July 28, 2006

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

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2006003 AND 05000412/2006003

Dear Mr. Lash:

On June 30, 2006, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Beaver Valley Power Station Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 24, 2006, with you and other members of your staff.

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

Based on the results of this inspection, the NRC has identified one (1) NRC-identified finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issue has been entered in the corrective action program, the NRC is treating the finding as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the findings in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Beaver Valley.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

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

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

Enclosures: Inspection Report 05000334/2006003; 05000412/2006003
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-334, 50-412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2006003 and 05000412/2006003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	April 1, 2006 through June 30, 2006
Inspectors:	P. Cataldo, Senior Resident Inspector D. Werkheiser, Resident Inspector G. Smith, Resident Inspector A. DeFrancisco, Reactor Inspector W. Holland, NRC Consultant, NRR P. Kaufman, Senior Reactor Inspector J. Lilliendahl, Reactor Inspector T. Moslak, Health Physicist T. O'Hara, Reactor Inspector L. Scholl, Senior Reactor Inspector R. Treadway, Resident Inspector
Approved by:	R. Bellamy, Ph.D., Chief Reactor Projects Branch 7 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2006003, IR 05000412/2006003; 04/01/2006 - 06/30/2006; Beaver Valley Power Station, Units 1 & 2; Problem Identification and Resolution.

The report covered a 3-month period of inspection by resident inspectors, regional reactor inspectors, and a regional health physics inspector. 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: Barrier Integrity

C Green. The inspectors identified an NCV of 10 CFR 50, Appendix B, Criterion 16, for failure to perform adequate operability evaluations for degraded components to assure off-site dose consequences are bounded in the radiological safety analysis for a Steam Generator Tube Rupture (SGTR) event. Specifically, some barrier integrity components (HCV-1MS-104 and 1MS-26, Atmospheric Steam Dump Valve(s), and Steam Generator Safety Valves) were degraded (leaking) and FENOC did not quantify and evaluate the current leakage regarding additional radiological dose consequences during a design basis accident (SGTR event). The licensee entered this deficiency into their corrective action program, assessed the magnitude of additional steam leakage that would be permitted before licensing basis dose results are exceeded, and determined the present leakage is bounded by current analyses.

This finding is more than minor because it was associated with the Structures, System, and Component (SSC), and Barrier Performance Attribute of the barrier integrity cornerstone and affected the objective of providing reasonable assurance that the physical design barrier (containment) protected the public from radionuclide releases caused by accidents or events (SGTR). The finding is of very low safety significance because although degraded, the leaking residual heat release valve and other components (e.g., safety valves and atmospheric dump valves) are not important to Large Early Release Frequency (LERF) and do not affect Core Damage Frequency (CDF). The cause of this finding is related to the corrective action program component of the Problem Identification and Resolution (PI&R) cross-cutting area, in that degraded components were not adequately evaluated to assure proper operability was determined. (Section 4OA2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period shut down for a significant refueling outage during which the reactor vessel closure head and all three steam generators were replaced. On April 19, 2006, reactor startup and low-power physics testing commenced, with 100% full power reached on April 23rd. On May 19th, power was reduced to 10%, and the turbine-generator taken off-line to perform turbine-shaft balancing. The Unit returned to full power on May 21st. On May 26th, a failure within Solid State Protection System (SSPS) required a shut down to comply with Technical Specifications (TS). On May 27th, the SSPS was returned to service, the reactor startup commenced, and full power was reached on May 28th. The Unit operated at or near full power for the remainder of the inspection period.

Unit 2 began the inspection period operating at 100% power, but automatically tripped off-line on April 2nd, due to a failure associated with the main unit generator exciter. Following repairs, the Unit returned to full power on April 7th. On April 11th, operators entered TS 3.0.3 and reduced power to 18%, due to an inadvertent wetting of the charcoal filter bed of the Supplemental Leak Collection and Release System. Following completion of repairs, the Unit was returned to full power on April 12th. The Unit down-powered for one hour to 86% on June 24th for turbine control valve testing. Upon return to full power, it continued to operate at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 samples)

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 hurricanes. The inspectors conducted interviews with various station personnel to gain insights into the station's hurricane 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 that the licensee was appropriately identifying and resolving weather-related equipment problems. In addition, the inspectors reviewed the readiness of the general area ventilation system (System 44F) during hot weather, which supports the Unit 1 emergency diesel generators and other associated loads.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns (4 samples)

a. Inspection Scope

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

- Unit 2 Emergency Diesel Generator 2-2 on May 10, 2006
- Unit 1 ‘A’ Low Head Safety Injection on May 23, 2006
- Unit 2 ‘B’ Quench Spray on June 5, 2006
- Unit 2 Chemical and Volume Control system on June 28, 2006

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 12 samples)

a. Inspection Scope

- Unit 1 (Fire Area CR-2, CR-4)
- Unit 1 (Fire Area CS-1)
- Unit 1 (Fire Area DG-1, DG-2)
- Unit 1 (Fire Area ES-1, ES-2)
- Unit 2 (Fire Area DG-1, DG-2)
- Unit 2 (Fire Area SB-1, SB-2)
- Unit 1 & Unit 2 (Fire Area IS-2)

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. 13. This review included FENOC’s control of transient combustibles and ignition sources; material condition of fire protection equipment including fire detection systems, water-based fire suppression systems, gaseous fire suppression systems, manual firefighting equipment and capability, passive fire

protection features, and the adequacy of compensatory measures for any fire protection impairments. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the licensee's internal flood protection measures for the Unit 2 deep pit area located in the safeguards building. This area is important from a risk significance standpoint since it contains pumps and valves associated with both trains of the Unit 2 Recirculation Spray System (RSS). This review included a plant walkdown and inspection of the deep pit area to evaluate hatch seals used to mitigate external flood sources, level alarm instrumentation, and the condition of penetrations that could communicate flood sources to the room. Additionally, the inspectors reviewed design information contained in the FSAR, Design Basis Documents, and applicable surveillance and operating procedures. Documents reviewed are listed in the Attachment.

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 thermal performance test associated with the Unit 1 'B' Charging pump lube oil cooler conducted on May 18, 2006, in accordance with 1BVT-2.30.7, "Charging Pump Lube Oil Cooler [1CH-7A,B, or C] Heat Exchanger Thermal Performance Testing," Rev. 0. The review included an assessment of the testing methodology and verified consistency with Electric Power Research Institute document TR-107397, "Service Water Heat Exchanger Testing Guidelines," March 1998. The inspectors reviewed the results against applicable acceptance criteria, and verified the inspection was consistent with Generic Letter 89-13,"Service Water System Problems Affecting Safety Related Equipment."

b. Findings

There were no findings of significance identified.

1R08 Inservice Inspection (71111.08P - 4 samples)

a. Inspection Scope

The inspector assessed the inservice inspection (ISI) activities using the criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The inspector reviewed documentation and interviewed personnel to verify that the activities were performed in accordance with the ASME Boiler and Pressure Vessel Code Section XI requirements. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The inspector reviewed a sample of condition reports to assess the licensee's effectiveness in problem identification and resolution. The specific ISI activities selected for review included:

- A visual examination (VT-2) of the pressurizer top head nozzles (A, B, C and D) and the spray nozzle (E)
- Boric acid corrosion control leak inspection results and mode hold resolution forms for the 'A' RHR pump discharge check valve and the RCP 1C seal supply drain valve
- UT erosion/corrosion examination of piping line 3"-RC-107-1502-Q1, located between motor-operated valve MOV-IRC-535 and the power-operated relief valve block valve, and associated condition report (CR)
- A liquid penetrant examination of an ASME Section XI IWB Category B-J pressure-retaining weld, DG-56-3-F-03, and associated condition report, and a Ten-Year Plan Change Request

The inspector interviewed the boric acid corrosion control program owner and sampled the photographic database of all examined areas to verify that visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components. The inspector also reviewed a sample of items on the mode hold list, as well as the procedures being used for visual inspection for evidence of boric acid leakage. The inspector confirmed that sampled condition reports were assigned corrective actions consistent with the requirements of the ASME Code and 10 CFR 50 Appendix B, Criterion XVI.

In addition, the inspector reviewed non-destructive examination (NDE) certifications for inspectors performing the above listed examinations, and for the Level III reviewer. The inspector also reviewed the procedures used to complete the reactor vessel lower head penetration inspections (bottom-mounted penetrations). The inspector observed the accessibility to the penetrations and the lower head insulation design factored into the design of the newly acquired camera and lighting equipment. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed Unit 1 licenced operator requalification training conducted on June 9, 2006. The inspectors evaluated licensed operator performance regarding command and control, implementation of normal, annunciator response, abnormal, and emergency operating procedures, communications, technical specification review and compliance, and emergency plan implementation. The inspectors evaluated the licensee staff training personnel to verify that deficiencies in operator performance were identified, and that conditions adverse to quality were entered into the licensee's corrective action program for resolution. The inspectors reviewed simulator physical fidelity to assure the simulator appropriately modeled the plant control room.

