May 9, 2006

Mr. James Lash
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 05000334/2006002 AND 05000412/2006002

Dear Mr. Lash:

On March 31, 2006, 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 April 27, 2006, 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, the NRC has identified two (2) NRC-identified findings, and one (1) self-revealing finding, of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they are entered in the corrective action program, the NRC is treating three of 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 7
Division of Reactor Projects

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

Enclosures: Inspection Report 05000334/2006002; 05000412/2006002
w/Attachment: Supplemental Information

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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.	05000334/2006002 and 05000412/2006002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	January 1, 2006 through March 31, 2006
Inspectors:	P. Cataldo, Senior Resident Inspector T. Moslak, Health Physicist D. Silk, Senior Emergency Preparedness Inspector G. Smith, Resident Inspector
Approved by:	R. Bellamy, Ph.D., Chief Reactor Projects Branch 7 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2006002, IR 05000412/2006002; 01/01/2006 - 03/31/2006; Beaver Valley Power Station, Units 1 & 2; Flood Protection Measures; Operability Determinations; Refueling and Outage Activities.

The report covered a 3-month period of inspection by resident inspectors, regional reactor inspectors, and a regional health physics inspector. Four Green findings 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: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation for failure to include seismic, safety-related valve pits for Unit 1 in the structural monitoring program of the maintenance rule as required by 10 CFR 50.65 (b). FENOC’s failure to monitor valve pit structures could have led to the failure to identify rain water, groundwater or piping leaks, as well as pipe and valve support degradation, potentially rendering the river water cross-connect valves unable to perform their required safety function. This finding was entered into the corrective action program for resolution. The licensee has inspected one of two valve pits, has scheduled the inspection of the other valve pit, and will be adding these structures into the appropriate plant procedures and processes to ensure the requisite inspections are performed.

  This finding was considered more than minor, because it was associated with the equipment performance attribute of the Mitigating System Cornerstone, and affected the availability and reliability of mitigating equipment. This finding was of very low safety significance since there never was a loss of function of the equipment in these structures. (Section 1R06)

- Green. The inspectors identified a self-revealing Non-Cited Violation against 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for inadequate corrective actions to resolve main steam safety valve (MSSV) component deficiencies in Unit 1. Specifically, the failure to internalize several years of industry operating experience impacted the initial lift setpoints of all main steam safety valves on the “C” main steam header, and would have led to higher lifting pressures for potentially the entire operating cycle. This finding was entered into the corrective action program for resolution. Subsequently, the licensee performed a root cause evaluation, replaced all five “C” main steam header MSSVs with improved materials less susceptible to the failure mechanisms encountered, and will perform a mid-cycle lift test as a proof test of the new materials.

The inspectors determined this finding is more than minor because it impacted the reliability and function of mitigating equipment important to safety. The inspectors determined that this finding is of very low safety significance, because there was no overall loss of function due to the redundant safety and atmospheric relief valves that remained capable of performing the necessary design basis function. A contributing cause to this finding is related to the identification subcategory of the problem identification and resolution cross-cutting area. Specifically, the failure to internalize several years of industry operating experience resulted in the oxidation condition that impacted the initial lift setpoints of all MSSVs on the “C” main steam header for potentially the entire operating cycle. (Section 1R15)

- Green. The inspectors identified a finding which involved the failure to adequately plan for entry into a reduced inventory condition during the Unit 1 refueling outage. This resulted in an increased exposure to a reduced “time to boil”. Controls were not in place to ensure that post drain-down required equipment was properly staged. Specifically, the reactor coolant system (RCS) drain-down was prematurely secured when it was discovered that the stud de-tensioners were not staged in containment to begin entry into reactor operating mode 6. Stud de-tensioner movement into the containment had been halted during the drain-down due to a suspension of crane operations as a result of high winds. The licensee entered this deficiency into their corrective action program for resolution. In addition, a trend review condition report was initiated to evaluate the shutdown risk impacts that resulted from this and other issues that arose during the outage.

  This finding is greater than minor because the licensee’s risk assessment failed to consider unusual external conditions that were present or imminent. This finding was determined to be a finding of very low safety significance because the event did not involve a loss of control or a reduction in mitigation capability. The cause of this finding is related to the cross-cutting element of human performance. (Section 1R20)

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 began the inspection period at 100 percent power. On February 13, 2006, the unit was taken off-line for a planned refueling outage (Section 1R20). The unit was defueled on February 22 with all fuel assemblies transferred to the spent fuel pool. Following replacement of all three steam generators and the reactor vessel head, fuel assemblies were reloaded into the reactor vessel on March 26. The unit remained in reactor operating mode 6 for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power. On March 9, the unit down-powered to 98 percent in order to remove impulse pressure from the turbine control scheme. During this transfer, the turbine control unexpectedly swapped to manual. Following troubleshooting and cause determination of the anomaly, the unit returned to full power operation on March 10 using turbine manual control (Section 1R14). The unit continued to operate at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors evaluated the adequacy of continued availability of the ultimate heat sink during cold weather conditions. This evaluation focused on the intake structure and considered the potential effects of ice and blockage concerns. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), and industry operating experience to verify that the intake structure would remain functional when challenged by extreme cold weather. The inspectors also performed a detailed walkdown of the intake structure to verify the adequacy of various weather protection features.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns (4 samples).

a. Inspection Scope

The inspectors performed four partial system walkdowns during this inspection period. The inspectors evaluated the operability of the selected train or system when the redundant train or system was inoperable or unavailable, or in abnormal alignments in support of outage activities. The inspectors verified correct valve positions and breaker alignments in accordance with the applicable procedures, and consistent with applicable chapters of the UFSAR.

- On January 30, 2006, the inspectors performed a walkdown of the Unit 1 ‘A’ train Low Head Safety Injection (LHSI) System while the ‘B’ train was out-of-service for planned maintenance.

- On February 28, during a refueling outage, the inspectors performed a walkdown of the the Unit 1 River Water (RW) system while the system was aligned to the ‘B’ supply header due to maintenance on a valve in the normal RW discharge line. This inspection was selected due to the abnormal alignment that directed the RW discharge through the ‘A’ header discharge line to the alternate intake structure.

- On March 6, the inspectors performed a walkdown of the Unit 2 ‘B’ train of Service Air (SA), while the ‘A’ train SA compressor was out-of-service for preventive maintenance.

