July 18, 2005

Mr. L. William Pearce
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2005006 AND 05000412/2005006

Dear Mr. Pearce:

On June 30, 2005, 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 11, 2005, 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 Commissions 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 one self-revealing finding which was of very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into the corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest anything 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 NRCs 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 610-337-5234 if you have any questions regarding this letter.

Sincerely,

/RA/

Ronald R. Bellamy, Ph.D., Chief
Projects Branch 7
Division of Reactor Projects

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

Enclosure: Inspection Report 05000334/2005006; 05000412/2005006 w/Attachments

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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.	05000334/2005006 and 05000412/2005006
Licensee:	First Energy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	P. Cataldo, Senior Resident Inspector D. Orr, Senior Resident Inspector D. Kern, Senior Resident Inspector E. H. Gray, Senior Reactor Inspector P. D. Kaufman, Senior Reactor Inspector T. Moslak, Health Physicist S. M. Pindale, Senior Reactor Inspector G. Smith, Resident Inspector G. Bowman, Reactor Inspector K. Diederich, Reactor Inspector A. Rosebrook, Project Engineer E. Knutson, Resident Inspector
Approved by:	R. Bellamy, Ph.D., Chief Projects Branch 7 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2005-006, IR 05000412/2005-006; 04/01/2005 - 06/30/2005; Beaver Valley Power Station, Units 1 & 2; Maintenance Rule Implementation.

The report covers a 3-month period of inspection by resident inspectors, a modification team inspection consisting of three regional inspectors and announced inspections by a regional health physics inspector, a senior reactor inspector, and two senior reactor inspectors during the Unit 2 refueling outage. One Green non-cited violation (NCV) was 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. A self-revealing, non-cited violation of the Unit 1 Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.1.1 was identified, in that an inoperable channel of the Over-Temperature Delta-Temperature (OTDT) Circuit was not placed in the tripped condition within six hours. Specifically, inadequate procedural steps within maintenance procedures resulted in the lead and lag switches of a circuit card in the OTDT channel of the Reactor Protection System (RPS) being left in the “OFF” position for several days following maintenance.

  This finding is greater than minor because it affected an attribute and objective of the Mitigating Systems Cornerstone, in that it reduced the reliability of a RPS component and thus reactivity control was degraded. Specifically, the lead and lag switches being left in the “OFF” position caused the loop 1 channel OTDT setpoint to be less responsive than required by TS. The finding is of very low safety significance because the affected channel of OTDT was still capable of causing a reactor trip and other trips were available to provide a backup to this safety function. (Section 1R12).

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period operating at 100 percent. On May 1, 2005, power was reduced to 90 percent in order to clean the 'B' and 'D' water boxes in the main unit condenser as part of the summer readiness program (Section 1R01). Following completion of the water box cleaning, the Unit was returned to full power on May 8. The Unit continued to operate at 100 percent until May 22, when the operators reduced power output to 90 percent to clean the 'A' and 'C' water boxes in the main unit condenser. Following completion of the water box cleaning, the Unit was returned to full power on May 29, where it continued to operate at a nominal value of 100 percent for the remainder of the inspection period.

Unit 2 began the inspection period operating at 100 percent power. On April 4, 2005, the unit was taken off-line for a planned refueling outage (Section 1R20). Following completion of this outage, the unit was taken critical on April 28. On May 2, the startup was delayed at approximately 98 percent power to complete repairs on the "A" separator drain receiver drain pump motor. Following completion of repairs, the Unit was returned to 100 percent on May 6, where it continued to operate for the remainder of the period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors reviewed the Beaver Valley Power Station (BVPS) design features and FENOC's implementation of procedures to protect risk significant mitigating systems from adverse weather effects due to high temperatures. The inspectors conducted interviews with various station personnel to gain insights into the station's warm weather readiness program and reviewed the status of various work orders categorized as warm weather preparation activities. The inspectors reviewed the corrective action program database, operating experience, and the Updated Final Safety Analysis Report (UFSAR), to determine the types of adverse weather conditions to which the site is susceptible, and to verify if the licensee was identifying and resolving weather-related equipment problems.

In addition, the inspectors reviewed the readiness of abnormal and emergency operating procedures, as well as emergency plan activation and classification requirements, for floods, earthquakes and other natural phenomenon that could occur in the summer months.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (IP 71111.02 - 18 samples)

a. Inspection Scope

The inspectors reviewed a sample of six safety evaluations for the initiating events, barrier integrity, and mitigating systems cornerstones to verify that changes and tests were reviewed and documented in accordance with 10 CFR 50.59 and, when required, NRC approval was obtained prior to implementation. The sample included safety evaluations for engineering change packages (ECPs), engineering calculations, and UFSAR changes. The inspectors assessed the adequacy of the safety evaluations through interviews with the cognizant plant staff and review of supporting information, such as calculations, engineering analyses, design change documentation, the UFSAR, and plant drawings. In addition, the inspectors reviewed the administrative procedures that control the screening, preparation, and issuance of safety evaluations to ensure that the procedures adequately implemented the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments."

The inspectors also reviewed a sample of 12 changes that the licensee had evaluated (using a screening process) and determined to be outside the scope of 10 CFR 50.59, therefore not requiring a full safety evaluation. The inspectors performed this review to evaluate whether the licensee's conclusions with respect to 10 CFR 50.59 applicability were appropriate. The sample of issues that were screened out included design changes, temporary alterations, procedure changes, and setpoint changes.

The safety evaluations and screenings were selected based on the safety significance of the affected structures, systems, and components (SSCs). A listing of the safety evaluations, safety evaluation screenings, and other documents reviewed is provided in Attachment 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 3 samples)

a. Inspection Scope

Partial System Walkdown. The inspectors performed three partial system walkdowns to verify system and component alignment and to note any discrepancies that would impact system operability. The inspectors verified selected portions of redundant or backup systems or trains were available while certain system components were out-of-service. The inspectors reviewed selected valve positions, the general condition of major system components, and electrical power availability. The partial walk-downs included the following systems:

- On May 12, 2005, the inspectors completed a walkdown of the Unit 1 ‘B’ and ‘C’ river water pumps during surveillance testing of the ‘A’ river water pump. The inspectors verified that the associated testing activities did not adversely affect redundant system components.

- On June 6, 2005, the inspectors performed a walkdown of the Unit 1 ‘B’ train Auxiliary Feedwater (AFW) system while the ‘A’ train was out-of-service for the performance of planned maintenance on MOV-1FW-151D, “’A’ train throttle valve to ‘B’ steam generator.”

- On June 13, 2005, the inspectors performed a walkdown of the Unit 1 ‘A’ and ‘C’ train Main Steam (MS) headers while the ‘B’ MS header atmospheric dump valve (ADV), PCV-MS-101B was out-of-service in order to replace a solenoid valve in the control circuit as well as perform other planned maintenance activities on the ADV.

b. <u>Findings</u>

No findings of significance were identified.

1R05 <u>Fire Protection</u> (71111.05Q - 9 samples)

a. <u>Inspection Scope</u>

The inspectors reviewed the Unit 1 Updated Fire Protection Appendix ‘R’ Review, Rev. 25, and the Unit 2 Fire Protection Safe Shutdown Report, Addendum 27, and selected the following nine fire areas for inspection:

- Unit 1 Main Exhaust Filter Bank 1
- Unit 1 Main Exhaust Filter Bank 2
- Unit 1 Fuel Building
- Unit 2 Cable Vault and Rod Control Area
- Unit 2 Condensate Polishing Building
- Unit 2 Turbine Building
- Unit 2 Turbine Building - Battery Room ‘2-6'
- Unit 2 Diesel Generator Cubicle - Orange
- Unit 2 Diesel Generator Cubicle - Purple

The inspectors reviewed the fire protection conditions of the fire areas listed above to verify compliance with criteria delineated in Administrative Procedure 1/2-ADM-1900, “Fire Protection,” Rev. 9. This review, for example, included FENOCs control of transient combustibles, material condition of fire protection equipment, and the adequacy of compensatory measures for any fire protection impairments.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed the internal flood protection features associated with the ‘C’ cubicle located within the Intake Structure. The inspectors performed a detailed walkdown of the area and reviewed the condition of various flood barriers, seals, sump pumps, alarm circuits, and other mitigating equipment. The inspectors also verified that the flood control features for the cubicle were consistent with the design and licensing basis of the plant, including the UFSAR, TS, Individual Plant Examination report, and other documents listed in Attachment 1.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of FENOC’s surveillance and control of heat exchanger performance by reviewing the results of a Unit 2 heat exchanger inspection for the 2-1 emergency diesel generator (EDG) jacket water cooling heat exchanger, 2EGS-E22A. The review included an assessment of work order (WO) 200099121 and the heat exchanger inspection report performed in accordance with 1/2 -ADM-2106, ”River/Service Water System Control and Monitoring Program,” Rev. 0. The inspectors reviewed the results, evaluated against applicable acceptance criteria, and verified the inspection was consistent with GL 89-13,”Service Water System Problems Affecting Safety Related Equipment.”

b. Findings

There were no findings of significance identified.

