHSI surveillance procedures (1OST-47.3F, and 1OST-47.3K) to ensure TS requirements were observed. The inspectors reviewed this LER and determined this was a more than minor violation of 10 CFR 50.59.

The finding is documented below. The licensee has documented this event in their corrective action program under CR 07-23462. This LER is closed.

b. <u>Findings</u>

<u>Inadequate 10 CFR 50.59 Review results in Condition Beyond Design Basis During Test.</u>

<u>Introduction</u>. The inspectors identified a Severity Level IV non-cited violation, in that the licensee did not perform an adequate safety evaluation in accordance with 10 CFR 50.59 associated with changing the periodicity of IST testing of valves MOV-1SI-890 A&B in May 2006. The review did not identify that the change allowed operation of these valves in Operational Modes that were prohibited by TS. Operation of these valves in Modes 1-4 placed the plant in an unanalyzed condition and may impact whether all design basis accident safety analysis conclusions described in the BVPS Unit 1 updated final safety analysis report (UFSAR) could be maintained if a Loss of Coolant Accident (LOCA) were postulated to occur while in this abnormal configuration. From May 2006 until July 2007, MOV-1SI-890 A&B were cycled nine times total.

<u>Description.</u> In June 2005, as part of an Extended Power Uprate Review, FENOC

personnel identified that additional hot leg injection flow was needed for Extended Power Uprate conditions (CR 05-04366). Engineering Change Package (ECP) 05-0280 was developed which credited MOV-1SI-890 A & B to open and Design Interface Evaluations (DIE) 2 and 16 recommended adding MOV-1SI-890 A&B to the BVPS IST program and identified these valves can only be stroked while the plant is in a shutdown condition. In January 2006, the appropriate IST procedures (1OST-47.3F rev 7 and 1OST-47.3K rev 10) were revised to require stroke time testing in the open direction at a refueling outage frequency (18 months).

IST program requirements state that all valves will be stroked timed quarterly unless justified. IST personnel contacted the operations department and questioned why this valve could not be cycled quarterly while on-line as other similar valves. Operations personnel were not able to provide any justification at that time. In April 2006, BVPS-1 IST Program Rev 20 was issued for stroke time testing of MOV-1SI-890 A&B on a quarterly basis per 1OST-47.3F and 1OST-47.3K. In May 2006, 1OST-47.3F rev 9 and 1OST-47.3K rev 11 were issued which revised stroke time testing to quarterly from a refueling outage frequency (18-month). 10 CFR 50.59 reviews were conducted for the IST program change, both surveillance procedure changes, and the ECP.

TS Surveillance 3.5.2.1 requires that valves MOV-1SI-890 “A” and “B” be verified closed with power to the valve operator control circuit removed, every 12 hours with the reactor in Modes 1-4. If either valve were to be open, then discharge from both LHSI pumps would be aligned to simultaneously inject into the Reactor Coolant System (RCS) Hot Legs and Cold Legs. This condition was not analyzed for and FENOC could not confirm that while in this abnormal arrangement, all design basis accident safety analysis conclusions described in the BVPS Unit 1 UFSAR would be maintained if a Loss of Coolant Accident (LOCA) were postulated to occur.

Thus, the revised procedure caused the plant to violate TS 3.5.2.1 because a TS revision had not been processed and approved. The 10 CFR 50.59 reviews on this issue were narrowly focused and only considered the IST program aspects of this change. The review did not adequately consider the impact of TS 3.5.2, TS 4.5.2, or ECP 05-0280 DIEs 2 and 16 which would have lead them to the TS requirements.

On July 13, 2007, during a review to develop a clearance boundary in preparation to perform a routine quarterly surveillance procedure at Unit 1, an operator questioned whether stroking valve MOV-1SI-890B per the Low Head Safety Injection System surveillance procedure was appropriate with the plant in Mode 1. From May 2006 to July 2007, these valves had been stroked quarterly with the plant in Mode 1. MOV-1SI-890A was stroked five times and MOV-1SI-890B was stroked four times. Each valve stroke was no longer than 30 minutes and was performed in accordance with the surveillance procedure. Therefore, the amount of time the LHSI system would have been inoperable due to testing was less than the allowed 72 hour TS outage time.

The licensee entered this issue into the corrective action program as CR 07-23462, conducted a root cause analysis and an extent of condition review, and revised the LHSI surveillance procedures (1OST-47.3F, 1OST-47.3K) to ensure TS requirements were being met. The licensee determined that this event was reportable and issued LER

05000334/2007-001, “Valve Testing Program Change Inadvertently Leads to Condition Beyond Design Basis During Test.”

FENOC identified the TS violation and properly reported it to the NRC. Normally, if a licensee-identified Green finding is a violation, it would be documented in the inspection report in Section 4OA7, “Licensee-Identified Violations.” However, if further inspection added significant value, then the finding is documented as an NRC-identified finding under the applicable section of the report. In this case, the NRC added value by evaluating the 10 CFR 50.59 violation, which resulted in the subsequent TS violations, identified three distinct 10 CFR 50.59 reviews which failed to identify the issue, and identified multiple points within each 10 CFR 50.59 review which should have identified the change was contrary to TS and created an unanalyzed condition.

The inspectors determined that the licensee’s inadequate 10 CFR 50.59 safety evaluation was a performance deficiency that should have reasonably been expected to have been foreseen.

Analysis. This performance deficiency affected the mitigating systems cornerstone and warranted a significance evaluation. Because this was a violation of 10 CFR 50.59, it was considered to be a violation which potentially impedes or impacts the regulatory process. Therefore, such violations are characterized using traditional enforcement process. In this case, the licensee failed to perform an adequate safety evaluation in accordance with 10 CFR 50.59 because the approved change proceduralized an operation that was prohibited by the plant’s TS. This change required prior approval from the NRC before its implementation. Comparing this item to the examples in NUREG 1600 Supplement I, ”Reactor Operations,” this finding is more than minor because NRC approval would have been required.

The inspectors completed a Significance Determination Review using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At Power Situations.” Using the Phase I Screening worksheet the finding was determined to be of very low safety significance (Green) since the finding did not represent an actual loss of safety function for greater than the TS allowed outage time. Comparing this item to the examples in NUREG 1600 Supplement I, this finding is similar to Item D.5, “Violations of 10 CFR 50.59 that result in conditions evaluated as having very low safety significance (i.e., green) by the SDP.” This is an example of a Severity Level IV violation.