- On March 21, the inspectors performed a walkdown of the Unit 2 ‘A’ train of LHSI, while the ‘B’ LHSI pump was out-of-service for preventive maintenance.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown. (71111.04S - 1 sample)

The inspectors conducted a detailed review of the alignment and condition of the Unit 1 Auxiliary Feedwater Water (AFW) System. This system was selected based on its risk significance and its associated frequency of inspector review. The inspectors reviewed plant drawings, abnormal operating procedures, and emergency operating procedures to determine proper equipment alignment. Condition reports associated with the AFW system were reviewed to verify that the licensee was adequately identifying and correcting system deficiencies; the inspectors also evaluated existing deficiencies to determine the impact, if any, on the AFW system operation. Additionally, the inspectors performed a detailed review of the AFW system health report and the design basis document to gain insights on any longstanding issues.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 11 samples)

a. Inspection Scope

Fire Area Walkdowns. The inspectors reviewed the Unit 2 Fire Protection Safe Shutdown Report, Addendum 18, and selected the following eleven risk significant areas for inspection:

- Alternate Intake Structure (Fire Area AIS-1)
- Cooling Tower Pump House (Fire Area CTP-1)
- Waste Handling Building (Fire Area WH-1)
- Waste Handling Building (Fire Area WH-2)
- Service Water Valve Pit (Orange) (Fire Area VP-1)
- Service Water Valve Pit (Purple) (Fire Area VP-2)
- Main Transformer (Fire Area TR-1)
- Unit Station Service Transformer TR-2C (Fire Area TR-2)
- Unit Station Service Transformer TR-2D (Fire Area TR-3)
- System Station Service Transformer TR-2B (Fire Area TR-4)
- System Station Service Transformer TR-2A (Fire Area TR-5)

The inspectors reviewed the conditions of the fire areas listed above to verify compliance with criteria delineated in Administrative Procedure 1/2-ADM-1900, "Fire Protection," Rev. 8. This review, for example, included FENOC's control of transient combustibles, material condition of fire protection equipment, and the adequacy of compensatory measures for any fire protection impairments. Additionally, where applicable, the inspectors reviewed completed surveillance tests conducted on various fire protection equipment to verify that appropriate acceptance criteria were met, and that adverse issues were entered into the corrective action program for resolution. See the Attachment for a list of associated documents that were reviewed during the inspection.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the licensee's flood protection measures with respect to external flooding. This review included flood control measures associated with valve pits (manhole) that contain Unit 1 River Water and Auxiliary River Water system components. This review utilized design basis information contained in the FSAR, Design Basis Documents, and applicable surveillance records.

b. Findings

Introduction. A Green, NRC-identified Non-Cited Violation was identified for failure to include seismic, and safety-related valve pits that contained equipment important to safety, into the structural monitoring program under the maintenance rule.

Description. During the Unit 1 refueling outage, work activities were performed in a valve pit that contains a cross-connect valve associated with the river water and auxiliary river water systems. The inspectors noted some standing water in the valve pit during the inspection, but concluded such small amounts would have no immediate effect on any equipment in the valve pit. Nonetheless, the inspector determined that the licensee was not performing structural and water intrusion inspections for this seismic, safety-related structure. Moreover, the inspector noted that similar structures (seismic, safety-related valve pits) exist at Unit 2, but unlike the Unit 1 structures, these were appropriately included in the structural monitoring program under the maintenance rule. Additionally, the Unit 2 structures were periodically inspected to ensure safety-related cables were not potentially damaged due to submergence from rainwater or groundwater in-leakage.

As a result, the licensee has completed the inspection of one of the two valve pits that contains MOV-1RW-116B (“B” train cross-connect valve), and no significant deficiencies were identified. Additionally, the licensee has scheduled an inspection of the remaining valve pit that contains the “A” train cross-connect valve., and has generated CR 06-01604, to capture this issue in the corrective action program for resolution.

Analysis. The failure to include a seismic, safety-related structure in the structural monitoring program of the maintenance rule, was determined to be a performance deficiency. The finding was more than minor, because it was associated with the equipment performance attribute of the Mitigating System Cornerstone, and affected the availability and reliability of mitigating equipment. FENOC’s failure to monitor valve pit structures could have led to the failure to identify rain water, groundwater or piping leaks, as well as pipe and valve support degradation, potentially rendering the river water cross-connect valves unable to perform their required safety function. This issue was a finding under the Mitigating Systems Cornerstone of the Reactor Oversight Process. The significance of this finding was evaluated using Appendix A, “At-Power Situations,” of the NRCs Significance Determination Process (Manual Chapter 0609). The inspectors determined that this finding was of very low safety significance (Green), because there was no overall loss of function due to the redundant cross-connect valves, there was no degradation of significance when the “B” train valve pit was inspected, and no major deficiencies were noted from past remote valve position indication test results.

Enforcement. 10 CFR 50.65(b) and (b)(1), states in part, that the scope of the maintenance rule shall include safety-related structures that are relied upon to remain functional during design basis events. Contrary to this, FENOC failed to incorporate two seismic, safety-related valve pits with important river water piping and valves, into the structural monitoring program of the maintenance rule. Because this deficiency was considered of very low safety significance, and was entered into the corrective action program for resolution as CR 06-01604, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000334/2006002-01, Failure to scope a seismic, safety-related structure into the maintenance rule structural monitoring program.**

1R11 Licensed Operator Requalification Program (71111.11 - 2 samples)

a. Inspection Scope

- The inspectors observed Unit 1 licensed operator requalification training conducted in the plant-reference simulator on January 23, 2006. The inspectors evaluated licensed operator performance regarding: effective communications, implementation of normal, abnormal and emergency operating procedures, command and control, technical specification compliance, and emergency plan implementation. The inspectors evaluated simulator fidelity to ensure major plant configurations or changes were captured in the simulator. Inspectors evaluated the staff evaluators during the training to verify identified deficiencies in operator performance were properly identified, and that any conditions adverse to quality were appropriately entered into the corrective action program for resolution.

- The inspectors reviewed the Unit 1 lesson plan for the Module #2 licensed operator requalification program, which involved emergency and abnormal operating procedure changes associated with the atmospheric containment conversion and replacement steam generator projects. The inspectors also reviewed the revised operator action times as well as revised setpoints instituted as a result of these two projects.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12 - 2 samples)

a. Inspection Scope

The inspectors evaluated follow-up actions for problems with selected structures, systems, and components (SSCs), and reviewed the performance history of these SSCs to assess the effectiveness of Beaver Valley's maintenance activities. The inspectors reviewed problem identification and resolution actions, as applicable, to determine whether plant staff had appropriately monitored, evaluated, and dispositioned the issues in accordance with station procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," and 1/2-ADM-2114, "Maintenance Rule Program Administration," Revision 0. In addition, the inspectors reviewed selected SSC classifications, scoped functions, and performance criteria and goals, as detailed in the System Basis Documents. The following conditions were evaluated:

- Unit 2 Diesel Driven Air Compressor Failed to Maintain Pressure (CR 05-08106);

- Recurrent failure to auto-stop of the Unit 2 chemical and volume control system's blender/totalizer, as detailed in CRs 05-08056, and 06-00010.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 7 samples)

a. Inspection Scope

The inspectors reviewed the scheduling and control of seven activities, evaluated the effect on overall plant risk, and evaluated the prescribed risk management actions, as applicable. This review was conducted using the criteria contained in 10CFR50.65(a)(4); 1/2-ADM-2033, "Risk Management Program," Rev. 3; NOP-WM-2001, "Work Management Process," Rev. 4; 1/2-ADM-0804, "On-Line Work Management and Risk Assessment," Rev. 4; 1/2-ADM-2114, "Maintenance Rule Program Administrative Procedure," Rev. 2; and Conduct of Operations Procedure 1/2OM-48.1.I, "Technical Specification Compliance," Rev. 18. The inspectors reviewed the following planned or emergent work activities:

- Unit 1 emergent "yellow" risk on January 7, 2006, due to the failure of a power supply module associated with the "A" train of solid state protection system.
- Unit 1 planned "yellow" risk on February 6, 2006, due to the performance of 1OST-33.10G, "System Station Service Transformer 1A Deluge Valve Test."
- Unit 2 emergent failure on March 11, 2006, of 2CNM-RV151, "Startup Feedwater Pump Suction Relief Valve," which rendered the pump inoperable.
- Unit 2 planned "yellow" risk on March 9, 2006, due to the performance of 2MSP-1.04-I, "Solid State Protection System Train 'A' Bi-Monthly Test," Rev. 29.
- Unit 2 planned "yellow" risk on March 13, 2006, due to installation of new, Unit 1 current and potential transformers in the switchyard, which disabled the 'B' system station service transformer, and the independent circuit between the offsite transmission network and the onsite class 1E distribution system.
- Unit 1 planned risk activity on March 15, 2006, associated with the disabling of the emergency switchgear ventilation, to perform various preventive maintenance activities on system ventilation dampers.
- Unit 1 planned activities, missile shield and concrete removal beginning in reactor operating mode 5.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14 - 2 samples)

a. Inspection Scope

The inspectors reviewed human performance during non-routine plant evolutions, to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety. The inspectors evaluated whether the evolution was properly implemented according to the applicable procedures and Technical Specification (TS) limiting condition for operations (LCOs).