1R08 Inservice Inspection (71111.08 - 3 Samples)

a. Inspection Scope

The inspection assessed the effectiveness of the licensee’s program for monitoring degradation of the reactor coolant system boundary. The inspection focused on four activities: the dissimilar metal weld and non-destructive examination (NDE) In-service

Inspection (ISI) program implementation; upper reactor pressure vessel head penetration inspection activities; steam generator eddy-current testing (ECT); and the boric acid corrosion control program.

For the dissimilar metal weld examination, the inspectors interviewed ultrasonic (UT) and visual test (VT) examination personnel and engineers to assess the planning and preparation for the activities. The inspectors reviewed training and qualification records to verify the licensee's personnel qualification process adequately prepared the assigned staff to perform the examination. Selected examination procedures and results were reviewed to determine whether they provided adequate guidance and examination criteria to implement the examination plan. Inspection documents were reviewed for indications that required corrective action.

The inspectors reviewed the welding and NDE records for one previously completed pressure boundary weld on a Class 1 system by examining BV-2RCS-REV21, "Replace Core Exit Thermocouple Nozzle E-13," and Beaver Valley Unit 2 WO 200036683, dated October 20, 2003. The inspectors evaluated this package to determine whether the canopy seal weld and NDE examination on the core exit thermocouple nozzle assembly were performed in accordance with the ASME Code.

The results of the inspectors' review of the licensee's upper reactor pressure vessel head penetration inspection activities are discussed in Section 4OA5.1 of this report.

The inspectors assessed the effectiveness of the licensee's steam generator tube ECT and repair program, procedures, and inspection activities for monitoring degradation of steam generator tubes. This assessment was based on the rules and regulations of the steam generator examination program, the Beaver Valley Unit 2 steam generator examination guidelines (ISIE1-8, Rev. 9), NRC Generic Letters, the Code of Federal Regulations 10 CFR 50, the Technical Specifications for Beaver Valley Unit 2, and ASME Boiler and Pressure Vessel Code Sections V and XI. Supporting the assessment were parts of Nuclear Energy Institute (NEI) 97-06, EPRI PWR steam generator examination guidelines, and the Beaver Valley Unit 2 steam generator degradation assessment (SG-SGDA-05-5) for the 2R11 refueling outage. The procedure for tube plugging (MRS-SSP-1027-DLW/DMW, Rev. 2) was also included in the review.

The inspectors assessed the licensee's ability to identify boric acid corrosion and leaks. The licensee's boric acid inspection procedures were reviewed to determine if they provided adequate scope and guidance on examination criteria and corrective actions when boric acid deposits are found. The inspectors conducted a boric acid walkdown of containment to verify that there were no active boric acid leaks and reviewed the licensee's boric acid walkdown results for indications of active boric acid leaks or boric acid corrosion of carbon steel components.

b. Findings

No findings of significance were identified.

Enclosure

1R11 Licensed Operator Requalification Program (71111.11 - 1 sample)

a. Inspection Scope

The inspectors observed licensed operator requalification training at the control room simulator. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario and implement the emergency plan. The inspectors observed the operators' simulator drill performance and compared it to the criteria listed in simulator scenario 1DRLS-E-3.004. The inspectors observed supervisory oversight, command and control, communication practices, and crew assignments to ensure they were consistent with normal control room activities. The inspectors observed operator response during the simulator drill transient and verified the fidelity of the simulator to the actual plant. The inspectors evaluated the training evaluators regarding the recognition of individual and operating crew errors, and subsequent post-training remediation actions. The inspectors attended the post-drill critique in order to evaluate the effectiveness of problem identification. Additional documents used for this inspection activity included:

- 1OM-6.4IF, Attachment 1, "Instrument Failure [LT-1RC-459(460)(461)]," Rev. 9
- 1OM-53A.1.E-0, "Reactor Trip or Safety Injection," Issue 1C - Rev. 6
- 1OM-53A.1.1-K, "Verification of Automatic Actions," Issue 1C - Rev. 2
- 1OM-53A.1.E-3, Steam Generator Tube Rupture," Issue 1C - Rev. 5
- 1OM-53C.4.1.24.1, "Loss of Main Feedwater," Rev. 3

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12 - 3 samples)

a. Inspection Scope

The inspectors evaluated the follow-up actions for problems with selected SSCs, and reviewed the performance history of these SSCs to assess the effectiveness of Beaver Valley's maintenance activities. The inspectors reviewed Beaver Valley's problem identification and resolution actions, as applicable, for these issues to evaluate whether plant staff had appropriately monitored, evaluated, and dispositioned the issues in accordance with station procedures and the requirements of 10 CFR 50.65(a)(1) and (a)(2), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." In addition, the inspectors reviewed selected SSC classification, performance criteria, and goals. The following three issues were reviewed:

- Multiple failures of Unit 2 feedwater flow control circuit cards
- Service Water System Leaks and 10 CFR 50.65(a)(1) evaluation
- CR-05-04249, "TM-1RC-412E Lead/Lag Switches As-Found out of Position"

Enclosure

b. Findings

Introduction. A Green, self-revealing NCV was identified for not placing an inoperable Over-Temperature Delta-Temperature (OTDT) channel in the tripped condition as required by TS LCO 3.3.1.1.

Description. On June 9, 2005, during the performance of 1MSP-6.35-I, “P-RC455 Pressurizer Pressure Protection Channel 1 Calibration,” Rev. 1, the reactor operator identified that the channel 1 OTDT setpoint reading was not as “dynamic” as the remaining two channels. A subsequent historical trend using data from the plant computer was performed by an Instrument and Control supervisor, which revealed a notable dampening of the signal beginning approximately June 1, 2005. This date corresponded to a previous calibration of the channel 1 average coolant temperature (TAVG) loop. Subsequent troubleshooting revealed the lead and lag switches were left in the ‘OFF’ position from this previous TAVG channel calibration completed on June 3, 2005. The two affected switches were subsequently returned to their correct position on June 10, 2005.

The licensee formed an event review team to determine the cause of the switch misposition. The team concluded that the mispositioned switches occurred during performance of a quarterly calibration of TAVG channel 1 conducted on June 1-3, 2005. In particular, during the calibration, which was performed under 1MSP-06-20-I, “DELTA-T TAVG Protection Instrument Channel I Test(T-RC412),” Rev. 2, as-found voltage readings necessitated adjustments to the circuit. However, these voltage adjustments required the use of an 18-month calibration procedure, 1MSP-6.38-I, “T-RC412 Delta T TAVG Protection Instrument Channel I Calibration,” Rev. 2. As a result, performance of selected sections of these two procedures, as written, resulted in the lead and lag switches remaining in the “OFF” position until their discovery on June 9, 2005.

Analysis. This issue is considered more than minor because it involved a performance deficiency and adversely affected the equipment performance attribute of the mitigating system cornerstone. Specifically, the reliability of the OTDT channel 1 was reduced due to the removal of the lead and lag time constants. FENOC's failure to restore the channel 1 TAVG lead and lag switches to their required position following calibration due to a faulty procedure is a performance deficiency. BVPS Unit 1 TS requires RPS OTDT trip setpoints to be set in accordance with a prescribed equation which includes time constants as specified in the Core Operating Limits Report (COLR). The COLR specifies the lead time constant to be greater than 30 seconds and the lag time constant to be less than 4 seconds. With both switches set to ‘OFF’, the time constants were both set to zero. Although the lag time constant was within this limit, the lead time constant was not in compliance with TS.

The significance of the finding was evaluated in accordance with Manual Chapter 0609, “Significance Determination Process.” Using Appendix ‘A’, “At-Power Situations,” the inspector determined that the Mitigating System cornerstone was affected because reactivity control was degraded. The affected mitigation system component for this finding is the solid state protection system (SSPS) of which the OTDT trip is a subcomponent. Based on the fact that 1) only the dynamic response of one channel of OTDT was affected by this performance deficiency and 2) the OPDT and low RCS pressure trip can provide backup reactor trip signals for the OTDT to trip the reactor, this finding screens to Green using the phase 1 screening process. Specifically, the SSPS did not incur a loss of function per Generic Letter 91-18.

Enforcement. TS LCO 3.3.1.1 requires three total channels of the OTDT functional unit. Action statement 7 for this LCO requires that when the number of operable channels is one less than the total (i.e. two), the inoperable channel must be placed in the tripped condition within six hours. Additionally, Table 3.3-1 of TS requires the lead and lag time constants for each OTDT channel to be greater than 30 seconds and less than 4 seconds, respectively as specified in the COLR. Contrary to the above, following maintenance on the channel 1 OTDT circuit, the lead time constant remained at zero and thus inoperable, yet the channel was not placed in the tripped condition within the following six hours. In fact, the channel remained inoperable and not in the tripped condition for approximately six days. This procedural deficiency resulted in less than adequate responsiveness in the OTDT trip setpoint calculator. Because this deficiency was of very low safety significance and has been entered into the corrective action program as CR 05-01630, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000334/2005-006-01, Inadequate Procedure Results in Incorrect Lead Time Constant in the Over Temperature Delta Temperature Reactor Trip Function.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 6 samples)

a. Inspection Scope

The inspectors reviewed the scheduling and control of six 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 10 CFR 50.65(a)(4); 1/2-ADM-2033, “Risk Management Program,” Rev. 3; NOP-WM-2001, “Work Management Scheduling 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 1/2-OM-48.1.I, “Technical Specification Compliance,” Rev. 18. This inspection activity represented six samples of the following work activities:

- Planned Unit 2 yellow shutdown risk based on the unavailability of spent fuel pool to supplemental leak collection ventilation flow paths due to maintenance clearances on April 13, 2005.