There is no cross cutting aspect for this finding, because it was determined that this finding is not reflective of current licensee performance.

Enforcement. 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides the basis for determination that the change, test, or experiment does not require a license amendment. Contrary to the above, in May 2006, the licensee conducted 10 CFR 50.59 reviews on three occasion for surveillance procedure and IST program changes which did not identify that it allowed operations of valves MOV-1SI-890 A&B in modes prohibited by TS. Therefore, a licensee amendment would have been required. This change was approved and implemented without the required TS amendment, which caused the licensee to be in violation of TS 3.5.2.1 on nine occasions from May 2006 to July 2007.

This violation was determined to be of very low safety significance, and the violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV violation. Because this non-willful violation was non repetitive, and was captured in the licensee's corrective action program (CR 07-23462), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000334/2007005-06, Inadequate 10 CFR 50.59 Review Results in Condition Beyond Design Basis During Test.**)

.2 Event Followup

Spent Fuel Pool (SFP) Fuel Up-Ender Cable Failure – October 12 - Unit 1:

a. Inspection Scope (1 sample)

On October 12, at 10:35 p.m., the stainless steel cable to Unit 1 fuel assembly up-ender (on the spent fuel pool side of containment) failed due to fatigue failure (see Section 1R20), while containing a new fuel assembly (FA) and irradiated rod cluster control assembly (RCCA). No visual damage was observed to the FA, RCCA, or surrounding equipment. There were no adverse radiological consequences from the event. The inspectors evaluated the response of station personnel and the evaluation of immediate consequence to this event. The inspectors verified that no entry into an emergency action level was warranted.

The station ceased refueling operations until a preliminarily event investigation was conducted. An Event Review Team (ERT) was assembled per 1/2-ADM-0703, Rev. 1, "Event Review." Divers performed an initial evaluation, subsequent cable replacement, and follow-up loose material cleaning. The inspectors assessed the licensee's immediate corrective actions and basis to recommence refueling. The inspectors reviewed the root cause report, evaluated the adequacy of short-term corrective actions, and verified appropriate measures were implemented to prevent recurrence. Refueling activities recommenced October 15, at 2:23 p.m.

b. Findings

No findings of significance were identified related to FENOC's immediate actions for this event. A green self-revealing NCV was identified for the up-ender cable failure. See Section 1R20 for additional details.

.3 Review of Personnel Performance during Non-Routine Operations

a. Inspection Scope (1 sample)

The inspectors reviewed one event that demonstrated personnel performance in coping with non-routine evolutions and transients. The inspectors observed operations in the control room and reviewed applicable operating and alarm response procedures, TS, plant process computer indications, and control room shift logs to evaluate the adequacy of FENOC's response to the following event:

- Unit 2: On November 15, at 12:53 p.m., during a planned replacement of the 26VDC power supply for Primary Process Rack 1, the redundant power supply de-energized, causing a loss of Process Rack 1. As a result, reactor coolant system letdown isolated and abnormal operating procedure (AOP) 2.7.1 “Loss of Charging or Letdown,” was entered. The crew established excess letdown and stabilized plant parameters. Appropriate TS LCOs were entered. Both power supplies were returned to service and post-maintenance testing was completed satisfactorily. Process Rack 1 was restored and normal reactor coolant charging letdown was established. The inspectors verified the event was entered into the corrective action program to resolve identified adverse conditions.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Temporary Instruction (TI) 2515/166 - Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

The inspectors performed the inspection in accordance with TI 2515/166, Rev. 1. The TI was developed to support the NRC review of licensee activities in response to NRC Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors.” Specifically, the inspectors verified implementation of the modifications and procedure changes was consistent with the proposed actions committed to in the GL response. The inspectors reviewed a sample of the licensing and design documents to verify that they were either updated or in the process of being updated to reflect the modifications, and the new requirements for containment sumps and debris generation sources. This included a sample of design change packages, drawings, testing and surveillance procedures, and calculations. The inspectors observed construction activities and performed several walkdowns of the strainer to verify it was installed in accordance with the approved design change package. Additionally, the inspectors walked down samples of piping inside containment to verify that the analyzed zone-of-influence during postulated loss-of-coolant accidents was appropriately considered. Finally, the inspectors walked down areas for potential choke-points that could prevent water from reaching the recirculation sump during a design basis accident.

b. Evaluation of Inspection Requirements

The TI requested the inspectors to evaluate and answer the following questions:

- Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 response?

  The inspectors verified that actions implemented by the licensee as described in response to GL 2004-02 were complete as related to the installation of the sump screen and evaluation of potential debris sources inside containment. Additionally, the inspectors found that procedures to programmatically control potential debris generation sources were updated appropriately. The inspectors noted that FENOC had not competed evaluation of downstream effects, or the effects of chemical precipitants on the strainer head loss at the time of the inspection.

- Has the licensee updated its licensing basis to reflect the corrective actions taken in response to GL 2004-02?

  The inspectors verified that changes to the facility or procedures as described in the UFSAR, and identified in FENOC's GL 2004-02 responses, were reviewed and documented in accordance with 10 CFR 50.59. Inspectors also verified that FENOC had obtained NRC approval prior to implementing changes that require such approval. Specifically, via license amendment 334, FENOC obtained NRC approval prior to implementing changes to the Recirculation Spray System start signal. Finally, the inspectors verified that FENOC was appropriately updating the Unit 1 licensing bases to reflect the modification and associated procedure changes in response to GL 2004-02.

The TI will remain open to allow for the review of portions of the GL response that have not been completed. Specifically, FENOC had not completed their downstream effects or chemical precipitant analyses. The results of these analyses have the potential to impact the final size of the strainer, licensing basis and programmatic procedures. Therefore, the inspection will be considered incomplete until the results are reviewed and accepted. FENOC plans to evaluate the strainer for adequacy once the test results that quantify head loss are known.

c. <u>Findings</u>

No findings of significance were identified.

.2 <u>Response to Confirmatory Action Letter (CAL) No. NRR-07-011: Unit 1</u>

a. <u>Inspection Scope</u>

Inspectors reviewed actions and commitments concerning inspection, monitoring, and mitigation of Alloy 82/182 pressurizer butt welds (CAL NRR-07-011) for Unit 1 documented in FENOC response letter L-07-031, dated February 23, 2007.

b. Findings and Observations

No findings of significance were identified. The committed actions of an enhanced reactor coolant leakage monitoring at appropriate thresholds was implemented prior to weld mitigation. During refueling outage 1R18, weld mitigation activities included weld overlays on three safety valve nozzles, spray nozzle, and relief valve nozzle on the pressurizer (see section 1R08 and 1R20).