The inspectors reviewed the Unit 2 lesson plan for the Module 4 current licensed operator requalification training period and observed sessions that covered Section 3 of the Improved Standard Technical Specifications. The inspectors verified recent plant and/or industry operating experience was incorporated into training activities. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12 - 3 samples)

a. Inspection Scope

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

- CR 05-07930, "Incomplete Actuation of K610A SSPS Slave Actuation Relay"
- CR-06-00860, "MCC1-E12 Feeder Breaker Fails To Auto-Close During 1OST-36.4"

- CR 06-03726, “HCV-1MS-104 and 1MS-26 Leak-by”

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 8 samples)

a. Inspection Scope

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

- On April 10, 2006, Unit 1 entered a planned “yellow” risk status due to the implementation of ECP No. 02-0269, Rev. 1, “Piping Changes to 1EE-P-2A and 2B Lube Oil Circulation Pump Discharge Lines” for the Unit 1 No. 1 EDG.

- On May 4, 2006, operators on Unit 2 identified a condition where local control of valve 2-SIS-HCV868A was lost. The valve is operated to establish an alternate charging path when required by emergency procedure ES-1.2, Post LOCA Cooldown and Depressurization. Work Order 200208344 was accomplished to correct the deficiency.

- On May 4, 2006, Unit 2 entered a planned “yellow” risk status to perform 2MSP-1.04-1, Solid State Protection System Train A Bi-Monthly Test,” Issue 4, Rev. 30; and to replace four universal logic cards in accordance with Work Order 200205488.

- On May 10, 2006, the inspectors reviewed the licensee evaluation (Green risk) and activities associated with an “emergent” test on the dedicated emergency feedwater pump, performed in accordance with 1OST-24.7, Rev. 13, “Dedicated Auxiliary Feed Pump [1FW-P-4] Test.”

- May 31, 2006, Unit 1 entered a planned "yellow" risk due to the replacement of the Refueling Water Storage Tank (RWST) Return Isolation Valve, PC-47.

- On June 5, 2006, the inspectors reviewed the licensee evaluation and preparations associated with a planned “yellow” risk status to accomplish maintenance activities on the Unit 2 Quench Spray Pump P21A.

- The inspectors reviewed condition reports CR-06-0266 and 06-02998, which addressed deficiencies in the shutdown risk assessment during the Unit 1 steam generator replacement outage.

- The inspectors reviewed CR 06-03863, which details operational challenges presented by a mechanical seal leak on 2CDS-P22A.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed two events that demonstrated personnel performance in coping with non-routine evolutions and transients. The inspectors observed operations in the control room and reviewed applicable operating and alarm response procedures, technical specifications, plant process computer indications, and control room shift logs to evaluate the adequacy of FENOC's response to these events. The inspectors also verified the events were entered into the corrective action program to resolve identified adverse conditions. Documents reviewed during the inspection are listed in the Attachment.

- Unit 1: On May 26, 2006, at 11:30 a.m., during performance of 1MSP-1.05-I, Solid State Protection System (SSPS) Train 'B' Bi-Monthly Test, the surveillance indicated unsatisfactory results while performing the Memories Function Test (CR-06-03525). Technical Specifications required a plant shutdown to Mode 3. The Unit was placed in Mode 3 as required, Solid State Protection System troubleshooting and repairs were completed with assistance from the nuclear steam supply system vendor (Westinghouse). All post-maintenance testing was completed satisfactory.

- Unit 2: On April 11, 2006, at 9:24 a.m., multiple unexpected Fire Protection System Deluge Valve actuations resulted in wetting of both charcoal main filter banks required by TS 3.7.8.1. As a result, TS 3.0.3 was entered, requiring a unit shutdown within the next 7 hours. Unit 2 power was reduced to 18 percent over the next 5 hours and preparations were made to disconnect the unit from the grid. The downpower was terminated at approximately 18%, due the licensee's receipt of a Notice Of Enforcement Discretion (NOED) from the NRC, which allowed for a fixed time window for restoration of the components.
URI 2006-003-01 was opened to track enforcement actions pending NRC review of the event.

  The inspectors monitored licensee activities in the control room including use of procedures to reduce power, technical specification adherence, NRC notification of the event, and operator actions during the power reduction. A post-event review of Unit 2 operator logs confirmed inspector observations during the event. The unexpected Fire Protection System Deluge Valve actuations occurrence was entered into the licensee's corrective action process for evaluation and resolution.

The inspectors noted that the licensee’s apparent cause was most likely due to a ground that occurred on the non-safety related 125VDC bus 2-5 / 2-6, which resulted in a surge that actuated certain fire protection relays. The plant risk associated with the inadvertent actuation of the fire protection deluge system is considered to be very low. Since no violation of NRC requirements were identified, beyond the TS implications associated with the NOED, this URI is closed. **URI 05000412/2006003-01, Beaver Valley Unit 2 NOED for Two Trains of SLCRS OOS (06-1-01).**

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 7 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 limiting conditions for operation (LCO) requirements and UFSAR design basis requirements were properly addressed. Documents reviewed are listed in the Attachment. This inspection activity represents seven samples of the following issues:

- An OD associated with one of the two upper snubbers on the Unit 1 ‘C’ Steam Generator (RC-HC-10C), as documented in CR 06-02913, regarding a hydraulic fluid leak first identified on April 16, 2006. The inspectors assessed the adequacy and acceptability of FENOC's conclusions in the OD that the current leak rate (1 drop every 2 minutes) is acceptable and that the snubber will remain fully functional. The inspectors evaluated the snubber reservoir level impact calculation, the coping strategy, and the conclusions of the seismic and combustible loading analyses.

- On May 8 - 10th, the inspectors reviewed an OD associated with Unit 1 Quench Spray System flow lines that were potentially not periodically tested or inspected, as documented in CR 06-03127. The inspectors assessed the adequacy and acceptability of FENOC's operability assessment and verified that appropriate technical issues were addressed.

- On May 9th, the inspectors reviewed licensee actions associated with potential reportability of issues identified in CRs 06-03134 and 06-03199. These CRs, addressed a charcoal sample iodine removal efficiency test failure on SLCRS Filter Bank 2HVS-FLTA205A, and charcoal sample testing for Filter Banks 2HVS-FLTA205A, 205B, 208B.

- On June 5th, the inspectors reviewed licensee actions associated with CR 06-03239, which included the adequacy of an engineering analysis of discharge pressure for 1-FW-P-4, during the performance of 1OST-24.7 on May 9, 2006.

- On June 7 and 8, 2006, the inspectors reviewed licensee actions associated with the operability determination for CR 06-03630, which detailed a pinhole leak on piping for Unit 2 vacuum breaker 2SWS-486.

- The inspectors reviewed the Engineering Evaluation associated with CR 06-02872, which addressed the licensee's exceedance of a 320 degree differential temperature limit for the pressurizer spray line. While the 322°F differential temperature resulted in a small, two degree exceedance of the limit in Section 8.4 of the License Requirements Manual (LRM), an evaluation was required and concluded that the out-of-limit condition did not affect the structural integrity of the pressurizer based on the bounding analysis of record, WCAP-15351, "Evaluation of Pressurizer Transients Based on Plant Operations for Beaver Valley Unit 1."

- The inspectors reviewed licensee actions associated with an Operational Decision Making Issue (ODMI) evaluation, due to indications that a potential loose part was located in the secondary side of the 'C' steam generator. The ODMI included data that indicated a potential loose part was present since steady-state 100 percent power operations was reached on April 27, 2006, following refueling outage No.17. The inspectors assessed the adequacy and acceptability of FENOC's review, as contained in the ODMI summary document "Continued Plant Operation With A Loose Part Indication In The Secondary Side Of 1RC-E-1C, Rev. 3 (including previous revisions). The review included applicable condition reports 06-03091, 06-03092, 06-03620, 06-03618 and 06-03709.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

The inspectors evaluated the design basis impact of a modification that installed piping changes to the Unit 1 No. 1 and 2 EDG lube oil circulating pump discharge lines, implemented in accordance with ECP 02-0269, Rev. 1, "Piping Changes to 1EE-P-2A & 2B Lube Oil Circulating Pump Discharge Lines." The inspectors reviewed the adequacy of the associated 10 CFR 50.59 screening, walked down the systems to verify that changes described in the package were implemented, and verified the post-modification testing was satisfactorily accomplished. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 9 samples)

a. Inspection Scope

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

- Emergency Diesel Generator (2EGS*EG2-2) repair & retest following control failure alarm during shutdown. (WO 200204018)

- 2OST-7.6, “Centrifugal Charging Pump Test [2CHS*P-21C],” Rev. 27, performed on April 26, following a planned inspection/replacement of the outboard pump bearing.

- 2MSP-1.04-I, “Solid State Protection System Train A Bi-Monthly Test,” Issue 4, Rev. 30, performed on May 4, and used to establish system conditions and perform post maintenance testing for replacement of four universal logic cards in accordance with notification 200205488.