- The inspectors evaluated the control room operator's performance during a power ascension while the main turbine electro-hydraulic control system was in manual in accordance with 2OM-26.4.C, "EH Control System Operation," Rev. 2.

- The inspectors evaluated control room operator's performance during operation of the chemical and volume control system's blender/totalizer while the system was in a degraded condition due to an auto-stop failure (See Section 1R12). Additionally, the inspector reviewed operating procedure 2OM-7.4.AR, "Blender Operation In Mode 1," Rev. 2, and verified that appropriate sensitivity regarding reactivity transients had been communicated to the control room operators.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed five conditions to determine whether proper operability determinations (OD), Basis For Continued Operations (BCO), or other assessments were performed. In addition, where applicable, the inspectors verified that TS and design basis requirements were properly addressed.

- The inspectors reviewed BCO 1-5-3, "1SI-P-1A Low Head Safety Injection (LHSI) Pump Seal Leak." This BCO evaluated a degraded condition on the Unit 1 'A' LHSI pump, which involved leakage of either one of the two tandem pump seals or the accumulator diaphram. The BCO concluded that the 'A' LHSI pump would continue to perform its intended safety function for a period of thirty days following a Design Basis Accident (DBA).

- The inspectors reviewed an OD documented in CR 06-02130, associated with a failed surveillance of the Unit1 'A' Quench Spray (QS) pump. The evaluation concluded that the QS pump was operable and the reduced hydraulic performance that contributed to the failed surveillance was associated with lower than normal refueling water storage tank level.

- The inspectors reviewed BCO 1-05-002, "SBLOCA Modeling Assumptions Impact on 10CFR50.46," which was initiated from a condition identified in 2005 under CR 05-05215, and subsequently terminated on February 9, 2006. This BCO evaluated the potential impact on peak cladding temperature and compliance with 10 CFR 50.46, due to the small break loss of coolant accident calculation methodology that did not include non-integer pipe break sizes.

- The inspectors reviewed an OD documented in CR 06-01654, associated with cracking that was identified on nylon sleeves of slave relay latch assemblies from the Unit 1 solid state protection system. The inspector reviewed extent of condition evaluations for Unit

2 conducted under CR 06-01721, as well as reports from Westinghouse (AR Relay Latch Attachment Fractures Report), and Beta Laboratories (Failure Analysis Report, ARLA Relay Latch Assembly).

- The inspectors reviewed the OD resulting from main steam safety valve (MSSV) testing conducted on February 12, 2006, for Unit 1, because all five MSSVs located on the "C" main steam header failed to lift at the specified pressure limits. The inspectors evaluated the results as documented in 1BVT-1.21.2, "Trevitest Method for Main Steam Safety Valve Setpoint Check," and detailed in CR 06-00794.

b. Findings

a. Introduction. A Green, self-revealing NCV was identified for inadequate corrective actions to resolve main steam safety valve (MSSV) component deficiencies. These deficiencies resulted in the failure of all five "C" main steam header MSSVs to lift at pressure limits during testing.

b. Description. On February 12, 2006, during lift-testing of Unit 1 MSSVs, all five safety valves located on the "C" main steam header exceeded the applicable TS-required lift pressures. Since the MSSVs also exceeded in-service testing acceptance criteria, the licensee expanded the testing to include the MSSVs located on the remaining main steam headers, however, these remaining MSSVs successfully passed the required testing. The licensee subsequently replaced certain components in the MSSVs that had failed lift testing with materials that had shown generally acceptable performance results in meeting testing requirements.

The cause of the initial as-found lift test failure was attributed to bonding of the valve disc and seat caused by the buildup of an oxide (corrosion) layer. Industry research efforts, as well as similar failures associated with the Dresser 3700 Series steam safety valves, which occurred at Davis-Besse in 1998 and 2000, could have been utilized to raise awareness and address the material deficiencies that existed at Beaver Valley. The corrective action recommended and implemented at Beaver Valley was to replace the nozzles (seats) and discs with chemically-treated (pre-oxidized) material that is less susceptible to the bonding, and included the use of Alloy X-750 for the valve seat.

Analysis. FENOC failed to identify and correct in a timely manner material deficiencies that prevented MSSVs from lifting at the appropriate lift setpoints, thereby impacting the reliability and function of mitigating equipment important to safety. The failure to consider industry operating experience to correct degraded components was determined to be a performance deficiency. As a result, the inspectors determined this issue was a finding under the Mitigating Systems Cornerstone of the Reactor Oversight Process. The significance of this finding was evaluated using Appendix A, "At-Power Situations," of the NRC's Significance Determination Process (Manual Chapter 0609). The inspectors determined that this finding was of very low safety significance (Green), because there was no overall loss of function due to the redundant safety and atmospheric relief valves that remained capable of performing the necessary design basis function. A contributing cause to this finding is related to the identification subcategory of the problem identification and resolution cross-cutting area. Specifically, the failure to internalize several years of industry operating experience resulted in the oxidation condition that impacted the initial lift setpoints of all main steam safety valves on the "C" main steam header for potentially the entire operating cycle.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, requires in part, that conditions adverse to quality are promptly identified and corrected. Contrary to this, FENOC failed to incorporate several years of available industry information to effect repairs or replacement of critical components of equipment important to safety, the main steam safety valves. These safety valves were potentially unable to lift at the prescribed setpoint for the entire cycle. Because this deficiency was considered of very low safety significance, and was entered into the corrective action program for resolution as CR-06-00794, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000334/2006002-02, Inadequate corrective actions to resolve main steam safety valve (MSSV) component deficiencies that were the subject of industry operating experience.**

1R19 Post-Maintenance Testing (71111.19 - 4 samples)

a. Inspection Scope

The inspectors reviewed and/or observed four post-maintenance tests (PMTs) to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component; and 3) the PMT was performed in accordance with applicable procedures. The following PMTs were observed:

- 2OST-15.1,"Primary Component Cooling Water Pump [2CCP*P21A] Test" Rev. 37, performed on December 22, 2005 as a retest following planned maintenance under Work Order (WO) 200127687.

- 2OST-36.2, "Emergency Diesel Generator [2EGS*EG2-2] Monthly Test," Rev. 48, performed on January 4, 2006, following speed control adjustment on the #2 Emergency Diesel Generator, conducted under WO 200165505.

- 2LCP-7-F168-I, "Boric Acid Blend Control Total Make-up Flow Loop 2CHS-F168 Calibration," Rev 2, performed on March 01, 2006, for the primary water totalizer on the blending station. The totalizer had been previously replaced following several failures.