4OA6 Management Meetings

.1 Inservice Inspection

On October 18, the inspector presented the inspection results to members of FENOC management and staff, at the conclusion of the inspection. The licensee acknowledged the conclusions and observations presented. The inspectors returned proprietary information reviewed during the inspection. No proprietary information is presented in this report.

.2 Access Control / ALARA Planning and Control

On October 19, the inspector presented the inspection results to members of FENOC management and staff. The licensee acknowledged the conclusions and observations presented. No proprietary information is presented in this report.

.3 TI 2515/166 – Unit 1 Pressurized Water Reactor Containment Sump Blockage

On October 29, the inspector presented the inspection results to members of FENOC management and staff, at the conclusion of the inspection. The licensee acknowledged the conclusions and observations presented. The inspectors returned proprietary information reviewed during the inspection. No proprietary information is presented in this report.

.4 Heat Sink Performance

On November 9, the inspector presented the inspection results to members of FENOC management and staff. FENOC management agreed that none of the information retained by the inspector was considered proprietary. No proprietary information is presented in this report.

.5 Quarterly Inspection Report Exit

On January 10, 2008, the inspectors presented the normal baseline inspection results to Mr. Peter Sena, Site Vice President, and other members of the licensee staff. The licensee acknowledged the conclusions and observations presented. The inspectors confirmed that proprietary information was not retained at the conclusion of the inspection period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Alberti	Steam Generator Program Owner
S. Baker	Site, Radiation Protection Manager
T. Bean	LOR Program Administrator
R. Boyle	System Engineer
A. Brunner	System Engineer
G. Caccianni	Design Engineering
J. Clark	Radiation Protection Health Services Technician
D. Craine	Manager Nuclear Security
P. Davis	Design Engineering
J. Fontaine	Supervisor, ALARA
M. Fox	Security Operations Supervisor
L. Freeland	Director Performance Improvement
J. Freund	Supervisor, Rad Operations Support
D. Girdwood	Radiation Protection, Quality Assessor
D. Grabski	Technical Services Engineering, ISI Coordinator
T. Heimel	Technical Services Engineering, NDE Level III
E. Hubley	Director of Site Maintenance
J. Kasunick	Project Manager
B. Klinko	System Engineering
R. Kuhn	Nuclear Specialist
J. Lebda	Supervisor, Radiation Protection Services
E. Loehlein	Technical Services Engineering, Alloy 600 Program Owner
C. Mancuso	Design Manager
B. Manko	System Engineering
M. Manoleras	Director, Engineering
J. Meyers	System Engineer
D. Mickinac	Regulatory Compliance
L. Miller	Fire protection Engineer
K. Mitchell	System Engineer
M. Mitchell	Supervisor, Maintenance Work Planning and Support
B. Murtagh	Design Engineering Supervisor
K. Ostrowski	Director, Site Operations
J. Rice	Senior Radiation Protection Technician
A. Ryan	Project Manager
P. Sena	Site Vice President
B. Sepelak	Supervisor, Regulatory Compliance
T. Sockaci	Supervisor, Engineering
M. Testa	Design Engineering
K. Troxler	Design Engineering
T. Westbrook	Design Engineering
J. Witter	Operations Support

Other Personnel

P. Aerts	MPR Associates, Inc. Engineer
M. Galler	Manager Welding Engineering, PCI Energy Services
L. Ryan	Inspector, Pennsylvania Department of Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000334/2007005-02	URI	Weld Overlays on Pressurizer Safety Nozzles Not Initially Qualified for P-1 Materials. (Section 1R08)

Open/Closed

05000412/2007005-01	FIN	Ineffective Corrective Action for Preventing Postponing of Safety Related Heat Exchanger Cleaning. (Section 1R07)
05000334/2007005-03	NCV	Failure to Comply with TS 3.8.1 Required Actions for One Offsite Power Source Inoperable. (Section 1R13)
05000334/2007005-04	NCV	Failure to Control Work Activity Results in Degraded ‘C’ Steam Generator Water Level Alarm Function. (Section 1R19)
05000334/2007005-05	NCV	Inadequate Inspection and Subsequent Failure of Fuel Transfer Up-Ender Cable. (Section 1R20)
05000334/2007005-06	NCV	Inadequate 10 CFR 50.59 Review results in Condition Beyond Design Basis During Test. (Section 4OA3)

Closed

05000334/2007-001-00	LER	Valve Testing Program Change Inadvertently Leads to Condition Beyond Design During Test (Section 4OA3)

Discussed

Temporary Instruction 2515/166	Pressurized Water Reactor Containment Sump Blockage (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Condition Reports

07-31997	07-31150	07-31073
07-31234	07-31139	

Section 1R04: Equipment Alignment

Procedures

1OST-36.2, Rev. 49, "Diesel Generator No. 2 Monthly Test"
1OST-36.7, Rev. 14, "Offsite to Onsite Power Distribution System Breaker Alignment Verification"
2OST-36.2, Rev. 53, "Emergency Diesel Generator [2EGS*EG2-2] Monthly Test"
2OST-36.7, Rev. 10, "Offsite to Onsite Power Distribution System Breaker Alignment Verification"

Drawings

8700-RM-430-1, Rev 24, "BVPS-1 Piping & Instrumentation River Water System"
8700-RM-436-1, Rev. 6, "Emergency Diesel Generator Air Start System"
8700-RM-436-2, Rev. 7, "Emergency Diesel Generator Fuel Oil System"
8700-RM-436-3, Rev. 2, "Emergency Diesel Generator Lube Oil System"
8700-RM-436-4, Rev. 4, "Emergency Diesel Generator Water Cooling System"

Clearances

1W04-34-IA-002	2W00-36-SM-011

Condition Reports

07-28237	07-28491	07-29481
07-28287	07-28510	07-29494

Section 1R05: Fire Protection

Drawings

10080-RM-433-1A, Rev. 16, "Fire Protection Water Distribution Network"
10080-RM-433-1B, Rev. 7. "Fire Protection Water- Misc Buildings

