UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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August 13, 2008

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

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2008003 AND 05000412/2008003

Dear Mr. Sena:

On June 30, 2008, 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 July 16, 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 four (4) self-revealing findings of very low safety significance (Green). Three of these findings were determined to involve a violation of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low significance is listed in this report. However, because of the very low safety significance and because the issues have been entered into the 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 05000334/2008003; 05000412/2008003
w/Attachment A: Supplemental Information
w/Attachment B: TI 2515/172 Reactor Coolant System Dissimilar Metal Butt Welds Documentation Questions for Beaver Valley 1 & 2

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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/
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-334, 50-412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2008003 and 05000412/2008003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	April 1, 2008 through June 30, 2008
Inspectors:	D. Werkheiser, Senior Resident Inspector D. Spindler, Resident Inspector P. Kaufman, Senior Reactor Inspector D. Kern, Senior Resident Inspector T. Moslak, Health Physicist A. Ziedonis, Reactor Inspector
Approved by:	R. Bellamy, Ph.D., Chief Reactor Projects Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2008003, IR 05000412/2008003; 04/01/2008 – 06/30/2008; Beaver Valley Power Station, Units 1 & 2; Operability Evaluations, Refueling and 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. Four Green findings, three of which were non-cited violations (NCVs), 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 self-revealing finding was identified for failure to properly coordinate clearance activities associated with testing for penetration 2X-46 during reduced reactor coolant system (RCS) level. A decision to post a clearance to support penetration testing resulted in the isolation of the make-up flow charging path to the reactor coolant system, resulting in an unexpected reduction of reactor coolant vessel level that was identified and stabilized within the established band. The licensee's immediate corrective actions were to stop work, perform system configuration verification, and re-evaluated in-progress and planned activities for plant safety impact. Long-term corrective actions include a change in procedures to not allow this type of penetration test in this plant configuration.

The finding is more than minor because it affects the configuration control attribute of the Initiating Events cornerstone and affects the shutdown equipment lineup needed for stable reactor vessel level control during reduced RCS level operations, a high risk evolution. The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, Checklist 3, Pressurized Water Reactor Cold Shutdown and Refueling Operation with RCS Open and Refueling Cavity Level < 23'. The inspectors reviewed station drawings and records of reactor vessel level indication during the event. The inspectors determined that although make-up flow was momentarily isolated, reactor vessel level was maintained, sufficient indication existed, and no actual loss of RCS inventory occurred. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance).

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC did not appropriately coordinate work activities for the existing plant conditions to ensure the operational impact on reactor vessel level while at a reduced water level was fully understood [H.3(b)]. (Section 1R20)

Green. A Green self-revealing NCV of TS 5.4.1.(a) was identified in that the licensee failed to take appropriate action to trip the main turbine as specified in 2OM-52.4.A, "Raising Power from 5% to Full Load Operation," Rev. 13 during an unexpected main turbine load increase that resulted in average reactor coolant temperature below the operational limit of 541F. The licensee has developed and implemented an operations department rapid improvement plan.

This finding was more than minor because it can be reasonably viewed as a precursor to a significant event. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter (IMC) 0609, Attachment 609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low risk significance.

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC failed to properly communicate critical parameters and limitations for personnel to perform work safely in a timely manner [H.1.(c)]. (Section 4OA3.3)

Green. A Green self-revealing NCV of TS 5.4.1.(a) was identified in that the licensee failed to properly enter and implement the appropriate abnormal operating procedure (AOP) for loss of main feedwater. The licensee has developed and implemented an operations department rapid improvement plan.

This finding was more than minor because it can be reasonably viewed as a precursor to a significant event. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter (IMC) 0609, Attachment 609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low risk significance.

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC failed to properly implement appropriate roles and authority for decision making during risk-significant decisions. [H.1.(a)]. (Section 4OA3.4)

Cornerstone: Mitigating Systems

Green. A self-revealing NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified in that the licensee failed to incorporate sufficient assembly detail into the maintenance procedure for the governor linkage on the Turbine-Driven Auxiliary Feedwater (TDAFW) pump. The required gaps and tightening criteria for the reassembly of the governor valve linkage were not included in the overhaul procedure resulting in jam nuts loosening, allowing the valve stem to rotate. Rotation of the valve stem caused an uncontrolled change in position of the governor valve position. This resulted in an unanticipated speed increase of the TDAFW during the performance of surveillance test 1OST-24.4 "Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2]." Corrective actions included a change to the maintenance procedure and the installation of spacer shims for the anti-rotation block.

This finding was 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

(IMC) 0609, Attachment 609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low risk significance.

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC did not maintain a complete, accurate, and up-to-date governor overhaul procedure in regards to actuator reassembly which resulted in speed control degradation to the TDAFW [H.2.(c)]. (Section 1R15)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at 100 percent reactor power throughout the inspection period.

Unit 2 began the inspection period at 100 percent power. On April 12, the unit was down-powered to 60 percent for planned testing prior to refuel outage (2R13). On April 14, the unit was shutdown to commence a refuel outage (2R13). On May 15, operators performed a reactor startup and achieved 15 percent power; operators shut down the unit due to a failed main turbine bearing on May 16. On May 22, the unit was restarted and maintained low power for turbine control system repairs (May 23 - 24). From May 25 - 29, the unit ascended to 95 percent power. The unit implemented the final phase 5 percent power uprate on June 3 and reached rated thermal power (2900MWt) on June 5. The unit remained at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Seasonal Susceptibility

a. Inspection Scope (1 offsite power grid reliability sample)

The inspectors evaluated Beaver Valley Power Station (BVPS) design features and FENOC's implementation of procedures to handle issues that could impact offsite and alternating current (AC) power systems. The inspectors reviewed FENOC's procedures and programs which discussed the operation and availability/reliability of offsite and alternate AC power systems during adverse weather. The inspectors verified that communication protocols between the transmission system operator and FENOC existed, and the appropriate information would be conveyed when potential grid stress and disturbances existed. The inspectors also verified that FENOC's procedures contained actions to monitor and maintain the availability/reliability of offsite and onsite power systems prior to and during adverse weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns (71111.04Q)

a. Inspection Scope (4 samples)

The inspectors performed four partial equipment alignment inspections during conditions of increased safety significance, including when redundant equipment was unavailable during maintenance or adverse conditions. The partial alignment inspections were also completed after equipment was recently returned to service after significant maintenance. The inspectors performed partial walkdowns of the following systems, including associated electrical distribution components and control room panels, to verify the equipment was aligned to perform its intended safety functions:

- On April 9, Unit 1 Auxiliary Feedwater (AFW) during an unplanned Limiting Condition of Operation (LCO) on the Turbine-Driven AFW Pump;
- On April 16, Unit 2 'A' train Service Water (SW) during 'B' SW maintenance;
- On April 27, Unit 2 Spent Fuel Pool (SFP) cooling during 'DF' electrical bus outage; and
- On June 9, Unit 1 'A' train River Water (RW) using the 'C' RW pump during 'B' train RW maintenance.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope (1 sample)

The inspectors completed a detailed review of the alignment and condition of the Unit 1 Refuel Water Storage Tank System (RWST). The inspectors conducted a walkdown of the system to verify that the critical portions, such as valve positions, switches, and breakers, were correctly aligned in accordance with procedures, and to identify any discrepancies that may have had an effect on operability.

The inspectors also reviewed outstanding maintenance work orders to verify that the deficiencies did not significantly affect the RWST system function. In addition, the inspectors discussed system health with the system engineer and reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

Quarterly Sample Review

a. Inspection Scope (6 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, Rev. 16, "Fire Protection." 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:

- On April 8, Unit 2 Cable Vault and Rod Control Area (Fire Area CV-3);
- On April 8, Unit 2 Valve Pit (West) (Fire Area VP-2);
- On April 14, Unit 2 Reactor Containment (Fire Area RC-1);
- On May 28, Unit 1 Primary Auxiliary Building Elevation 722 (Fire Area PA-1G);
- On May 28, Unit 2 Cable Spread Room (Fire Area CB-2); and
- On June 30, Unit 1 CO2 Storage / PG Pump Room (Fire Area CO-2).

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08)

a. Inspection Scope (1 sample)

From April 21-May 15, 2008, the inspectors conducted a review of FENOC's implementation of risk-informed in-service inspection (ISI) program activities for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries for Beaver Valley Unit 2 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. The inspector also conducted a review of TI 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds for Beaver Valley Unit 1 and Unit 2, program implementation activities. The inspector reviewed documentation, observed in-process non-destructive examinations (NDE) and interviewed inspection personnel to verify that the activities were performed in accordance with the ASME Boiler and Pressure Vessel Code Section XI requirements.

Non-Destructive Examination (NDE) Activities and Welding Activities

Dye Penetrant (PT), Ultrasonic Testing (UT), Eddy Current Testing (ECT), Visual Testing (VT), and Radiographic Testing (RT) Activities Reviewed:

Reactor pressure vessel (RPV) lower head (VT-2) bare metal visual inspection (BMI) video (sampled some of the 50 penetrations that were examined), reactor vessel upper head visual inspection video (VT-2), automated UT examination of reactor vessel nozzle-to-shell and nozzle-to-safe end welds and automated UT examination of reactor pressure vessel head penetration control rod drive mechanism (CRDM) nozzles, eddy current testing (ECT) examinations of three steam generators, PT examinations performed of the Unit 2 reactor vessel head penetration weld overlays on penetrations #16, #56, and #61 that were installed during 2R12, and RT films and data sheets of Unit 2 charging system valve 2CHS-FCV122, welds 2CHS-288-F03-A and 2CHS-120-F01-A.

Repair/Replacement Consisting of Welding:

Repair/replacement activity associated with Unit 2 charging system valve 2CHS-FCV122, class 2 valve, was being replaced with engineering change package ECP 07-0357-01 and the weld overlay to Unit 2 reactor with vessel head penetration #51 J-groove weld were reviewed by the inspector to ensure that welding and applicable NDE activities were performed in accordance with ASME Code requirements.

Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

The inspector directly observed a sample of in-process Unit 2 reactor pressure vessel head and vessel head penetration control rod drive mechanism (CRDM) nozzle ultrasonic testing (UT), supplementary eddy current testing (ECT) examinations and repair activities during the 2R13 refueling outage. A circumferential indication approximately 0.280" long and 0.146" in depth was identified on the outside diameter (OD) of CRDM penetration tube #51 in reactor pressure vessel (RPV) head at the toe of the J-groove weld. The indication was mitigated by a weld overlay repair of the J-groove weld area to prevent what was evaluated to be caused by Primary Water Stress Corrosion Cracking (PWSCC). Post-repair dye penetrant (PT) examinations of the repaired region identified no indications (PT-white).

The inspector reviewed the certifications of the NDE technicians performing the weld overlay examinations, as well as certifications of the welders performing the weld overlay on the upper reactor head penetration #51 J-groove weld repair. The inspector also reviewed a sample of the remote bare metal visual (VT-2) examination and video tapes of the RPV head surface and 360 degrees around each of the 65 CRDM penetrations and verified that no boric acid leakage had been observed on the upper reactor head surface, specifically around penetration #51.

Reactor Pressure Vessel Lower Head Penetration Nozzle Inspection Activities

The inspector verified the VT-2 inspection results of the bare metal visual inspection (BMI) of the Unit 2 reactor pressure vessel lower head penetration nozzles that was conducted by FENOC personnel during 2R13 by reviewing portions of the video tapes and visual inspection documentation record results (BOP-VT-08-015) of the BMI inspection. No boric acid leakage was noted around the annulus area on any of the 50 penetrations.

Boric Acid Corrosion Control (BACC) Inspection Activities

The inspector discussed the boric acid control program controlled by Beaver Valley station procedure NOP-ER-2001, Boric Acid Corrosion Control Program, Rev. 7, with the boric acid corrosion control program owner and sampled photographic inspections of boric acid found on safety significant piping and components inside Unit 2 Containment during Mode 3 walkdowns. Direct observations by the resident inspectors verified that the visual inspections were performed in accordance with the procedure and checklists which emphasized the areas and locations where boric acid leaks could cause degradation of safety significant components and that deficient conditions were identified and documented.

A sample of engineering evaluations/corrective actions associated with these boric acid deficiencies and a sample of these items on the Unit 2 mode hold list were reviewed by the inspector. The inspector confirmed that condition reports were assigned corrective actions consistent with the requirements of the ASME Code and 10 CFR 50 Appendix B Criterion XVI. The inspector specifically reviewed the following evaluations:

CR 08-38367/2IIS-344 Incore Thimble Blockage at Location H-03; CR 08-39097/2RCS-44 Loop ‘A’ Bypass Flow Isolation Valve, and CR 08-38234/2RHS-E21B ‘B’ Residual Heat Removal Heat (RHR) Exchanger. The inspector noted that 2RCS-44 valve (CR-06-8271) and 2RHS-E21B ‘B’ RHR heat exchanger (CR-06-7743) had boric acid leakage identified during the previous 2R12 outage and are identified for re-inspection as part of the Unit 2 boric acid monitoring program each outage. FENOC’s corrective actions to these types of long standing boric acid leakage conditions were not extensive or aggressive enough to prevent further degradation of 2RCS-44, a Kerotest valve; consequently a temporary modification had to be installed during 2R13 to prevent the ongoing boric acid leakage and further valve stem thread degradation.

Steam Generator (SG) Tube Inspection Activities

The inspector reviewed the Beaver Valley Unit 2 steam generator Eddy Current Testing (ECT) tube examinations, and applicable procedures for monitoring degradation of SG tubes to verify that the steam generator examination activities were performed in accordance with the rules and regulations of the SG examination program, Beaver Valley Unit 2 steam generator examination guidelines, NRC Generic Letters, 10 CFR 50, Technical Specifications for Beaver Valley Unit 2, Nuclear Energy Institute (NEI) 97-06, EPRI PWR steam generator examination guidelines, and the ASME Boiler and Pressure Vessel Code Sections V and XI. The review also included the Beaver Valley Unit 2 steam generator degradation assessment (SG-CGME-08-14) and steam generator Cycle 13 operational assessment for 2R13 refueling outage.