- 2OST-24.4, "Steam Driven Auxiliary Feed Pump [2FWE*P22] Quarterly Test," Rev.56, which included the post-maintenance stroke test following steam admission valve 2MSS-SOV105B solenoid coil replacement, completed under WO 200173319.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 1 partial sample)

a. Inspection Scope

The inspectors observed selected Unit 1 outage activities to determine whether shutdown safety functions (e.g. reactor decay heat removal, spent fool pool cooling, and containment integrity) were properly maintained as required by TS and plant procedures. The inspectors evaluated specific activities and performance attributes, including operator performance, outage management daily meetings, clearance activities and configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors also evaluated the following activities:

- Reactor plant shutdown and cooldown, including evaluation of cooldown rates;
- Pre-Outage Shutdown Safety Review;
- Refueling operations;
- Restart readiness meetings and mode hold resolution discussions;
- Clearance execution;
- Draining the RCS to support refueling operations;
- Spent fuel pool cooling system operation;
- Maintenance of boration flowpaths;
- Coordination of electrical bus work and minimization of shutdown risk;
- Reactor coolant system level instrumentation during periods of reduced inventory to verify appropriate configuration;
- Initial mode 4, as-found boric acid walkdown inside containment;
- Containment closure contingencies and procedures;
- 1OST-7.11, "CHS And SIS Operability Test - Train B - PAF 06-00380," Rev. 18;
- Loss of river water event on March 21, 2006.

b. Findings

Introduction. A Green, NRC-identified finding was identified for failure to consider the effects of external conditions (high winds) that were present or imminent prior to, and during RCS drain down activities. As a result, Unit 1 was exposed to a reduced "time to boil" for a longer than expected time frame.

Description. On February 17, 2006, at 0502, the Unit 1 operators began a drain-down of the RCS from a starting point of visible level in the pressurizer, to approximately 6 inches below the reactor vessel flange, to facilitate vessel head removal for refueling activities. Prior to the drain-down, the time to boil was approximately 36 minutes. The typical drain-down sequence drains the RCS from a visible level in the pressurizer to a level just below the reactor flange, followed by the de-tensioning and removal of the reactor closure head studs and nuts using specialty equipment to facilitate the removal of the reactor vessel closure head. The vessel and cavity are then re-flooded so that refueling operations can commence. However, at 0655, the operators secured the drain-down approximately 6 inches from the target level due to a report that the stud de-tensioning equipment had not yet been moved into containment. This equipment had not been moved into containment due to high winds that prevented further crane operation outside of containment. RCS level was stabilized at 100 inches with an associated time to boil of approximately 21 minutes. Following a reduction in winds, crane operation resumed and the stud de-tensioning equipment was transferred into containment. RCS drain down recommenced at 1417, and the RCS level was drained to approximately 6 inches below the reactor vessel flange as targeted. As defined by TS, Mode 6 was entered when the first vessel stud was less than fully tensioned, which occurred at 1449.

Analysis. The inspectors determined that the licensee's failure to account for the effects of external conditions, e.g., high winds, that were predicted (National Weather Service) to occur, and its ultimate impact on crane operations (which ultimately impacted completion of head removal preps prior to draining the reactor vessel) to be a performance deficiency. Specifically, Generic Letter 88-17, "Loss of Decay Heat Removal," describes recommended actions that include "Develop and implement procedures that cover reduced inventory operation and that provide an adequate basis for entry into a reduced inventory condition." The finding was determined to be more than minor because the licensee's risk assessment did not consider the impact of external weather conditions during the drain-down, and thus, failed to consider the potential loss of the outside crane. The inspectors reviewed previous outage records and noted that mode 6 was typically entered within two hours of reaching the targeted RCS level for head removal. Minimizing the time from when drain-down begins to the time the reactor and cavity can be reflooded limits the exposure to a reduced "time to boil" condition and ultimately minimizes shutdown risk.

The inspectors evaluated the finding using Manual Chapter, 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that the finding was not a loss of control as defined by Table 1, and was evaluated using Checklist 3. The inspectors determined that all required procedures, equipment and processes were in place with the exception of item II.B.(2), "Outage schedule delays to the extent practical going to reduced inventory conditions when decay heat load is high." The inspectors determined that the licensee's decision to drain-down in parallel with the movement of the required equipment into containment was ultimately unsuccessful, since the drain-down was not completed until hours later when the equipment was finally transferred inside containment. However, the inspectors determined that a quantitative analysis was not required because the finding 1) did not increase the likelihood of a loss of RCS inventory, 2) did not increase the likelihood of a loss of residual heat removal (RHR) flow, and 3) did not degrade the licensee's ability to recover RHR once it was lost. Therefore, this finding is of very low safety significance (Green). A contributing cause to this finding is related to the organization subcategory of the human performance cross-cutting area. Specifically, the failure to account for high winds, impacted the sequence of activities regarding the RCS drain-down activities and the transition to mode 6, and is evidence of inadequate planning and scheduling related to the transfer of de-tensioning equipment into containment. **FIN 05000334/2006002-03, Failure To Consider External Events During Reactor Coolant System Drain-down Activities.**

Enforcement. No violation of regulatory requirements occurred.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

The inspectors observed and/or reviewed the following seven operational surveillance tests (OSTs). This review included an in-service test and a containment isolation valve surveillance. The inspectors verified that the equipment or systems were capable of performing their intended safety functions and ensured compliance with related TS, UFSAR, and procedural requirements:

• 1OST-15.2, Rev. 16	Reactor Plant Component Cooling Water Pump Operating Surveillance Test
• 1OST-30.12B, Rev. 23	Train ‘B’ Reactor Plant River Water System Full Flow Test
• 1OST-36.4, Rev. 21	Diesel Generator No. 2 Automatic Test
• 2OST-1.12E, Rev. 17	Safeguards Protection System Train B Miscellaneous Go Test
• 1OST-13.7D, Rev. 1	1B Recirculation Spray Pump Auto Start Test
• 1OST-10.4, Rev. 15	Residual Heat Removal System Valve Exercise (IST Surveillance)
• 1OST 1.3, Rev. 14	Containment Isolation Trip Test, CIA Train A and Train B (CIV Test)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed the following temporary modification (TM) based on risk significance. The TM and associated 10CFR50.59 screening was reviewed against the system design basis documentation, including the FSAR and the TS. The inspectors verified the TM was implemented in accordance with Administrative Procedure, 1/2-ADM-2028, “Temporary Modifications,” Rev. 3.

- Unit 1 TM 1-06-002, Rev. 0, “1R17 Temporary Penetrations Modification for Containment Cables.”

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04 - 1 sample)

a. Inspection Scope

The inspector conducted an in-office inspection on February 3 and March 2, 2006, of recent changes to the Beaver Valley emergency plan and implementing procedures. The inspector evaluated whether these changes were made in accordance with 10 CFR 50.54(q), which the licensee had determined did not decrease the effectiveness of the Emergency Plan and concluded that the changes continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR 50. During this inspection, the inspector conducted a sampling review of the changes which could potentially result in a decrease in effectiveness. This review does not constitute an approval of the changes and, as such, the changes are subject to future NRC inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 9 samples)

a. Inspection Scope

During the period February 27 - March 3, 2006, the inspector conducted activities to verify that the licensee was properly implementing physical, administrative, and engineering controls for access to locked high radiation areas and other radiologically controlled areas during the Unit 1 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant TSs, and the licensee's procedures.