- 1MSP-6.14-I, “P-457 Pressurizer Pressure Channel III Test,” Issue 4, Rev. 9, performed on May 11, following replacement of four relays in accordance with notification 200133455.

- 1OST-24.7, “Dedicated Auxiliary Feed Pump [1FW-P-4] Test,” Rev. 13, performed on May 12, following planned maintenance of the pump.

- Stroke testing of Residual Heat Release Valve (HCV-1MS-104) after corrective maintenance to valve positioner on June 7th.

- 1MSP-21.24-I, “P-1MS486 Loop 2 Steamline Pressure Protection Channel 4 Calibration,” Issue 4, Revision 4, performed on June 20, which included adjustment of lead/lag amplifier dynamic time constants.

- 1/2CMP-75-MCP-5E, Rev. 1, “Electrical Test Procedure for Inspection, Verification, and Calibration Testing of 480V Motor Control Center Motor Circuit

Protectors," performed on May 16. This activity replaced the breaker module for the Unit 2 emergency diesel generator No. 2, lube oil heater, which had experienced problems controlling temperature, and was performed in accordance with work order 200210437.

- 1MSP-1.05-1, "Solid State Protection System (SSPS) Train 'B' Bi-Monthly Test," performed on May 26, 2006, following installation of a new A412 logic card during SSPS surveillance.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 3 Samples)

.1 Unit 1 Refueling Outage (1R17)

a. Inspection Scope

The inspectors observed selected Unit 1 outage activities to determine whether shutdown safety functions (e.g. reactor decay heat removal, spent fool pool cooling, and containment integrity) were properly maintained as required by TS and plant procedures. The inspectors evaluated specific performance attributes including operator performance, communications, and instrumentation accuracy. The inspectors reviewed procedures and/or observed selected activities associated with the Unit 1 refueling outage. The inspectors verified activities were performed in accordance with procedures and verified required acceptance criteria were met. The inspectors also verified that conditions adverse to quality identified during performance of selected outage activities were identified as required by the licensee's corrective action program. Documents reviewed are listed in the Attachment. The inspectors also evaluated the following activities:

- Containment Structural Integrity Test
- High Head Safety Injection Full Flow Test
- AFW Check Valve 1FW-156C Reverse Flow Test
- Heatup rate monitoring during startup
- Containment Sump Inspection
- Final Containment Walkdown and Closeout Inspection
- Control rod testing.
- Initial Criticality
- Reactor Startup
- Low power reactor physics testing.
- Plant startup and heatup

- The inspectors observed the pre-test briefing and selected test activities for 1-BVT-2.1.1, “Control Rod Cold Plant Exercise and Data Collection” test, Issue 1, Revision 0, performed on April 11 and 12.

- The inspectors observed selected management review activities associated with restart readiness of Unit 1, following completion of the Unit 1 R17 refueling activities. The restart readiness review meeting was accomplished as required by NOBP-OM-4010, “Restart Readiness for Plant Outages” Rev. 3 on April 13. The purpose of the review, in part, was to assure to station management that the plant’s material condition, programs/processes, and staff members are ready for startup and safe, reliable operation after completion of outage activities.

- The inspectors observed selected test activities on April 13 and 14, and reviewed the completed test procedure, 1BVT 1.47.2, “Containment Type A Test”, Revision 5. The inspectors verified that test data documented an acceptable “As-Left” leakage rate of 0.03053 percent weight per day.

b. Findings

No findings of significance were identified.

.2 Unit 1 Forced Outage

a. Inspection Scope

The inspectors reviewed licensee performance during a forced outage following a Unit 1 TS required shutdown on May 26, 2006, due to a SSPS surveillance test failure. The inspector reviewed compliance to TS requirements and approved procedures, conduct of outage risk evaluations, configuration control, and maintenance of key safety functions. Documents reviewed during the inspection are listed in the Attachment. During this forced outage, the inspectors monitored FENOC’s control of the outage activities listed below:

- Shutdown risk evaluation
- Startup scheduling
- Reactor Startup and Criticality
- Plant Startup
- Power Ascension

b. Findings

No findings of significance were identified.

.3 Unit 2 Forced Outage

a. Inspection Scope

The inspectors reviewed licensee performance during a forced outage following a Unit 2 turbine/reactor trip on April 2, 2006. The inspector reviewed compliance to TS requirements and approved procedures, conduct of outage risk evaluations, configuration control, and maintenance of key safety functions. Documents reviewed during the inspection are listed in the Attachment. During this forced outage, the inspectors monitored FENOC's control of the outage activities listed below:

- Shutdown risk evaluation
- Startup scheduling
- Reactor Startup and Criticality
- Plant Startup
- Power Ascension

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

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

- 1OST-7.4, "Centrifugal Charging Pump Test [1-CH-P-1A]," Rev. 34 performed on May 2

- 1OST-36.1, "Diesel Generator No. 1 Monthly Test," Rev. 44 performed on May 3

- 2OST-24.4, "Steam Driven Auxiliary Feed Pump [2FWE*P22] Quarterly Test," Rev. 56

- 2OST-21.7, "Main Steam Trip Valve [2MSS*AOV101A,B and C] Full Closure Test," Rev. 11 (Containment Isolation Valve Sample)

- 2OST-2.4A, "Quadrant Power Tilt Ratio Manual Calculation," Rev. 4

- 2OST-6.2A, "Computer Generated Reactor Coolant System Water Inventory Balance," Rev. 22 (Leak Rate Sample)

- 2OST-36.2, “Emergency Diesel Generator [2EGS*EG2-2] Monthly Test,” Rev. 48

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 samples)

a. Inspection Scope

The inspectors reviewed the following temporary modifications (TM) based on risk significance. The TM and associated 10CFR50.59 screening were reviewed against the system design basis documentation, including the UFSAR and the TS. The inspectors verified the TMs were implemented in accordance with Administrative (ADM) Procedure, 1/2-ADM-2028, “Temporary Modifications,” Rev. 5. Documents reviewed are listed in the Attachment.

- Temporary modification to the emergency response facility Outside Air Damper (D-3) in accordance with ECP-06-0142, Rev. 0, completed on April 26. For this activity, the inspectors walked down the systems to verify that changes described in the package were actually implemented, and verified the post-modification testing was satisfactorily accomplished.

- Temporary modification to allow operation of the Bypass Regulator of the Vital Bus Uninterruptible Power Supply (UPS) UPS-VITBS2-3, with the Metal Oxide Varister (MOV) 82 disconnected, completed under TMOD 2-05-14.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed a Unit 2 emergency plan mini-drill conducted on May 25, 2006. Operator and event personnel performance regarding event notifications were specifically evaluated. The inspector evaluated the simulator-based drill that involved multiple safety-related component failures and plant conditions that warranted emergency plan activation, emergency facility activation, and escalation through the event classification of General Emergency. The licensee counted this evolution toward Emergency Preparedness Drill/Exercise Performance (DEP) Indicators, therefore, the inspectors reviewed the applicable event notifications to determine whether they were appropriately credited, and properly evaluated consistent with Nuclear Energy Institute

(NEI) 99-02, Rev. 2, "Regulatory Assessment Performance Indicator Guideline." The inspectors attended the Technical Support Center critique, and reviewed all other critiques to ensure appropriate adverse conditions were appropriately entered into the Corrective Action Program. Other documents utilized in this inspection include the following:

- 1/2-ADM-1111, "NRC EPP Performance Indicator Instructions," Rev. 2
- EPP/I-1b, "Recognition and Classification of Emergency Conditions," Rev. 7
- 1/2-EPP-I-2, "Unusual Event," Rev. 23
- 1/2-EPP-I-3, "Alert," Rev. 21
- 1/2-EPP-I-4, "Site Area Emergency," Rev. 21
- 1/2-EPP-I-5, "General Emergency," Rev. 22

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 8 samples)

a. Inspection Scope

During the period April 3 - 7, 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 1 refueling outage and Unit 2 power operations. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant technical specifications, and the licensee's procedures. This inspection activity represents completion of eight samples relative to this inspection area.

Plant Walkdown and RWP Reviews

During the Unit 1 refueling outage, the inspector identified exposure-significant work activities being conducted in the reactor building and primary auxiliary building. Specific work activities included removal of scaffolding and temporary shielding from the reactor building, installation of a ventilation system on the reactor head, installation of level sensing lines and insulation on the steam generators, air-operated valve repair/testing, and various outage support activities. The inspector reviewed radiation survey maps and radiation work permits (RWP) associated with these activities to determine if the associated controls were acceptable.