- Planned Unit 2 yellow risk due to maintenance and testing associated with the “23A” auxiliary feedwater pump on May 2, 2005.

- Emergent risk evaluation following the May 8, 2005, failure of the Unit 2 vital inverter No. 2.

- Planned Unit 2 yellow risk on June 13, 2005, due to deluge testing and associated switchyard circuit breaker testing on the dedicated, offsite power system transformer 2B.

- Planned Unit 1 yellow risk on June 30, 2005, due to solid state protection system testing.

- Planned Unit 2 yellow risk on June 30, 2005, due to deluge testing on the dedicated, offsite power system transformer 2A.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed a non-routine plant evolution involving the isolation of a dedicated off-site power source, the 2B station service system transformer. The inspectors evaluated personnel performance during the evolution, to verify that applicable operating procedures were adhered to and that control room and in-field briefs adequately addressed the scope of the evolution. The inspectors also reviewed plant operating logs, verified technical specification compliance, reviewed operating shift summaries, and verified that adverse conditions were entered into the corrective action rogram.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 samples)

a. Inspection Scope

The inspectors evaluated the technical adequacy of selected operability determinations (OD), Basis For Continued Operations (BCO), or operability assessments, to verify that determinations of operability were justified, as appropriate. In addition, the inspectors verified that TS LCO requirements and UFSAR design basis requirements were properly addressed. This inspection activity represented six samples of the following issues:

- The inspectors reviewed the adequacy and scope of an operability assessment associated with CR 05-03772. This CR detailed a non-conservative valve factor that was previously applied to evaluate the performance of feedwater isolation trip valves, 2FWS-HYV157 A, B, and C. The valve factor affected valve stroke time and ability to close under differential pressure.

- The inspectors reviewed the adequacy and scope of an operability determination associated with CR 05-03674. This CR addressed a time critical manual action to secure a Unit 1 charging pump operating without river water cooling during postulated fire scenarios. The time available for manual actions was significantly less than had been assumed in the BVPS-1 Appendix R Report.

- The inspectors reviewed the adequacy and scope of an operability assessment associated with CR 05-03161. This CR involved a 2-2 emergency diesel generator exciter modification that necessitated a testing methodology change for inservice testing of the air start system solenoid valves.

- The inspectors reviewed an OD associated with the Unit 1 high head safety injection (HHSI) pumps, as documented in CR 05-02162, regarding a concern that a new calculation methodology could violate safety analysis assumptions associated with HHSI flow during a design basis accident. The inspectors assessed the adequacy and acceptability of FENOC's conclusion in the OD that the reactor coolant pump seal injection flow calculation would not cause the HHSI flow to fall below the minimum credited in the accident analysis.

- The inspector evaluated operability impacts to vital power supplies following the failure of the #2 vital inverter at Unit 2, on May 8, 2005. In particular, the inspector evaluated FENOCs extent of condition reviews regarding similar inverters that exist at both units due to the identification of failed capacitors during the investigation. The inspector evaluated technical specification compliance, as well as the risk significance associated with the inoperability as detailed in Section 1R13.

- The inspectors evaluated the adequacy and scope of the operability assessment associated with the Unit 1 emergency diesel generator (EDG). Specifically, the inspectors evaluated stress calculations that supported a positive operability determination following the identification of a broken support for lube oil piping of the #1 EDG that was originally identified by FENOC in October 2003, as detailed in CR-03-03956.

b. <u>Findings</u>

No findings of significance were identified.

1R16 <u>Operator Workarounds</u> (71111.16 - 2 samples)

a. <u>Inspection Scope</u>

The inspectors reviewed the current Unit 1 Operator Work-Arounds (OWAs), which also included Operator Challenges and Control Room Deficiencies. Only one OWA currently exists, involving manual voltage control of the main unit generator. The inspector reviewed the impact of this OWA relative to the ability of the operators to implement abnormal and emergency operating procedures, and assessed whether the functional capability of the system or human reliability was affected in response to an initiating event.

Additionally, the inspectors reviewed the cumulative effects of OWAs to determine the overall impact on mitigating systems and the operator's ability to respond to transients and accidents, and whether it could impact the reliability of the affected systems. The inspector also reviewed the overall impact of current operator challenges and control room deficiencies as defined by BVBP-OPS-0002, Rev. 10, "Operator Work-Arounds, Operator Challenges, and Control Room Deficiencies," and verified that adverse conditions were entered into the corrective action program for resolution.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

1. Annual Review (71111.17A - 1 sample)

a. Inspection Scope

The inspectors evaluated the design basis impact of a permanent modification to the Unit 2 digital radiation monitoring system. Engineering Change Package (ECP) 02-0018, "Installation of N–16 monitors at BV-2," Rev. 1, involved the installation of three N-16 radiation monitors on the three Unit 2 main steam lines to provide a more responsive method to monitor primary-to-secondary steam generator tube leakage. The inspector (1) reviewed the design adequacy of ECP 02-0018, (2) reviewed operational impacts and controls during implementation, (3) reviewed the adequacy of post-modification testing, and (4) reviewed the resulting procedural changes.

b. Findings

No findings of significance were identified.

2. Biennial Review (IP 71111.17B - 5 samples)

a. Inspection Scope

The inspectors reviewed selected permanent plant modification packages, calculations, setpoint changes, and engineering evaluations to verify that the design bases, licensing bases, and performance capability of risk-significant SSCs have not been degraded through plant modifications.

Plant changes were selected for review based on plant risk insights and included SSCs associated with the initiating events, barrier integrity, and mitigating systems cornerstones. The inspection included walkdowns of selected plant systems and components, interviews with plant staff, and review of applicable documents, including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action documents, the UFSAR, TS, and system design basis documents (DBDs).

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument setpoints, uncertainty calculations, electrical coordination, electrical loads analysis, and equipment environmental qualification. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For each modification, the 50.59 screenings or evaluations were reviewed as described in Section 1R02 of this report. The inspectors verified that procedures, DBDs, and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that the as-built configuration was accurately reflected in the design documentation, and that post-modification testing was adequate to ensure the SSCs would function properly.

A listing of documents reviewed is provided in Attachment 1.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 samples)

a. Inspection Scope

The inspectors reviewed and/or observed selected 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, consistent with the applicable design and licensing bases and associated safety functions; and 3) the PMT was performed in accordance with applicable procedures, with appropriately calibrated testing equipment, and the system was returned to normal system alignment following acceptable testing results. The following five PMTs were observed:

- 2OST-11.2, “Low Head Safety Injection Pump(2SIS*P12B) Test” Rev 21 performed on April 11, 2005 as a retest following maintenance on the rotating element of the pump.

- 1OST-1.10, “Cold Shutdown Valve Exercise Test,” Rev 30, performed on May 11, 2005, for the 1A steam generator atmosphere dump valve, PCV-1MS-101A. Maintenance involved control solenoid replacement, comprehensive air operated valve testing, and packing gland follower nut retorque.

- Portions of 2MSP-36.19-M,“#1 Emergency Diesel Generator Inspection” Issue 4 Rev 9 and 2MSP-36.29-M, “#1 Emergency Diesel Generator Filter, Strainer, Heat Exchanger and Woodward Governor Maintenance” Issue 4 Rev 14 performed to retest the ‘2 -1' Emergency Diesel Generator following inspection and maintenance.

- 1OST-30.6A, Rev. 16, “Reactor Plant River Water Pump 1C Test on Train A Header,” performed on June 5, 2005, and 1BVT-2.30.3, Rev. 6, “River Water Pump [1WR-P-1C] Head Capacity Curve.” This PMT followed the overhaul of the “C” reactor plant river water pump under work order 200075036, engineering change package (ECP) 03-0589, and 1-CMP-—30-001, Rev. 4, “River Water Pump Overhaul.”

- 1OST-1.10, “Cold Shutdown Valve Exercise Test,” Rev. 30, performed on June 13, 2005, following replacement of a control circuit solenoid valve associated with PCV-1MS-101B, “1B S/G Atmos Dump.”

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 2 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 technical specifications (TS) and plant procedures. The inspectors evaluated specific performance attributes including operator performance, clearance activities and configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors also evaluated the following activities:

- Pre-Outage Shutdown Safety Review
- Refueling operations
- Restart readiness meetings and mode hold resolution discussions
- Plant startup and heatup, including initial criticality following refueling.
- Low power reactor physics testing.
- Clearance execution
- Simulated emergency closure of the containment hatch.
- Reactor plant shutdown and cooldown, including evaluation of cooldown rates.
- Draining the RCS to support refueling operations.
- Spent fuel pool cooling system operation
- Verified maintenance of boration flowpaths
- Coordination of electrical bus work and minimization of shutdown risk
- Performed a walkdown of the reactor coolant system level instrumentation during periods of reduced inventory to verify appropriate configuration.