Eddy current testing of all tubes in the three SG was conducted during the 2R13 outage. The inspector reviewed plant specific SG information, tube inspection criteria, integrity assessments, degradation modes, and tube plugging criteria. The inspector discussed the in-process ECT inspection activities with the data management and data acquisition personnel and resolution analysts and observed a sample of the tubes being examined from each of the three SGs. Examination data records for selected tubes from each of the generators and the characterization and disposition of the identified flaws were reviewed by the inspector to verify the SG inspection program was implemented in accordance with the SG examination program.

A circumferential outside diameter stress corrosion cracking (ODSCC) flaw was identified at a free-span ding in tube (R23 C58 at 06H +30”) in the “C” steam generator. Even though the indication did not meet the EPRI requirements for in-situ pressure testing, in-situ pressure testing of this particular tube was conducted and it passed the pressure leakage test. The ding is believed to have been caused during original tube installation. FENOC considers this a new degradation mechanism for the Beaver Valley Unit 2 steam generators, even though it was listed as a potential degradation mechanism in the 2R13 degradation assessment report.

FENOC participated in an outage conference call with NRR on April 25 to discuss Unit 2 steam generator examination results obtained and the status of eddy current inspections up to that time. FENOC expanded the SG inspections to include additional sampling in the Plus Point special interest area during 2R13. Results from the expanded Plus Point sampling inspections of free-span ding pairs (523 locations) did not reveal any additional indications. The inspector confirmed the SG eddy current inspections, testing, and documentation activities were conducted in accordance with Beaver Valley Unit 2 steam generator examination guidelines, station and vendor procedures, and EPRI guidelines. A total of 84 SG tubes required plugging.

Problem Identification and Resolution

The inspector reviewed a sample of CRs related to ISI and Materials Reliability Program MRP-139 program activities to assess FENOC's effectiveness in problem identification and resolution and determined that deficiencies are being appropriately identified and adequately entered into and resolved by the corrective action program.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

Resident Inspector Quarterly Review

a. Inspection Scope (2 samples)

The inspectors observed two samples of Unit 2 licensed operator just-in-time (JIT) training in preparation for reactor and plant startup on May 6 and May 23. 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 physical fidelity to assure the simulator appropriately modeled the plant control room. 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.

1R12 Maintenance Rule Implementation (71111.12Q)

Routine Maintenance Effectiveness Inspection

a. Inspection Scope (3 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 08-38049, “The Maintenance Rule (a)(1) criteria was exceeded when the Unit 1 “A” SWGR chiller tripped and not able to restart”;

- CR 08-41178, “The Unit 1 1AE Bus Shutdown Panel control switch for 1CH-P-1C functional failure”; and

- CR 07-21571, “Fire Detection Not Restored within Time Requirement from ADM-1900.”

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (7 samples)

The inspectors reviewed the scheduling and control of seven 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:

- Week of March 31, Unit 1 risk assessment with Emergency Diesel Generator (EDG) and AFW testing planned;
- On April 8 and 9, Unit 1 Maintenance Risk Assessment with Turbine-Driven AFW out of service;
- On April 13, Unit 2 Yellow plant safety risk due to EDG engine-barring during ‘A’ train of Recirculation Spray system out of service;
- On May 6, Unit 2 shutdown risk during reactor coolant drain down to install vessel head;
- On May 21, Unit 2 Maintenance Risk Assessment with EDG 2-2 out of service due to fuel oil tubing repairs;
- On May 30, Unit 1 Maintenance Risk Assessment due to ‘A’ offsite transformer relay calibration coincident with safety-related river water maintenance preparation;
- On June 12, Unit 1 Maintenance Risk Assessment due to activities related to switchyard maintenance with ‘B’ train off-site power unavailable.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (7 samples)

The inspectors evaluated the technical adequacy of selected immediate operability determinations (IOD), prompt operability determinations (POD), or functionality assessments (FA), to verify that determinations of operability were justified, as appropriate. In addition, the inspectors verified that TS LCO requirements and Updated

Final Safety Analysis Report design basis requirements were properly addressed. Documents reviewed are listed in the Attachment. This inspection activity represents seven samples of the following issues:

- On April 4, issues regarding 'A' Recirculation Spray System (RSS) heat exchanger (HX) operability due to 2SWS-1064 drain valve stuck as documented in CR 08-37757;
- On April 8, Unit 1 turbine-driven auxiliary feedwater pump failing its surveillance test due to high turbine speed as documented in CR 08-37921;
- On April 12, issues regarding procedural steps for Unit 1 Quench Spray valves MOV-1QS-100A and MOV-1QS-101A and their impact on RWST operability, as documented in CR 08-37489;
- On April 16, evaluated licensee's extent of condition assessment regarding motor-operated valve key failure of Unit 2 2SWE-MOV116B as documented in CRs 08-39942, 08-38648, and 08-38265;
- On April 25, the inspectors evaluated the licensee's assessment of operability for issues regarding the Unit 2 source range nuclear instrument (N32) detector characteristic curve as documented in CR 08-38416;
- On May 9, the POD developed to address reduced "B" Service Water (SW) flow to the RSS and EDG heat exchanger identified during the full-flow test, as documented in CRs 08-37272 and 08-38017; and
- On May 12, issues identified regarding 'B' train Quench Spray (QS) flow indicator (FI-1QS-103) readings at no-flow as documented in CR 08-40190.

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 incorporate sufficient detail to properly assemble and tighten the governor linkage for the Turbine Driven Auxiliary Feedwater (TDAFW) pump. This resulted in an unexpected turbine speed change because the governor valve stem movement rotated due to loosened jam nuts.

Description. On April 8, during the planned performance of surveillance test 1OST-24.4 "Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2]," it was noted that the speed of the turbine was 4350 RPM, which is outside the acceptance criteria range of 4212 to 4252 RPM. Upon subsequent investigation, it was determined that the jam nut between the linkage arm and the valve disk had loosened, which allowed the control valve disk to open more and allow more steam to be admitted to the turbine.

The Auxiliary Feedwater system is safety-related and consists of two electric motor driven centrifugal pumps and one steam turbine driven centrifugal pump (i.e., TDAFW). The system is designed to deliver water to all three steam generators from the Primary Plant Demineralized Water Storage Tank in the event of feedwater system isolation. The turbine for the TDAFW utilizes a hydraulic actuator and linkage to the control valve for the turbine. The control valve stem exits the valve body horizontally and is threaded along the stem to allow for adjustment. There is a jam nut with a setscrew on each side of the actuator linkage to hold the control valve stem in the correct position with the governor linkage arm.

In October 2007, the turbine governor valve was overhauled and re-assembled with its actuator and linkage per maintenance procedure 1/2-CMP-M-24-001, “Auxiliary Feed Pump Turbine Governor Valve Overhaul.” However, the procedure was insufficient in that it did not detail the reassembly with required shims and gaps to attain and maintain a tight fit for the actuator linkage and components that held the valve stem in place. The failure to correctly assemble the actuator allowed the locking nuts to loosen and the stem to rotate. This changed the position of the governor valve and caused the speed increase. The failure to incorporate sufficient detail into the maintenance procedure for proper assembly is considered a performance deficiency.

Corrective actions included consultation with the vendor and a revision to the maintenance procedure. The linkage component dimensions were verified and documented. This information identified the need to install spacer shims for the anti-rotation block to clear jam nut interference with the actuator lever. Also, as recommended by the vendor, the valve stem threads were flattened and dimpled to support proper engagement of lever nut setscrews. No deficiencies were identified during an extent of condition review for the Unit 2 TDAFW terry turbine linkage.

Analysis. This finding was 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. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with inspection manual chapter (IMC) 0609, Attachment 609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the finding was determined to be of very low risk 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 did not maintain a complete, accurate, and up-to-date governor overhaul procedure in regards to actuator reassembly, which resulted in speed control degradation to the TDAFW [H.2.(c)].

Enforcement, 10CFR 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 April 2008, FENOC failed to prescribe the required assembly directions to ensure the governor linkage would remain sufficiently tightened on the TDAFW pump turbine. This resulted in a degraded speed control for the TDAFW pump turbine. Because this deficiency is considered to be of very low significance (Green), and was entered into the corrective action program (CR-08-37921) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000334/2008003-01, Inadequate Maintenance Procedure Results in Unexpected Terry Turbine Speed Increase)

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope (1 sample)

The inspectors reviewed the following temporary modification (TMOD) based on risk significance. The TMOD and associated 10 CFR 50.59 screening were reviewed against the system design basis documentation, including the UFSAR and the TS. The inspectors verified the TMODs were implemented in accordance with Administrative (ADM) Procedure, 1/2-ADM-2028, "Temporary Modifications," Rev. 6. Documents reviewed are listed in the Attachment.

- ECP-08-0237, Temporary Modification to Seal Weld Bonnet Cap for BV-2RCS-44, Rev. 4. The TMOD specified a seal weld between the cap and body of the valve and a temporary leak repair clamp to the yoke to encapsulate any leak.

.2 Permanent Plant Modifications

a. Inspection Scope (2 samples)

The inspectors evaluated the design basis impact of the modifications listed below. The inspectors reviewed the adequacy of the associated 10 CFR 50.59 screening, verified that attributes and parameters within the design documentation were consistent with required licensing and design bases, as well as credited codes and standards, and walked down the systems to verify that changes described in the package were appropriately implemented. The inspectors also verified the post-modification testing was satisfactorily accomplished to ensure the system and components operated consistent with their intended safety function. Documents reviewed are listed in the Attachment.

- ECP-07-0327, Replacement BV2 EDG K1 Relays, Rev. 0. Replacement of the Unit 2 Emergency Diesel Generator voltage regulator K1 relays with functionally equivalent devices to improve the reliability of the relay; and

- ECP-05-0343, 2FWE-P22 Turbine Driven Auxiliary Feedwater Pump Mechanical Seal Upgrade, Rev. 1. Pump design was revised to install mechanical seals in place of the original packing gland seals to improve reliability and reduce historical pump unavailability due to maintenance of the packing gland seals.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (9 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 April 9 and April 19, cleaning of the Unit 2 ‘B’ train service water (SW) flow-side of the ‘D’ Recirculation Spray System heat exchanger and subsequent full-flow test;
- On April 9, installation and alignment of the Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) Pump turbine governor linkage per ½-CMP-M-24-001 and subsequent testing per 1OST-24.4;
- On April 24 and 25, installation of AC shutdown contactor relay 2AK1 on EDG 2-2;
- On April 28, repair and retest of frayed wiring for pressurizer power operated relief valve operator, 2RCS-PCV455D, per work order (WO) 200295817;
- On May 2, Unit 2 EDG 2-2 maintenance window closeout during refueling outage 2R13;
- On May 4, Unit 2 operators performed ‘A’ and ‘C’ RSS pump automatic start circuit testing per 2OST-13.3A and 2OST-13.5A after implementing coincident logic additions (WO 200252480);
- On May 12, retest of Unit 2 Pilot Operated Relief Valves (PORV) by operators per 2OST-6.8 after outage maintenance activities;
- On May 14, 2008, operators performed 2OST-24.4A, Steam Driven Auxiliary Feed Pump [2FWE*P22] Full Flow Test, Rev. LUC 08-01078 (18) following maintenance including installation of a replacement pump impeller, modification of the pump seals, and overhaul of the turbine governor valve (2FWE-TGV22); and
- On May 21, installation and retest of wire sheathing in Unit 2 EDG 2-2 generator (WO200323467, ECP 08-0305).

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

Unit 2 Refueling Outage (2R13)

a. Inspection Scope (1 sample)

The inspectors observed selected Unit 2 outage activities conducted from April 14–May 25 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 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 identified as required by the licensee’s corrective action program. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment:

- Coordination of electrical bus work, emergency diesel generator auto-load tests;
- Service water system piping inspections and full-flow tests;
- Monitoring of decay heat removal processes;
- Installation of Containment Sodium Tetra-borate baskets;
- Containment and Containment Sump Walkdowns;
- Refueling activities, including split-pin replacement, fuel handling and inspection;
- Observed licensee inspection of all fuel-handling cables and devices;
- Reactor coolant system draindown and vessel head lift;
- Reactor vessel lower internals lift and dose monitoring;
- Control rod drive split-pin replacements;
- Drain down of reactor coolant;
- 2C14 core map / fuel assembly verification;
- Final containment walkdown;
- Reactor start-up and low power physics testing;
- Control rod drop measurement and testing;
- Reactor and plant start-up and heat-up, (May 15-16);
- Plant and Reactor shutdown and repair of high pressure turbine bearing;
- Reactor and Plant start-up, (May 22-23); and
- Balance-of-plant walkdown during power ascension, (May 24-27).

The inspectors observed containment structure test activities on May 9-11, and reviewed the completed test report for the reactor containment building integrated Type A leakage test (2BVT-1.47.2). The inspectors verified that test data documented an acceptable "as left" leakage rate of 0.0587 percent weight per day.

The inspectors also observed selected management review activities associated with restart readiness of Unit 2, following completion of the 2R13 refueling activities. The restart readiness review meeting was accomplished as required by NOBP-OM-4010, "Restart Readiness for Plant Outages," Rev. 4 on May 15 and May 21. The purpose of the review, in part, was to assure to station management that the plant's material condition, programs/processes, and staff members are ready for startup, and then safe, reliable operation after completion of outage activities.

b. Findings

Deficient Control of Clearance Posting Interrupts Reactor Coolant Charging Path while Vessel Water Level Drained below the Flange

Introduction. A self-revealing Green finding was identified for failure to properly coordinate maintenance and operational activities associated with type 'C' testing for penetration 2X-46 while reactor coolant system level was drained to the vessel flange. A decision to post a clearance to support penetration testing resulted in the isolation of the make-up flow charging path to the reactor coolant system, resulting in an unexpected reduction of reactor coolant vessel that was identified and stabilized within the established band.