The inspector toured accessible radiological controlled areas, including the Unit 1 reactor building, the Unit 1 and Unit 2 primary auxiliary buildings, Unit 1 fuel handling

building, Unit 2 condensate polishing building, Unit 1 decontamination building, Unit 1 waste handling building, and the old steam generator/reactor head storage mausoleum. With the assistance of the ALARA Supervisor, the inspector performed independent surveys of selected areas in these buildings to confirm the accuracy of survey maps and the adequacy of postings. The inspector confirmed that environmental dosimeters were properly located on the mausoleum and that the building was secured.

In evaluating 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 RWP No. 106-8005, Steam Generator Replacement Project (SGRP) scaffolding removal; RWP 106-8006, SGRP restoration; RWP 106-8011, SGRP temporary shielding removal; RWP 106-8017, reactor head reassembly; RWP 106-4009, reactor cavity draindown/decontamination; and RWP 106-4003, outage mechanical maintenance.

The inspector reviewed RWPs and associated instrumentation and engineering controls for potential airborne radioactivity areas located in the reactor building and primary auxiliary building. The inspector confirmed that no worker received an internal dose that exceeded 10 mrem due to airborne radioactivity when performing outage-related tasks. The inspector reviewed personnel contamination event reports and the dose assessment methodology for tasks potentially resulting in internal exposures to confirm the accuracy of the results. Tasks reviewed included reactor building demobilization, insulation removal, and decontamination activities.

Problem Identification and Resolution

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

Jobs-In-Progress

The inspector observed aspects of various activities to confirm that radiological controls, such as required surveys, area postings, job coverage, and pre-job RWP briefings were implemented; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological conditions. The inspector attended a supervisory meeting that addressed determining the cause and reducing the frequency of personnel contamination events.

High Risk Significant - LHRA and VHRA Controls

Keys to locked high radiation areas (LHRA) and very high radiation areas (VHRA) for Unit 1 and Unit 2 were inventoried, and accessible LHRAs and VHRAs were verified to be properly secured and posted during plant tours.

The inspector discussed with radiation protection supervision the adequacy of physical and administrative controls for performing work in high radiation areas including the movement of spent resin for processing and disposal. The inspector verified that any changes to relevant procedures did not substantially reduce the effectiveness and level of worker protection and evaluated the adequacy of pre-requisite communications and authorizations.

Radiation Worker Performance

The inspector observed radiation worker and radiation protection technician performance during the installation of a ventilation system on the Unit 1 reactor head, installation of insulation on the steam generators, and the conduct of various outage support activities. The inspector determined that the individuals were aware of current radiological conditions and access controls, and that the skill level was sufficient with respect to the potential radiological hazards and the work performed.

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

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 4 Samples)

a. Inspection Scope

During the period April 3 - 7, 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 reasonably achievable (ALARA) for tasks conducted during the Unit 1 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 four samples relative to this inspection area.

Radiological Work Planning

The inspector reviewed pertinent information regarding current exposure trends for 2006 and on-going Unit 1 outage activities to assess current performance and outage exposure challenges. The inspector compared actual exposure with forecasted estimates.

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

The inspector reviewed the refueling outage work scheduled during the inspection period and the associated work activity dose estimates. Scheduled work included removal of temporary shielding from the reactor building, reactor head reassembly, and scaffolding removal/modification in support of steam generator restoration.

The inspector reviewed the 1R17 outage dose summary reports, detailing worker estimated and actual exposures through April 6th for SGRP-related activities and balance-of-plant tasks.

The inspector evaluated the effectiveness of exposure mitigation requirements specified in RWPs and ALARA Plans and compared actual worker cumulative exposure with estimated dose. The inspector attended a post-job ALARA de-briefing for reactor cavity decontamination. The inspector also reviewed in detail, those work activities where actual cumulative dose approached performance goals; e.g. upper reactor internals lift (RWP#106-4023, ALARA Plan 06-1-14), insulation replacement on steam generators (RWP#106-8003, ALARA Plan 06-1-25), SGRP scaffolding installation (RWP 106-8005, ALARA Plan 06-1-27), reactor cavity draindown/decontamination (RWP#106-4009, ALARA Plan 6-1-08) and reactor reassembly (RWP 106-4019, ALARA Plan 6-1-11). Additionally, the inspector reviewed the following post-job ALARA reviews that evaluated the effectiveness of radiological controls and identified lessons learned in performing specific tasks:

- RWP#106-4019, ALARA Plan #06-1-11, Reactor Disassembly/Reassembly
- RWP#106-4022, ALARA Plan #06-1-42, Refueling Operations
- RWP#106-4023, ALARA Plan #06-1-14, Upper Internals Lift
- RWP#106-4024, ALARA Plan #06-1-15, Reactor Core Off-Load/Reload
- RWP#106-4033, ALARA Plan #06-1-54, AOV Testing on PCV-RC-455A &B
- RWP#106-4040, ALARA Plan #06-1-43, Replace 1CH-1
- RWP#106-4044, ALARA Plan #06-1-47, Repair RC-261
- RWP#106-8006, ALARA Plan #06-1-28, Welding of New Steam Generators
- RWP#106-8007, ALARA Plan #06-1-60, SGRP Pipe End Decon
- RWP#106-8014, ALARA Plan #06-1-36, Transport of old SG’s & Reactor Head
- RWP#106-8016, ALARA Plan #06-1-38, Disassembly Old Reactor Head
- RWP#106-4031, ALARA Plan #06-1-18, Pressurizer Inspections

The inspector evaluated the departmental interfaces between radiation protection, engineering, operations, maintenance crafts, and contractors to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing the Manager-Radiation Protection, the Senior Nuclear Specialist-ALARA, and the Supervisor-ALARA; reviewing ALARA Committee meeting minutes; reviewing Nuclear Oversight field observation reports; and attending an outage planning meeting.

The inspector compared the person-hour estimates provided by various departments and contractors with actual work activity time requirements and evaluated the accuracy of these estimates. Specific jobs reviewed included temporary shielding removal, SGRP scaffolding removal, and reactor reassembly.

The inspector determined if work activity planning included the use of temporary shielding, system flushes, and operational considerations to further control dose. The inspector evaluated the effectiveness of temporary shielding installed on various components and the effects of reactor coolant system flushes/filtration.

The inspector reviewed personnel contamination event reports and dose assessments for selected personnel who became contaminated while performing outage tasks. The inspector reviewed the effectiveness of the licensee's methods for controlling airborne radioactivity concentrations through the use of temporary ventilation systems.

Verification of Dose Estimates and Exposure Tracking Systems

The inspector reviewed the assumptions and basis for the annual site collective exposure estimate and the revised Unit 1 outage dose projection.

The inspector reviewed the licensee's method for adjusting exposure estimates and re-planning work when emergent work or expanded job scope was encountered. The inspector attended an outage planning meeting, reviewed recent actions of the Station ALARA Committee in monitoring and controlling dose allocations, and interviewed site staff regarding actions to be taken when actual dose approached estimated dose.

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

The inspector reviewed dose and dose rate alarm reports to determine if the dosimeter set points were appropriate for the work area radiological conditions.

Job Site Inspection and ALARA Control

The inspector observed various outage activities being performed for re-assembling the reactor head, temporary shielding removal, and scaffolding removal, to verify that radiological controls, such as required surveys, job coverage, pre-job HRA briefings, and contamination controls were implemented; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological controls.

The inspector reviewed the exposure of individuals in selected work groups, including contractors, radiation protection, and maintenance crafts to determine if 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 1 source term. Through review of survey maps and interviews with the Senior Nuclear Specialist-ALARA, the inspector evaluated recent source term measurements and control

strategies. Specific strategies being employed at Unit 1 included zinc addition, shutdown chemistry controls, system flushes, and temporary shielding.

Problem Identification and Resolution

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

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.0 - 6 Samples)

a. Inspection Scope

During the period April 24 - 28, 2006, the inspector conducted the following activities to verify that the licensee's radioactive material processing and transportation programs complied with the requirements of 10 CFR 20, 61, and 71; and Department of Transportation (DOT) regulations contained in 49 CFR 170-189.

Radioactive Waste System Walkdown

The inspector walked down accessible portions of the Unit 1 and Unit 2 radioactive liquid and solid waste collection/processing systems with the Operations Services Field Supervisor in charge of radwaste processing/shipping. The inspector evaluated if the systems and facilities were consistent with the descriptions contained in the UFSAR and Process Control Program (PCP), evaluated the general material conditions of the systems and facilities, and identified any changes to the systems. The inspector reviewed the current processes for transferring radioactive resin/sludge to shipping containers, subsequent de-watering, and shipment to a waste processor.

The inspector discussed with the Field Supervisor the status of non-operational, retired-in-place, radioactive waste processing equipment, and the administrative and physical controls for various components in these systems. The inspector determined that no significant changes have been made to radwaste processing systems since the last inspection, conducted in April 2004.

Areas visually inspected included the radwaste storage areas located within the Unit 1 and Unit 2 protected area, and those outside the protected area in the Interim Waste Storage Facility and old steam generator/reactor head storage building. The inspector

reviewed storage container inventories, container inspection records, and associated procedures; and verified that the storage containers were properly labeled and the area appropriately posted.