Condition Reports

07-31271	07-31198	07-27588	07-26555	07-21508

Pre-Fire Plans

U2 SB-5	U2 SB-9	U2 FB-1 & FB-1A	U2 PT-1

Other
2DBD-33B, Rev. 7, "Design Basis Document for Fire Protection System."
10080-B-085, Rev. 12, "BVPS-2 Fire Hazards Analysis"
Green Tag #36898
NOTF 60038870
WO 20026599

Section 1R07: Heat Sink Performance

Condition Reports

03-10036	05-07862	07-29819 *	07-29825 *
04-05808	07-13112	07-29820 *	07-29826 *
04-06901	07-29472	07-29822 *	07-29827 *
05-07809	07-29480	07-29823 *	07-29876 *
05-07842	07-29792 *	07-29824 *	07-29900 *

* Initiated as a result of this inspection

Procedures
2OM-30.2.A, Service Water System Precautions and Limitations, Rev. 11
2OM-30.4.A, Service Water System Operating Procedure, Rev. 16
1/2OST-30.19A, Main Intake Structure 'A' Bay Silt Check and Bay Cleaning, Rev. 8
NOP-SS-2101, Engineering Program Management, Rev. 3
1OM-53C.4.1.30.2, River Water/Normal Intake Structure Loss, Rev. 6
1/2-ADM-1738, Closed Loop and River Water Systems Monitoring Program, Rev. 2
1OM-30.1.B, River Water System Description, Rev. 5
1OM-30.2.B, River Water System Setpoints, Rev. 8
1/2-ADM-2106, River/Service Water System Control and Monitoring Program, Rev. 2

Calculations
10080-N-795, Minimum Tube Wall Thickness and Maximum Tube Plugging for 2EGS-E21A/B, Rev. 0
10080-N-800, Minimum Service Water Flow Requirements for U2 EDGs, Rev. 0
10080-N-805, Minimum Tube Wall Thickness and Maximum Tube Plugging for 2EGS-E22A/B, Rev. 0
8700-DMC-1548, Recirculation Spray Heat Exchanger Inputs to MAAP Containment Analysis, Rev. 0
8700-DMC-2353, Tube Plugging Limits for Recirculation Spray Heat Exchangers at Beaver Valley Unit 1, Rev. 3
10080-N-779, Main Intake Bay Silt Buildup Limits, Rev. 0
10080-N-829, Tube Plugging Criteria for 2CCP-E21A/B/C, Rev. 0

Test/Surveillance Results
Eddy Current Examination Reports, Dated 3/1/00, 10/1/03, 10/1/04, 10/1/07
PGT-2002-1714, Thermal Performance Test Data Evaluation, Rev. 0
Main Intake Structure Silt Check and Bay Cleaning, Dated 4/37/06, 4/3/07, 7/3/07, 7/10/07
2EGS-E21A/E22A Visual Inspection Reports, Dated 9/27/03, 4/5/05, 10/6/06
2EGS-E21B/E22B Visual Inspection Reports, Dated 10/19/06
1RS-E-1C Visual Inspection Reports, Dated 10/26/04, 3/1/06, 10/2/07

2CCP-E21B Visual Inspection Reports, Dated 12/17/05, 9/18/06, 5/23/07

System Health Reports

Unit 1 River Water System Health Reports, Quarters 2006-4, 2007-1, 2007-2
Unit 2 Service Water System Health Reports, Quarters 2006-4, 2007-1, 2007-2
GL 89-13 Program Health Report, Quarters 2007-1, 2007-2, 2007-3

Vendor Manuals

2504.110-12B-001, Primary Component Cooling Water Heat Exchangers, Rev. D
08700-04.021-0011, Recirculation Spray Water Coolers, Rev. E
438531-0, Specification Sheet for Unit 2 EDG Intercooler, Dated 2/18/76
438541-0, Specification Sheet for Unit 2 EDG Jacket Water Cooler, Dated 2/18/76

Drawings

8700-RM-0430-001, P&ID River Water System, Sheets 1-6, Rev. 29
10080-RM-430-1, P&ID Service Water System, Sheets 1-5, Rev. 31
10080-RT-136C, Tube Sheet Map for 2EGS-E22A, Rev. 1
10080-RT-136A, Tube Sheet Map for 2EGS-E21A, Rev. 1
10080-RT-115B, Tube Sheet Map for 2CCP-E21B, Rev. 1
10080-RT-113, Tube Sheet Map for RS-E-1C, Rev. 6

Work Orders

200059595	200071998	200124617	200207728
200059649	200099121	200150031	
200060006	200113851	200188876	

Miscellaneous

BV-SA-04-07, Heat Exchanger Program Self-Assessment, Dated 7/22/04
Duquesne Light Company letter, J. Sieber to NRC, dated 6/27/91, Followup to GL 89-13
Duquesne Light Company letter, J. Sieber to NRC, dated 1/29/90, Response to GL 89-13
FENOC letter, L. Pearce to NRC, dated 5/13/05, Commitment Change for GL 89-13

Section 1R08: Inservice Inspection

Miscellaneous

Wesdyne Beaver Valley Unit 1 Pressurizer Structural Weld Overlay Project Final Examination Report, Outage 1R18
Wesdyne Overlay Ultrasonic Testing Examination Indication Report Data Sheets for Pressurizer Spray Nozzle and PORV Nozzle, dated 10/17/2007
Liquid Penetrant Examination Report Results
ECP No. 06-0236-04, Pressurizer Nozzles Weld Overlay - PZR Safety EWOL (RC-99-1-E-03/RC-99-1-01, Rev. 0
10CFR 50.59 Screen No. 07-03184 for ECP No. 06-0236
PCI Energy Services Weld Overlay for Pressurizer Safety Nozzle RC-97-1-OL-01 Repair Traveler
Ultrasonic Testing Report No. SR-A-003, dated 10/8/07
1R18 Steam Generator Degradation Assessment (SG-CDME-07-24), dated September 5, 2007
Response to Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections
Pressurizer Safety Nozzle Overlay Profiles, A, B, C