Description. On May 6, 2008, Unit 2 was in Mode 6, with the reactor coolant system (RCS) level maintained approximately six inches below the Reactor Vessel Flange for reactor head installation. The unit was in a Yellow shutdown risk due to the refueling cavity drained and RCS loops isolated. RCS temperature was 80 degrees Fahrenheit

(F), RCS pressure was atmospheric, and the calculated time to boil was approximately 80 minutes. No charging pumps where in operation. RCS level was maintained by level and pressure in the volume control tank (VCT). Flow from the RCS is returned to the VCT using a Residual Heat Removal (RHR) pump (VCT Float). The amount of water returned to the VCT is controlled by a flow control valve. Another control valve is utilized to control the amount of water added to the RCS. The fill header to the RCS taps into the charging header upstream of valve 2CHS-28, which must remain open to provide flow through the charging header.

A valve integrity test was to be performed to satisfy in-service testing for penetration 2X-46 on May 2, and is normally scheduled after the High-Head Safety Injection (HHSI) Full Flow Test, when the refueling cavity is filled. The HHSI test was delayed due to needed repairs associated with several throttles valves for this system, which delayed the penetration test. The penetration test was not rescheduled through Outage Central. Once the HHSI test was completed on May 4, reactor vessel level was lowered to the flange. The penetration testing crew attempted to obtain the clearance which would allow testing penetration 2X-46. This request was rejected by the Primary Work Window Manager since the clearance would isolate make-up to the RCS. However, over the next two days, due to activities to support fill and venting of the fill header (a prerequisite for the penetration test), it was incorrectly determined that the clearance could be posted without affecting the VCT float path. This decision was not challenged by the clearance personnel or control room operators. On May 6, the first valve to be closed was 2CHS-28, which isolated make-up flow to the RCS. The flow was interrupted which caused the RCS level to drop approximately 2 inches (91.5 to 88.9 inches) before control room operators isolated RHR letdown to stabilize level. The operators maintained the established band of 94 – 87 inches. After determining the direct cause of the problem, the clearance was removed and VCT float reestablished.

The licensee immediately secured activities to investigate the causes leading to this event. The inspectors reviewed station procedures and evaluated plant conditions. The licensee performed system configuration verification and re-evaluated in-progress and planned activities for plant safety impact. Long-term corrective actions include a change in procedures to not allow this type of penetration test in this plant configuration. The inspectors determined that maintenance activities and reduced RCS level operations were not properly coordinated to ensure reactor vessel level remained protected and that changes were understood by the operating crew. Issues related to this event are documented in condition reports 08-39835, 08-39875, and 08-39892.

The inspectors determined that station personnel's failure to properly coordinate maintenance and operations activities while in a reduced RCS level was a performance deficiency. Operations, Outage, and Clearance Desk personnel authorized maintenance personnel to post a clearance affecting RCS make-up without properly identifying the impact on a critical operational parameter (RCS level). Consequently, reactor vessel level unexpectedly lowered while the plant was in an elevated shutdown risk condition.

Analysis. This issue affected the configuration control attribute of the Initiating Events cornerstone and was more than minor because this configuration control error affected the shutdown equipment lineup needed for stable reactor vessel level control during reduced RCS level operations, a high risk evolution. The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, Checklist 3, "Pressurized Water Reactor Cold Shutdown and Refueling Operation with RCS Open and Refueling Cavity Level < 23." The inspectors reviewed station drawings and records of reactor vessel level indication during the event. The inspectors determined that although make-up flow was momentarily isolated, reactor vessel level was maintained, sufficient indication existed, and no actual loss of RCS inventory occurred. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance).

The cause of this finding is related to the cross-cutting area of human performance, in that FENOC did not appropriately coordinate work activities for the existing plant conditions to ensure the operational impact on reactor vessel level while at a reduced water level understood [H.3(b)].

Enforcement. This issue does not constitute a violation of NRC requirements. The finding was of very low safety significance (Green) and FENOC documented this issue in corrective action program condition report 08-39835, 08-39875, and 08-39892. (**FIN 05000412/2008003-02, Deficient Control of Clearance Posting Interrupts Reactor Coolant Charging Path while Vessel Water Level Drained below the Flange)**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (6 samples: 1 isolation valve, 1 leak rate, and 4 routine.)

The inspectors observed pre-job test briefings, observed selected test evolutions, and reviewed the following completed Operation Surveillance Test (OST) and Maintenance Surveillance (MSP) packages. 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 6 activities were reviewed:

- On April 3, 2OST-30.13A, Rev. 25, "Train A Service Water Full Flow Test;"
- On April 15, 2OST-36.4, Rev. 26, "Emergency Diesel Generator [2EGS* EG 2-2] Automatic Test;"
- On May 4, 2OST-11.14B, Rev. 25, "'B' Train High Head Safety Injection Flow Test;"
- On May 11, 2BVT-1.47.2, Rev. 2, "Containment Type A Leak test;"
- On May 12, 2OST-06.08, Rev. 15 "Pressurizer PORV Stroke Test;" and
- On June 30, 2OST-06.02A,, Rev. 26, "72 Hour Reactor Coolant System Water Inventory Balance."

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed a Unit 2 licensed-operator annual simulator evaluation conducted on April 3, 2008. Senior licensed-operator performance regarding event classifications and notifications were specifically evaluated. The inspectors evaluated the 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 Alert. The licensee planned to credit 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. 3, "NRC EPP Performance Indicator Instructions;"
- 1/2-ADM-1111.F01, Rev. 2, "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. 31, "Unusual Event;"
- 1/2-EPP-I-3, Rev. 29, "Alert;"
- 1/2-EPP-I-4, Rev. 29, "Site Area Emergency;" and
- 1/2-EPP-I-5, Rev. 30, "General Emergency."

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (11 samples)

During the period April 21-24, 2008, the inspector 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 radiological controlled areas during the Unit 2 refueling outage (2R13). Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant TSs, and the licensee's procedures. This inspection activity represents completion of 11 samples relative to this inspection area.

Plant Walkdown and RWP Reviews

- During the Unit 2 refueling outage, the inspector identified exposure significant work activities in the Reactor Containment Building (RBC). Specific work activities included reactor head inspection/repairs, core barrel removal, containment sump modifications, steam generator tube eddy current inspection/plugging, and steam generator secondary side cleaning. The inspector reviewed radiation survey maps and radiation work permits (RWP) associated with these activities to determine if the associated controls were acceptable.

- The inspector toured accessible radiological controlled areas in the Unit 2 RBC. With the assistance of the ALARA Coordinator, the inspector performed independent surveys of selected areas to confirm the accuracy of survey maps and the adequacy of postings.

- In evaluating the RWPs, the inspector reviewed electronic dosimeter dose/dose rate alarm set points to determine if the set points were consistent with the survey indications and plant policy. The inspector verified that the workers were knowledgeable of the actions to be taken when the dosimeter alarms or malfunctions. Work reviewed included scaffolding erection (RWP 208-5039), core barrel removal (RWP 208-5026), steam generator secondary sludge lancing (RWP 208-5015), and steam generator platform support (RWP 208-5017).

- The inspector reviewed RWPs and associated instrumentation and engineering controls for potential airborne radioactivity areas. The inspector confirmed that no worker received an internal dose in excess of 10 mrem due to airborne radioactivity when performing outage related tasks. The inspector reviewed the whole body counting results and dose assessment methodology for tasks resulting in internal exposures that were less than 10 mrem to confirm the accuracy of the results.

Problem Identification and Resolution

- The inspector reviewed elements of the licensee's corrective action program related to controlling access to radiological controlled areas, completed since the last inspection of this area, to determine if problems were being entered into the program for resolution. Included in this review were the dose and dose rate alarm reports and personnel contamination reports to determine if regulatory limits or performance indicator criteria were exceeded.

- The inspector reviewed Condition Reports, and associated corrective actions, recent Nuclear Quality Assessment field observation reports, and the fourth quarter 2007 and first quarter 2008 Nuclear Oversight Performance Report to evaluate the threshold for identifying, evaluating, and resolving problems in implementing the ALARA program.

Jobs-In-Progress

- The inspector observed aspects of various ongoing activities to confirm that radiological controls, such as required surveys, area postings, job coverage, and pre-job RWP briefings were conducted; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological conditions. The inspector attended the pre-job RWP briefing for reactor core barrel removal.

High Risk Significant - LHRA and VHRA Controls

- Keys to locked high radiation areas (LHRA) and very high radiation areas (VHRA), stored at the containment control point were inventoried and accessible LHRAs were verified to be properly secured and posted during RBC tours.

- The inspector discussed with radiation protection supervision the adequacy of physical and administrative controls for performing work in potentially VHRAs, including the movement of reactor in-core detectors to their storage locations and spent fuel transfers. The inspector verified that any changes to relevant procedures did not substantially reduce the effectiveness and level of worker protection and evaluated the adequacy of prerequisite communications and authorizations.

Radiation Worker Performance

- The inspector observed radiation worker and radiation protection technician performance during reactor core barrel removal, reactor head inspections, and spent fuel transfers. The inspector determined that the individuals were aware of current radiological conditions, access controls, that the skill level was sufficient with respect to the potential radiological hazards and the work involved.

- The inspector reviewed condition reports, related to radiation worker and radiation protection errors, and personnel contamination event reports to determine if an observable pattern traceable to a similar cause was evident.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (17 samples)

The inspector 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 tasks being conducted during the 2R13 refueling outage. This inspection represents completion of 17 samples relative to this inspection area.

Radiological Work Planning

- The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing outage activities to assess current performance and outage exposure challenges. The inspector determined the site's 3-year rolling collective average exposure.

- The inspector reviewed the refueling outage work scheduled during the inspection period and the associated work activity dose estimates. Scheduled work reviewed included reactor core barrel removal, steam generator secondary side cleaning, reactor head inspections/repair, spent fuel transfers, and various in-containment support activities.

- The inspector reviewed procedures associated with maintaining worker dose ALARA and with estimating and tracking work activity specific exposures.

- The inspector reviewed the 2R13 dose summary reports, detailing worker estimated and actual exposures.

- The inspector evaluated the exposure mitigation requirements specified in RWPs and ALARA Plans (AP), and compared actual worker cumulative exposure with estimated dose for tasks associated with these activities. The inspector reviewed those work activities whose estimated cumulative exposure exceeded 5 person-rem. These jobs included reactor head inspection/repair (RWP 208-5058), reactor core barrel removal (RWP 208-5056), containment sump modification (RWP 208-5055), steam generator sludge lancing (208-5015), and installing permanent steam generator platforms (RWP 208-5067).

- The inspector evaluated the departmental interfaces between radiation protection, engineering, operations, and maintenance crafts to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing the Manager-Radiation Protection, the Senior Nuclear Specialist-ALARA, and the Supervisor, Radiation Protection Services; reviewing ALARA Committee meeting minutes; reviewing Nuclear Oversight field observation reports; and attending daily departmental turnover meetings, and an ALARA Committee meeting, regarding elevated dose for reactor head repairs.

- The inspector compared the person-hour estimates provided by the maintenance planning and other work groups with actual work activity time requirements and evaluated the accuracy of these estimates. Specific jobs reviewed included reactor head inspection/repair and containment sump modification activities.

- The inspector determined if work activity planning included the use of temporary shielding, system flushes, and operational considerations; e.g., filling steam generators during dose intensive tasks, to further control dose. The inspector examined temporary shielding installed to support steam generator maintenance and radwaste staging in containment.

- The inspector reviewed personnel contamination event (PCE) reports, whole body counting data and related calculations for internal dose assessments for selected personnel. The inspector reviewed the effectiveness of the licensee’s methods for controlling airborne radioactivity concentrations through the use of temporary ventilation systems.

Verification of Dose Estimates and Exposure Tracking Systems

- The inspector reviewed the assumptions and basis for the annual site collective exposure estimate and the Unit 2 refueling outage dose projection.

- The inspector reviewed the licensee’s method for adjusting exposure estimates, and re-planning work, when emergent work or expanded job scope was encountered. The inspector attended an Outage ALARA Committee meeting for reactor head inspection/repair, and reviewed recent actions of the committee in monitoring and controlling dose allocations.

- The inspector reviewed the licensee’s exposure tracking system (HIS-20) to determine whether the level of detail, exposure report timeliness and dissemination was sufficient to support the control of collective exposures. Included in this review were departmental dose compilations, specific RWP dose summaries, and individual exposure records.

Job Site Inspection and ALARA Control

- The inspector observed maintenance and operational activities being performed for reactor head eddy current inspections and core barrel removal to verify that radiological controls, such as required surveys, job coverage, pre-job briefings, and contamination controls were implemented; personnel dosimetry was properly located; and that workers were knowledgeable of work area radiological conditions.

- The inspector reviewed the exposure of individuals in selected work groups, including mechanical maintenance, radiation protection, and electrical maintenance to determine if supervisory efforts were being made to equalize dose among the workers.

Source Term Reduction and Control

- The inspector reviewed the status and historical trends for the Unit 2 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 by the licensee included shutdown chemistry controls, increased letdown flow, use of macro-porous resin, system flushes, and temporary shielding.

Declared Pregnant Workers (DPW)

- The inspector reviewed the procedural controls implemented for declared pregnant workers and determined that no DPWs were employed to support the 2R13 outage.

Problem Identification and Resolution

- The inspector reviewed elements of the licensee's corrective action program related to implementing radiological controls to determine if problems were being entered into the program for timely resolution.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety [PS]

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope (6 Samples)

During the period June 16 - 19, 2008, the inspector conducted the following activities to verify that the licensee's radioactive material processing and transportation programs complied with the requirements of 10 CFR 20, 61, 71; and Department of Transportation (DOT) regulations 49 CFR 170-189.

Radioactive Waste Systems Walkdown

The inspector walked down accessible portions of the radioactive liquid processing systems and site radwaste storage areas with the Field Supervisor-Operations/Radwaste and the Supervisor Radwaste/Transportation, respectively. During the tour, the inspector evaluated if the systems and facilities were consistent with the descriptions contained in the UFSAR and the Process Control Program (PCP), evaluated the general material conditions of the systems and facilities, and identified any changes to the systems. The inspector reviewed the current processes for transferring radioactive resin/sludge to shipping containers, and the subsequent de-watering process.

Also during this tour, the inspector walked down portions of radwaste systems that are no longer in service or abandoned in place, and discussed with the Radwaste Systems Engineer, the status of administrative and physical controls for these systems including components of the radwaste evaporators and solidification equipment. As part of this review, the inspector reviewed actions taken by the Plant Health Committee in evaluating abandoned systems.