Waste Characterization and Classification

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

- the radio-chemical sample analysis results for various Unit 1 and Unit 2 radioactive waste streams, including spent resins, dry active waste, and mechanical filters

- the development of scaling factors for hard-to-detect radionuclides from the radio-chemical analysis

- methods and practices to detect changes in waste streams

- classification and characterization of waste relative to 10 CFR 61.55 and to DOT shipment subtype per 49 CFR 173

Shipment Preparation

The inspection included a review of radioactive waste program documents, shipment preparation procedures, and in-progress activities for regulatory compliance, including:

- attending a pre-job briefing for transferring a highly radioactive Unit 1 spent filter (CH-FL-2) from the transfer bell to the shielded storage container, and measuring contact and one meter dose rates from the container in preparation for characterizing the contents for shipment, on April 25, 2006

- attending a pre-job briefing and subsequently observing technicians installing a closure lid on a High Integrity Container (HIC), in preparation for shipping the HIC to a waste processor, following de-watering of the spent resin

- review of radioactive material shipping logs for 2004, 2005 and 2006

- review of certificates-of-compliance for in-use shipping casks

- verification that training was provided to appropriate personnel responsible for classifying, handling and shipping radioactive materials, in accordance with NRC Bulletin 79-19, and 49 CFR 172 Subpart H

Shipping Records

The inspector selected six (6) of the highest activity shipments made since the last inspection, and reviewed records associated with these non-excepted shipments made during 2005 (Nos. 3260, 3247, 3290, 3300, 3307, and 3308). The following aspects of the radioactive waste packaging and shipping activities were reviewed:

- implementation of applicable shipping requirements including proper completion of manifests
- implementation of specifications in applicable certificates-of-compliance, for the approved shipping casks including limits on package contents
- classification of radioactive materials relative to 10 CFR 61.55 and 49 CFR 173
- labeling of containers
- radiation and contamination survey of packages
- placarding of transport vehicles
- conduct of vehicle checks
- providing of driver emergency instructions
- completion of shipping papers

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 6 samples)

a. Inspection Scope

The inspectors sampled licensee submittals for three Performance Indicators (PI) listed below for Unit 1 and Unit 2. The inspectors reviewed portions of the operational logs and PI data developed from monthly operating reports, and discussed methods for compiling and reporting the PIs with cognizant engineering and licensing personnel. To verify the accuracy of the PI data reported during this period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 2, were used for each data element.

Reactor Coolant System (RCS) Activity

The inspectors reviewed the PI for RCS activity to verify that the proper dose equivalent Iodine-131 was reported and that it was below the TS limit. Inspectors reviewed data for each unit from December 2004 to April 2006.

Reactor Coolant System (RCS) Leak Rate

The inspectors reviewed the PIs for RCS leak rate to verify that the maximum identified leakage did not exceed the TS value and that it was properly reported. Inspectors reviewed data for each unit from December 2004 to April 2006.

Unplanned Power Changes per 7000 Critical Hours

The inspectors reviewed the PIs for unplanned power changes per 7000 critical hours, to verify that power changes greater than 20 percent had been properly reported as specified in NEI 99-02, Rev. 2. The inspectors verified the accuracy of the reported data through reviews of Licensee Event Reports, monthly operating reports, plant operating logs, and additional records. The inspectors reviewed data from June 2004 to April 2006.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Review of Problem Identification and Resolution

a Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed a daily screening of items entered into FENOC's corrective action program. This review was accomplished by reviewing summary lists of each CR, attending screening meetings, and accessing FENOC's computerized CR database.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors reviewed site trending results for the time period July through December, 2005, to determine if trending was appropriately evaluated by FENOC. This review covered the site trending program under FENOCs Integrated Performance Assessment process detailed in Self-Assessment BV-SA-06-079, 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 40A2.1 and .7), and (3) not indicative of a more significant safety concern. Additionally, the inspectors verified the performance of site trending against NOP-LP-2001, Rev. 13, "Condition Report Process", and NOBP-LP-2018, Rev. 01, "Integrated Performance Assessment /Trending." The inspectors also reviewed quarterly Quality Assurance reports and issues captured in the Activity Tracking database to identify issues and trends to evaluate during the inspection.

b. Findings

No findings of significance were identified.

.3 Annual Sample Review (71152 - 1 sample)

CR-03-05-06503 - Installed Reducing Bushing On 2CHS-E25B Was Not The Correct Material

a. Inspection Scope

The inspectors selected condition report (CR) 05-06503 for detailed review, which was initiated in September 2005, and documented that a steel bushing had been incorrectly installed on the lube oil cooler to the Unit 2 'B' Charging Pump that required a stainless steel bushing. The inspector reviewed the adequacy and appropriateness of corrective actions to address the installation of incorrect material on a safety-related component. Documents that were reviewed for this inspection are located in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The inspector noted that the CR had no associated corrective actions beyond the documentation that the steel bushing was replaced with the required stainless steel bushing. The inspector identified that various site program and human performance barriers failed to preclude the installation (and later the identification) of a steel bushing in a component that required stainless steel in accordance with applicable drawing and design documentation. For example, Material & Revision Request No. 4094, utilized during replacement of the bushing from February 2002, contained observations that some site drawings did not detail the reducing bushings, and that they were rusty and

weeping moisture, and needed to be replaced in 2R09. The inspectors determined that these were missed opportunities, since design documentation from the vendor manual clearly identified the bushing material as stainless steel. Moreover, since all other site drawings referenced in the work orders during the 2002 and 2005 replacement did not detail the bushing or the material type, the inspector concluded that the vendor manual for the lube cooler could have been reviewed during the work planning process to ensure the correct component was installed on the safety-related charging pump. The inspectors verified that the issue was placed into the corrective action program as CR 06-03948 for resolution.

.4 Access Controls and ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed 25 Condition Reports, 13 Nuclear Oversight Field Observation Reports, and a Radiological Quality Assessor report to evaluate the threshold for identifying, evaluating, and resolving problems in implementing radiological controls. This review was conducted against the criteria contained in 10 CFR 20, TSs, and the licensee's procedures.

b. Findings

No findings of significance were identified.

.5 Radioactive Material Processing and Transportation

a. Inspection Scope:

The inspector reviewed sixteen (16) Condition Reports, five (5) Quality Assessment Field Observations, and a quarterly Nuclear Oversight Assessment Report relating to radioactive material processing and shipment. Through this review, the inspector assessed the licensee's threshold for identifying problems, and the promptness and effectiveness of the resulting corrective actions. This review was conducted against the criteria contained in 10 CFR 20.1101(c), TSs, and the licensee's procedures.

b. Findings

No findings of significance were identified.

.6 Beaver Valley Unit 1 Steam Generator Replacement Inspection

a. Inspection Scope

The inspectors reviewed condition reports (CRs) and a sample of self-assessments associated with 10 CFR 50.59 Safety Evaluations and plant modification issues. This review ensured that FENOC was identifying, evaluating, and correcting problems

associated with these areas and that the planned or completed corrective actions for the issues were appropriate.

Additionally, the inspectors reviewed CRs related to the SGRP, including project engineering, fabrication and shop testing, transportation and receiving, lifting operations, removal operations, piping insulation removal and pipe cutting, radiological control procedures, physical security, cutting and restoration of the containment construction opening, and the initiation of non-conformance reports by contractors, equipment vendor, and licensee employees.

b. Findings

No findings of significance were identified.

.7 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. Specific focus was placed on immediate corrective actions (operability determinations) for CR 06-03726, "HCV-1MS-104 and 1MS-26 Leak-by." In addition, related corrective actions for CR 06-03664 and CR 06-03904 were reviewed.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion 16, for failure to perform adequate operability evaluations for degraded components to assure off-site dose consequences are bounded in the radiological safety analysis for a SGTR event.

Description. On June 6, 2006, condition report (CR) 06-03664 was initiated due to identified leakage past valve 1MS-26, while the licensee was investigating steam leakage to atmosphere past the downstream residual heat release valve HCV-1MS-104. These two valves provide a boundary isolation function during a steam generator tube rupture (SGTR) event in accordance with 1OM-53A.1.E-3(ISS1C). On June 7, 2006, CR 06-03726 was initiated to evaluate the off-site dose effects from this leakage past HCV-1MS-104, following a SGTR. The CR also requested the evaluation account for any current leakage past the SG atmospheric dump valves and SG safety valves. Operators reviewed requirements contained in the License Requirements Manual specification LR 6.1 and the associated bases, and concluded that HCV-1MS-104 was operable.