- Final containment inspection prior to restart and FENOC's completed closeout and debris inspection and associated report.
- External inspection of the containment sump and evaluated the cleanliness and integrity of the sump relative to its design function.
- Completed containment sump inspection performed by FENOC.

b. Findings

There were no findings of significance identified.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

The inspectors reviewed and/or observed portions of the following surveillance tests, and compared test data with established acceptance criteria to verify the systems demonstrated the capability of performing their intended safety functions. The inspectors reviewed the applicable sections of the UFSAR, verified that the applicable systems or components maintained operational readiness consistent with the design bases, and verified compliance with applicable TSs, and that adverse conditions were entered into the corrective action program for resolution. This inspection activity represented seven samples, which included:

- 1OST-30.2, "Reactor Plant River Water Pump 1A Test," Rev. 35
- 2OST-49.2, "Shutdown Margin Calculation" Rev. 13
- 2OST-30.13B, Rev. 17, "Train B Service Water System Full Flow Test," completed on March 22, 2005.
- 1MSP-2.03-I, Rev. 16, "Power Range Neutron Flux Channel N41 Refueling Calibration," conducted on April 11, 2005.
- 2OST-36.1, Rev. 45, "Emergency Diesel Generator [2EGS*EG2-1] Monthly Test," conducted on May 11, 2005.
- 1OST-24.4, Rev. 30, "Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2]"
- 2OST-11.14A, Rev. 14, "LHSI Full Flow Test"

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors selected one temporary modification (TM) for review based on risk significance. The TM and associated 10 CFR 50.59 screening were reviewed against the system design and licensing basis documentation, including the UFSAR and the TS. The inspectors verified the TM was implemented in accordance with Administrative (ADM) Procedure, 1/2-ADM-2028, "Temporary Modifications," Rev. 3.

The inspectors reviewed the Unit 2 TM, 2-05-003, Rev. 0, "2IAS-1 Jumper." This TM removed check valve 2SAS-5, as well as an adjacent 3 inch diameter pipe spool piece and installed a temporary pipe manifold at the upstream pipe flange of the service air tank discharge check valve, 2IAS-1. The purpose of this TM was to facilitate maintenance on check valve 2IAS-1, while maintaining a service and instrument air in service.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors observed an emergency event training evolution conducted at the control room simulator to evaluate emergency procedure implementation, event classification, event notification, and protective action recommendation development. The event scenario involved plant conditions which warranted declaration of an Unusual Event and subsequent escalation to an Alert event classification. The licensee counted this training evolution for evaluation of Emergency Preparedness Drill/Exercise Performance (DEP) Indicators. The inspectors observed the training critique to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. Additionally, the inspectors verified the DEP performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 2. Additional documents reviewed for this inspection activity are listed in Attachment 1.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 10 Samples)

a. Inspection Scope

During the period April 18-22, 2005, 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 radiologically controlled areas during the Unit 2 refueling outage and Unit 1 power operations. 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 10 samples relative to this inspection area.

Completion of these 10 samples in conjunction with the 11 (power operations related) samples, completed during the period February 14-18, 2005, completes the 71121.01 annual inspection requirement of 21 samples.

Plant Walkdown and RWP Reviews

- During the Unit 2 refueling outage, the inspector identified exposure significant work activities in the Containment Building and Fuel Handling Building. Specific work activities included reactor head installation, containment demobilization, containment air recirculating fan (2HVR-FN201A) repair, and fuel transfer system inspection/repair. 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 2 containment building, and the Unit 1&2 primary auxiliary buildings and fuel handling buildings. With the assistance of radiation protection personnel, 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 setpoints to determine if the setpoints 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 connecting reactor head thermocouples (RWP 205-5019), replacing the fuel transfer system canal blank flange (RWP 205-5020), and removing steam generator maintenance equipment (RWP 205-5015).

- The inspector reviewed RWPs and associated instrumentation and engineering controls for potential airborne radioactivity areas located in the containment and primary auxiliary buildings. 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 the dose assessment methodology for tasks resulting in internal exposures that were less than 50 mrem to confirm the accuracy of the results. Tasks reviewed, involving airborne radioactivity, included thimble tube cleaning, reactor cavity decontamination, and steam generator eddy current testing.

Problem Identification and Resolution

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

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 briefings for selected tasks including reconnecting reactor head thermocouples, containment equipment demobilization, and insulation installation.

High Risk Significant - LHRA and VHRA Controls

- Keys to locked high radiation areas (LHRA) and very high radiation areas (VHRA) stored at the control point were inventoried. Accessible LHRAs were verified to be properly secured and posted during Units 1&2 plant 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 also evaluated the adequacy of prerequisite communications and authorizations.

Radiation Worker Performance

- The inspector observed radiation worker and radiation protection technician performance during containment equipment demobilization, reactor head installation, and connecting reactor head thermocouples. 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 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 - 8 samples)

a. Inspection Scope

During the period April 18-22, 2005, 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 refueling outage. Implementation of these controls was reviewed against the criteria contained in

10 CFR 20, applicable industry standards, and the licensee's procedures. This inspection represents completion of eight (8) samples relative to this inspection area.

Completion of these 8 samples in conjunction with the 7 samples inspected in November 1-5, 2004 (Report Nos. 05000334/2004006 and 05000412/2004006) completes the biennial inspection requirement of 15 samples for 71121.02.

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 containment air recirculating (CAR) fan repair, reactor reassembly, and transfer canal blind flange installation.

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

  The inspector reviewed the 2R11 dose summary reports, detailing worker estimated and actual exposures, through April 21, 2005, for outage-related tasks.

- The inspector evaluated the exposure mitigation requirements specified in RWPs and ALARA Reviews (AR), and compared actual worker cumulative exposure with estimated dose for tasks associated with these activities. The inspector reviewed in detail those work activities whose actual cumulative exposure exceeded 80% of the estimated dose; e.g., reactor head installation, which resulted in a work stoppage until the outage ALARA Committee discussed the cause(s) for the elevated dose with job supervisors and projected what additional dose was needed to complete the task. Jobs reviewed by the inspector included reactor head installation (AR 05-2-17), CAR fan repair (AR 05-2-40), replacing the transfer canal blind flange (AR 05-2-16), steam generator sludge lancing (AR 05-2-43), and replacing supplemental neutron shielding (AR 05-2-44).

  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; attending daily departmental turnover meetings, and an ALARA Committee meeting, regarding elevated dose for reactor head installation.

  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 CAR fan repair and outage mechanical maintenance 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 replanning work, when emergent work or expanded job scope was encountered. The inspector attended an Outage ALARA Committee meeting for reactor head installation, 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 installation and containment demobilization 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, system flushes, and temporary shielding.

Declared Pregnant Workers (DPW)

- The inspector reviewed the procedural controls implemented for declared pregnant workers and reviewed the associated records for three (3) DPWs employed to support the Unit 2 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. Details of this review are contained in Section 4OA2 of this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

1. ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed fifteen (15) Condition Reports, and associated corrective actions, recent Nuclear Quality Assessment field observation reports, and the fourth quarter 2004 Nuclear Oversight Assessment Report to evaluate the threshold for identifying, evaluating, and resolving problems in implementing the ALARA program. This review was conducted against the criteria contained in 10 CFR 20, Technical Specifications, and the licensee's procedures.

b. Findings

No findings of significance were identified.

2. Inservice Inspection

a. Inspection Scope

The inspectors reviewed a sample of condition reports that identified problems in the area of in-service inspection. The inspectors verified that problems were accurately recorded in the condition reports and that the corrective actions taken were appropriate.

b. Findings

No findings of significance were identified.

3. Annual Sample Review (71152 - 1 sample)

a. Inspection Scope

The inspectors selected condition reports (CR) associated with service water system pressure indicator (PI) failures for detailed review. The subject PIs, 107A and 107B, provide a local indication of service water supply pressure to the control room air conditioning unit condensers (local indication only). There were eight PI failures over roughly a five year period, and all appeared to be the result of a pressure surge. The inspectors reviewed the CRs (including CR 05-01247, initiated to evaluate a recent PI failure and to collectively address previous failures) to ensure that the full extent of the issues was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. As part of this review, the inspectors interviewed station personnel, reviewed related documents, and conducted plant walkdowns. The inspectors also evaluated the CRs against the licensee's corrective action program requirements as stated in procedure NOP-LP-2001, "Condition Report Process" and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action".

b. Findings and Observations

No significant findings were identified associated with the reviewed sample; however, the inspectors identified that the licensee's efforts to identify the cause and correct the PI failures have not been fully effective. Specifically, the licensee has not confirmed the source of the apparent pressure surge in the system, nor have they prevented continued PI failures. While some system monitoring was performed coincident with specific activities (e.g., testing, service water pump/system manipulations) to identify whether water hammer conditions existed, pressure transients large enough to damage the PIs did not occur during monitoring. In addition, the licensee developed and implemented a modification that replaced the model of the installed PIs with a type less susceptible to pressure surges. However, subsequent to replacing both PIs, additional failures still occurred.