Plant Walkdown and RWP Reviews

- During the Unit 1 refueling outage, the inspector identified exposure-significant work activities being conducted in the Reactor Building and Primary Auxiliary Building. Specific work activities included removal of the Old Reactor Vessel Closure Head (ORVCH) and transporting it to the on-site storage facility, severing of the Old Steam Generator (OSG) primary and secondary piping, and various outage support activities.

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, including the Unit 1 reactor building and the Unit 1 primary auxiliary building, fuel handling building, and waste handling building. With the assistance of the licensee Senior Nuclear Specialist - ALARA, the inspector performed independent surveys of selected areas in these buildings to confirm the accuracy of survey maps and the adequacy of postings.

- In evaluating 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 for tasks being conducted under selected RWPs. Work reviewed included:

RWP No. 106-8005	Steam Generator Replacement Project (SGRP) scaffolding erection;
RWP 1016-8014	SGRP OSG and ORVCH transport to the storage facility;
RWP 106-8006	SGRP Piping Severance; and
RWP 106-1009	Construction Activities in High Radiation Areas (HRA).

- The inspector reviewed RWPs and associated instrumentation and engineering controls for potential airborne radioactivity areas located in the reactor building and primary auxiliary building. The inspector confirmed that no worker received an internal dose (in excess of 50 mrem) due to airborne radioactivity when performing outage related tasks. The inspector reviewed Personnel Contamination Event reports and the dose assessment methodology for tasks potentially resulting in internal exposures to confirm the accuracy of the results. Tasks reviewed included OSG piping severance, insulation removal, and decontamination activities.

<u>Problem Identification and Resolution</u>

- The inspector reviewed elements of the licensee’s corrective action program related to controlling access to radiologically controlled areas, to determine if problems were being entered into the program for resolution. Details of this review are contained in Section 4OA2 of this report.

<u>Jobs-In-Progress</u>

- 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 a pre-job briefing for erecting scaffolding in high radiation areas and evaluated the visual aids associated with this briefing.

High Risk Significant - LHRA and VHRA Controls

- Keys to locked high radiation areas (LHRA) and very high radiation areas (VHRA), for Unit 1, were inventoried and accessible LHRAs and VHRAs were verified to be properly secured and posted during Unit 1 plant tours.

- The inspector discussed with radiation protection supervision the adequacy of physical and administrative controls for performing work in high radiation areas, including the movement and storage of in-core detectors and irradiated 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 the removal of the ORVCH from containment, OSG pipe severance, and various radiation protection activities. The inspector determined that the individuals were aware of current radiological conditions, access controls, and that the skill level was sufficient with respect to the potential radiological hazards and the work performed.

- The inspector reviewed condition reports, related to radiation worker and radiation protection technician 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 - 5 Samples)

a. Inspection Scope

During the period February 27 through March 3, 2006, the inspector conducted 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 conducted during the Unit 1 refueling outage, including reactor vessel head and steam generator replacements. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee’s procedures.

Radiological Work Planning

- The inspector reviewed pertinent information regarding site cumulative exposure history for 2005, current exposure trends for 2006, and on-going Unit 1 outage activities to assess current performance and outage exposure challenges. The inspector determined the site's 3-year rolling average collective exposure.

- The inspector reviewed the specialized steam generator replacement project radiation protection action plans (SGRP-RPAP-01 thru 18) that have been implemented for maintaining personnel exposure ALARA, in preparation for removing and replacing the reactor vessel closure head and the steam generators.

- The inspector reviewed the refueling outage work scheduled during the inspection period and the associated work activity dose estimates. Scheduled work included removal of the ORVCH from containment and transport to the on-site storage facility, OSG piping severance, and scaffolding erection/modification in support of OSG removal.

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

- The inspector reviewed the 1R17 outage dose summary reports, detailing worker estimated and actual exposures, through March 2, 2006, for SGRP related activities and balance-of-plant tasks.

- The inspector evaluated the effectiveness of exposure mitigation requirements specified in RWPs and ALARA Plans and compared actual worker cumulative exposure with estimated dose. The inspector reviewed the following work activities whose actual cumulative dose was approaching the estimated dose:

RWP106-8003, ALARA Plan 06-1-25	Insulation removal from steam generators channel heads
RWP 106-8005, ALARA Plan 06-1-27	SGRP scaffolding installation
RWP 106-4003, ALARA Plan 06-1-03	Code safety valve work
RWP 106-4019, ALARA Plan 6-1-11	Reactor disassembly

Additionally, the inspector reviewed the following post-job ALARA Reviews, that identified lessons learned in performing specific tasks:

RWP #106-8016, ALARA Plan #06-1-38	ORCH-Disassembly
RWP #106-4033, ALARA Plan #06-1-54	Air Operated Valve Testing PCV-RC-455A&B

RWP # 106-8006, ALARA Plan #06-1-28   SGRP Piping Severance

- The inspector evaluated the departmental interfaces between radiation protection, engineering, operations, maintenance crafts, and contractors 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-ALARA; reviewing ALARA Committee meeting minutes; reviewing Nuclear Oversight field observation reports; and attending an ALARA Committee meeting.

- The inspector compared the person-hour estimates provided by various departments and contractors with actual work activity time requirements and evaluated the accuracy of these estimates. Specific jobs reviewed included pressurizer inspections, SGRP scaffolding installation, and ORVCH disassembly.

- The inspector verified if work activity planning was effective in further dose control, which included the use of temporary shielding on various components, reactor coolant system flushes, and other operational considerations, e.g., filling steam generators during dose intensive tasks.

- The inspector reviewed personnel contamination event (PCE) reports and dose assessments for selected personnel who were contaminated while performing outage tasks. 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 1 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 on SGRP scaffolding challenges, reviewed recent actions of the committee in monitoring and controlling dose allocations, and interviewed site staff regarding actions to be taken when actual dose approached estimated dose.

- 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 various SGRP activities being performed for ORVCH removal, OSG piping severance, and SGRP scaffolding installation, to verify that radiological controls, such as required surveys, job coverage, pre-job HRA briefings, and contamination controls were implemented; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological controls.

- The inspector reviewed the exposure of individuals in selected work groups, including contractors, radiation protection, and maintenance crafts, to determine if 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 1 source term. Through review of survey maps and interviews with the Senior Nuclear Specialist-ALARA, the inspector evaluated recent source term measurements and control strategies. Specific strategies being employed at Unit 1 included zinc addition, shutdown chemistry controls, system flushes, and temporary shielding.

Declared Pregnant Workers

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

Problem Identification and Resolution

- The inspector reviewed elements of the licensee's corrective action program related to implementing the ALARA program to determine if problems were being entered into the program for timely resolution. Condition Reports related to dose/dose rate alarms, programmatic dose challenges, and effectiveness in predicting and controlling worker dose were reviewed. Details of this review are contained in Section 4OA2 of this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 4 samples)

a. Inspection Scope

Unplanned Scrams and Scrams with Loss of Normal Heat Sink

The inspectors reviewed the Unit 1 and Unit 2 performance indicators for unplanned scrams per 7000 critical hours, to verify that scrams had been properly reported as specified in NEI 99-02, Rev. 1 and Rev. 2. The inspectors verified the accuracy of the reported data through reviews of Licensee Event Reports, monthly operating reports, plant operating logs, and additional records. The inspectors reviewed 2 years of data (January 2003 to December 2005) for unplanned scrams.