The inspector visually inspected various radioactive material storage locations with a Radiation Protection Specialist, including areas of the Waste Handling Buildings, Decontamination Facilities, outside yard locations within the Protected Area, and the Old Steam Generator Storage Facility to evaluate material conditions and accuracy of inventories.

Waste Characterization and Classification

The inspection included a selective review of the waste characterization and classification program for regulatory compliance, including:

- Radio-chemical sample analytical results for various radioactive waste streams;
- Development of scaling factors for hard-to-detect radio-nuclides from radio-chemical data;
- Methods and practices to detect changes in waste streams; and
- Characterization and classification of waste relative to 10 CFR 61.55 and the determination of DOT shipment subtype per 49 CFR 173.

Shipment Preparation

The inspection included a review of radioactive waste program records, shipment preparation procedures, training records, and observations of jobs-in-progress, including:

- Reviewing radwaste and radioactive material shipping logs for calendar years 2006, 2007, and 2008;
- Verifying that training was provided to appropriate personnel responsible for classifying, handling, and shipping radioactive materials, in accordance with Bulletin 79-19 and 49 CFR 172 Subpart H;
- Verifying that appropriate NRC (or agreement state) license authorization was current for shipment recipients for recent shipments; and
- Verifying compliance with the relevant Certificates-of-Compliance and related procedures for shipping casks.

Shipment Records

The inspector selected and reviewed records associated with five Type B shipments of radioactive material made since the last inspection of this area. The shipments were Nos. B-3503, B-3577, B-3590, B-3598, and B-3664. The following aspects of the radioactive waste packaging and shipping activities were reviewed:

- Implementation of applicable shipping requirements including proper completion of manifests;
- Implementation of specifications in applicable certificates-of-compliance, for the approved shipping casks, including limits on package contents;
- Verification that dewatering criteria was met;
- Classification of radioactive materials relative to 10 CFR 61.55 and 49 CFR 173;
- Labeling of containers relative to package dose rates;
- Radiation and contamination surveys of the packages;
- Placarding of transport vehicles;
- Conduct of vehicle checks;
- Providing of emergency instructions to the driver;
- Completion of shipping papers; and
- Notification by the recipient that the radioactive materials have been received and disposed of.

Identification and Resolution of Problems

The inspector reviewed the 2007 Annual Radioactive Effluent Release Report, relevant Condition Reports, a Nuclear Oversight Audit, Quality Field Observation reports, and quarterly performance reports. Through this review, the inspector assessed the licensee's threshold for identifying problems, and the promptness and effectiveness of the resulting corrective actions. This review was conducted against the criteria contained in 10 CFR 20.1101(c) and the licensee's procedures.

b. Findings

No findings of significance were identified

4. OTHER ACTIVITIES **[OA]**

4OA1 Performance Indicator Verification (71151) (Total - 6 samples)

a. Inspection Scope

The inspectors sampled licensee submittals for Performance Indicators (PI) listed below for both Unit 1 and Unit 2. The inspectors reviewed portions of various logs and reports specified and PI data developed from monthly operating reports, and discussed methods for compiling and reporting the PIs with cognizant engineering and licensing 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, Rev. 5, "Regulatory Assessment Indicator Guideline," were used for each data element. Documents reviewed during this inspection are listed in the Attachment.

.1 Cornerstone: Mitigating Systems (2 samples)

Safety System Functional Failure [MS05] (Units 1 & 2)
The inspectors reviewed the PI for safety system functional failures by to verify conditions that prevented or could have prevented the fulfillment of safety functions were properly reported. The inspectors reviewed logs, maintenance rule records, work orders, and event reports. Inspectors reviewed failure data from October 2007 to March 2008.

.2 Cornerstone: Barrier Integrity (4 samples)

Reactor Coolant System (RCS) Activity [BI01] (Units 1 & 2)
The inspectors reviewed the PI for RCS activity to verify that the proper dose equivalent Iodine-131 was reported and that it was below the TS limit. Inspectors reviewed data from April 2007 to March 2008.

Reactor Coolant System Leak Rate [BI02] (Units 1 & 2)
The inspectors reviewed the PI for RCS leak rate to verify that the maximum identified leakage did not exceed the TS value and that it was properly reported. Inspectors reviewed data from April 2007 to March 2008.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

Daily Review of Problem Identification and Resolution

a Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for followup, 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.

4OA3 Event Followup (71153 - 5 total samples)

.1 Unit 2: Drop of Control Rod Drive Shaft (CRDS) onto Rod Cluster Control Assembly (RCCA) during Refueling Outage Installation

a Inspection Scope

On May 2, 2008, a CRDS was dropped into its associated guide tube from approximately 30 inch height due to a malfunction of the Baron drive shaft handling tool. The CRDS struck the spider body of the RCCA located in fuel assembly R57, at core position P10.

This was the third transfer of a CRDS from the storage stand to the reactor head upper internals after split-pin replacements had been completed during refueling outage 2R13.

The inspectors verified that operations and refueling personnel responded in accordance with procedures, and equipment responded as intended, by reviewing control room narrative logs, completed procedures, corrective action program condition reports, and interviews of involved personnel. The inspectors reviewed the immediate actions of the involved drive shaft handling crew and the support by the vendor to evaluate the condition of affected reactor components. The inspectors monitored the vendor's camera inspection of the affected RCCA and CRDS and reviewed video inspection data of the surrounding area. The inspectors did not observe any abnormal configurations, but noted two small areas where the oxide layer was removed on the RCCA spider body where the CRDS struck squarely. No loose or foreign material was observed.

The inspectors reviewed FENOC's evaluation of the vendor inspection and justification for continued component use. The inspectors attended licensee challenge meetings of the vendor's evaluation. All affected components were determined to be acceptable for use. The Baron handling tool was quarantined for further investigation. The subsequent installations of CRDS's were successfully performed using the licensee's handling tool. This issue is documented in condition report 08-39693.

b. Findings

No findings of significance were identified.

.2 Unit 2: Main Generator High Pressure Turbine (HPT) Bearing Damage during Initial Startup

a Inspection Scope

On May 16, during initial main turbine roll following refueling outage 2R13, the licensee identified a damaged bearing on the main turbine as indicated by high metal and oil temperatures during turbine roll at 550 rpm. Further investigation confirmed a damaged #2 main turbine bearing associated with the high-pressure turbine (HPT), which had recently been installed during the refueling outage. No additional main turbine damage was identified. The reactor was critical, at approximately 15 percent power. The licensee tripped the main turbine, secured the secondary steam plant, and performed a reactor plant shutdown to Mode 3.

The inspectors verified that operations personnel responded in accordance with procedure and that equipment performed as intended by observing crew operations and plant shutdown, reviewing procedures and operating logs, and interviewing involved personnel. Technical specifications and condition reports were also reviewed. The inspectors performed a walkdown of control room panels and secondary plant equipment to verify equipment status and plant parameters. The licensee identified several concurrent conditions contributing to the cause of the bearing failure, but attributed the failure mainly to bearing overload. The bearing was replaced and the main turbine tested satisfactory after plant startup on May 22. This issue is documented in condition report 08-40485.

b. Findings

No findings of significance were identified.

.3 Unit 2: Inadvertent Main Turbine Load Increase at Low Power due to Electro-Hydraulic Control (EHC) Malfunction

a Inspection Scope

On May 23, at 00:19 am, after placing Unit 2 on-line, operators transferred main turbine control into First Stage Pressure Feedback Mode (First Stage In) and experienced an instantaneous load increase of 80MWe. This increase in turbine load caused Reactor Coolant System (RCS) temperature and pressure to lower in response. Operators immediately responded by lowering turbine load to restore RCS parameters. The plant was stabilized after approximately ten minutes without control rod motion. The operators continued to experience main turbine control difficulty and eventually placed the control system in manual. The licensee determined that two Electronic Hydraulic Controller (EHC) cards had failed and required replacement. The unit was taken offline on May 24 at 4:20 am for card replacement and returned to service on May 24 at 9:10 pm.

The morning following the event, inspectors conducted control panel walkdowns, reviewed crew statements, and trended plant data from the transient. Appropriate TS LCOs were entered and actions taken within the allowed times. The minimum observed RCS pressure was 2100 psig, and minimum RCS average temperature was 536 degrees Fahrenheit (F). RCS average temperature lowered below 541F for approximately 2 ½ minutes.

b. Findings

Introduction. A Green self-revealing NCV of TS 5.4.1.(a) was identified in that the licensee failed to take appropriate action to trip the main turbine as specified in 2OM-52.4.A, "Raising Power from 5% to Full Load Operation," Rev. 13, during an unexpected main turbine load increase that caused average reactor coolant temperature to lower below the operational limit of 541F.

Description. On May 23, at 00:19 am, after placing Unit 2 on-line, operators transferred main turbine control into First Stage Pressure Feedback Mode (First Stage In) and experienced an instantaneous load increase of 80MWe. This increase in turbine load caused Reactor Coolant System (RCS) temperature and pressure to lower in response. Operators immediately responded by lowering turbine load to restore RCS parameters. The plant was stabilized after approximately ten minutes. The operators continued to experience main turbine control difficulty and eventually placed the control system in manual. The licensee determined that two Electronic Hydraulic Controller (EHC) cards had failed and required replacement. The unit was taken offline on May 24 at 4:20 am for card replacement and returned to service on May 24 at 9:10 pm.

The morning following the event, inspectors conducted control panel walkdowns, reviewed crew statements, and trended plant data from the transient. The minimum observed RCS pressure was 2100 psig, and minimum RCS average temperature was 536 degrees Fahrenheit (F). RCS average temperature lowered below 541F for approximately 2 ½ minutes.

The unit had recently been started and entered Mode 1 on May 22 at 5: 27 pm, with the unit placed online at 11:57 pm. Power operations were governed by 2OM-52.4.A, "Raising Power from 5% to Full Load Operation," Rev. 60, which states in Precaution and Limitation (P&L):

- #6, "The turbine shall be tripped if Tavg drops to 541°F."

A review of general operating instructions 2OM-52.2.A, "Precautions, Limitations and Setpoints," Rev. 13, identified similar precautions and limitations (P&L):

- P&L 31, in part: "When above the point of adding heat (5E-6 amps), if the lowest operating loop temperature (Tavg) drops below 541F, the turbine should be immediately tripped."
- P&L 38: "The turbine shall be tripped if Tavg drops to 541F."

The inspectors discussed this issue with senior licensee management and observed department stand-down actions to communicate station expectations. In response to this event, the licensee has developed and implemented an operations department rapid improvement plan. The licensee documented this issue in CR 08-40753. The failure to take appropriate action to trip the main turbine as specified in 2OM-52.4.A, "Raising Power from 5% to Full Load Operation," Rev. 60 during an unexpected main turbine load increase that caused average reactor coolant temperature to lower below the operational limit of 541F is considered a performance deficiency.

<u>Analysis.</u> This finding was more than minor because it can be reasonably viewed as a precursor to a significant event. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter (IMC) 0609, Attachment 609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low risk 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 properly communicate critical parameters and limitations for personnel to perform work safely in a timely manner. [H.1.(c)].

<u>Enforcement</u>, TS 5.4.1(a) states, in part, that written procedures shall be properly established and implemented for process monitoring and power operations. Contrary to the above, on May 23, 2008, operators failed to take appropriate action to trip the main turbine as specified in 2OM-52.4.A, "Raising Power from 5% to Full Load Operation," Rev. 60, during an unexpected main turbine load increase that caused average reactor coolant temperature to lower below the operational limit of 541F. Because this deficiency is considered to be of very low significance (Green), and was entered into the corrective action program (CR-08-40753) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 5000412/2008003-03, Failure to Properly Implement Operating Procedure during Plant Startup)**

.4 Unit 2: Steam Generator Level Transient and P14 Actuation on High Water Level

a Inspection Scope

On May 24, while making preparations to roll the main turbine after control system repairs, operators observed on the control panel a decreasing trend on steam generator water level (SGWL) for all steam generators. Operators took manual control of SGWL and commenced actions to restore SGWL. During these actions, a SGWL transient ensured, which resulted in reaching a Hi-Hi SGWL alarm setpoint and subsequent feedwater isolation setpoint (P-14, 92.2% level). During the transient, reactor power was approximately 16 percent, the main turbine was off-line, and primary temperature was controlled via the steam dumps to the main condenser. The 'B' main feedwater pump was operating. The feedwater isolation caused the running main feed pump to trip and subsequent automatic start of the two safety-related motor-driven auxiliary feedwater pumps (MDAFW).

The crew entered the alarm response procedure (ARP) for high SGWL. Sequence of Events computer recorded a SG Level Hi-Hi Turbine Trip, Low-Low Tave K636 Trip, and a P14 Feedwater Isolation 'B' main feed pump auto stop. Within two minutes, and SGWL indicating within the control band, operators successfully reset the feedwater isolation (P-14) and restored normal feedwater to the steam generators by restarting the 'B' main feed pump. The licensee notified the NRC of the automatic start of the MDAFW in accordance with 10CFR50.72(b)(3), (Event Notification #44239). Inspectors were onsite immediately following the event and conducted control panel walkdowns, interviews, plant data and document reviews to assess plant conditions, personnel, and equipment performance associated with the event, and licensee corrective actions.

b. Findings

Introduction. A Green self-revealing NCV of TS 5.4.1.(a) was identified in that the licensee failed to properly enter and implement the appropriate abnormal operating procedure (AOP) for loss of main feedwater.

Description. On May 24, operators in the control room observed steam generator water levels (SGWL) trending down. Reactor power and steam flow were approximately 16 percent power in preparation for placing the main turbine on-line. Primary temperature control was via the steam dumps to the main condenser and the 'B' main feedwater pump was operating, supplying feedwater to the three steam generators. SGWL is normally controlled and maintained utilizing water from the feed and condensate system through electric motor driven feed pumps and air operated flow control valves. The flow control valves are normally controlled in automatic. However, if the need arises, the flow control valves can be controlled remotely in manual from the control room.