The inspectors noted that, in response to individual PI failures, the licensee conducted service water system walkdowns to confirm that no additional system components were adversely affected; and that the operability of the service water system was maintained. The inspectors independently conducted a walkdown of portions of the service water system and did not identify related deficiencies. The cause of this deficiency has not been identified and the issue has not been effectively corrected in accordance with procedure NOP-LP-2001 and 10 CFR 50, Appendix B, Criterion XVI. In response to this concern, the licensee initiated CR 05-04530 to address weaknesses related to corrective action effectiveness. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the Enforcement Policy.

4. 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 this section; however, for reviews that entailed large number of CRs, these are more appropriately documented in the Attachment.

b. Findings and Observations

No significant findings were identified associated with the reviewed samples; however, the inspectors identified instances where the licensee's efforts to resolve previously identified problems have not been fully effective. The inspector identified a minor violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," associated with a broken lube oil tubing support on the #1 EDG. Specifically, the inspector determined that inadequate corrective actions were implemented because the apparent, root or contributing causes of the recurrent, broken supports for Unit 1 EDG lube oil tubing (CR-02-00067 from January 2002) were not fully investigated by FENOC, and could have prevented the broken support identified in October 2003, which was subsequently repaired in June 2005. This finding is considered minor because it did not impact operability of the EDG and therefore cornerstone objectives, did not affect performance indicator thresholds, would not become a more safety significant concern, and was not associated with risk assessment and risk management actions. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

5. 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 hard copies of each condition report, attending various daily screening meetings, and when necessary, by accessing the licensee's computerized corrective action program database.

b. Findings

No findings of significance were identified.

6. Semi-Annual Review of PI&R Trends

a. Inspection Scope

The inspectors reviewed site trending results to determine if trending was appropriately evaluated by FENOC. This review covered FENOC's trending program to verify that existing trends were (1) appropriately captured and scoped by applicable departments, (2) consistent with the inspectors' assessment from the daily CR and inspection module reviews (Section 40A2.4 and 5), and (3) not indicative of a more significant safety concern. Additionally, the inspectors verified the performance of site trending against NOP-LP-2001, Rev. 11, "Condition Report Process", and NOBP-LP-2018, Rev. 00, "Integrated Performance Assessment /Trending."

b. Findings

No findings of significance were identified.

7. Permanent Plant Modifications and Evaluation of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed corrective action documents associated with 10 CFR 50.59 and plant modification issues to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the corrective actions for the issues were appropriate. The inspectors also reviewed a sample of audits and self-assessments related to 10 CFR 50.59 and plant modification activities.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

1. Temporary Instruction 2515/150, Revision 3: Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)

a. Inspection Scope

The inspectors reviewed the licensee's activities to detect circumferential cracking of reactor pressure vessel (RPV) head penetration nozzles in response to NRC Order EA-03-009, and as specified by NRC temporary instruction (TI) 2515/150, Rev. 3. This included interviews with visual examination personnel, reviews of qualification records and procedures, observation of the in-process visual examination and review of selected video tape records of the reactor vessel closure head visual examination. The inspectors independently viewed a sample set of 14 out of the total 65 penetrations examined by the plant staff. In accordance with TI 2515/150, the inspectors verified that deficiencies and discrepancies associated with the reactor coolant system structures and the examination process, if identified, would be placed in the licensee's corrective action process. The specific reporting requirements of TI 2515/150 are documented in Attachment 2.

b. Findings

No findings of significance were identified.

2. Temporary Instruction 2515/152, Revision 1: Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)

a. Inspection Scope

The inspectors assessed the effectiveness of the licensee's reactor pressure vessel (RPV) lower head vessel penetration nozzle inspection in detecting small amounts of boric acid deposits on the lower head penetration nozzles and detecting RPV lower head degradation. The inspection consisted of interviews with visual test (VT) examination and engineering personnel. The VT training and qualification records were reviewed to verify the licensee's personnel qualification process adequately prepared the assigned staff to perform the examinations. The inspectors also reviewed the examination procedures to determine the adequacy of the guidance and examination criteria to implement the examination plan.

The inspectors examined samples of the results of bare metal visual (BMV) examination of the 50 bottom mounted instrumentation (BMI) nozzles through the RPV lower head surface, and the annular space around each entry boss perforation of the 50 BMI tubes through the lower head to determine whether there were any signs of leakage of primary coolant through the head-to-tube annulus, along the surface of the tubing, or on the surface of the lower reactor head. From samples of magnified video taken of four quadrants for each tube at each intersection between the lower shell perforations and BMI tubes, the inspectors reviewed the BMV examination results. The inspectors also compared these examination results to the results of the previous BMV examination, conducted in September 2003. The specific reporting requirements of TI 2515/152 are documented in Attachment 3.

b. Findings

No findings of significance were identified.

3. Temporary Instruction 2515/160, Revision 0: Pressurizer Penetration Nozzles and Steam Space Piping Connections In U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

a. Inspection Scope

This inspection consisted of a review of the licensee's in-service inspection activities associated with the Beaver Valley Unit 2 pressurizer penetration nozzles and steam space piping connections made from Alloy 82/182/600 materials to determine whether examination of these components was implemented in accordance with the licensee's response to NRC Bulletin 2004-01. All bare metal visual (BMV) examination activities identified in TI 2515/160 were conducted for Beaver Valley Unit 2.

The inspectors evaluated the implementation of the inspection of pressurizer penetration nozzles and steam space piping connections made from Alloy 82/182/600 materials by direct field observation of as-found conditions at five of the six pressurizer penetration nozzles, reviewing related visual inspection and boric acid corrosion control procedures, reviewing a sample of current and previous non-destructive examination inspection records of the pressurizer penetration nozzles fabricated from Alloy 82/182/600 material, and reviewing digital photographs of the as-found conditions of the pressurizer penetration nozzles. The specific inspection requirements of TI 2515/160 are documented in Attachment 4.

b. Findings

No findings of significance were identified.

4. TI 2515/163 - Operational Readiness of Offsite Power

a. Inspection Scope

The inspector performed Temporary Instruction 2515/163, "Operational Readiness of Offsite Power." The inspector collected and reviewed licensee procedures and supporting information pertaining to the offsite power system specifically relating to the areas of offsite power operability, the maintenance rule (10 CFR 50.65), and the station blackout rule (10 CFR 50.63). The inspectors reviewed this data against the requirements of 10 CFR 50.63; 10 CFR 50.65; 10 CFR 50 Appendix A General Design Criterion 17, "Electric Power Systems"; and Plant TSs. This information was forwarded to the US NRC Office of Nuclear Reactor Regulation for further review.

b. Findings

No findings of significance were identified.

5. Temporary Instruction 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR 20, 71, and Department of Transportation regulations contained in 49 CFR 173. The inspector interviewed licensee personnel and determined that the licensee had undergone refueling/defueling activities between January 1, 2002 and present, but had not packaged and shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

On July 11, 2005, the inspectors presented the inspection results to Mr. William Pearce and other members of his staff. The licensee had no objections to the NRC's observations. The inspector confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT 1: Supplemental Information
ATTACHMENT 2: Reporting Requirements for Temporary Instruction 2515/150, Revision 3: Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Order EA-03-009)
ATTACHMENT 3: Reporting Requirements for Temporary Instruction 2515/152, Revision 1: Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)
ATTACHMENT 4: Inspection Requirements for Temporary Instruction 2515/160, Revision 0: Pressurizer Penetration Nozzles/steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Alberti	Steam Generator Eddy-Current Test Supervisor
G. Cacciani	Staff Nuclear Engineer
J. Clark	Radiation Protection Health Services Technician
J. Clutter	Senior Radiation Protection Technician
T. Cosgrove	Director, Site Engineering
K. Frederick	Senior Consultant
G. Freeh	VT Level III
J. Freund	Supervisor, Rad Operations Support
F. Gardner	Design Engineer
D. Grabski	Staff Nuclear Engineer (ISI Program Owner)
R. Haddock	Shift Manager
T. Heimel	Staff Nuclear Specialist (NDE Level III)
S. Honavec	Supervisor, Plant System Engineering
C. Hrelec	Senior Radiation Protection Technician
J. Lash	Plant Manager
J. Lebda	Senior Nuclear Specialist, Dosimetry
C. Leclrc	Radiation Protection Supervisor
F. Lipchick	Senior Nuclear Specialist
E. Loehlein	Advanced Nuclear Engineer (Alloy 600 Program Owner)
C. Mancuso	Supervisor, Mechanical and Structural Engineering
M. Manoleras	Manager, Design Engineering
D. McBride	System Engineer
F. Oberlitner	Senior Nuclear Engineer
W. Pearce	Vice President, FENOC
R. Pucci	Senior Nuclear Specialist, ALARA Coordinator
M. Ressler	Staff Nuclear Engineer
P. Sena	Manager, Site Operations
B. Sepelak	Supervisor, Regulatory Compliance
J. Sipp	Manager, Radiation Protection
J. Wilbur	Radiation Protection Operations Field Coordinator
W. Williams	Senior Nuclear Engineer (Boric Acid Program Owner)