Scrams with Loss of Normal Heat Sink

The inspectors reviewed the Unit 1 and Unit 2 performance indicators for scrams with loss of normal heat sink to verify that scrams had been properly reported as specified in NEI 99-02, Rev. 1 and Rev. 2. The inspectors verified the accuracy of the reported data through reviews of Licensee Event Reports, monthly operating reports, plant operating logs, and additional records. The inspectors reviewed 12 quarters of data (January 2003 to December 2005) for scrams with loss of normal heat sink.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Inspection Module Problem Identification and Resolution (PI&R) Review

a. Inspection Scope

The inspectors reviewed various CRs associated with the inspection activities captured in each inspection module of this report. During this review, the inspectors assessed the fundamental ability of the licensee to identify adverse conditions, and verified the licensee had entered these issues into the corrective action program for resolution. Where applicable, CRs reviewed during the inspection are documented under each module, or under Section 40A2; however, for reviews that entailed large number of CRs, these are more appropriately documented in the Attachment.

b. Findings

No findings of significance were identified.

Enclosure

.2 Daily Condition Report Review

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 follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing each condition report, attending daily screening meetings, and accessing the licensee's computerized corrective action program database.

b. Findings

No findings of significance were identified.

.3 Cross-References to PI&R Findings Documented Elsewhere

Section 1R15 describes a finding for failure to incorporate several years of available industry information to correct deficient material associated with the Unit 1 main steam safety valves. Consequently, since these valves were installed with the deficient material during the preceding refueling outage, they were potentially unable to lift at the prescribed setpoint for the entire cycle 17 operating period. A contributing cause to this finding is related to the identification subcategory of the problem identification and resolution cross-cutting area.

.4 Access Controls and ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed 24 Condition Reports, ten Nuclear Oversight Field Observation Reports, and three Radiological Quality Assessor reports to evaluate the threshold for identifying, evaluating, and resolving problems in implementing radiological controls. This review was conducted against the criteria contained in 10 CFR 20, TSs, and the licensee's procedures.

b. Findings

No findings of significance were identified.

.5 Annual Sample Review (71152 - 1 sample)

a. Inspection Scope

The inspectors reviewed licensee corrective actions in response to the potential for fire induced cable damage to result in spurious opening of the power operated relief valves (PORVs) identified in CR-03-08460, "Inadequate Fire Protection Safe Shutdown Analysis of High-Low Pressure Interface." The inspector also reviewed the associated probable cause evaluation to ensure recommended corrective actions were appropriate and consistent with the identified cause, that they were adequate to prevent recurrence, and would reasonably address the underlying factors that led to the issue. The inspector also sampled a number of corrective actions, such as the performance of engineering calculations and deviation request changes, to verify they had been properly performed and implemented.

b. Findings and Observations

No findings of significance were identified.

4OA3 Event Response

(Closed) LER 05000334/2005002-02. Latent Fire Protection Issue Regarding Possible Loss of Reactor Coolant System Makeup Function

During an engineering review conducted in 2004, and ultimately determined in November 2005, the licensee identified a non-compliance with the current licensing basis, in that a potential vulnerability would exist following the loss of all component cooling water pumps due to a fire. This vulnerability was traced back to an incorrect assumption contained in the initial safe shutdown analysis report. Specifically, FENOC failed to ensure that appropriate procedural guidance was provided in post-fire/post-trip procedures to limit elevated seal return water from negatively impacting a running charging pump, and could have led to the inability to achieve and maintain safe shutdown conditions. The licensee performed a root cause evaluation, submitted a licensee event report, and revised appropriate procedures to alleviate the identified safe shutdown concern. This issue was considered more than minor because (1) it was associated with the procedure quality attribute of the mitigating systems cornerstone, and (2) it affected the cornerstone objective because without the procedure changes that were made, safe shutdown conditions would not have been met and would have led to core damage. The risk significance and enforcement aspects are discussed in Section 4OA7 of this report. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Section 1R20 describes a finding associated with minimizing “time to boil” during a refueling outage on Unit 1. This finding exhibited human performance cross cutting aspects in the organization subcategory due to inadequate planning and scheduling of the RCS drain down.

4OA5 Other

.1 Reactor Vessel Head Replacement Inspection (71007)

a. Scope

During the period February 27 through March 3, 2006, the inspector verified that the reactor vessel head removal activities maintained adequate nuclear and radiological safety by reviewing radiation protection program controls, planning, and preparations using applicable portions of inspection procedures 71121.01 and 71121.02. Activities related to the removal and storage of the Old Reactor Vessel Closure Head (ORVCH) were evaluated with respect to

ALARA planning, dose estimates/dose tracking, exposure controls including use of temporary shielding, contamination controls, radioactive materials management, emergency contingencies, and project staffing/training.

Details of this inspection are contained in Section 2OS1 and 2OS2 of this report.

b. Findings

No findings of significance were identified.

.2 Steam Generator Replacement Inspection (50001)

a. Scope

During the period February 27 through March 3, 2006, the inspector verified that steam generator removal and replacement activities maintained adequate nuclear and radiological safety by reviewing radiation protection program controls, planning, and preparations using applicable portions of inspection procedures 71121.01 and 71121.02. Activities related to the removal of the ORVCH were evaluated with respect to ALARA planning, dose estimates/dose tracking, exposure controls including use of temporary shielding, contamination controls, radioactive materials management, emergency contingencies, and project staffing/training.

Details of this inspection are contained in Section 2OS1 and 2OS2 of this report.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 05000412/2004-002-01 Potential for Spurious Opening of PORVs Due to Fire Induced Hot Shorts

The potential for spurious opening of the PORVs due to fire induced cable failures was identified by the licensee and documented in condition report 03-08460. The unresolved item was opened pending the results of an industry initiative to perform additional testing and evaluations of the effects of fires on cables and the likelihood of resultant fire induced faults. Independent of the industry initiative, FENOC assumed the postulated failures would occur and performed additional engineering calculations to evaluate the effects of a spurious PORV opening. The results of the analysis showed that the ability to safely shut down following a fire would not be adversely affected by a spurious PORV opening. The safe shutdown analysis was revised and updated to reflect the results of the calculations. The failure to identify and address the spurious opening of the PORVs was the result of inadequate circuit analysis during the initial fire safe shutdown program development. As such, this issue is considered to be a violation of license condition 2.F, which requires that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report. However, this issue was determined to be a minor violation because when analyzed the issue was shown to not adversely impact the safe shutdown capability. Based on review of the supporting engineering analysis and corrective actions (section 4OA2), this item is closed.

4OA6 Management Meetings

On April 27, 2006, the inspectors presented the inspection results to you, and other members of your staff. The inspector confirmed that proprietary information was not provided or examined during the inspection.