Following discussions with plant personnel, CR 06-03903 was initiated on June 21, 2006, to account for the additional SGTR dose consequences for current leakage past HCV-1MS-104, 1MS-26, and the remaining atmospheric dump and safety valves above the leakage already assumed in the radiological safety analysis. The CR, in part,

referenced LR 6.1 Bases and noted the Bases also addressed fully closing on demand as one of the functional requirements of HCV-1MS-104. Operations, after discussions with engineering, documented in CR 06-03903 that HCV-1MS-104 was operable, but degraded. CR 06-03903 further documented that current leakage past degraded components (HCV-1MS-104 and 1MS-26, Atmospheric Steam Dump Valve(s), and Steam Generator Safety Valves) had not been quantified. Stone and Webster was contacted to assess the magnitude of additional steam leakage that would be permitted following a SGTR, and stay within licensing and design bases limits.

The inspectors determined that the licensee had not fully evaluated nor estimated current leakage for degraded components (HCV-1MS-104 and 1MS-26, Atmospheric Steam Dump Valve(s), and Steam Generator Safety Valves) since restart of Unit 1. Furthermore, leakage estimates for degraded components that would be associated with a ruptured steam generator prior to the Unit 1 refueling/SG replacement outage were several times greater than the volumetric flow rate of 45.2 cubic feet per minute. Stone and Webster concluded on June 27, that the allowable leakage via the available components in question, on a ruptured steam generator could be a maximum of 2.8 lbs/sec, and still be bounded by the current licensing basis.

Analysis. The issue involved a performance deficiency in that FENOC failed to perform an adequate Operability Determination that recognized the degraded condition of the leaking valves. Had this degradation been fully recognized, the assessment would have accounted for existing leakage past the Unit 1 Residual Heat Release Valve, Atmospheric Steam Release Valve(s), and Main Steam Safety Valves, to assure off-site dose consequences were bounded in the radiological safety analysis for a SGTR event.

The inspectors evaluated this finding in accordance with IMC 0609, Appendix H, since there was the potential of increasing the Large Early Release Frequency (LERF) without affecting Core Damage Frequency (CDF), and therefore a Type B finding. Since this finding is not associated with structures, systems and components important to LERF, this finding is considered to be of very low safety significance, Green.

A contributing cause to this finding is related to the corrective action program component of the PI&R cross-cutting area, in that degraded components were not adequately evaluated to assure proper operability.

Enforcement. 10 CFR 50, Appendix B, Criterion 16, Corrective Action, requires in part, that measures shall be established to assure that conditions adverse to quality such as failures, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, degraded conditions (valve leakage) identified for valves HCV-1MS-104 and 1MS-26 in CRs 06-03664 and 06-03726 were not quantified or evaluated for leakage impact (additional radionuclide release) above the leakage already assumed in the radiological safety analysis of record. In addition, current leakage past Atmospheric Steam Release Valve(s), and Main Steam Safety Valves had

not been quantified and evaluated for leakage impact (additional radionuclide release) above that already assumed in the radiological safety analysis of record.

Because this violation was of very low safety significance and FENOC entered this violation into their corrective action program, the violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC enforcement policy. **NCV 05000334/2006003-02, "Failure to perform an adequate operability determination for current leakage past main steam safety, dump, and residual release valves."**

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000412/2005001-01. Containment Isolation Valve Relay Failure Unknowingly Leads to Technical Specification Noncompliance

Supplement 00 to this LER was reviewed and closed under inspection report 05000334(412)/2005008. The failure was documented as a Green finding under NCV 05000412/2005007-01, "Failure to Demonstrate Effective Maintenance on the Unit 2 TDAFW Steam Admission Valves." Based on the review of supplement 01, no additional findings of significance or NRC violations were identified. This LER is closed.

.2 (Closed) LER 05000334/2006001-00. Main Steam Safety Valve Relief Tests Exceeded Technical Specification Required Setpoint Tolerance

The inspectors reviewed the circumstances surrounding the cause of the February 12, 2006, failure of the Unit 1 main steam safety valves (MSSVs) to lift at the required setpoints during testing. The underlying cause was dispositioned as a non-cited violation and was documented in NRC inspection report 50-334/2006002-02, and the inspector evaluated the adequacy of the corrective actions identified in the subject LER. Since no other violation of NRC requirements was identified, this LER is closed.

.3 (Closed) LER 05000412/2006001-00. Turbine-Generator Trip Due to Loss of Generator Excitation Power Results in Reactor Trip

The inspectors reviewed the circumstances surrounding the cause of the April 2, 2006 event, actions taken by the licensee in response to the trip, as well as corrective actions to address identified adverse conditions contained in the referenced LER and the corrective action program. This review included evaluation of the sequence of events log, applicable operating and abnormal procedures, available printouts and parameter trending data, as well as plant-reference simulator scenarios used to validate the cause of the trip and other equipment performance verifications. The inspector reviewed the corrective actions contained in the LER and the corrective action program, which were selected to address the missing support block that subsequently led to fatigue failure in the turbine-generator exciter field coil support plate, and ultimately, the loss of field and the plant trip. The inspector determined that no findings of significance were identified and there were no violation of NRC requirements. This LER is closed.

.4 (Closed) LER 05000412/2006002-00. Entry into Technical Specification 3.0.3 Due to Inoperability of Both Trains of the Supplemental Leak Collection and Release System

The inspectors reviewed the circumstances surrounding the event on April 11, 2006, (See Section 1R14), which involved entry into TS 3.0.3, the corrective actions taken by the licensee due to the inadvertent deluge actuation, and subsequent regulatory notifications. The inspector verified that appropriate reportability criterion were implemented during the initial notification and followup LER. This issue was entered into FENOC's corrective action program as CR 06-03754. The inspector determined that no findings of significance were identified and no violations of NRC requirements occurred. This LER is closed.

4OA5 Other

.1 Beaver Valley Unit 1 Replacement Reactor Vessel Closure Head (71007)

a. Inspection Scope

The inspectors reviewed the Beaver Valley Unit 1 Replacement Reactor Vessel Closure Head (RRVCH) using the guidance in NRC Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection."

FENOC elected to replace the Beaver Valley Unit 1 RVCH during the 1R17 Spring 2006 refueling outage due to demonstrated susceptibility of Alloy 600 control rod drive mechanism (CRDM) nozzles and UNS W86182 weld filler material in the existing RVCH due to primary water stress corrosion cracking. The design of the RRVCH is similar to the old RVCH except for the replacement of the Alloy 600 nozzle material and weld material with a new and improved Primary Water Stress Corrosion Cracking (PWSCC) resistant material (Alloy 690).

The RRVCH was manufactured by Equipos Nucleares, S.A. (ENSA) in Maliaño, Spain. The RRVCH was made from a one-piece hemispherical forging meeting the material requirements of ASME Code Section II, Part A, SA-508, Class 3. The RRVCH includes 48 CRDM Alloy 690 penetration pressure housing assemblies that were shrunk fit into the RRVCH and attached with alloy 152/52 filler material partial penetration welds, 8 spare control rod drive mechanism housings with threaded flanges with capped head adapters, 56 thermal sleeves, 56 guide funnels for the thermal sleeves, four instrumentation port head adapters with threaded flanges, and four instrumentation port housing guide funnels. In addition, the nozzles for the Reactor Vessel Head Vent System (RVHVS) nozzle and Reactor Vessel Level Instrumentation System (RVLIS) are attached to the RRVCH and are also constructed from Alloy 690 material.

Replacement CRDMs for the RRVCH were manufactured by Curtiss-Wright Electro-Mechanical Corporation (EMD). In addition, a new simplified head assembly (SHA) was designed for the RRVCH and manufactured by the Penn State Tool & Die Corporation.

Design and Planning

The inspectors verified that the RRVCH-related design changes and modifications to components described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59. The inspectors also reviewed the adequacy of 10 CFR 50.59 applicability reviews, screening evaluations, and safety evaluations for the design changes, modifications, and procedure changes.

The inspectors reviewed applicable design documents related to the components being replaced. The inspectors compared these changes to the original reactor vessel (RV) that was designed in accordance with Westinghouse General Equipment Specification 676413, General Reactor Vessel, with Addendum Equipment Specification 678801, Beaver Valley Unit No. 1 Reactor Vessel, and the American Society Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition, through the Winter 1968 Addenda per Westinghouse Specification requirements and the new components designed in accordance with FENOC Specification 8700-CGS-0035, Westinghouse Equipment Specification 676413, and the American Society Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1989 Edition.

The inspectors conducted in-office and onsite reviews of engineering change packages, engineering calculations, analyses, design specifications, material specifications, piping specifications, component specifications, installation specifications, certified design reports, and drawings for the Beaver Valley Unit 1 RRVCH, SHA, and CRDMs, to assess the technical adequacy of the design changes and to verify that the design bases, licensing bases, and the performance capability of the modified components were not degraded through the modifications.

The design and fabrication of the RRVCH, SHA and CRDMs were specified by FENOC in certified procurement specifications. Westinghouse performed the analyses, calculations or evaluations necessary to support the 10 CFR 50.59 evaluations of the RRVCH, SHA, and CRDMs. The inspectors reviewed the RRVCH design in ECP 03-0541-01, Replacement Reactor Vessel Closure Head, SHA design in ECP 03-0295-01, Simplified Reactor Vessel Head Assembly-CRDM and CETNA Head Port Adapter Section, and CRDM design in ECP 04-0432-01, Replacement Control Rod Drive Mechanisms. The engineering change packages for the RRVCH, SHA, and CRDMs include:

- Evaluations and/or analyses to show that all applicable acceptance criteria are met with the RRVCH, SHA, and CRDMs.