Operators took manual control of the feedwater regulating valves because it appeared that automatic level control was not responding properly. This is an expected operator response. The operators attempted to control SGWL, but during these actions, a significant SGWL transient ensured. This resulted in SGWL reaching a Hi-Hi SGWL alarm setpoint and subsequent feedwater isolation setpoint (P14, 92.2%). The feedwater isolation caused the running main feed pump to trip and an automatic start of the two safety-related motor-driven auxiliary feedwater pumps (MDAFW). The crew entered the alarm response procedure (ARP) for Hi-Hi SGWL (2OM-24.4.AAK). The Sequence of

Events computer recorded a SG Level Hi-Hi Turbine Trip, Low-Low Tave K636 relay trip, and a P-14 Feedwater Isolation 'B' main feed pump auto stop. Since the turbine was not on-line, no turbine trip occurred. The crew evaluated the P-14 actuation as a result of unexpected steam generator swell and set feedwater demand to zero to lower SGWL.

The resultant isolation and main feedwater pump trip met the entry conditions for Abnormal Operating Procedure 2OM-53C.4.2.24.1 "Loss of Main Feedwater." Entry conditions for Loss of Main Feedwater (2OM-53C.4.2.24.1) were discussed by the crew, but the decision was made by the crew to transition to a main feedwater recovery procedure (2OM-24.4.N, Feedwater System Operation After Hi-Hi SG Level Trip) based on lowering SGWL and probable cause of the P-14 actuation. Reports were also received from field operators of abnormal operation of feedwater pump recirculation valves. The crew key-locked open both recirc valves (2FWR-FCV-150A&B). Within two minutes, and SGWL indicating within the control band, operators successfully reset feedwater isolation (P-14) and restored normal feedwater to the steam generators by restarting the 'B' main feed pump. Had the crew entered the Loss of Main Feedwater AOP, the procedure would have directed the crew to perform a manual reactor trip.

The expected hierarchy of procedures would have Emergency Operating Procedures (EOP) and Abnormal Operating Procedures (AOP) take priority over Annunciator Response Procedures (ARP) as prescribed in 1/2OM-48.2-C, "Adherence and Familiarization to Operating Procedures," and NOP-OP-1002, "Conduct of Operations" once entry conditions have been identified. Based on observed plant conditions, procedure response priority, and other plant complications the crew would be expected to have prioritized the Loss of Feedwater AOP over the Feedwater Isolation ARP and initiated a manual reactor trip. The failure of the operating crew to properly enter and implement the appropriate abnormal operating procedure (AOP) for loss of main feedwater is considered a performance deficiency. The licensee documented this issue in CR 08-40825.

The licensee immediately stabilized the plant and established an event response team. The crew was relieved of licensed duties, interviewed, and subsequently remediated. The inspectors discussed this issue with senior licensee management and observed department stand-down actions and remediation training for the relieving crew. The training focused on recent station events and a re-focus on operator fundamentals and procedure hierarchy. The licensee has developed and implemented an operations department rapid improvement plan. The inspectors will continue to follow the progress of the licensee's plan.

<u>Analysis.</u> This finding was more than minor because it can be reasonably viewed as a precursor to a significant event. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter (IMC) 0609, Attachment 609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low risk 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 properly implement appropriate roles and authority for decision making during risk-significant decisions. [H.1.(a)].

Enforcement. TS 5.4.1(a) states, in part, that written procedures shall be properly established and implemented for Loss of Feedwater or Feedwater System Failure. Contrary to the above, on May 24, 2008, operators failed to properly enter and implement the appropriate abnormal operating procedure (AOP) for loss of main feedwater. Because this deficiency is considered to be of very low significance (Green), and was entered into the corrective action program (CR-08-40825) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000412/2008003-04, Failure to Properly Implement Abnormal Operating Procedure during Plant Startup)**

.5 (Closed) LER 05000334/2007-002. Undetected Loss of 138 kV 'A' Phase to System Station Service Transformer Leads to Condition Prohibited by Plant Technical Specification

On November 3, 2007, the licensee 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 the 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. The licensee has initiated a root cause investigation under CR 07-30614. Corrective actions include removal of the metering units, interim changes to the offsite power availability surveillance test and switchyard walkdowns. The NRC reviewed this event when it occurred and issued a Green NCV (NCV 05000334/2007005-03, Failure to Comply with TS 3.8.1 Required Actions for One Offsite Power Source Inoperable) in NRC Inspection Report 05000334& 05000412/2007005.

The inspectors reviewed this LER and no additional findings were identified. The licensee has documented this event in their corrective action program under CR 07-30614. This LER is closed.

4OA5 Other

.1 Unit 2 Extended Power Uprate (IP 71004)

a. Inspection Scope

The inspectors observed selected plant testing and other power ascension activities during the implementation of the final two phases (+2.5%, +2.5%) (2770 MWt to 2900 MWt) of a planned 3-phase extended power uprate totaling approximately 8% power. Inspectors observed and/or reviewed selected plant changes and testing prior to the power ascension that began on June 3, 2008, as well as post-100% power activities and reviewed selected plant data to determine if significant plant anomalies occurred, and to ensure plant behavior was as predicted by simulator and analysis data.

The inspectors also reviewed operator actions, applicable procedure changes, and reviewed selected plant design changes and other inspection activities conducted under the normal baseline inspection program, to ensure an adequate sample of risk-significant attributes required by the governing procedure were evaluated. Specific inspections already completed and credited in past NRC inspection reports, as well as those credited in the current report can be found in the Attachment.

b. Findings

No findings of significance were identified.

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

During Unit 1 refueling outage 1R18 (October 2007), 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.

On October 2, 2007, during installation of weld overlays on pressurizer safety nozzles, FENOC and the contractors 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 "P3" 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 P3" material.

The NRC has completed its investigation of this issue and has reviewed the final evaluation of FENOC's assessment of the use of the referenced procedure for P-1 material. The inspectors determined this deficiency to be a licensee identified non-cited violation which is documented in Section 4OA7 of this report.

.3 (Closed) NRC Temporary Instruction 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)

a. Inspection Scope

The inspectors performed an inspection in accordance with Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage, Revision 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 that the implementation of the modifications and procedure changes was consistent with the actions discussed in FENOC's letters to the NRC, dated December 20, 2007 and February 14, 2008.

The inspectors reviewed the TS and the UFSAR to verify that required changes to the TS had been approved by the NRC, and that the UFSAR had been or was in the process of being updated to reflect the plant changes. Additionally, the inspectors reviewed a sample of procedures to verify that they were updated to reflect programmatic changes to the facility. The inspectors also reviewed a sample of work orders to verify that specific work had been performed to meet FENOC's GL commitments. Finally, the inspectors discussed details of the containment sump modification with engineers to verify design control of the modification process. Documents reviewed are listed in the attachment. Portions of the TI were performed during the 2006 refueling outage at Unit 2, and the 2007 refueling outage at Unit 1, which verified the physical modifications to the containment sump. The results of those inspections were documented in Inspection Report Nos. 05000412/2006005 and 05000334/2007005.

b. Evaluation of Inspection Requirements

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

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

The inspectors verified that FENOC has implemented, or was in the process of implementing, the plant modifications and procedure changes committed to in their GL 2004-02 responses. The inspections previously performed in 2006 and 2007 verified the implementation of the sump screen modifications as related to the GL. During this inspection, the inspectors verified that procedures were updated as related to programmatic controls of potential debris sources, and inspections for containment coating degradation. The inspectors verified that modifications to address downstream effects had been performed on both units. However, at the time of inspection, inspectors noted that additional downstream effects evaluations were ongoing at each unit. The inspectors noted that FENOC was still performing chemical effects testing and evaluating the need for further corrective actions on Unit 1.

2. 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. The inspectors also verified that FENOC had obtained NRC approval prior to implementing those changes that required such approval. Specifically, FENOC had obtained NRC approval prior to implementing changes to the recirculation spray system start signal on both units. Additionally, FENOC plans to submit a License Amendment Request for changes to the Unit 2 chemical addition system buffer agent. Finally, the inspectors verified that FENOC has a process in place to update the Beaver Valley Unit 2 licensing bases accordingly, contingent upon license amendment approval. FENOC plans to provide a follow-up supplemental response to the GL by August 30, 2008, which will discuss chemical effects testing, and chemical effects and downstream effects analyses.

Based on the inspectors' review of the hardware modifications, procedure changes, and licensing bases changes, the inspection requirements of the Temporary Instruction are complete and the TI is closed at Beaver Valley Units 1 and 2. In a letter dated February 29, 2008, NRR approved FENOC's request to extend the completion date for the remaining analyses and licensing activities required for GL 2004-02 compliance until no later than the end of the Unit 1 Refueling Outage in the Spring of 2009 (1R19). As of this inspection, the remaining activities include completion of downstream effects analyses on both units, completion of chemical effects testing and any required updates to calculations for Unit 1, completion of any corrective actions that may be required due to chemical effects testing on Unit 1, submission of a License Amendment Request associated with the chemical addition system buffer agent for Unit 2, and implementation of the buffer agent change-out upon approval of the Unit 2 license amendment.

The TI-2515/166 inspection results, as well as any results of sampling audits of licensee actions will be reviewed by the NRC as input, along with the Generic Letter (GL) 2004-02 responses to support closure of GL 2004-02 and Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance." The NRC will notify FENOC by letter of the results of the overall assessment as to whether GSI-191 and GL 2004-02 have been satisfactorily addressed at Beaver Valley Power Station. Completion of TI-2515/166 does not necessarily indicate that FENOC has finished all testing and analyses needed to demonstrate the adequacy of their modifications and procedure changes. As noted above, FENOC has obtained approval of a plant-specific extension that allows for completion of testing, analyses, and later implementation of plant modifications. FENOC will confirm completion of all corrective actions to the NRC. As part of the process described above to ensure satisfactory resolution of GL 2004-02 and GSI-191, the NRC will track all such yet-to-be-performed items identified in the TI-2515/166 inspection reports, track the items to completion, and may choose to inspect implementation of some or all of them.

c. Findings

No findings of significance were identified.

.4 (Closed) NRC TI 2515/172, RCS Dissimilar Metal Butt Welds (DMBW)

a. Inspection Scope

The Temporary Instruction, TI 2515/172 provides for confirmation that owners of pressurized-water reactors (PWRs) have implemented the industry guidelines of the Materials Reliability Program (MRP) -139 regarding nondestructive examination and evaluation of certain dissimilar metal welds in reactor coolant systems containing nickel based alloys 600/82/182. The TI requires documentation of specific questions in an inspection report. The questions and responses are included in Attachment "B" to this report.

In summary, Beaver Valley Units 1 and 2 have MRP-139 applicable Alloy 600/82/182 RCS welds. Unit 1 has five pressurizer nozzle welds that were preemptively mitigated using full structural weld overlays (FSWOL) during the Fall 2007 1R18 refueling outage (RFO). Unit 2 has six pressurizer nozzle welds that were also preemptively mitigated by FSWOL during the Fall 2006 2R12 RFO and three hot and three cold leg pipe to reactor pressure vessel nozzle DMBW connections that were examined from the inside volumetrically by ultrasonic testing and on the inside diameter (ID) surface by eddy current during the Spring 2008 2R13 RFO. No indication of cracking was found.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Access Control / ALARA Planning and Control

The inspector presented the inspection results of 2S01 and 2S02 to Mr. Kevin Ostrowski, Director of Site Operations, and other members of FENOC staff, at the conclusion of the inspection on April 24, 2008. The licensee acknowledged the conclusions and observations presented. No proprietary information is presented in this report.

.2 Inservice Inspection and TI 2515/172, RCS Dissimilar Metal Butt Welds (DMBW)

The inspector presented the ISI inspection results to Mr. Kevin Ostrowski, Director of Site Operations, and other members of the FENOC staff at the ISI inspection briefing on May 8, 2008, and results of the TI 2515/172 inspection for Beaver Valley Unit 1 and Unit 2 were presented at the conclusion of the inspection on May 15, 2008 by phone to Brian Sepelak, Supervisor, Licensing & Compliance. The licensee acknowledged the conclusions and observations presented. Some proprietary information was reviewed during this inspection and was either returned or properly destroyed, but no proprietary information is presented in this report.

.3 TI 2515/166, Pressurized Water Reactor Containment Sump Blockage

The inspector presented the inspection results to Mr. Mark Manoleras, Director of Engineering, and other members of the FENOC staff at the debrief on June 18, 2008, The licensee acknowledged the conclusions and observations presented. Some proprietary information was reviewed during this inspection and was either returned or properly destroyed, but no proprietary information is presented in this report.

.4 Radioactive Material Processing and Transportation

The inspector presented the inspection results of 2PS2 to Mr. Kevin Ostrowski, Director of Site Operations, and other members of FENOC staff, at the conclusion of the inspection on June 19, 2008. The licensee acknowledged the conclusions and observations presented. No proprietary information is presented in this report.

.5 Quarterly Inspection Report Exit

On July 16, 2008, the resident inspectors presented the normal baseline inspection results to Mr. Peter P. Sena, III, Beaver Valley Site Vice President, and other members of the licensee staff. The licensee acknowledged the findings and observations presented. The inspectors confirmed that proprietary information was not retained at the conclusion of the inspection period.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meest the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a non-cited violation (NCV).

- 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” requires that, “Measures shall be established to assure that special processes, including welding, heat treating and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.”

  Contrary to 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” the licensee failed to assure that a special welding process was performed in accordance with the applicable ASME Code 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. Failure to comply with ASME Code welding requirements could result in flaws within reactor coolant system piping welds. This issue was documented in the licensee’s corrective action program as Condition Report 07-27664. The safety significance of this issue was considered very low, since there were no recordable indications in the weld overlay, no adverse consequences were identified, the procedure deficiency was found prior to returning the component or system to service, and the welding process was subsequently qualified. Failure to perform welding of Class 1 welds in compliance with the applicable American Society of Mechanical Engineers (ASME) Construction Codes (ASME Section III ‘65 edition, Winter ‘66 addenda, ASME Section IX, latest edition) requirements is considered a licensee-identified violation (Green), Non-Cited Violation of 10CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes.”