NRC Personnel:

P. Cataldo	Senior Resident Inspector
G. Smith	Resident Inspector

LIST OF ITEMS, OPENED, CLOSED, AND DISCUSSED

Open/Closed

05000334/2005006-01	NCV	Inadequate Procedure Results in Incorrect Lead Time Constant in the Over Temperature Delta Temperature Reactor Trip Function (Section 1R12)

LIST OF DOCUMENTS REVIEWED

Sections 1R02 and 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

Engineering Change Packages

ECP-02-0009	Revise the Time the Emergency Diesel Generators Can Operate Without Cooling to Appendix R, Rev. 1
ECP-02-0150	Replacement of Borg Warner Actuators, Rev. 0
ECP 03-0563	2SWS-STRM Service Water Strainer Replacement, Rev. 1
ECP 04-0219	Unit 1 River Water System Minimum Operating Point Changes, Rev. 0
ECP-04-0447	Changes to Containment Accident Analysis Parameters, Rev. 0

10 CFR 50.59 Safety Evaluations

SE 03-03200	Install Pipe Spools with Flanges to Facilitate Removal of 2SWS-106 and 2SWS-107, Rev. 0
SE 03-04080	Small Break LOCA Re-Analysis and UFSAR Changes, Rev. 0
SE 03-04128	Small Break LOCA PCT Re-Analysis, 50.46 PCT Rack-Up Sheet, and UFSAR Changes, Rev. 0
SE 04-02259	Temporarily Remove and Reinstall Roof Plug from the Unit 2 Safeguards Building while in Mode 1, Rev. 0
SE 04-02359	Elimination of Post-DBA Hydrogen Control System Requirements/Abandon-in-Place BV1 Hydrogen Recombiners, Rev. 0
SE 04-04434	Construction of New Access Openings in the Intake Structure, Rev. 0

10 CFR 50.59 Safety Screens

03-02930	Borg Warner Actuator Replacement for Unit 2 Modulating Valves, Rev. 0
03-03434	Loss of All Emergency 4 kV Power, Rev. 0
03-03796	Filling and Venting the Safety Injection System, Rev. 0
03-04352	Credits Manual Actions to the Service Water Seal Strainer (2SWS-STRM47 and 48) and Adds a Repair Activity to Restore Power to the Strainer Motor, Rev. 1
04-00192	4160 V Emergency Bus 1AE-1E7 Overcurrent Trip, Rev. 0
04-01042	BV2 Main Transformer Fire Detection System Modification, Rev. 0
04-01150	Unit 1 Charging Pump Cubicle “C” Return Air Register and Damper Replacement, Rev. 0

04-01170	GE AK Breaker and Trip Device Upgrade - Unit 1 480V Buses, Rev. 0
04-01530	Unit 1 River Water System Minimum Operating Point Changes, Rev. 0
04-03837	Upgrade Containment Sump Penetrations and Gaps, Rev. 0
04-03629	Changes to Containment Accident Analysis Parameters, Rev. 0
04-03986	Opening Steam Generator Main Feedwater Isolation Valve Without Power, Rev. 0

Procedures

1/2MI-75-Packing-1M	Valve Packing Instruction, Rev. 8
1OM-53A.1.2-T	Opening One Steam Generator Main Feedwater Isolation Valve Without Power, Rev. 0
NOP-CC-2003	Engineering Changes, Rev. 6
NOP-CC-2004	Design Interface Reviews and Evaluations, Rev. 2
NOP-LP-4003	Evaluation of Changes, Test, and Experiments, Rev. 1

Calculation Documents

8700-DMC-1352	Emergency Diesel Generator Operating time with Loss of River Water, Rev. 0
8700-DMC-3136	River Water Pump Minimum Operating Point, Rev. 3
8700-US(B)-269	Beaver Valley Power Station Unit 1 Recirculation Spray Heat Exchanger Analysis Using TREMOLO PC 3.0.1, Rev. 0
DSC-6756	Installation of New Access Openings and Opening Covers in Intake Structure Elevation 705' Slab, Rev. 0

Condition Reports

01-03263	04-06673	05-00370	05-04157*
02-06899	04-06674	05-00488*	05-04158*
02-06954	04-06896	05-00489*	05-04220*
02-07068	04-07214	05-01875	05-04488*
03-08148	04-08160	05-02443	05-04489*
03-08641	04-08839	05-03103	05-04492*
04-01109	04-09568	05-03523	05-04517*
04-01664	04-09591	05-04038*	05-04519*
04-01705	04-09804	05-04056	
04-01761	04-09823		

(“*” denotes condition reports that were generated as a result of NRC inspection)

Drawings

1485-011-01	Beaver Valley Power Station 3" Strainer Assembly

Self-Assessments and QA Audits

SA-05-080	10 CFR 50.59 Reviews, January through March 2005
SA-05-089	Engineering Changes, January through March 2005

Miscellaneous Documents
Beaver Valley Unit 1 and 2 Probabilistic Risk Assessment Update Report
Company Nuclear Review Board 50.59 Subcommittee Meeting Minutes from May 11, 2005
Plant On-Site Review Committee Meeting Minutes from December 3, 2004

Work Orders

Order 200025024, 2SVS-HCV 104-OPER, RH Release Valve Actuator
Order 200024991, 2SVS-PCV 101A-OPER, 21A SG Atmos STM Dump Valve
Order 200025016, 2SVS-PCV 101C-OPER, 21C SG Atmos STM Dump Valve
Order 200024998, 2SVS-PCV 101B-OPER, 21B SG Atmos STM Dump Valve
Order 200029762, 2FWE-HCV 100B-OPER, 21C SG Aux FW Throttle Valve A
Order 200029768, 2FWE-HCV 100D-OPER, 21B SG Aux FW Throttle Valve A
Order 200029774, 2FWE-HCV 100F-OPER, 21A SG Aux FW Throttle Valve A

Section 1R04: Equipment Alignment

Drawings

8770-RM-421-1, “Main Steam System,” Rev. 15
8770-RM-424-2, “Feedwater System,” Rev. 11

Procedures

1OM-21.3.B.1, “Valve List - 1MS,” Rev. 15
1OM-21.3.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

Miscellaneous

1DBD-24B, Design Basis Document for Auxiliary Feedwater, Rev. 7

Section 1R06: Flood Protection Measures

CR-05-04492, “Packing Leakage at 2SWS-STRM48"
CR-05-04519, “Evaluate Need for Drip Collecter On Service Water Strainer”
Surveillance Test 1/2OST-30.21B, Rev.1, “Group 2 Flood Door Seal System Operability Check”
Operating Manual Figure No. 41D-2, Issue 5, “Turb & Service Bldg & Yard Drains”
Calculation 8700-DMC-3443, Rev. 2, “BVPS Intake Structure Cubicles Internal Flood Analysis”
Calculation 12241-211-B265, Rev. 6, “Flooding Analysis Outside Containment”

Section 1R07 Heat Sink Performance

1/2 -ADM-2106, “River/Service Water System Control and Monitoring Program.” Rev 0

Generic Letter 89-13, “Service Water System Problems Affecting Safety Related Equipment.” Rev 0

<u>Section 1R08: Inservice Inspection</u>

<u>Procedures</u>
1/2-ADM-2112, Boric Acid Corrosion Control, Rev. 3
ISIE1-8 BVPS, Unit 2 Steam Generator Examination Guidelines, Rev. 9.
MRS-SSP-1027-DLW/DMW, Mechanical Ribbed Plugging of Steam Generator Tubes, Rev. 2
MRS-SSP-1510, RPV Head Penetration Remote Visual Inspection for BV Unit 2, Rev. 1
MRS-SSP-1520, BMI Visual Inspection for Beaver Valley Unit 2, Rev. 1
NDE-UT-321, Ultrasonic Examination of Austenitic Pipe Welds, Rev. 3
NDE-VT-510, Visual Inspection for Evidence of Boric Acid Leakage, Revision 11
NOP-ER-2001, Boric Acid Corrosion Control Program, Rev. 4

<u>Other Documents</u>
A5.5511E, Eye Examination Certification Form
Calculation 10080-DMC-0849, Rev. 0, EDY Calculation for the BVPS Unit 2 RPV Head
Core Location Map, page 19 of SSP-1510
ECT Data Analyst Training & Performance Demonstration for BV Unit 2 Refuel Outage 11
Letter L-03-190, dated 12/9/2003 for BV U2 RPV 60 day inspection report for RFO-10
Order 200093842, BV-2RCS-REV. 21, Reactor Vessel, Rev. 0
Personnel Qualifications, 2R11 RPV Head Inspection Qualifications and Certification, 4/1/05
QC Receiving Inspection Records (4/1/05 & 4/4/05) - RPV inspection personnel qualifications
RTL A5.611A, Certificate of Qualification Form
SG-SGDA-05-5, BVPS Unit 2 2R11 Steam Generator Degradation Assessment, 3/9/05