- Reviews of the plant Technical Specifications, UFSAR, SERs, and emergency operating procedures to identify changes that could be required by use of the RRVCH, SHA, and CRDMs.

The inspectors also reviewed the original design drawings of the Beaver Valley Unit 1 reactor vessel and reactor vessel head. Based on the drawings and dimensional data

collected from the original components, the information was reconciled to the replacement components design dimensions and as-built drawings were developed by ENSA. The inspectors verified that the RRVCH conformed to as-built design drawings and there were no fabrication deviations from design.

Lifting/Rigging and Transportation RRVCH

The adequacy of the lifting and rigging equipment associated with Beaver Valley Unit 1 RRVCH were tested and/or evaluated to verify that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting/rigging equipment and supporting structures.

The inspectors reviewed the procedures for heavy lifting, inspection and testing of the cranes and lifting equipment. The inspectors verified that the capability of the lifting equipment had been inspected, tested, and/or evaluated through engineering calculations and analyses. The review focused on applicable lifting and handling procedures FENOC 1/2-ADM-0819, Handling of NUREG 012 Heavy Loads, including Bechtel's work plan and inspection records (WPIR) C-RCR-147, Handling RVCHs Outside Containment and C-RCR-148, Rigging and Handling Of The RVCHs Inside Containment.

The inspectors reviewed the potential impact of load handling activities on the reactor core, spent fuel cooling, and other plant support systems, and the consequence of any impact loading of structures, systems, and components due to a RVCH drop accident. The inspectors reviewed the Beaver Valley Unit 1 polar crane used to handle the RVCH and determined that the polar crane is not single failure proof, does not have both mechanical and electrical stops, and there is no load drop analysis as described in recommendations of phase 2 of NUREG-0612, Control of Heavy Loads at Nuclear Power Plants. The NRC provided relief in committing to phase 2 of NUREG-0612 in GL 85-11, which stated there is enough defense in depth in phase 1 in which phase 2 is not required, but recommended. The inspector verified FENOC's commitments to Phase 1 of NUREG-0612, which included all of the phase 1 guideline requirements in place regarding good practices for crane operations and load movements. The inspectors verified that the guidelines were properly implemented by reviewing the following: heavy load safe load path drawings for lifts inside Beaver Valley Unit 1 Containment, procedures for load handling operations, training and certification records of crane operators, special lifting devices, and FENOC's periodic preventive mechanical and electrical maintenance, inspections, and tests.

Removal and Replacement RVCHs

The inspectors verified the temporary modification needed for containment access to support the RRVCH and steam generator replacement activity. The structural modifications and restoration of the temporary construction opening in the Beaver Valley Unit 1 containment were inspected and documented under the steam generator replacement inspection activity.

The inspectors reviewed activities associated with removal and replacement of the RVCHs. The inspectors reviewed portions of the preparation, including direct field observation of load testing of the outside lift system used to move the RVCHs and steam generators into and out of containment. The inspectors also observed portions of the lifting, rigging, down-ending, and transporting of the RVCHs into and out of containment.

Fabrication Inspections

The inspectors performed direct field observations of the assembly and life cycle testing of the CRDMs at the Curtiss-Wright Electro-Mechanical Corporation (EMD) facility in Cheswick, Pennsylvania. The inspectors also conducted field observations of in-process automated canopy seal welding of CRDM latch housings-to-head adapter flanges, and reviewed liquid penetrant examination sheets of canopy seal welds 900312-02, FW-02, through 900312-13, FW-13. No relevant indications were identified in any of the canopy seal welds.

The inspectors performed reviews of FENOC Specification 8700-DGS-0035 and Westinghouse Equipment Specification 676413 for the Beaver Valley Unit 1 RRVCH, to verify that the specified material, design, fabrication, inspection, examination, testing, certification, documentation, and functional requirements were consistent with the requirements of the American Society Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1989 Edition.

The inspectors reviewed the manufacturing data package reports for the Beaver Valley Unit 1 RRVCH. The data packages contained ASME Code Data Report Form –2, certified material test reports (CMTR), hydro test report, hydro test certificate, certificate of compliance, heat treatment records, post-weld stress relief heat treatment records, weld records, quality control receipt inspections, non-conformance reports, corrective actions, non-destructive examinations, and weld material acceptance tests for the manufacture of the Beaver Valley Unit 1 RRVCH. In addition, the inspectors reviewed the manufacturing reports for the CRDMs and SHA under purchase orders 55101746 and 55101403. The inspectors verified that the authorized nuclear inspector (ANI) inspected the replacement parts, reviewed the manufacturing reports, and certified that the components were fabricated and tested in accordance with ASME Code, Section III, Division 1.

Pre-Service Inspection (PSI) and Baseline Inspections

The baseline examinations that provide the data for future in-service inspections and dissimilar metal examinations for CRDM, vent line, reactor vessel level instrumentation system (RVLIS) line, and serve as a pre-service inspection (PSI) in accordance with ASME Section XI requirements for dissimilar metal welds, as well as the First Revision to NRC Order EA-03-009, consists of: (1) automated inside diameter UT and eddy current (ET) examination of 48 CRDM penetrations, RVLIS line penetration, and reactor head vent line penetration; (2) outside diameter and J-groove weld eddy current (ECT) examination of 48 CRDM penetrations, RVLIS penetration and the vent line penetration;

(3) under head visual test (VT) examination of all J-groove welds and penetration outside diameters; (4) top of head bare metal VT examination of all penetrations; and (5) under head PT examination of all penetration to head J-groove welds using "PT White" acceptance criteria.

An inspection was conducted on-site to evaluate the licensee's baseline examination of the Beaver Valley Unit 1 RRVCH. The inspectors reviewed results from the examination of the reactor vessel head penetration nozzles and the J-groove welds, and assessed the effectiveness of the NDE procedures and welding procedures.

The inspectors visually inspected Beaver Valley Unit 1 RRVCH and its penetrations. No abnormalities were observed. The inspectors observed the licensee's contracted NDE personnel performing eddy current testing (ECT) examinations of the wetted surface area of the reactor vessel head penetration nozzles and J-groove welds. At the time of the inspection, ultrasonic testing (UT) data acquisition had already been completed, and UT data analysis was ongoing. The UT and ECT examinations were performed as a baseline inspection using the NRC Order EA-03-009 requirements.

The inspectors also reviewed the NDE examination results including UT and ECT of the reactor vessel head penetration nozzles and the J-groove welds. The UT analysis showed a total of five weld volumetric indications (WVIs) and 14 weld interface indications (WIIs). The licensee's analysts, including a UT Level III analyst, reviewed the data and concluded that the indications were neither reportable nor recordable in accordance with their NDE procedures. However, the data was collected and stored as a baseline for future ISI inspection reference. ECT result analysis also showed no recordable indications.

The inspectors' review of selected documentation, including the UT and ECT procedures, and UT and ECT data sheets of the Beaver Valley Unit 1 replacement head penetration nozzles and the J-groove welds was conducted to ensure that NDE examinations were performed according to qualified procedures. The inspectors also reviewed the automatic welding procedure related to welding of the reactor vessel head penetrations during the original fabrication.

<u>Post-Installation Verification and Testing</u>

The inspectors verified that the post-maintenance testing (PMT) of the installed component replacements RVCH, CRDMs, and SHA cooling system and rod control system were conducted in accordance with approved procedures and verified the functional testing confirmed the design and established baseline measurements. Specifically, procedures 3BVT-11.60.11, CRDM Cooling System Air Flow Test, 1-BVT-1.1.9, Rod Control Timing Verification Test, 1-BVT-1.1.1, Rod Drop Time Test at Hot Standby, 1OST-6.9, RCS Vent System Test, and 1MSP-6.80-1, RVLIS Train A and B Sensor Calibration were reviewed to verify the testing acceptance criteria was met.