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Alberti	Steam Generator Program Owner
S. Baker	Manager - Radiation Protection
R. Brosi	Director, Performance Improvement
A. Brunner	Systems Engineer, Radwaste
J. Clark	Radiation Protection Health Services Technician
P. Davis	Staff Nuclear Engineer
J. Dobo	Senior Radiation Protection Technician
J. Fontaine	ALARA Supervisor
J. Freund	Supervisor, Rad Operations Support
B. Furdak	First Energy Oversight
D. Grabski	ISI Coordinator
T. Heimel	NDE Level III
J. Hill	Bartlett Manager
E. Hubley	Director, Site Maintenance
C. Kellar	Manager, Compliance
J. Lebda	Senior Nuclear Specialist, Dosimetry
E. Loehlein	Alloy 600 Program Owner
M. Manoleras	Director, Engineering
J. Mauck	Regulatory Compliance
D. McGee	Field Supervisor, Operations/Radwaste
C. Miller	Senior Radiation Protection Technician
K. Ostrowski	Director, Site Operations
J. Patterson	Engineer
M. Pergar	Nuclear Oversight Supervisor
D. Price	Supervisor, Nuclear Project Engineering
R. Pucci	Senior Nuclear Specialist, ALARA Coordinator
J. Saunders	Supervisor, Radwaste and Transportation
P. Sena	Site Vice President
B. Sepelak	Supervisor, Regulatory Compliance
T. Sockaci	Principal Consultant, Design Engineering
M. Testa	Principal Consultant, Design Engineering
J. Wilbur	Radiation Protection Operations Field Coordinator
W. Williams	BACC Program Owner

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open/Closed

05000334/2008003-01	NCV	Inadequate Maintenance Procedure Results in Unexpected Terry Turbine Speed Increase. (Section 1R15)
05000412/2008003-02	FIN	Deficient Control of Clearance Posting Interrupts Reactor Coolant Charging Path while Vessel Water Level Drained below the Flange. (Section 1R20)
05000412/2008003-03	NCV	Failure to Properly Implement Operating Procedure during Plant Startup. (Section 4OA3.3)
05000412/2008003-04	NCV	Failure to Properly Implement Abnormal Operating Procedure during Plant Startup. (Section 4OA3.4)

Closed

05000334/2007-002	LER	Undetected Loss of 138 kV 'A' Phase to System Station Service Transformer Leads to Condition Prohibited by Plant Technical Specification. (Section 4OA3.5)
05000334/2007005-02	URI	Weld Overlays on Pressurizer Safety Nozzles Not Initially Qualified for P-1 Materials. (Section 4OA5.2)
05000334, 412/2515/166	TI	Pressurized Water Reactor Containment Sump Blockage. (Section 4OA5.3)
05000334, 412/2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds. (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
1/2OM-53C.4A.35.1, “Degraded Grid,” Rev. 4
NOP-OP-1003, “Grid Reliability Protocol,” Rev. 0
NOP-OP-1007, “Risk Determination,” Rev. 5

Section 1R04: Equipment Alignment

Drawings
RM-0082A, Fuel Pool Cooling and Purification (Unit 2 Flow)
RM-0420-001, Fuel Pool Cooling and Purification (Unit 2 VOND)

RM-0047A through F, Circulating and Service Water Piping (Unit 2 Flow)
RM-0430-002/003, Service Primary Cooling (Unit 2 VOND)
RM-0513-01/02, Containment Depressurization System (Unit 1 Flow)
RM-0413-001/002, Containment Depressurization System (Unit 1 VOND)

Other
1-DBD-13, Rev. 15, Containment Depressurization System
1-DBD-24B, Rev. 10, Auxiliary Feedwater System
1-DBD-30, Rev. 15, River Water, Auxiliary River Water and Raw Water Systems
2-DBD-30, Rev. 16, Service Water Systems
2-DBD-20, Rev. 8, Fuel Pool Cooling and Purification System
BV1 Reactivity Impact Determination 2008-02, dated June 11, 2008

Section 1R05: Fire Protection

Condition Reports
08-39094

Other
Pre-Fire Plans for CV-3, VP-2, RC-1, PA-1G, CB-2, CO-2
Fire Protection Analysis No. 10080-B-085, Rev. 12
Updated Fire Protection Appendix R Analysis for Unit 1, Rev. 26
Fire Protection Safe Shutdown Report for Unit 2, Rev. 28

Section 1R08: Inservice Inspection

Procedures
MRS-SSP-1511, Reactor Vessel Head CRDM Penetration Repair at Beaver Valley Unit 2
Unit 1/2, NDE-GP-106, Reactor Vessel Head Inspection, Rev. 1
Unit 1/2, NDE GP-105, Evaluation of PSI/ISI Flaw Indications, Rev. 9
Westinghouse Procedure WDI-UT-013, IntraSpect UT Analysis Guidelines, Rev. 13
1/2 ADM-2039 Beaver Valley ISI 10-Year Plans, Rev. 6
1/2 ADM-0801, ASME Section XI Repair/Replacement Program, Rev. 6
1/2-ADM-2112, Boric Acid Corrosion Control, Rev. 3

Certifications
PCI Welder Performance Paperwork Certifications PCI Energy Services, 35 welders

CRs

08-36128	08-38820	08-39282	08-39425	08-39717	08-39807
08-39638	08-39097	08-38225	08-38234		

Other
SG-CDME-08-14, BV Unit 2 Steam Generator Degradation Assessment 2R13 Refueling Outage April 2008, dated 3/11/2008
Unit 2, Interval 3, Period 3, Refueling Outage 2R13, RV Examination Summary, April 30, 2008
Visual Examination for Boric Acid Detection Report, BOP-VT-08-015, dated 4/18/2008
RCS*REV21-N-23 Ultrasonic Examination Indication Reports, VEN-08-1032
RCS*REV21-N-24 Ultrasonic Examination Indication Reports, VEN-08-1033

RCS*REV21-N-25 Ultrasonic Examination Indication Reports, VEN-08-1034
RCS*REV21-N-26 Ultrasonic Examination Indication Reports, VEN-08-1035
RCS*REV21-N-27 Ultrasonic Examination Indication Reports, VEN-08-1036
RCS*REV21-N-28 Ultrasonic Examination Indication Reports, VEN-08-1037
Eddy Current Examination Report, Unit 2 2R13, Steam Generators - April 2008
Radiographic Examination, BOP-RT-08-005, 2CHS-FCV-122, dated 5/2/2008
Radiographic Examination, BOP-RT-08-006, 2CHS-FCV-122, dated 5/2/2008
WPS 8F AU-GTAW, ASME IX Welding Procedure Specification, Rev. 6
WPS 8 MN-GTAW/SMAX, ASME IX Welding Procedure Specification, Rev. 15
Ultrasonic Report Data Sheet Penetration No. 51 DMW-R13-OH01-51-03
NOP-ER-2001, Boric Acid Corrosion Control Program, Rev. 7
SG-CDME-07-8, BV Unit 2 Steam Generator Cycle 13 Operational Assessment, February 2007, dated 2/13/2007

Section 1R12: Maintenance Rule Implementation

Condition Reports
08-37206 08-37252 08-37573

Procedures
NOP-ER-3004-03, Maintenance Preventable Functional Failure Evaluation

Other
CA 08-37206-03
NOTF 600458153

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures
½-ADM-0804, On-Line Risk Assessment and Management
NOP-OP-1007, Risk Determination

Condition Reports
08-41044 08-37686 08-40655

Other
Unit 1 & Unit 2 Weekly Maintenance Risk Summary for the weeks of March 30, April 7, May 21, May 30, June 12, 2008
Unit 1 Shift Operating Logs dated March 31, April 8-9, May 30, June 12, 2008
Unit 2 Shift Operating Logs dated April 13, April 26, May 21, 2008

Section 1R15: Operability Evaluations

Drawings
8700-RM-412-1, Rev. 19, “Valve Oper. No. Diagram Containment Depressurization System”

Procedures
1ICP-13-FI103, FI-QS-103 Quench Spray Pumps Recirculation Flow Indicator Calibration
1OST-47.3G, Rev. 8 effective 3/28/08

1OST-47.3G, Rev. 7 effective 10/30/07
1OST-47.3G, Rev. 6 effective 10/18/07
1OST-47.3G, Rev. 5 effective 3/2/07
2MSP-2.16-I, Rev. 9, Nuclear Instrumentation Source Range N32 Neutron Detector Channel Calibration
2OST-30.13B, Rev. 23, “Train B Service Water System Full Flow Test
2OST-36.3, Rev. 28, Emergency Diesel Generator [2EGS*EG2-1] Automatic Test

Technical Specifications
3.5.2 3.5.4 3.6.7

Condition Reports

08-37489	08-38416	08-42156	08-37757	08-38648	08-38265
08-39942	08-38016	08-38017	08-38180		

Work Orders and Notifications

600457128	600458849	200252872	200318826	200319013	200319076

Other
2DBD-13, Design Basis Document for Containment Depressurization System, Rev. 8
2DBD-30, Design Basis Document for Service Water System, Rev. 14
BVPS-1 Operator Rounds L5 logs, dated April 8, 2008
BVPS-1 UFSAR Chapter 14, Rev. 23
BVPS-2 Shift Logs, dated April 16, 2008
BVPS-2 PRA Notebook for Containment Depressurization System, Rev 3B
ECP 07-0077-ID-14, MOV Key Replacement, Rev. 0
Failure Analysis Report for 2SWE-116B key failure, dated May 2, 2008
Weak-Link Analysis for 2SWE-MOV116B, dated April 28, 2008

Section 1R18: Plant Modifications

Condition Reports
08-39497 08-39837 08-39594

Drawings
10080-RM-45B, “Unit 2 Flow Diagram Auxiliary Feedwater Piping,” Rev. 0
W-D-9909-(1), Rev. J, “2-inch Series 1500 Y-Type Globe Valve”

Procedures
1/2CMP-75-LEAK REPAIR-1M, Issue 4 Rev 15, On-Line Leak Repair
2OM-54.3.PAB2, “Unit 2 PAB Tour,” Rev. 30
2OST-24.4, “Steam Driven Auxiliary Feed Pump [2FWE*P22] Quarterly Test,” Rev. 62
2OST-24.4A, “Steam Driven Auxiliary Feed Pump [2FWE*P22] Full Flow Test,” Rev. 18

Work Orders
200319895 – 2RCS44, Clamp Installation
200321485 – 2RCS44, Leak Repair

Other
ECP-05-0343, “2FWE-P22 Turbine Driven Auxiliary Feedwater Pump Mechanical Seal

Upgrade," Rev. 0 and Rev. 1
Engineering Evaluation Request for 2RCS-44, per ECP 08-0237 (NOTF 600462826)

Section 1R19: Post-Maintenance Testing

Procedures
2OST-30.13B, "Train B Service Water Full Flow Test", Rev. 23
2OST-36.2, "Emergency Diesel Generator [2EGS*EG2-2] Monthly Test," Rev. 54
2DB2-36A, "Design Basis Document for Emergency Diesel Generator System," Rev. 8
1/2 –CMP-M-24-001, "Auxiliary Feed Pump Turbine Governor Valve Overhaul," Rev. 6
2ICP-24-FI155, "Auxiliary Steam Generator Feed Pump (2FWE*P22) Discharge Flow Indicator 2FWE-FI155 Calibration," Rev. 1
2ICP-24-PI155, "Auxiliary Steam Generator Feed Pump (2FWE*P22) Discharge Pressure Indicator 2FWE-PI155 Calibration," Rev. 4
2ICP-24-PI156, "Auxiliary Steam Generator Feed Pump (2FWE*P22) Suction Pressure Indicator 2FWE-PI156 Calibration," Rev. 2
2MSP-6.68-I, "Reactor Overpressurization PORV PCV 455C Setpoint Test"
2MSP-24.35-I, "2FWE-F100A, Auxiliary Feedwater Flow Loop Calibration," Rev. 7
2MSP-24.36-I, "2FWE-F100B, Auxiliary Feedwater Flow Loop Calibration," Rev. 7
2MSP-24.37-I, "2FWE-F100C, Auxiliary Feedwater Flow Loop Calibration," Rev. 8

Technical Specifications
TS 3.0.5, 3.8.1
TS Basis 3.8.1

Work Orders & Notifications

200249585	200249476	200286543	200151145	200252480	600464932

Condition Reports

08-39430	08-39006	08-38950	08-38903	07-38704	08-38688
08-39535	08-40164	08-40194			

Other
Problem Solving Plan, Rev. 0
ASME OM Code-2001 ISTB Inservice Testing of Pumps in Light-Water Reactor Nuclear Power Plants

Section 1R20: Refueling and Outage Activities

Procedures and Surveillances
2OM-6.4.I, Rev. 2, "Draining the RCS for Refueling"
2OM-47.4.B, Rev. 6, "Personnel Air Lock Operations"
2OM-49.4.H, Rev. 10, "Movement of Spent Fuel Pool Crane Checklist"
2OM-51.4.I, Rev. 3, "Station Shutdown-Preparation for Entering Refueling (Mode 6)"
2OST-6.2A, Rev. 25, "Computer Generated RCS Water Inventory Balance"
2OST-7.8, Rev. 10, "Boric Acid Storage Tank and RWST Level and Temperature Verification"
2OST-47.3.E, Rev. 5, "Verification of Administrative Closure Controls for Containment / Fuel Building during Refueling"
2OST-49.3, Rev. 11, "Refueling Operations Prerequisites"

2RP-2.6, Issue 0, Rev. 4, “Remove Reactor Vessel Studs/Clean”
2RST-2.1, Issue 1, Rev. 8, “Initial Approach to Criticality After Refueling”
AOP-2.6.5, Shutdown LOCA
AOP-2.10.1, RHR System Loss
AOP-2.36.1, Loss of All AC while Shutdown
IPTE - Draining Down the RCS for Refueling
RWP 308-3002
2OST-49.2, “Shutdown Margin Calculation”, performed on May 10, 2008
2OST-11.18, “Low Head Safety Injection Pump Boric Acid Flowpath Verification”
NOBP-OP-0007-01,”IPTE Worksheet for Reactor Vessel Lower Internals Removal,” Rev. 0
NOBP-OP-0007-02, “Pre-Evolution Briefing for Lower Internals Removal,” Rev. 0
2RP-3.27, “Refueling Procedure Lower Internals Assembly Removal/Installation,” Rev.3
1/2 CMP-47, “Contingency Hatch Closure –IM,” Rev. 0