Procedures
NDE-VT-500, General Requirements for Visual Examination, Rev. 12
NDE-VT-502, Leakage Examination Requirements, Rev. 8
NDE-VT-510, Visual Inspection for Evidence of Boric Acid Leakage, Rev. 14
NDE-VT-513, Visual Examination of the Reactor Vessel Bottom Mounted Instrumentation (BMI) Nozzles, Rev. 2
NOP-ER-2001, Boric Acid Corrosion Control Program, Rev. 4
NOP-CC-5002, Control of Special Processes, Rev. 1
1/2-ADM-2112, Boric Acid Corrosion Control, Rev. 3
1/2-ADM-2039, Beaver Valley ISI 10-Year Plans, Rev. 6
1/2-ADM-0801, ASME Section XI Repair/Replacement Program, Rev. 5
PDI-UT-8, Generic Procedure for the Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds, Rev. F
WPS 3-8/52-TB MCGTAW-N638, ASME IX Welding Procedure Specification Base Metals, Rev. 7, for P-3 Material
WPS 1-8/52-TB MCGTAW-N638, ASME IX Welding Procedure Specification Base Metals, Rev. 0 (PQR 781, Rev. 0) for P-1 Material
MRS-SSP-2100, BV Unit 1 Structural Weld Overlay Field Service Procedure, Rev. 1
WCAP-16739-P, BV Unit 1 Pressurizer Nozzles Structural Weld Overlay Qualification, Rev. 0

Certifications
Welder Certifications PCI Energy Services, Identifications M-1337; M-1217; M-1458

Condition Reports

07-15333	07-17961	07-26980	07-27135	07-27418	07-27424
07-27542	07-27543	07-27658	07-27664	07-27809	07-27964
07-28088	07-28221	07-28223	07-28604	07-28845	

Section 1R11: Licensed Operator Requalification Program

Procedures & Forms
½-ADM-1106, Rev. 15, “Drill/Exercise Scenario Development, Preparation and Conduct”
½-ADM-1111.F01, Rev. 2, “Emergency Preparedness Performance Indicators Classification/Notification/PARS”, dated 11/15/07
½-ADM-EPP-IP-1.1.F01, Rev. 1, “Initial Notification Form,” dated 11/15/07
NRC Form 361, Rev 12-2000, “Reactor Plant Event Notification Worksheet,” for drill, dated 11/15/2007

Condition Reports
07-30239

Other
Green Team Mini-Drill #4 Controller – Evaluator Manual, dated November 15, 2007

<u>Section 1R12: Maintenance Rule Implementation</u>

<u>Procedures</u>
1OST-7.11B, Rev 4, “CHS and SIS Operability Test – Train B”

<u>Other</u>
Maintenance Rule Failure Review Evaluation 07-27188-02, BV-4KVS-1DF-1F15
BVPS-1 Shutdown Risk Assessment, September 26, 2007
Notification 415092
Work Order 600417796, 600415020
Tagout SW00-36-SM-011

<u>Section 1R13: Maintenance Risk Assessment and Emergent Work Control</u>

<u>Procedures</u>
1OST-36.7, Rev. 14, “Offsite to Onsite Power Distribution System Breaker Alignment Verification”
2OST-36.7, Rev. 10, “Offsite to Onsite Power Distribution System Breaker Alignment Verification”

<u>Surveillances</u>
1OST-36.7, Rev. 14, “Offsite to Onsite Power Distribution System Breaker Alignment Verification”, dated 11/29, 11/28, 11/27, 11/21, 11/14, 11/13, 11/3, 10/31 (2007)
2OST-36.7, Rev. 10, “Offsite to Onsite Power Distribution System Breaker Alignment Verification”, dated 11/30/2007

<u>Diagrams</u>
8700-RE-1A, “BVPS 1 – Main One-Line Diagram,” Sh 1, dated May 01, 2006
8700-RE-1B, “BVPS 1 – Main One-Line Diagram,” Sh 2, dated May 01, 2006
8700-RE-1C, “BVPS 1 – Equipment One-Line Diagram,” dated October 22, 2006
8700-RE-1D, “BVPS 1 – 4160V One-Line Diagram”, Sh 1, dated May 09, 2003
8700-RE-1GA, “BVPS 1 – ERFS Transformers 3A & 3B,” dated April 21, 1996
9445010B – Kulhman Electric Dimensional Diagram for Metering Outfit KA-145, dated May 09, 2005

<u>Work Orders</u>

200239138	200240128	200241110	200208609	600424573

<u>Condition Reports</u>

07-27573					
07-31562	07-31109	07-30764	07-30730	07-30724	07-30614
07-30165	07-27509	07-27439	07-14560		

<u>Other</u>
1DBD-36B, Rev. 7, “Design Basis Document for 4.16kV Power Distribution System”
BV1-BVA-24, Rev. 2, Relay Setting Sheet for 46-V108
BVPS-1 Shift Operator Logs dated November 1 through November 29, 2007

BVPS-1 Temporary Log for SSST (Bus 1A) Tap Changer in Manual, dated November 14 – November 27, 2007
BVPS-1,2 Standing Order 07-011/Compensatory Actions to perform Switchyard Walkdowns during 1OST-36.7 and 2OST-36.7
Engineering Analysis of Electrical System Response for As-Found degraded 'A' Phase on TR-1A SSST Primary, dated December 1, 2007
Failure Analysis Report, A Phase – 138kV Lead, dated December 14, 2007
Plant Information (PI) Display Trends for TR-1A current and voltage, dated October 22, 2007 through November 29, 2007
Station Risk Evaluation of Loss of 'A' train off-site power concurrent with 'A' EDG outage, dated December 14, 2007
VTM 8700-01.014-0040, Instruction Book for SL Core Form Power Transformer, dated May 1971

<u>Technical Specifications</u>
ITS 3.8.1

<u>Section 1R15: Operability Evaluations</u>

<u>Surveillance</u>
1OST-30.12A, Rev. 24, "Train A Reactor Plant River Water System Full Flow Test", dated September 20, 2007

<u>Calculation</u>
8700-DMC-3534, Rev. 0 & 1, "Beaver Valley Power Station Unit 1 River Water Model Development and Benchmark"

<u>Condition Reports</u>
07-26849 07-27544 07-28465

<u>Work Orders</u>
200215340 200215728

<u>Other</u>
Letter from ABB, Inc. to FIRSTENERGY, dated September 25, 2007: "Electroswitch Corp. L-2 Auxiliary Switch Assemble – 10 CFR Part 21 Notification"
Test Report No. 1VAF200012D0022, September 12, 2007, Mechanical Life Test Florence Engineering Test Lab L-2 Spacer Bushing Test