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 meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation:

TS 6.8.1.f, requires in part, that procedures related to the Fire Protection Program, be implemented and maintained. Contrary to this, FENOC failed to ensure that appropriate procedural guidance was provided in post-fire/post-trip procedures to limit elevated seal return water from negatively impacting the charging pump, and could have led to the inability to achieve and maintain safe shutdown conditions. The licensee entered this issue into the corrective action program as CR-05-07518, has completed a root cause evaluation, and implemented appropriate actions to preclude the event, which included the necessary procedure changes. The regional Senior Reactor Analyst performed a phase 3 analysis using information provided by the licensee from the plant's fire PRA. A phase 3 analysis was necessary because the issue involved multiple fire areas, including the main control room, which is not easily evaluated using IMC 0609, Appendix F, "Fire Protection SDP". The result of the phase 3 analysis was that the finding was of very low safety significance (Green).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

E. Anderson	First Energy Fleet Radiation Protection Manager
S. Baker	Site, Radiation Protection Manager,
R. Boyle	Staff Nuclear Engineer
J. Clark	Radiation Protection Health Services Technician
G. Davie	Manager, Training
P. Dearborn	Staff Nuclear Engineer
J. Fontaine	Supervisor, ALARA
J. Freund	Supervisor, Rad Operations Support
D. Girdwood	Supervisor Technical Training
D. Gratta	Senior Nuclear Engineer
R. Hansen	Manager, Nuclear Oversight
A. Hartner	Shift Manager
M. Helms	Health Physics Shift Manager
C. Hrelec	Senior Radiation Protection Technician
S. Janes	Senior Radiation Protection Technician
H. Kahl	Design Engineering
J. Lash	Site Vice President
J. Maracek	Regulatory Affairs
R. Mende	Director, Site Operations
J. Miller	Fire Protection Engineer
B. Paul	Senior Nuclear Specialist
P. Pauvlinch	Rapid Response Supervisor
R. Pucci	Senior Nuclear Specialist, ALARA Coordinator
K. Schweikart	Staff Nuclear Engineer
B. Sepelak	Supervisor, Regulatory Compliance
J. Sipp	Manager, Chemistry
J. Witter	Shift Manager
K. Wolfson	Superintendent, Nuclear Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000412/2004002-01	URI	Potential for Spurious Opening of PORVs Due to Fire Induced Hot Shorts.
05000334/2005-002	LER	Latent Fire Protection Issue Regarding Possible Loss of Reactor Coolant System Makeup Function.

Open/Closed

05000334/2006002-01	NCV	Failure to scope a seismic, safety-related structure into the maintenance rule structural monitoring program. (Section 1R06)
05000334/2006002-02	NCV	Inadequate corrective actions to resolve main steam safety valve (MSSV) component deficiencies that were the subject of industry operating experience. (Section 1R15)
05000334/2006002-03	FIN	Failure To Consider External Events During Reactor Coolant System Drain-down Activities. (Section 1R20)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Drawings

8700-RM-411-1, “Safety Injection System,” Rev. 18
8700-RM-430-1, “River Water System,” Rev. 26
8700-RM-430-2, “River Water System,” Rev. 18
8700-RM-430-3, “River Water System,” Rev. 19
8700-RM-424-1, “Feed Water System,” Rev. 14
8700-RM-424-2, “Feed Water System,” Rev. 11
10080-RM-411-1, “Low/High Head Safety Injection,” Rev. 12
10080-RM-434-1A, “Station Service Air Plant/Distribution,” Rev. 10
10080-RM-434-12, “Station Instrument Air,” Rev. 20
10080-RM-434-11, “Instrument Air Standby Train,” Rev. 1

Procedures

1OM-30.4.AB, “Operation of the Reactor Plant River Water System With Normal Return Flow Path to Unit 1/2 Blowdown Isolated,” Rev. 7
1OM-11.3.B.1, “Valve List - 1SI,” Rev. 14
1OM-11.3.C, “Power Supply and Control Switch List,” Rev. 8
1OM-24.3.B.1, “Valve List - 1FW,” Rev. 18
1OM-24.3.C, “Power Supply and Control Switch List,” Rev. 12
1OM-53A.1.2-I “Response to Loss of AFW Pump Suction Flow,” Rev. 0
1OM-53A.1.2-H “Makeup to PPDWST [1WT-TK-10],” Rev. 2
1OM-53A.1.2-K “Dedicated AFW Pump [1FW-P-4] Startup,” Rev. 1
1OM-53A.1.2-J “Feeding the Steam Generators From the Condensate System,” Rev. 1

1OST-24.10, “Auxiliary Feedwater System Monthly Verification,” Rev. 7

2OM-11.3.B.1, “Valve List - 2SIS,” Rev. 9
2OM-11.3.C, “Power Supply and Control Switch List,” Rev. 9
2OM-34.3.B.1, “Valve List - 2SAS,” Rev. 15
2OM-34.3.C, “Power Supply and Control Switch List,” Rev. 18

Condition Reports

05-01577	05-08063	05-04640
04-01276	05-04849	05-02584

Miscellaneous

Unit 1 and Unit 2 Operator Narrative Logs
PRA Notebook, “Auxiliary Feedwater System,” Rev. 0
Design Basis Document for Auxiliary Feedwater System, Rev. 8

Section 1R05: Fire Protection

2OST-33.10B, “Main Transformer Deluge Valve Test,” Rev. 0
2OST-33.10C, “Unit Station Service Transformer 2C Deluge Valve Test,” Rev. 0
2OST-33.10D, “Unit Station Service Transformer 2D Deluge Valve Test,” Rev. 0
2OST-33.10E, “System Station Service Transformer 2A Deluge Valve Test,” Rev. 0
2OST-33.10F, “System Station Service Transformer 2B Deluge Valve Test,” Rev. 0

Section 1R06: Flood Protection Methods

1/2MI-75-MANHOLE-1E, “Inspection of Manholes for Water Induced Damage,” Rev. 4
1/2-ADM-2016, “General Area Structural Inspections,” Rev.1
2BVT-1.33.7, “Flood Seals Visual Inspection,” Rev. 2
1BVT-1.33.07, “Flood Seals Visual Inspection,” Rev. 3
NPDAP 8.6, “BVPS Configuration Management Program,” Rev. 4
Maintenance Rule System Basis Document, Unit 1 Structures, Rev. 7

Section 1R20: Refueling and Other Outage Activities

Procedures

1OM-6.4.N, “Draining the RCS for Refueling,” Rev. 18
1OM-6.4.AC, “Hydrogen Peroxide Additions to RCS,” Rev. 4
1OM-6.4.BH, “Filling the RCS With the Temporary Reactor Vessel Cover Installed,” Rev. 0
1OM-20.4.P, “Draining the Reactor Vessel Using the Temporary Reactor Vessel Cover,” Rev. 1
1/2-RP-3.16, “Core Unload,” Rev. 9
NOP-ER-2001,” Boric Acid Corrosion Control Program,” Rev. 5
NDE-VT-510, “Visual Inspection for Evidence of Boric Acid Leakage,” Rev. 13
1OM-37.5.B.7, “Table 37-7 480 Volt Motor Control Center Loads,” Rev. 29
NOP-OP-1005, “Shutdown Safety,” Rev. 8

Condition Reports

06-02344	06-00981	06-01010
0-02154	06-01157	06-01165

Miscellaneous

Unit 1 Operator Narrative Logs
Unit 1 Shutdown Risk Assessments
Unit 1 Pre-Outage Shutdown Safety Report, Rev. 0
Clearance 1SM-20-OPS-002
Clearance 1EBV-11-EBM-010
Clearance 1R17-20-MNE-001A