Condition Reports

08-40134	08-38539	08-40416	08-40353	08-40407	08-40639
08-40436	08-40433	08-40332	08-40420	08-40363	08-40400*
08-40397*	08-40345	08-40162	08-40170	08-40274	08-40279
08-40169	08-40152	08-40217	08-39508	08-39699	08-39799
08-39869	08-39771	08-39396	08-39693	08-39717	08-39770
08-38539	08-39393	08-39416	08-39250	08-38921	08-39102

Other
2BVT-1.47.2, “Containment Type A Leak Test,” Rev. 2
2OM-50.4.L, RCS and Pressurizer Spray Heatup Data and Plots, dated May 11-16, 2008
2R13 Outage Handbook
ANSI/ANS-56.8-1994, Section 5.11, “Reporting of Type A Test Results”
Unit 2 Plant Computer Cooldown Data tables and plots, dated April 14, 2008

Section 1R22: Surveillance Testing

Procedures
2OST-30.13A, “Train A Service Water Full Flow Test,” Rev. 25
2OST-36.4, “Emergency Diesel Generator [2EGS* EG 2-2] Automatic Test,” Rev. 26
ERS-ATC-95-007, “Mixing Calculations,” Rev. 2

Condition Reports

08-37754	08-37676	08-40134

Technical Specifications
ITS 5.5.12

Other
WO 200295408 – Reactor Coolant Inventory Balance
Plant Computer Output for 2OST-06.02A dated June 30, 2008
Plant Computer Output from 2OST-11.14B dated May 3, 2008
NOTF 600462321
Reactor Containment Building Integrated Leakage Rate Test Report, performed May 19, 2008
2BVT-11.47.1 “Containment Local Leak Rate Monitoring Program”

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

Procedures:
Access Control to Radiological Significant Areas/ALARA Planning & Controls
1/2-ADM-1601, "Radiation Protection Standards," Rev. 15
1/2-ADM-1611, "Radiation Protection Administrative Guide," Rev. 9
1/2-ADM-1621, "ALARA Program," Rev. 3
1/2-ADM-1630, "Radiation Worker Practices," Rev. 10
1/2-ADM-1631, "Exposure Control," Rev. 5
1/2-HPP-3.02.003, "Decontamination Control," Rev. 8
1/2-HPP-3.02.004, "Area Posting," Rev. 4
1/2-HPP-3.04.002, "Bioassay Administration," Rev. 5
1/2-HPP-3.05.001, "Exposure Authorization," Rev. 4
1/2-HPP-3.07.002, "Radiation Survey Methods," Rev. 5
1/2-HPP-3.07.013, "Barrier Checks," Rev. 3
1/2-HPP-3.08.003, "Radiation Barrier Key Control," Rev. 10
1/2-HPP-3.08.006, "Shielding," Rev. 1
BVBP-RP-0003, "Dosimetry Practices," Rev. 4
BVBP-RP-0013, "Radiation Protection Risk Assessment Process," Rev. 2
BVBP-RP-0016, "Survey Requirements During Plant Transients," Rev. 0
BVBP-RP-0020, "RP Job Coverage General Guidance," Rev. 6
NOP-WM-7001, "ALARA Program," Rev. 1
NOP-WM-7002, "Operational ALARA Program," Rev. 1
NOP-WM-7003, "Radiation Work Permit," Rev. 3
NOP-WM-7015, "Respiratory Protection Program," Rev. 2
NOP-WM-7017, "Contamination Control Program," Rev. 1
NOP-WM-7021, "Radiological Postings, Labeling, and Markings," Rev. 2
NOP-WM-7025, "High Radiation Area Program," Rev. 0
BVBP-RP-0024, "Remote Monitoring," Rev. 1

Nuclear Oversight Reports:
Quality Field observations: BV320083281, BV220083304, BV320083318
Fourth Quarter 2007 and First Quarter 2008 Nuclear Oversight Performance Reports

Condition Reports :

08-39238	08-39243	08-39242	08-39241	08-39239	08-38345
08-38474	08-38349	08-36257	08-38029	08-38667	08-38665
08-38681	08-38644	08-39668	08-38494	08-38413	08-38301
08-38352	08-38435				

ALARA Council Meeting Minutes:
Meeting Nos: 2R13- 2 08-14 08-12 08-11

Radiation Work Permits/ALARA Plans:
RWP 208-5015/08-2-18, Sludge Lance Operations/ASCA Cleaning Secondary Side Inspection,
RWP 208-5017/5061 /08-2-20, Steam Generator Eddy Current, Tube Plugging
RWP 208-5039/ 08-2-31, Scaffolding

RWP 208-5067/08-2-72, Install Permanent Scaffolding Platforms
RWP 208-5018/08-2-21, Steam Generator Channel Head Work
RWP 208-5055/08-2-33, RBC Sump Modification
RWP 208-5027/08-2-38, Split Pin Replacement

Miscellaneous Reports:
Primary Demineralizer Use Plan for 2R13
2R13 Outage ALARA Plan
2R13 Steam Generator Maintenance Project ALARA Plan
2R13 Under Reactor Head Repair ALARA Plan
2R13 Areas with Restricted Access Due to Radiological Conditions
2R13 Shielding Plan
2R13 Personnel Contamination Event Mitigation Plan

Section 2PS2: Radioactive Material Processing and Transportation

Procedures:
1/2-PCP-1.01, "Process Control Program," Rev. 2
NOP-OP-2002, "Shipment of Radioactive Material/Waste," Rev. 5
BVBP-RP-0022, "Temporary On-Site Storage of Radioactive Waste," Rev. 1
1OM-18.4.AF(ISS3, "Unit 1 Dewatering of High Integrity Containers)," Rev. 2
2OM-18.4Y, "Unit 2 Dewatering Shipping Containers," Rev. 6
1OM-17.4AH, "Unit 1 Resin Transfer of LW Demineralizer (1LW-I-2)," Rev. 0
2OM-18.4F, "Flushing of any Group III Ion Exchanger Resin to a HIC," Rev. 4
BVBP-SITE-0010, "Abandoned In Place Equipment," Rev. 1

Nuclear Oversight Reports/Audits:
BV-PA-07-03, "Third Quarter Oversight Quarterly Performance Report" (2007)
BV-PA-07-01, "First Quarter Oversight Quarterly Performance Report" (2007)
MS-C-07-08-03, "Quality Assurance Audit Report"

Nuclear Oversight Field Observation Reports:

BV120073254	BV120062608	BV120062616	BV120062760
BV220062810	BV320072908	BV320073054	BV120073104

Shipping Manifests:
Shipment No. B-3577, Resin, Type B
Shipment No. B-3503, LSA-II, Type B
Shipment No. B-3664, Resin ,Type B
Shipment No. B-3598, Resin, Type B
Shipment No, B-3590, Resin, Type B

Condition Reports:

07-24068	07-24898	07-24887	07-21313	07-17090	07-13230
07-29227	08-40705	08-36759	08-33782	08-40135	06-05175
06-09977	06-09606	06-06931	08-42055		

Radwaste Systems Drawings:
Unit 1 Liquid Waste Disposal System, Dwg No. 8700-RM-417-3

Unit 2 Solid Waste Disposal Piping, Dwg. No. 1000-RM-418-1

Miscellaneous Documents:
RadWaste and Radioactive Material Shipping Logs for 2006, 2007, and 2008
2007 Beaver Valley Annual Radioactive Effluent Release Report
Radwaste/Transportation Training Records for selected personnel
10 CFR 61 Reports for 2006, 2007, and 2008
Designation of Authorized Radioactive Material Shippers

Materials for Bulletin 79-19 Training
GEN-USDOT-FEN-01, Lesson Plan for USDOT Regulations General Awareness
RP-RADSHIPPING-FEN, Lesson Plan for Radioactive Material Packaging, Transport, & Disposal

Section 4OA3: Event Response

Condition Reports

08-40825	08-40826	08-40485	08-40659	08-40678
08-40488	08-39693	08-39835	00-03293	

Drawing
10080-RM-424-1, Rev. 11, Main Feedwater System

Procedures
2BVT 1.47.5, “Type C Leak Test”
2OM-24.1.D, Rev. 6, Steam Generator Feedwater System Description, I&C, Fig 24-06C
2OM-24.4.AAK, Rev. 14, STM GEN 21A LEVEL High/Low
2OM-24.4.N, Rev. 12, FEEDWATER SYSTEM OPERATION AFTER HI-HI SG LEVEL TRIP
2OM-52.4.A.IV.B & C
2OM-52.4.R.1.F & S
2OM-53C.4.2.24.1, Rev. 2, Loss of Main Feedwater
BVBP-OPS-024, Transient Response Guidelines, Rev. 1
NOP-OP-1002, Rev. 4, Conduct of Operations

Other
BVPS-2, 2C14 Core Map, dated May 2, 2008
BVPS-2 Shift Operating Logs, dated May 2 – 6; May 16, May 22, May 24, 2008
Evaluation of Fuel Assembly R57, dated May 5, 2008
Evaluation of RCCA and CRDS, dated May 5, 2008
Event Notification #44239, dated May 24, 2008
Event Response Immediate Investigation for CRDS drop, May 2 – 6, 2008
Event Timeline for CR 08-40825, dated May 24, 2008
Field Anomaly Report DM-08-02, dated May 3, 2008
Failure Modes Effects Analysis for Unit 2 Turbine Bearing Failure, dated May 19, 2008
Mode Hold Resolution Form Data for CR 08-39693
Operations Department Stand-down and Refocus on Operator Fundamentals, dated May 24, 2008
Unit 2 Plant Computer Data and Traces, dated May 24, 2008
Unit 2 Turbine Monitoring Plan, dated May 21, 2008

Section 4OA5: Other Activities

Calculations
10080-DSC-6762, Sodium Tetraborate Basket Evaluation, Rev. 0

Condition Reports (* denotes NRC identified during this inspection)
08-38875 08-42073*

Procedures
1/2-ADM-1801, Containment Cleaning Program, Rev. 0
1/2-ADM-2060, Containment Coatings Inspection and Assessment Program, Rev. 0

Notifications
600462438 600463700

Work Orders

200252480	200252734	200260979	200260983	200260986	200260988
200276381	200276394	200276395	200276396	200308421	200320538

Miscellaneous
Vendor Technical Information Review Form 2702.250-000-012, Report on Beaver Valley Unit 2 Containment Building Walkdowns for Emergency Sump Strainer Issues, Including Outage 2R13, Rev. B
ECP 06-0227-02, Add RWST Level Interlock to RS Pump Start for BV2
ECP 07-0062, 2R13 Insulation Removal and Replacement
ECP 07-0083, High Head Safety Throttle Valve Modification
ECP 08-0065, Installation of Sodium Tetraborate Baskets
Letter from U.S. NRC to FENOC: Extension Request Approval RE: Generic Letter 2004-002, dated 02/29/2008
Letter from FENOC to U.S. NRC: Generic Letter 2004-002, Request for Extension of Completion Date for Corrective Actions, dated 02/14/2008
Letter from FENOC to U.S. NRC: Generic Letter 2004-002, Request for Extension of Completion Date for Corrective Actions, dated 12/20/2007

Section 4OA5: Other Activities / Unit 2 Extended Power Uprate (IP 71004)

Procedures / Surveillances / Post Maintenance Tests
2BVT 1.6.1, "Reactor Coolant System Total Flow Measurement", Issue 1, Rev. 14, completed June 2 (>95% power) and June 11 (100%), 2008
2-SPT-52-40441-3, "Escalation to EPU Uprate Power (2900 MWt), Rev. 1, completed June 24, 2008
2LCP-5-DIR1C-I, Issue 4, Rev. 2, Analog Computer Point Checks of Data input Rack 1, Chassis- C, dated April 29, 2008
2FWS-F497, Loop 3 Feedwater Flow Channel III Calibration, Issue 4, Rev. 12, dated April 13, 2008
2MSS-P447 First Stage Pressure Protection Channel IV Calibration, Issue 4, Rev. 9, dated April 17, 2008
Unit 2 Uprate: Thermal Expansion and Restraint Walkdowns for 2770MWt, 2835MWt, and 2900MWt Operating Conditions Summary Report, dated June 24, 2008

Engineering Changes
ECP 02-0190-02: Replacement Pressure Transmitter and Indicator for 1st Stage HP Pressure
ECP 04-0440-02: Phase 2 & 3 (+5%) Power Uprate Ascension Testing – Unit 1 & Unit 2
ECP 05-0013-01: 2900 MWt Power Uprate implementation – NSSS Instrumentation Rescaling Changes for Unit 2
ECP 05-0013-ID-02: Secondary Power Calorimetric Algorithm and OTDT/OPDT Changes for Unit 2.