<u>Section 1R19: Post-Maintenance Testing</u>

<u>Procedures & Forms</u>
1CMP-6RC-LT-TEMP-1A-3I, "Temporary RCS Level Indication for Refueling – A Loop"
1CMP-6RC-REFL-LVL-1C-3I, "Temporary RCS Level Indication for Refueling – C Loop"
1/2-ADM-2028.F02, Rev. 2, "Jumper/Lifted Lead Tag Index," Unit 1, System 6, Sept – Oct 2007

<u>Drawings</u>
1081H94, Rev. H, "Solid State Protection System", Sheets 18 through 26

Work Orders

600416728	600416259	200208641
600416727	200240002	200208640

Condition Reports
07-29570 07-29487

Other
BVPS-1 & 2 Shift Operating Narrative Logs dated October 31 – November 2, 2007
Human Performance 'NewsFlash', dated November 1, 2007

Section 1R20: Refueling and Outage Activities

Procedures
1BVT-1.1.1, Rev. 4, "Rod Position Indication System Calibration Verification and Control Rod Drop Test"
1BVT 2.1.1, Issue 1, Rev. 0, "Control Rod plant Exercise and Data Collection"
1OM-6.4.AO, Rev. 20, "Isolating and Draining a Reactor Coolant Loop"
1OM-20.4E, Rev. 31, "Draining The Refueling Cavity"
1OM-50.4D, Rev. 49, "Reactor Startup From Mode 3 to Mode 2"
1OM-50.4L, Rev. 18, "Plant Heatup From Mode 6 to Mode 3"
1OM-50.4L, Rev. 18, "Plant Heatup From Mode 6 to Mode 3, Data Sheet 2: RCS Heatup / Cooldown Determination"
1OM-52.4.R.1.F, Rev. 14, "Station Shutdown from 100% Power to Mode 5", Data Sheet 2: RCS Cooldown Determination Tables.
1OST-11.14A, Rev. 19, "LHSI Full Flow Test"
1OST-47.2B, Rev. 6, "Containment Closeout Inspection"
1OST-49.2, Rev. 22, "Shutdown Margin Calculation (Plant Shutdown) (Updated for Cycle 18)"
1MSP-9.04-M, Rev. 8, "Containment Sump Inspection"
1RP-3.2, Issue 0, Rev. 3, "Fuel Transfer System"
1RP-3.26, Rev. 7, "Refueling Procedure Upper Internals Assembly Installation"
1RP-3.28, Rev. 4, "Lower Internals Assembly Removal / Installation"
1RST-2.1, Rev. 11, "Initial Approach to Criticality After Refueling"
1RST-2.2, Rev. 10, "Core Design Check Test"
NOBP-OM-4010, Rev. 4, "Restart Readiness for Plant Outages"
NOBP-WM-5003, Rev. 1, "FENOC Rigging and Lifting Manual"
NOP-OP-1005, Rev. 10, "Shutdown Defense in Depth"
NOP-WM-5003, Rev. 1, "Rigging, Lifting, and Load Handling"

Drawings
8700-02.102-0050, Rev. A, "General Arrangement Transfer System"
Cable Drive Installation, Transfer System – BVPS1, Rev. 1

Work Orders

Repetitive Task 10001	99-0201123-000	200285260	600426477

Condition Reports

07-27682	07-27576	07-27954
07-27637	07-30964	07-27936
07-27783	07-28968	07-27910
07-27680	07-28967	07-27878
07-27750	07-28946	07-27859
07-27573	07-28938	07-27857
07-27577	07-28906	07-27509
07-27599	07-28854	07-27439
07-27606	07-28783	07-27422
07-27659	07-28471	03-10165
07-27644	07-28468	03-10148
07-27591	07-28252	

Other
1R18 Outage Handbook
100-Hour Safety & Human Performance Standdown Notes, September 27 & 28, 2007
8700-02.102-0010, UE&C Instruction Manual Cable Drive Fuel Transfer System
ANSI B30.2-1976, "Overhead and Gantry Cranes"
ANSI B30.20, American National Standard for Lifting Devices
ANSI N14.6-1978, American National Standard for Special Lifting Devices Book III
BV-SA-07-096, "BV1-R18 Safety & Human Performance Snapshot Self-Assessment," Dec '07
BVPS-1 Shift Operating / Refueling Logs dated Oct 12 – 15, 2007
Failure Analysis Report for Failed Cable from Spent Fuel Pool Upender from BVPS-1, 11/19/07
Failure Mode Analysis for 07-28471, dated October 13, 2007
NUREG-0612
Primavera Schedule, 1R18
Westinghouse Field Anomaly Report FAR DL-07-81 / 83
Westinghouse Justification for continued use of Unit 1 Containment Side Fuel Upender, dated October 15, 2007
Westinghouse NF-DL-07-14, Rev. 1, "Beaver Valley Power Station Unit 1 Cycle 19 Redesign Core Loading Plan"

Section 1R22: Surveillance Testing
Condition Reports
07-27784
07-27763
07-27582
07-27493

Technical Specification
ITS 3.0, Surveillance Requirement Applicability
ITS 3.3.3, Post Accident Monitoring Instrumentation

Other
BVPS Unit 2 Operation Logs dated October 3, 2007
BVPS Units 1 and 2 controlled copy of Technical Specifications, Sections 3.0 and 3.

Section 1R23: Temporary Plant Modifications

Condition Reports
07-28510

Regulatory Applicability Determination and 10 CFR 50.59 Screens
07-04766, performed as per Engineering Change Package 07-0315

Other
Engineering Change Package 07-0315
NOP-OP-1009-01, Rev. 00, “Prompt Operability Determination Form” associated with CR 07-28510

Sections 2OS1 Access Control to Radiologically Significant Areas and 2OS2 ALARA Planning and Controls