UT Pipe Weld Examination Report UT-05-1001, 2CH-356-1B (Butt Weld - 4" pipe to reducing elbow weld on the discharge of Charging Pump 2CHS-P21A), 4/6/05

UT Pipe Weld Examination Report UT-05-1002, 2CH-279-F500A (Butt Weld - 3" pipe to orifice weld on the mini-flow line of Charging Pump 2CHS-P21A), 4/6/05

UT Pipe Weld Examination Report UT-05-1003, 2QSS-004-15B (Butt Weld - 8" pipe to elbow weld on the discharge piping of Quench Spray Pump 2QSS-P21A), 4/6/05

<u>Condition Reports</u>
03-09784
03-09907
03-09962
03-09991
03-10166
03-10596
03-10642
04-07864
04-08066
04-08080

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

2DBD-01, “Reactor Control and Protection System,” Rev. 7
Condition reports (CR) 02-08801, 03-10778, 05-01209, 05-03442, 05-03507

Maintenance rule (a)(1) disposition review, Unit 2 Process Control System
Maintenance rule basis document, Unit 2 Process Control System, Rev. 3
1/2-ADM-2114, “Maintenance Rule Program Administrative Procedure,” Rev. 2

Section 1R15: Operability Evaluations

Condition Reports
05-03772 04-03207 05-03674 05-03161 05-04066

Procedures and other Documents
1/2-ADM-2113, Rev. 1, “Operability Determination and Basis For Continued Operation”
1OM-56C.4 Rev. 2, C.4.A Rev. 4, C.4.C Rev. 23, C.4.D Rev .25, and C.4.F-11 Rev. 7, “Alternate Safe Shutdown From Outside Control Room”
2OST-36.2, Revs. 46 & 47, “Emergency Diesel Generator (2EGS*EG2-2) Monthly Test”
Unit 1 Drill - 1OM56C Alternate Safe Shutdown from Outside the Control Room (time line and conclusions for drill performed on March 23, 2001)
Calculation 8700-DMC-3509, Rev. 0, Add. 1, “HHSI Pump Oil Temperature Following Loss of RW”
BVPS Updated Final Safety Analysis Report Sections 15.1.5 and 15.2.8
Inservice Testing (IST) Program for Pumps and Valves, Issue 2, Revision 11
Unit 1 Operating Manual Figure No. 36-3, Issue 2, “Lube Oil System.”

Section 1R19: Post-Maintenance Testing

Work Orders
200026493 200026497 200071838 200065777

Condition Reports
05-03693 05-03732 05-03803 05-04164

Procedures
NOP-WM-1005, Rev 0, “Work Management Order Testing Process”
1/2-ADM-2111, Rev 2, “Inservice Testing (IST) Program Administrative Process”
DG-0026, Rev 0, “Post Maintenance Test Requirements Desktop Guide”
2MSP-36.19-M,”#1 Emergency Diesel Generator Inspection” Issue 4 Rev 9
2MSP-36.29-M ,”#1 Emergency Diesel Generator Filter, Strainer, Heat Exchanger and Woodward Governor Maintenance” Issue 4 Rev 14
2OST-11.2,”Low Head Safety Injection Pump(2SIS*P12B) Test” Rev 21
2OST-11.14A,”Low Head Safety Injection System Full Flow Test” Rev 14

Other
Supplier Verification Report #10262-S010, “River Water Pump Wet End Refurbishment”

Section 1R20: Refueling and Outage Activities

NOP-OP-1005, “Shutdown Safety”, Rev 8
2RST-2.1,”Initial Approach to Criticality after Refueling” Issue 1 Rev 7

2RST-2.2,”Core Design Check Test” Issue 1 Rev 7
2OM-50.4.D,”Reactor Startup from Mode 3 to Mode 2" Rev 45
2OM-50.4.M,”Station Startup-Mode 5 to Mode 3" Rev 7
Estimated Critical Position Data Sheets.
2CMP-47--EquipHatch-1-MME, Rev. 6, “Removal and Reinstallation of Containment Equipment Hatch and Escape Air Lock.”
2OM-52.4.R.1.F, Rev. 4, “Refueling Station Shutdown From 100% Power To Mode 5.”
2OM-6.4.I, Rev. 18, “Draining The RCS For Refueling.”
NOBP-OM-4010, Rev. 01, “Restart Readiness For Plant Outages.”
BVBP-SITE-0045, Rev. 00, “Restart Readiness For Plant Outages.”
2OST-47.2B, Rev. 2, “Containment Close Out Inspection.”
2MSP-9.04-M, Rev. 4, “Containment Sump (2DAS-TK204) Inspection,” and associated work order 200095842.

Section 1R22: Surveillance Testing

BVPS Unit 1 Inservice Testing (IST) Program for Pumps and Valves, Rev. 4
Technical Specification 3/4.7.4, “Reactor Plant River Water System”
2OST-49.2, “Shutdown Margin Calculation” Rev 13

Section 1EP6: Drill Evaluation

1OM-6.4IF, Attachment 1, “Instrument Failure [LT-1RC-459(460)(461)],” Rev. 9
1OM-53A.1.E-0, “Reactor Trip or Safety Injection,” Issue 1C - Rev. 6
1OM-53A.1.1-K, “Verification of Automatic Actions,” Issue 1C - Rev. 2
1OM-53A.1.E-3, “Steam Generator Tube Rupture,” Issue 1C - Rev. 5
1/2-ADM-1111, “NRC EPP Performance Indicator Instructions,” Rev. 1
EPP/I-1A, “Recognition and Classification of Emergency Conditions,” Rev. 5
1/2-EPP-I-3, “Alert,” Rev. 19

Sections 2OS1: Access Control to Radiologically Significant Areas

Procedures

1/2-ADM-1601, Rev 10	Radiation Protection Standards
1/2-ADM-1611, Rev 6	Radiation Protection Administrative Guide
1/2-ADM-1621, Rev 3	ALARA Program
1/2-ADM-1630, Rev 6	Radiation Worker Practices
1/2-ADM-1631, Rev 5	Exposure Control
1/2-HPP-3.02.003, Rev 3	Decontamination Control
1/2-HPP-3.02.004, Rev 4	Area Posting
1/2-HPP-3.05.001, Rev 3	Exposure Authorization
1/2-HPP-3.07.002, Rev 3	Radiation Survey Methods
1/2-HPP-3.07.013, Rev 2	Barrier Checks
1/2-HPP-3.08.001, Rev 8	Radiological Work Permit
1/2-HPP-3.08.005, Rev 4	ALARA Review Program
BVBP-RP-0003, Rev 2	Dosimetry Practices

Miscellaneous Reports

Unit 1 and Unit 2 Radiation Protection Department Shift Logs

Section 2OS2 ALARA Planning and Controls

Procedures

1/2-ADM-1601, Rev 10	Radiation Protection Standards
1/2-ADM-1611, Rev 6	Radiation Protection Administrative Guide
1/2-ADM-1621, Rev 3	ALARA Program
1/2-ADM-1630, Rev 6	Radiation Worker Practices
1/2-ADM-1631, Rev 5	Exposure Control
1/2-HPP-3.02.003, Rev 3	Decontamination Control
1/2-HPP-3.02.004, Rev 4	Area Posting
1/2-HPP-3.05.001, Rev 3	Exposure Authorization
1/2-HPP-3.07.002, Rev 3	Radiation Survey Methods
1/2-HPP-3.07.013, Rev 2	Barrier Checks
1/2-HPP-3.08.001, Rev 8	Radiological Work Permit
1/2-HPP-3.08.005, Rev 4	ALARA Review Program
BVBP-RP-0003, Rev 2	Dosimetry Practices

Section 4OA2 Problem Identification and Resolution

Nuclear Oversight Reports
NQAR Field observations: BV320052070, BV120052004, BV120052047, BVBV220052005, BV220052059, BV320052028, BV220052085, BV220052076

Fourth Quarter 2004 Nuclear Oversight Assessment Report

Condition Reports
05-01177, 05-02312, 05-02313, 05-02614, 05-02268, 05-02291, 05-02254, 05-02773, 05-02542, 05-02545, 05-02614, 05-02648, 05-02816, 05-02869, 0407545

ALARA Council Meeting Minutes
Meeting Nos: 05-06, 05-05, 05-04, 05-03, 05-02

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ADM	Administrative Procedure
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
AR	ALARA Reviews
ASME	American Society of Mechanical Engineers
BCO	Basis for Continued Operation
BMI	Bottom Mounted Instrumentation
BMV	Bare Metal Visual
BVPS	Beaver Valley Power Station

CAR	Containment Air Recirculating fan
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	Condition Reports
CRDM	Control Rod Drive Mechanism
DBD	Design Basis Documents
DEP	Drill/Exercise Performance
DPW	Declared Pregnant Workers
ECP	Engineering Change Packages
ECT	Eddy-Current Testing
EDG	Emergency Diesel Generator
EDY	Effective Degradation Years
EPRI	Electric Power Research Institute
FENOC	First Energy Nuclear Operating Company
GL	Generic Letter
HHSI	High Head Safety Injection
HRA	High Radiation Area
I&C	Instrumentation and Control
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
LHRA	Locked High Radiation Areas
MS	Main Steam
NCV	Non-cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	Operability Determinations
OPDT	Over Pressure Delta Temperature
OTDT	Over Temperature Delta Temperature
OWA	Operator Work-Arounds
PCE	Personnel Contamination Event Report
PI	Pressure Indicator
PMT	Post-Maintenance Tests
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
SSPS	Solid State Protection System
TAVG	Average Coolant Temperature
TI	Temporary Instruction
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