Miscellaneous
1/2-ADM-1359.F06, “Unit 2 Simulator to Plant Comparison at 103% Uprate”, Rec 3 with SDR No. 6259, 6277, 6278 on June 15, 2007
BV2 Reactivity Plan – Power Change Calculation for RTP Power Ascension dated, June 3 & June 5, 2008
BV-PORC-08, Review of BVPS-2 EPU Uprate 97.75% (2835MWt) Test Results, dated June 4, 2008

Work Orders
WO 200227803: Master work order to ascend Unit 2 from 2770 MWt to 2900 MWt
WO 200252810: Delta-T Tavg Summer
WO 200320187: RTD Amplifier Loop 2 T-HOT 1
WO 200252809: Steam Dump Temp Summer (and other BOP rescaling)

Condition Reports

08-41168	08-41153	08-41596	08-41453	08-41356	08-40438
08-38963	08-39761	08-39653	08-34102	08-34780	08-34925
07-29506	07-20695	06-11626			

Inspection Procedure	Title	Inspection Report	Description and 71004 Section
71004	Power Uprate	08-03	BV2 Simulator to Plant Comparison at 108% Uprate (2900MWt) (2.02.d&e)
		08-03	BV2 EPU Post-100% +100hour (6/10) plant data. (2.02.d&e)
		08-03	BV2 EPU 2835MWt (6/3) and 2900MWt (6/5) heat balance data (2.02.d&e)
		08-03	BV2 Main Generator parameter trend data at 2900MWt (6/5) (2.02.d&e)
		08-03	BV2 Process Temperature operational data at 2900MWt (6/5) (2.02.d&e)
		08-03	BV2 RCS Total Flow Measurement data at 2900MWt (6/11) (2.02.d&e)
		08-03	BV2 PORC review of interim EPU power level (2835MWt) test data (6/4) (2.02.g)
		08-03	BV2 RCS Chemisty Data at 2900MWt (6/6) (2.02.e)
		08-03	BV2 Health Physics Radiation Surveys outside containment at 2900MWt (5/29-6/15) (2.02.e)
		08-03	BV2 Thermal Expansion and Restraint Walkdown data (6/3, 6/5 & 6/10) (2.02.e)
71111.11, 71111.20		08-03	BV2 Operator Startup Just-in-time training (5/6 & 5/23) (2.02.d)
71111.17, 71111.20 71004		08-03	BV2 High Pressure Turbine Replacement (2R13) (2.02.b)
71111.17, 71111.20, 71004		08-03	BV2 ‘A’ & ‘C’ Main Feedwater Regulating Valve & Actuator replacement (2R13) (2.02.b)
71111.17, 71111.20 71004		08-03	Various Instrumentation Rescaling (2R13) (2.02.b)

LIST OF ACRONYMS

ADM	Administrative Procedure
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
BCO	Basis for Continued Operations
BMI	Bare Metal Visual Inspection
BVPS	Beaver Valley Power Station
CFR	Code of Federal Regulations
CR	Condition Report(s)
CRDM	Control Rod Drive Mechanism
DMBW	Dissimilar Metal Butt Welds
DOT	Department of Transportation
ECT	Eddy Current Testing
FENOC	First Energy Nuclear Operating Company
FSWOL	Full Structural Weld Overlay
GL	Generic Letter
GSI	Generic Safety Issue
HRA	High Radiation Area
ID	Inside Diameter
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
LCO	Limiting Conditions for Operations
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MRP	Materials Reliability Program
MT	Magnetic Particle Testing
MR	Maintenance Rule
NDE	Non Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OD	Outside Diameter
ODSCC	Outside Diameter Stress Corrosion Cracking
OST	Operations Surveillance Test
PCE	Personnel Contamination Event Report
PCP	Process Control Program
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Testing
PORV	Pilot Operated Relief Valve
PT	Dye Penetrant Testing
PWR	Pressurized-water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QS	Quench Spray
RCA	Radiological Controlled Area

RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSS	Recirculation Spray System
RT	Radiographic Testing
RW	River Water
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SG	Steam Generator
SW	Service Water
TDAFW	Turbine-Driven Auxiliary Feedwater
TI	Temporary Instruction
TMOD	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VHRA	Very High Radiation Area
VT	Visual Testing

ATTACHMENT "B"

TI 2515/172 Reactor Coolant System (RCS) Dissimilar Metal Butt Welds (DMBW) Documentation Questions for Beaver Valley Units 1 and 2

Introduction:

Temporary Instruction, TI 2515/172 provides for confirmation that owners of pressurized-water reactors (PWRs) have implemented the industry guidelines of the Materials Reliability Program (MRP) -139 regarding nondestructive examination and evaluation of certain dissimilar metal welds in the reactor coolant systems containing nickel-based Alloy 600/82/182 materials. MRP-139 categorizes welds according to their mitigation status. The TI requires documentation of specific questions in an inspection report. The questions and responses are included in this Attachment "B".

In summary, the Beaver Valley Units 1 and 2 are Westinghouse three loop plants. Each unit has MRP-139 applicable Alloy 600/82/182 RCS welds. Beaver Valley Unit 1 has fewer MRP-139 applicable Alloy 600/82/182 RCS welds than Unit 2. Unit 1 has one 4" pressurizer spray nozzle-to-safe-end weld, and four 6" pressurizer safety and relief nozzle-to-safe end welds, which are MRP-139 applicable Alloy 600/82/182 that were previously categorized as H-welds based upon un-inspectability due to configuration limitations. All of the Unit 1 pressurizer nozzle-to-safe end welds were preemptively mitigated via full structural weld overlay (FSWOL) during the Fall 2007 1R18 RFO and are now categorized as F-welds following FSWOL based upon no pre-overlay inspection and are assumed cracked. FENOC Letters L-07-039 & L-07-073, "Beaver Valley Power Station, Unit 1, Proposed Alternative to American Society of Mechanical Engineers Code Section XI Repair Requirements (Request No. BV1-PZR-01)," both dated March 31, 2007 were submitted for the Unit 1 pressurizer nozzle weld overlays. Following application of the FSWOL, ultrasonic examinations performed by Performance Demonstration Initiative (PDI) qualified technicians did not identify any recordable indications in the outer 25% of dissimilar metal welds.

Unit 2 has three 29" reactor vessel outlet hot leg nozzle-to-safe end welds (2RCS*REV21-N-24, N-26, N-28) and three 27.5" reactor vessel inlet cold leg nozzle-to-safe end welds (2RCS*REV21-N-23, N-25, N-27) DMBW connections which were examined from the inside volumetrically by automated ultrasonic testing and on the ID surface by eddy current in the Spring 2008 2R13 refueling outage (RFO). No indication of cracking was found on any of these welds. Unit 2 also has one 14" pressurizer surge line nozzle-to-safe end weld, one 4" pressurizer spray nozzle-to-safe end weld, and four 6" pressurizer safety and relief nozzle-to-safe end welds, which are MRP-139 applicable Alloy 600/82/182 that were previously categorized as Primary Water Stress Corrosion Cracking (PWSCC) H-welds based upon un-inspectability due to configuration limitations. All of the Unit 2 pressurizer nozzle-to-safe end welds were preemptively mitigated via FSWOL during the Fall 2006 2R12 RFO and are now categorized as F-welds following FSWOL based upon no pre-overlay inspection and are assumed to be cracked. FENOC Letter L-06-038, "Beaver Valley Power Station, Unit 2, Proposed Alternative to American Society of Mechanical Engineers Code Section XI Repair Requirements (Request No. BV2-PZR-01)," dated March 31, 2006 was submitted for the Unit 2 pressurizer nozzle weld overlays. Following application of the FSWOL, ultrasonic examinations performed by PDI qualified technicians did not identify any recordable indications in the outer 25% of these dissimilar metal welds.

a. For MRP-139 baseline inspections:

Qa1. Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance?

A. Yes. For Unit 1, the pressurizer spray nozzle weld, three pressurizer safety nozzle welds and the relief nozzle weld were preemptively mitigated via FSWOL during the Fall 2007 1R18 RFO. For Unit 2, ultrasonic volumetric examinations were done from the inside weld diameter and eddy current (ET) examinations were done of the inside weld surface area on the three hot leg and three cold leg piping to vessel nozzle-to-safe end welds during the Spring 2008 2R13 refueling outage (RFO). The Unit 2 MRP-139 applicable Alloy 600/82/182 pressurizer surge line, spray line, and three safety and relief nozzle welds were preemptively mitigated via full structural weld overlay (FSWOL) during the Fall 2006 2R12 RFO. No pre-overlay inspections were performed prior to the weld overlays on either unit. Following application of the FSWOL, ultrasonic examinations were performed by PDI qualified personnel on each unit.

Qa2. Is the licensee planning to take any deviations from the MRP-139 baseline inspection requirements of MRP-139? If so, what deviations are planned and what is the general basis for the deviation? If inspectors determine that a licensee is planning to deviate from any MRP-139 baseline inspection requirements, NRR should be informed by email as soon as possible.

A. Yes, in lieu of baseline UT inspections of the Unit 1 and Unit 2 pressurizer MRP-139 applicable Alloy 600/82/182 welds which were considered uninspectable due to configuration limitations, FENOC elected to preemptively mitigate these pressurizer nozzle welds using FSWOL during 1R18 RFO and 2R12 RFO.

b. For each examination inspected, was the activity:

Qb1. Performed in accordance with the examination guidelines in MRP-139 Section 5.1 for unmitigated welds or mechanical stress improved welds and consistent with NRC staff relief request authorization for weld overlaid welds?

A. Yes. For Unit 1 the inspector directly observed several of the weld overlays and UT examinations performed on the pressurizer dissimilar metal weld overlays during the Fall 2007 1R18 RFO, which is documented in NRC inspection report 05000334/2007005. For Unit 2 the inspector reviewed the UT examination records of the three hot leg and three cold leg piping to vessel nozzle-to-safe-end welds that were examined during the Spring 2008 2R13 RFO. The inspector also directly observed manual UT of the Unit 2 pressurizer surge nozzle weld overlay during 2R12 RFO, which is documented in NRC inspection report 05000412/2006005.

Qb2. Performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

A. Yes. The certifications of the non-destructive examination (NDE) technicians performing the manual, phased array PDI UT examinations of the Unit 1 pressurizer weld overlays were reviewed by the inspector and the technicians were PDI qualified.

Qb3. Performed such that deficiencies were identified, dispositioned, and resolved?

A. No material deficiencies were identified during these NDE examinations.

<u>c. For each weld overlay inspected, was the activity:</u>

Qc1. Performed in accordance with ASME Code welding requirements and consistent with NRC staff relief requests authorizations? Has the licensee submitted a relief request and obtained NRR staff authorization to install the weld overlays?

> A. For Unit 1 the inspector directly observed several of the weld overlays applied to the pressurizer dissimilar metal welds during the Fall 2007 1R18 RFO. The weld overlays were performed in accordance with FENOC Letters L-07-039 & L-07-073, "Beaver Valley Power Station, Unit 1, Proposed Alternative to American Society of Mechanical Engineers Code Section XI Repair Requirements (Request No. BV1-PZR-01). However, during installation of Unit 1 weld overlays on the three pressurizer safety valve nozzle-to-safe end welds, it was identified that the weld overlay procedure did not meet ASME Code welding requirements which is documented in NRC inspection report 05000334/2007005 as an unresolved item (URI 05000334/2007005-02, Procedure used for Weld Overlays on Pressurizer Safety Nozzles was not Initially Qualified for P-1 Material), which was subsequently closed in this report.

Qc2. Performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

> A. Yes. Welder performance paperwork qualification records were reviewed to verify that the welders performing the Unit 1 pressurizer weld overlays during 1R18 RFO were qualified to weld under the qualification ranges listed on each specific welder performance record which was summarized in a Table, "1R18 Pressurizer Weld Overlay PCI Welder Performance," Revision 1. Manual-driven, encoded phased array PDI-UT examinations were performed on the Unit 1 pressurizer nozzle weld overlays. The NDE examiners performing the post weld overlay UT examinations consisted of one Level II and two Level III examiners who were PDI qualified technicians.

Qc3. Performed such that deficiencies were identified, dispositioned, and resolved?

> A. Deficiencies were being properly identified, dispositioned, and resolved during the Unit 1 and Unit 2 pressurizer weld overlays as demonstrated by the above example of the welding procedure deficiency that was found during the installation of weld overlays on the three Unit 1 pressurizer safety valve nozzle-to-safe end welds.

<u>d. For each mechanical stress improvement used by the licensee during the outage, was the activity performed in accordance with a documented qualification report for stress improvement processes and in accordance with demonstrated procedures? Specifically:</u>

Qd1. Are the nozzle, weld, safe end, and pipe configurations, as applicable, consistent with the configuration addressed in the SI qualification report?

> A. N/A, mechanical stress improvement was not used on either Unit.

Qd2. Does the SI qualification report address the location radial loading is applied, the applied load, and the effect that plastic deformation of the pipe configuration may have on the ability to conduct volumetric examinations?

> A. N/A

Qd3. Do the licensee's inspection procedure records document that a volumetric examination per the ASME Code, Section XI, Appendix VIII was performed prior to and after the application of the SI?

A. N/A

Qd4. Does the SI qualification report address limiting flaw sizes that may be found during pre-SI and post-SI inspections and that any flaws identified during the volumetric examination are to be within the limiting flaw sizes established by the SI qualification report.

A. N/A

Qd5. Performed such that deficiencies were identified, dispositioned, and resolved?

A. N/A

<u>e. For the inservice inspection program</u>:

Qe1. Has the licensee prepared an MRP-139 inservice inspection (ISI) program? If not, briefly summarize the licensee's basis for not having a documented program and when the licensee plans to complete preparation of the program.

A. Yes. FENOC has an MRP-139 ISI program for Unit 1 and Unit 2, which is separate from the ASME Section XI ISI program/Risk-Informed ISI programs. However, even though the MRP-139 ISI program is separate, the welds in the MRP-139 program are included in the Unit 1 and Unit 2 ASME Section XI ISI program/Risk-Informed ISI program.

Qe2. In the MRP-139 ISI program, are the welds appropriately categorized in accordance with MRP-139? If any welds are not appropriately categorized, briefly explain the discrepancies.

A. Yes, the DM welds in each Unit are appropriately categorized in accordance with MRP-139.

Qe3. In the MRP-139 ISI program, are the ISI frequencies, which may differ between the first and second 10-year intervals after the MRP-139 baseline inspection, consistent with the ISI frequencies called for by MRP-139?

A. Yes, the ISI frequencies are consistent with the MRP-139 frequencies.

Qe4. If any welds are categorized as H or I, briefly explain the licensee's basis for the categorization and the licensee's plans for addressing potential PWSCC.

A. N/A, currently there are no welds that are categorized as H or I at Beaver Valley Units 1 or 2. However, previously before the pressurizer weld overlays, the pressurizer nozzle welds on each Unit were categorized as H-welds based upon uninspectability due to configuration limitations.

Qe5. If the licensee is planning to take deviations from the ISI "requirements" of MRP-139, what are the deviations and what are the general bases for the deviations? Was the NEI 03-08 process for filing deviations followed?

A. No. Additional ISI deviations are planned for any ISI of the welds to MRP-139. Proposed Alternative to American Society of Mechanical Engineers Code Section XI Repair Requirements (Request No. BV1-PZR-01 and BV2-PZR-01) provided the needed deviations to the ISI requirements for the pressurizer weld overlays on each Beaver Valley Unit.