Procedures

1/2-ADM-1601, Rev 15	Radiation Protection Standards
1/2-ADM-1611, Rev 9	Radiation Protection Administrative Guide
1/2-ADM-1621, Rev 3	ALARA Program
1/2-ADM-1630, Rev 10	Radiation Worker Practices
1/2-ADM-1631, Rev 5	Exposure Control
1/2-HPP-3.02.004, Rev 4	Area Posting
1/2-HPP-3.04.002, Rev 5	Bioassay Administration
1/2-HPP-3.05.001, Rev 4	Exposure Authorization
1/2-HPP-3.07.002, Rev 5	Radiation Survey Methods
1/2-HPP-3.07.013, Rev 3	Barrier Checks
1/2-HPP-3.08.001, Rev 8	Radiological Work Permit
1/2-HPP-3.08.003, Rev 10	Radiation Barrier Key Control
1/2-HPP-3.08.005, Rev 4	ALARA Review Program
1/2-HPP-3.08.006, Rev 1	Shielding
BVBP-RP-0003, Rev 4	Dosimetry Practices
BVBP-RP-0013, Rev 2	Radiation Protection Risk Assessment Process
BVBP-RP-0020, Rev 6	RP Job Coverage General Guidance
NOP-WM-7001, Rev 0	ALARA Program
NOP-WM-7002, Rev 0	Operational ALARA Program
NOP-WM-7003, Rev 0	Radiation Work Permit
NOP-WM-7017, Rev 0	Contamination Control Program
NOP-WM-7021, Rev 1	Radiological Postings, Labeling, and Markings

Quality Assurance Assessments
Quality Assurance Audit Report MS-C-07-08-03, Radiation Protection & Radwaste Processing
Quality Field Observation Reports (Radiation Protection) January 2007 through September 2007

Departmental Self-Assessments
BV-SA-07-078, SOER 01-1, Unplanned Radiation Exposures
BV-SA-07-125, Radiation Protection
BV-SA-07-002, Radiation Protection, Bench Marking Brunswick Station
BV-SA-07-036, TLD Processing Program

BV-SA-07-090, Remote Monitoring Technology
BV-SA-07-067, Radiation Protection

Condition Reports

71121.01 Related:	07-25888, 07-27105, 07-28476, 07-27325, 07-27066, 07-27023, 07-27027, 07-27055, 07-26688, 07-24516, 07-19631, 07-18375, 07-17788, 07-27985, 07-28081, 07-28001, 07-28174, 07-28490, 07-27945, 07-27914, 07-28012, 07-27973, 07-27972, 07-27922, 07-25885, 07-27141, 07-28394, 07-27525, 07-27601, 07-27325 07-27313, 07-27141, 07-28786
71121.02 Related:	07-27243, 07-27110, 07-26705, 07-25584, 07-28148, 07-28234, 07-28486, 07-27942, 07-25986, 07-28154, 07-27619, 07-27491, 07-27191, 07-27391, 07-27349, 07-27301, 07-27167, 07-28786

ALARA Plans & related Work-in-Progress /Post-Job Reviews
Sump Modification ECP 05-0361 (07-01-33)
Pressurizer Weld Overlay (07-01-56)
Pressurizer Weld Overlay Support Activities (07-01-36)
Scaffolding Construction in Unit -1 Reactor Building (07-01-31)
Steam Generator Support (07-01-20)
Reactor Disassembly/Reassembly (07-01-22)
Diving Operations (07-01-59)

ALARA Committee Meeting Minutes
Meeting Nos. 1R18-1/2/3/4/5/6/7 and 1R18-14

ALARA Reports
1R18 Outage ALARA Plan
EPRI Standard Radiation Monitoring Program - Unit 1 Source Term Measurements

Section 4OA2: Identification and Resolution of Problems

Condition Reports

00-02665	01-08369	02-06542	03-10353	04-09432	06-02686
06-02954	06-06472	06-06777	06-06867	06-06985	06-07416
06-07416	06-08633	06-08699	07-13839	07-30306	07-30344
07-30480	07-30720	07-30729	07-30808	07-31563	

Miscellaneous
Beaver Valley Unit 1 Chemical and Volume Control System Health Report (Quarters 2007-3, 2007-2, 2007-1, 2006-4)
Beaver Valley Unit 2 Chemical and Volume Control System Health Report (Quarters 2007-3, 2007-2, 2007-1, 2006-4)
BVBP-OPS-0002,"Operator Work-Arounds, Operator Burdens, and Control Room Deficiencies" Rev 11
Operator Workaround, Burden, and Control Room Deficiency Tracking Report Dated 12/18/07
Beaver Valley Management Alignment and Ownership Meeting Report Dated 12/18/07

Condition Reports

0600430946	0600343638	0600374859	0600374860	0600389589	0600412110
0600422912	0600430946	0600369131	0600348134	0600055702	0600369130
0600377063					

Section 4OA3: Event Response

Miscellaneous

BVPS-1 Shift Operating Logs, dated October 10 – 15, 2007
BVPS-2 Shift Operating Logs, dated November 15 - 16, 2007
LER 05000334/2007-001, "Valve Testing Program Change Inadvertently Leads to Condition Beyond Design Basis During Test." Rev 0
Training Presentation Slides for "10CRF50.46 ECCS Acceptance Criteria" following EPU.
Engineering Change Package 05-0280,"Simultaneous Hot and Cold Leg SI Recirculation (LHSI to Hot Legs) Rev 0
Regulatory Applicability Determination 05-04109," 10 CFR 50.59 Review for ECP-0280" Rev 0
Procedure 1OST-47.3F "Containment Isolation and ASME Section XI Test" Revs 7,9,11, and 12 and associate 10CFR 50.59 Screenings.
Root Cause Analysis Report,"MOV-SI-890B Stroke Alignment Per 1OST-47.3F Not Found in FSAR Accident Analysis" Dated 9/6/2007

Condition Reports

05-04366
07-23462
07-30245
07-30247
07-30251

Section 4OA5: Other Activities

Calculations

8700-DMC-1650, Rev. 1, "Beaver Valley, Power Station Unit 1 - Volume of Insulation and Debris Inside the Containment Building
8700-DMC-1651, Rev. 0, Addendum 1, "Containment Coatings Walkdown"
8700-DMC-1652, Rev. 0, Addendum 1, "Beaver Valley Power Station - Unit 1 HELB Debris Generation Calculation"
8700-DMC-1653, Rev. 0, Addendum 1, "Beaver Valley Station Unit 1 GSI-191 Containment Recirculation Sump Evaluation: Debris Transport Calculation"
8700-DMC-3575, Rev. 0, "Proof of Absence of Vortices Above Reactor Building Emergency Sump Strainers"
8700-US(B)-263, Rev. 3, Addendum 1, "Assessment of Beaver Valley Unit 1 Containment Response for Design Basis Accidents For Containment Atmospheric Conversion Project"
8700-US(B)-265, Rev. 3, "Assessment of Beaver Valley Units 1 and 2 Containment Response for Small and Intermediate Accidents For Containment Atmospheric Conversion Project"
3 SA-096.060, Rev. 1, CCI Calculation - Beaver Valley Unit 1 Reactor Building Emergency Sump Strainers Head Loss Calculation
680/41366, Rev. 1, CCI Calculation - Beaver Valley ECCS Suction Strainer Large Size Head Loss Test Report