Drawings

8700-RM-420-1, “Fuel Pool Cooling and Purification System,” Rev. 6
8700-RM-410-1, “Residual Heat Removal System,” Rev. 11
8700-RE-22AW, “Instrument Power Supplies in the Vertical and Benchboards,” Rev. 8
8700-RE-22CX, “RHR - Temperature and Flow Loop Diagrams TRB-RH604, TRB-RH606, FT-RH605, HIC-RH758,” Rev. 5
8700-RE-22E, “Boric Acid Blend System Control Loop Diagrams, FT-CH113, FT-CH168,” Rev. 10
8700-RE-3AH, “Wiring Diagram Benchboard Section A, Sheet 32,” Rev. 9

Section 1EP4: Emergency Action Level (EAL) Revision Review

Beaver Valley Power Station Emergency Preparedness Plan:
- Section 1, Definitions, Rev 16
- Section 2, Scope and Applicability, Rev 15
- Section 3, Summary, Rev 15
- Section 4, Emergency Conditions, Rev 19
- Section 5, Emergency Organization, Rev 20
- Section 6, Emergency Measures, Rev 23
- Section 7, Emergency Facilities and Equipment, Rev 21
- Section 8, Maintaining Preparedness, Rev 20

1/2 -EPP-I-1a/b, Recognition and Classification of Emergency Conditions, Rev 6
1/2-EPP-I-2, Unusual Event, Rev 21
1/2-EPP-I-3, Alert, Rev 20
1/2-EPP-I-4, Site Area Emergency, Rev 20
1/2-EPP-I-5, General Emergency, Rev 21
1/2-EPP-IP-1.1, Notifications, Rev 33
1/2-EPP-IP-1.2, Communication and Dissemination of Information, Rev 20
1/2-EPP-IP-1.6, Emergency Operations Facility Activation, Operation and Deactivation, Rev 18
1/2-EPP-IP-3.5, Traffic and Access Control, Rev 11
1/2-EPP-IP-4.1, Offsite Protective Actions, Rev 20
1/2-EPP-IP-6.2, Termination of the Emergency and Recovery, Rev 13
1/2-EPP-IP-7.1, Emergency Equipment Inventory and Maintenance Procedure, Rev 18

1/2-EPP-IP-9.4, Activation, Operation, and Deactivation of the Joint Public Information Center (JPIC), Rev 15

Section 2OS1/2OS2: Access Control to Radiologically Significant Areas/ALARA Planning & Controls

Procedures

1/2-ADM-1601, Rev 13	Radiation Protection Standards
1/2-ADM-1611, Rev 8	Radiation Protection Administrative Guide
1/2-ADM-1621, Rev 3	ALARA Program
1/2-ADM-1630, Rev 9	Radiation Worker Practices
1/2-ADM-1631, Rev 5	Exposure Control
1/2-HPP-3.02.003, Rev 8	Decontamination Control
1/2-HPP-3.02.004, Rev 4	Area Posting
1/2-HPP-3.04.002, Rev 5	Bioassay Administration
1/2-HPP-3.05.001, Rev 4	Exposure Authorization
1/2-HPP-3.07.002, Rev 4	Radiation Survey Methods
1/2-HPP-3.07.013, Rev 3	Barrier Checks
1/2-HPP-3.08.001, Rev 8	Radiological Work Permit
1/2-HPP-3.08.003, Rev 9	Radiation Barrier Key Control
1/2-HPP-3.08.005, Rev 4	ALARA Review Program
BVBP-RP-0003, Rev 3	Dosimetry Practices
BVBP-RP-0013, Rev 2	Radiation Protection Risk Assessment Process
NOP-WM-7001, Rev 0	ALARA Program
NOP-WM-7002, Rev 0	Operational ALARA Program
NOP-WM-7003, Rev 0	Radiation Work Permit
NOP-WM-7017, Rev 0	Contamination Control Program
NOP-WM-7021, Rev 0	Radiological Postings, Labeling, and Markings

Nuclear Oversight Field Observation Reports (10):
BV120062503, BV120062466, BV120062483, BV120062486, BV120062447, BV120062492, BV120062510, BV120062509, BV120062481, BV120065446

Condition Reports (24):
06-01115, 06-01060, 06-01005, 06-00977, 06-00948, 06-00871, 06-00854, 06-01402, 06-01393, 06-01436, 06-01394, 06-00977, 06-01305, 06-00978, 06-00930, 06-01113, 06-01029, 06-01003, 06-01005, 06-01259, 06-00921, 06-01497, 06-01534, 06-00921

ALARA Work-In-Progress Reviews:
RWP #106-8000, ALARA Plan #06-1-22, SGRP Management Tours
RWP #106-8003, ALARA Plan #06-1-25, Insulation Removal from B & C steam generators
RWP#106-8005, ALARA Plan #06-1-27, SGRP scaffolding installation
RWP#106-8010, ALARA Plan #06-1-32, SGRP staging of ventilation & vacuums
RWP#106-4003, ALARA Plan #06-1-03, Code Safety Valve Work
RWP#106-4036, ALARA Plan #06-1-27, Reactor Head scaffolding installation
RWP#106-4019, ALARA Plan #06-1-11, Reactor Disassembly/Reassembly

ALARA Post-Job Reviews:

RWP #106-4019, ALARA Plan #06-1-11, Reactor Disassembly
RWP #106-4033, ALARA Plan #06-1-54, AOV Testing on PCV-RC-455A &B
RWP #106-8006, ALARA Plan #06-1-28, SGRP Piping Severance

ALARA Committee Meeting Minutes:

Meeting Nos.: 06-01, 06-02, 06-03, , 1R17-01, 1R17-02

Miscellaneous Reports:

1R17 Outage ALARA Plan
Steam Generator & Reactor Vessel Head Replacement Radiation Protection Action Plans (SGRP-RPAP-01 thru 18)

Section 4OA2: Identification and Resolution of Problems

Calculations

10080-DEC-0254, BV2 Evaluation of Potential for PORV Spurious Actuations Due to Fire Induced Electrical Hot Shorts, Rev. 0
10080-DBC-0820, Beaver Valley Power Station Unit 2 Loss of Offsite Power + Stuck Open Pressurizer PORV Analysis, Rev. 0
10080-DBC-0820, Beaver Valley Power Station Unit 2 Loss of Offsite Power + Stuck Open Pressurizer PORV Analysis, Addendum No. 1, Rev. 0

Procedures

20M-56B, Safe Shutdown Following a Serious Fire in the Cable Vault Building, Rev. 11
20M-56C.4.C, Alternate Safe Shutdown From Outside Control Room - NCO Procedure, Rev. 16
20M-56C.4.D, Alternate Safe Shutdown From Outside Control Room - Nuclear Operator #1 Procedure, Rev. 19

Miscellaneous

2OM-56B Safe Shutdown Procedures Manual Operator Actions Timeline Study, dated January 11, 2004
2OM-56C Safe Shutdown Procedures Manual Operator Actions Timeline Study, dated January 11, 2004

LIST OF ACRONYMS

BVPS	Beaver Valley Power Station
CFR	Code of Federal Regulations
CR	Condition Reports
FENOC	First Energy Nuclear Operating Company
HRA	High Radiation Area
NRC	Nuclear Regulatory Commission
ORVCH	Old Reactor Vessel Closure Head
OSG	Old Steam Generator
OST	Operations Surveillance Test
PCE	Personnel Contamination Event Report
PI	Performance Indicator
PI&R	Problem Identification and Resolution
RWP	Radiation Work Permit
SGRP	Steam Generator Replacement Project
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VHRA	Very High Radiation Area
WO	Work Order