UT	Ultrasonic Test
VHRA	Very High Radiation Area
VT	Visual Test
WO	Work Order

ATTACHMENT 2

Reporting Requirements for Temporary Instruction 2515/150, Revision 3: Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Order EA-03-009)

The questions listed in the Reporting Requirements for TI 2515/150 are addressed below for the visual (video) examination method used during the outage.

a.4. The visual test (VT) examinations were performed by individuals who were qualified as Level II, VT-2 examiners with specific training on the appearance of boric acid deposits from degradation found at other plant sites. Their training and qualification certification were reviewed and documented in the task related records.

a.2. The VT examination was performed in accordance with approved procedure MRS-SSP-1510, Rev. 1.

a.3. The equipment used and methods of tracking the examination locations were capable to identify, disposition, and resolve deficiencies.

a.4. The visual resolution and technique were capable of identifying the PWSCC and/or RPV head corrosion phenomena described in the Order. The video and written records of the process provide the ability to review the condition of the four quadrants around each of the Control Rod Drive Mechanisms (CRDM).

b. The physical condition of the RPV head (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions) was, with a few minor exceptions, clean.

c. Small boron deposits, as described in NRC Bulletin 2001-01, could be identified and characterized.

d. No material deficiencies (i.e., cracks, corrosion, etc.) items of concern were identified that required repair. At several locations where areas of deposits were present, these were compared to the visual presentations made during the examination of the previous outage and determined to be acceptable.

e. The insulation geometry provided some impedance to examination of the outer positioned CRDMs, although a full view of these was obtained.

f. The site prepared a calculation 10080-DMC-0849 to establish the Effective Degradation Years per Order EA-03-009 for the RPV head. The basis for the temperatures for susceptibility ranking calculation were measurements from the Beaver Valley Unit 1 RPV head.

g. Only visual examination was performed during this April 2005 outage.

h. Procedures existed to identify potential boric acid leaks from pressure-retaining components above the RPV head (Part 6.0 of Procedure 1510).

I. There were no indications of boric acid flow from above the RPV head, and therefore, no follow-on examinations were required.

Summary. No issues of boric acid leakage were identified by the visual examination. The visual inspection process was monitored by the site staff and the final documentation including the visual records were reviewed by the site Level III for visual examinations.

ATTACHMENT 3

Reporting Requirements for Temporary Instruction 2515/152, Revision 1: Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)

The questions listed in the Reporting Requirements for TI 2515/152 are addressed below for the visual (video) examination method used during the outage.

a.1. The visual test (VT) examination was performed by qualified and knowledgeable personnel with certification to ASME, Section XI, Level II and Level III for visual examiners. In addition, Level II and Level III examiners had received training in this type of inspection. The training included a review of industry experiences, lessons learned, inspection results and procedure requirements.

a.2. The VT examination was performed using an approved and adequate procedure. The procedure specified the extent of the inspection required, provided documentation requirements and provided clear inspection standards and acceptance criteria on which personnel were trained.

a.3. The examinations were adequate to identify, disposition, and resolve deficiencies. A detailed systematic visual examination of tubing quadrants was made at each penetration. The VT examination documentation included a video and written record.

a.4. The examinations performed were capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion.

b. The VT examination method was sufficient to identify and characterize small boric acid deposits, if present.

c. The VT examination was conducted with a video camera and associated monitor.

d. The VT examination provided complete coverage around the nozzles (each nozzle was divided into four quadrants).

e. The physical condition of the RPV lower head was acceptable and did not have any boric acid deposits at the interface between the vessel and the penetrations. As indicated during the prior RPV lower head inspection (NRC Inspection 50-412/2003-004; Fall 2003), there was a thin white residual powdery substance on the lower head surface. There was also evidence of slight discoloration markings in the annulus between the tubing and boss on 29 of the 200 quadrants examined (50 BMIs, four quadrants each), requiring additional evaluation. The licensee attributed this substance to residue from protective coating (e.g., tape) that was likely applied during RPV head initial fabrication.

Following the performance of the VT examination during this outage, the licensee mechanically cleaned the discolored areas (31 quadrants, 19 BMI penetrations), and then reexamined the cleaned areas to provide a new baseline for subsequent inspections.

f. No material deficiencies (i.e., cracks, corrosion) were identified that required repair.

g. There were no significant impediments to the examination.

h. There were no required follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV lower head.

I. The licensee did not take any samples of the discolored areas during this outage. However, they sampled and chemically analyzed the substance identified during the prior refueling outage, which identified no evidence of boron.

j. As stated in item e. above, the licensee mechanically cleaned previously identified white residue/discoloration from the lower head and BMI tubing. Following the cleaning, these areas were reexamined.

k. The licensee concluded that the previously identified white residue and discoloration were residue from protective coating (e.g., tape) that was likely applied during RPV lower head fabrication/shipping, based on chemical analysis and location of the residue (none was identified at the annulus between the BMI tubing and boss).

Summary. No issues of boric acid leakage were identified by the visual examination. The visual examination process was monitored by the site staff and the final examination records were reviewed by the site Level III for visual examinations.

ATTACHMENT 4

Inspection Requirements for Temporary Instruction 2515/160, Revision 0: Pressurizer Penetration Nozzles/steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

The items listed in the Inspection Requirements for TI 2515/160 specific to the Bare Metal Visual Examination (Section 02.03) are addressed below.

At Beaver Valley Unit 2, a bare metal visual (BMV) examination was performed of 100% of the pressurizer penetration nozzles fabricated from Alloy 82/182/600 material to verify the absence of boric acid crystals and to verify the integrity of the pressurizer shell. A total of six penetration nozzles fabricated from Alloy 82/182/600 material were examined during this Unit 2 refueling outage. The only impediment to the BMV examinations was the installed insulation around the six penetrations. After removing the insulation, the observed penetrations were free of material that could adversely affect viewing the pressurizer penetrations. No evidence of boric acid leakage or material deficiencies (i.e., cracks, corrosion, etc.) were identified at any penetration nozzle. The inspections conducted at Beaver Valley Unit 2 by qualified NDE technicians were consistent with the licensee's bulletin response dated July 27, 2004, and supplemental response dated September 29, 2004.

a. The examination method used to inspect the Alloy 82/182/600 pressurizer penetrations was visual test (VT) examination. The qualification records of the individual performing the VT examinations were reviewed to ascertain whether the qualification records properly reflected the employer's name, person certified, activity qualified to perform, effective period of certification, signature of employer's designated representative, basis used for certification, annual examination of visual acuity and color vision and periodic recertification.

b. Direct bare metal VT examinations of the Alloy 82/182/600 pressurizer penetrations was performed by a licensee NDE technician certificated to ASME, Section XI, Level II or Level III for visual examiners and qualification and training requirements described in ASME Section XI. The NDE technician performing the VT examinations was qualified and in conformance with licensee inspection procedures. Therefore, the individual was capable of identifying and characterizing small boron deposits around the pressurizer penetration nozzles. As found conditions of the six penetrations were also recorded by digital photographs.

c. The examination procedures for boric acid corrosion control and visual inspection for evidence of boric acid leakage were adequate to identify and characterized, small boric acid deposits representing reactor coolant leakage, in the penetration nozzle or steam space piping components.

d. The inspectors did not directly observe the licensee's performance of the VT examinations. All six pressurizer penetration nozzle inspections were conducted by a qualified licensee NDE technician using direct visual examinations. The VT examinations of the six pressurizer penetration nozzles were implemented in conformance with licensee inspection procedures and ASME code requirements. The boric acid corrosion control and visual inspection procedures provided specific actions to be implemented if boric acid deposits were identified. The inspection procedures were determined to be adequate to identify, resolve, and dispose of identified deficiencies.

e. A sample of current and previous NDE records of the pressurizer penetration nozzles fabricated from Alloy 82/182/600 material were examined. Digital photographs of the as-found condition of the six pressurizer penetration nozzles VT inspected during this refueling outage were also examined by the inspectors.

f. The inspectors independently examined the physical condition of five Alloy 82/182/600 penetration nozzles and steam space piping components. The nozzles were free of debris, dirt, and boron deposits. However, the PORV nozzle had a ring of white residue below the heat trace and a chemical analysis was conducted which showed no indication of boric acid in the sample. The only viewing obstruction was that insulation had to be removed from the six penetration nozzles to perform the direct VT visual examinations.

g. Six penetration nozzles were visually examined 360E around the circumference and no boric acid deposits were identified.

h. No boric acid deposits were identified.

i. No anomalies, deficiencies, or discrepancies associated with reactor coolant system structures or the examination process were identified.