Q.003.84.782, Rev. 1, CCI Calculation - Beaver Valley ECCS Suction Strainer Large Size Head Loss Test

Condition Reports

04-07056	07-28965	07-28936	07-28867	07-28533	07-28610
07-28507	07-28263	07-28180	07-28277	07-28102	07-28106
07-28049	07-27681	07-27776	07-27602	07-27684	07-27536
07-26845	07-26745	07-26081	07-25783	07-25782	07-22591
07-21760	07-27079	07-27058	07-26942	07-27099	07-27532
07-21194	07-13276	07-13276	07-22029		

Drawings

8700-06.060-0027, Rev. 0, "Beaver Valley Unit 1 Layout"
8700-06.060-0028, Rev. 0, "Module Standard, Strainer Assembly"
8700-06.060-0029, Rev. 0, "End Module, Strainer Assembly"
8700-06.060-0055, Rev. 1, "Suction Duct
8700-06.060-0113, Rev. 0, "Box - Channel Assembly Drawing"
RM-0001D, Rev. 0, "Mac. Loc. Reactor Cont. Sht. 4"
RM-0001E, Rev. 0, "Mac. Loc. Rector Cont. Sht. 5"
RM-0001G, Rev. 0, "Mac. Loc. Reactor Cont. Sht. 7"
RM-0413-0002, Rev. 0, "Valve Oper. no. Diag, Containment Depressurization Sys."
RM-0513-02, Rev. 0, "Flow Diagram, Containment Depressurization Sys."
RS-0016Y, Rev. 0, "Recirc. Pump Sup's & Screens, Sh.1, Reactor Cont."
RS-0016Z, Rev. 0, "Recirc. Pump Sup's & Screens, Sh.2, Reactor Cont."

Modifications

ECP 05-0361 ECP 06-0227 ECP 06-0247

Procedures

1BVT 1.47.1, Rev. 9, "Containment Structural Integrity Test"

Miscellaneous

BV1 Control Room Logs, dated October 19, 2007
L-03-117, FENOC Letter to USNRC: 60-day Response to Bulletin 2003-01, dated August 8, 2003
L-05-034, FENOC Letter to USNRC: Response to Generic Letter 2004-002, dated March 4, 2005
L-05-146, FENOC Letter to USNRC: Response to Generic Letter 2004-002, dated September 6, 2005
L-06-020, FENOC Letter to USNRC: Response to Generic Letter 2004-002, dated April 3, 2006
L-06-145, FENOC Letter to USNRC: Response to Generic Letter 2004-002, dated September 29, 2006
L-06-171, FENOC Letter to USNRC: Response to Generic Letter 2004-002, dated December 21, 2006
L-07-017, FENOC Letter to USNRC: License Amendment Request Nos. 334 and 205, dated February 9, 2007
L-07-095, FENOC Letter to USNRC: "Response to a Request for Additional Information (RAI) dated July 3, 2007 in Support of License Amendment Request Nos. 334 and 205," dated August 8, 2007
L-07-105, FENOC Letter to USNRC: "Supplemental Information for License Amendment Request Nos. 334 and 205," dated August 23, 2007
NOTF 600370347

UFSAR, Rev. 23, Section 6.1
UFSAR, Rev. 23, Section 6.3
UFSAR, Rev. 23, Section 6.4
UFSAR, Rev. 23, Section 7.1
UFSAR, Rev. 23, Section 7.3
USNRC Generic Letter 2004-002: “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors”
USNRC Letter to FENOC: Request for Additional Information with regard to Generic Letter 2004-002 Responses, dated February 9, 2006
USNRC Letter to FENOC: “Beaver Valley Power Station, Unit No. 1 - Issuance of Amendment Re: Changed to the Recirculation Spray System Pump Start Signal due to the Containment Sump Screen Modification,” dated October 5, 2007
USNRC Letter to Holders of Licenses for PWRs: “Alternate Approach for Responding to the Nuclear Regulatory Commission Request for Additional Information Letter Re: Generic Letter 2004-002,” dated March 28, 2006

Observation of Training Required by Confirmatory Order EA-07-199

BV-L-07-128, Actions Required By Confirmatory Order EA-07-199, dated 09/20/2007
Confirmatory Order EA-07-199, dated August 15, 2007
Insurance Claim/Regulatory Timeline Handout, dated 11/13/2007
NOP-TR-1004-01, Rev. 00, “FENOC Attendance Sheet”; Regulatory Sensitivity Training at BVPS on 11/13/2007
Regulatory Sensitivity Training Book, SAP Business Event 62191207
Regulatory Sensitivity Power Point Handout, dated 11/13/2007

LIST OF ACRONYMS

ADM	Administrative Procedure
ALARA	As Low As is Reasonably Achievable
AP	ALARA Plan
ASME	American Society of Mechanical Engineers
BCO	Basis for Continued Operations
BVPS	Beaver Valley Power Station
CA	Corrective Action
CAP	Corrective Action Program
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CR	Condition Report(s)
DIE	Design Interface Evaluations
DPW	Declared Pregnant Worker
ECP	Engineering Change Package
ECT	Eddy Current Test
EDG	Emergency Diesel Generator
FENOC	First Energy Nuclear Operating Company
GL	Generic Letter
GSI	Generic Safety Issue
HRA	High Radiation Area
IMC	Inspection Manual Chapter

IP	Inspection Procedure
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss of Coolant Accident
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LHRA	Locked High Radiation Area
MSP	Maintenance Surveillance Package
MT	Magnetic Particle Testing
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulation
OD	Operability Determinations
OST	Operations Surveillance Test
OWA	Operator Work Around
PCE	Personnel Contamination Event Report
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Testing
PT	Liquid Penetrant
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RW	River Water
RWP	Radiation Work Permit
SDP	Significance Determination Process
TI	Temporary Instruction
TMOD	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VHRA	Very High Radiation Area