The inspectors also reviewed results of completed procedure 1OST-6.1, Primary Side In-service Leak Test to verify that no reactor coolant system (RCS) leakage was

observed from the RRVCH, RCS vent, and RVLIS piping and components during the Containment walkdowns conducted at Mode 3 normal operating pressure and temperature (NOP/NOT) by certified VT-2 Level 2 examiners.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Reactor Vessel Head Replacement Inspection</u> (71007)

a. <u>Inspection Scope</u>

During the period April 3 - 7, the inspector verified that the reactor vessel head removal activities maintained adequate nuclear and radiological safety by reviewing radiation protection program controls, planning, and preparation in specific areas using applicable portions of inspection procedures 71121.01 and 71121.02. Activities related to the removal, transport, and storage of the old Reactor Vessel Closure Head were evaluated with respect to ALARA planning, dose estimates/dose tracking, exposure controls including use of temporary shielding, contamination controls, radioactive materials management, emergency contingencies, and project staffing/training. Details of this inspection are contained in Section 4OS1 and 4OS2 of this report.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>Steam Generator Replacement Inspection</u> (50001)

a. <u>Inspection Scope</u>

During the period April 3 - 7, the inspector verified that steam generator removal and replacement activities maintained adequate nuclear and radiological safety by reviewing radiation protection program controls, planning, and preparation in specific areas using applicable portions of inspection procedures 71121.01 and 71121.02. Activities related to the removal, transport, and storage of the old steam generators were evaluated with respect to ALARA planning, dose estimates/dose tracking, exposure controls including use of temporary shielding, contamination controls, radioactive materials management, emergency contingencies, and project staffing/training.

Details of this inspection are contained in Section 4OS1 and 4OS2 of this report.

b. <u>Findings</u>

No findings of significance were identified.

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

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

.5 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

a. Inspection Scope

The objective of TI 2515/165, “Operational Readiness of Offsite Power and Impact on Plant Risk,” was to gather information to support the assessment of nuclear power plant operational readiness of offsite power systems and impact on plant risk. The inspector evaluated licensee procedures against the specific offsite power, risk assessment, and system grid reliability requirements of TI 2515/165.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on April 3, 2006.

b. Findings

No findings of significance were identified.

.6 To verify that engineering evaluations and design changes associated with steam generator (SG) replacement are completed in conformance with requirements in the facility license, the applicable codes and standards, licensing commitments, and the regulations

a. Inspection Scope

The inspectors reviewed applicable Engineering Change Packages (ECP's) associated with the replacement of the steam generators. All ECP's were reviewed in accordance with the requirements of IP 71111.02, Evaluation of Changes, Tests and Experiments and IP 71111.17B, Permanent Plant Modifications.

The inspectors reviewed administrative procedures that controlled the screening, preparation, and issuance of the Safety Evaluation (SE) to ensure that the procedures adequately covered the requirements of 10 CFR 50.59.

The inspectors reviewed applicable procedures, work practices, and documentation of the steam generator replacement project (SGRP), and assessed the control of safety-related activities associated with the major phases of the SGRP. These activities included restoration of the pressure boundaries of the reactor coolant system (RCS), secondary systems, and containment systems, exclusion of foreign materials, and plant modifications that could affect plant risk during the replacement activities or subsequent plant operation.

The inspectors reviewed procedures that governed the cutting and machining of weld preparations at existing main steam (MS), feedwater (FW) and reactor coolant system (RCS) connections on the old steam generators (OSG). The inspectors examined welding procedures, welder qualifications, weld filler metal selection, non-destructive test procedures, examiner qualifications, acceptance criteria and test results for compliance with the requirements of the ASME Section XI. The inspectors examined training and qualification records of selected personnel performing welding and non-destructive examination. The inspectors reviewed a sample of Work Plan and Inspection Records (WIPIR) based on their risk significance, which covered the welding and installation of FW, MS and RCS piping, fittings and pipe supports. The work planning and control documents were also examined to ensure specified work tasks were appropriate and that provisions and "hold points" to check and close out the activities at the completion of work were appropriate.

Specific work activities inspected included observation of the qualification of welders using the cadweld process for the welding of re-bar to close the equipment opening cut in the containment building. The inspectors observed joint preparation, sleeve installation, firing, removal, cleaning and testing of a sample of the qualification welds. Also, the inspectors observed the machining of weld end preparations on selected samples of MS, FW and RCS piping spools and assessed the dimensional accuracy for comparison with specification requirements. The inspectors observed the shop welding of various piping spools in the pipe fabrication shop prior to their movement to the field

for installation. The inspectors observed the field fit-up, tack welding and final welding of a sample of these components to their respective steam generator nozzles.

One field weld from the MS, FW and RCS piping was selected by the inspectors for an in-depth inspection, beginning with the severance at the generator nozzle to the re-attachment to the steam generator, including in-process and final non-destructive testing of the new welds. The final non-destructive tests for the acceptance of these risk significant welds involved surface, radiographics and ultrasonic testing. The inspectors reviewed the test activities and inspection reports to assess the completeness of the examination processes. Also, the inspectors conducted interviews with engineers, construction, and examination personnel responsible for implementing these activities.

In addition to field activities, the inspectors reviewed the non-destructive test results of the pre-service examinations of the replacement steam generator (RSG) shell, nozzle and attachment welds, and the baseline examination of the generator tubes. The inspectors also reviewed the SG Degradation Assessment developed in advance of the pre-service tube inspections (PSI) to be performed with respect to the limited range of degradation mechanisms that may pertain to the as-built condition.

The inspectors reviewed the security considerations associated with vital and protected area barriers that may have been affected by the steam generator replacement activities, and conducted a review of the licensee’s compensatory measures. The inspectors also conducted a walkdown and review of security boundaries affected by the steam generator replacement project and conducted a review of the licensee’s compensatory measures.

The inspection of the containment construction opening included a review of the licensee’s plans, procedures, and processes to create a sufficiently large and safe access to the interior of the containment structure. This review included discussions with cognizant technical and management personnel, visual inspections of the work-in-progress, and review of applicable documents. The review of documents included the Design Basis of the containment as described in the UFSAR and Design Change Packages, which included the 10 CFR 50.59 safety evaluation for the modification, relevant design and the construction drawings developed for the opening, and the structural analysis of the containment structure. The work control package for the demolition of concrete, removal of reinforcing bars, and the plan and process for restoring the containment structural integrity after the SG/Reactor Head replacement were also reviewed. In addition, the inspectors evaluated the control of safety-related materials during demolition and restoration of structural concrete, including the reinforcing bar splicing methods and materials. The review also included the licensee’s application of industry codes and standards in the development of the restoration process of concrete and reinforcing bars.

The inspectors observed the restoration, welding and NDE of the containment liner, the re-welding of reinforcing bars, and the placement of the concrete to close the temporary construction opening. The inspectors monitored the sampling, testing, and curing of the concrete to verify that adequate strength was achieved prior to containment Type A

certification test. Additionally, the inspectors reviewed a sampling of the licensee's test reports from the IWE inspections performed on the interior of the containment and outside of the construction opening.

b. <u>Findings</u>

No findings of significance were identified.

.7 <u>To verify that SG removal and replacement activities maintain adequate nuclear and radiological safety</u>

a. <u>Inspection Scope</u>

The inspector evaluated the adequacy of the rigging, lifting and handling procedures that would be used to control the movement of the steam generators. The inspectors verified that the licensee had adequately addressed the potential impact on existing plant SSCs of both units due to the movement of these heavy loads.

The inspectors verified that temporary electrical power to support construction efforts inside the Unit 1 containment were independent from Unit 1 and Unit 2 operational electrical loads.

The inspectors reviewed the following radiation protection program attributes utilizing the applicable portions of inspection procedures IP 71121.01 and 71121.02:

- ALARA planning
- Dose estimates and dose tracking.
- Exposure controls including temporary shielding.
- Contamination controls.
- Radiological material management.
- Radiological work plans and controls.
- Emergency contingencies.
- Project staffing and training plans.
- Radiological safety plans for storage of retired steam generators and components.

b. <u>Findings</u>

No findings of significance were identified.

.8 <u>Verify that the SG post-installation test program is technically adequate, in conformance with requirements, and satisfactorily implemented</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the pre-service inspection of new welds for the replacement steam generators, the testing and restoration of the containment construction opening,

and the results of the containment Type “A” leakage test. The licensee will be performing the following tests to ensure that the plant modifications have been successfully completed:

- RCS flow verification
- RCS leakage testing
- SG secondary side leakage testing
- Calibration and testing of instrumentation affected by SG replacement
- SG performance testing - moisture carryover tests

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

The inspector presented the1R08 inspection results to Mr. James Lash, Site Vice President, and other members of the licensee staff, at the conclusion of the inspection on March 30, 2006. The licensee acknowledged the conclusions and observations presented.

On April 7, 2006, the inspector presented 2OS1/2OS2 inspection results to Mr. Mende and other members of his staff. FENOC acknowledged the findings.

On July 24, 2006, the inspectors presented the normal baseline inspection results to Mr. James Lash, and other members of the licensee staff. The inspector confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Alberti	Senior Nuclear Specialist
S. Baker	Site Radiation Protection Manager
R. Bisbee	Supervisor, Nuclear Performance Improvement
R. Bologna	Manager, Site Operations
R. Boyle	Staff Nuclear Engineer
S. Buffington	Staff Nuclear Engineer
G. Cacciani	Staff Nuclear Engineer
A. Castagnacci	Supervisor, Nuclear HP Services
J. Clark	Radiation Protection Health Services Technician
C. Custer	SGRP
G. Davie	Manager, Training
P. Dearborn	Staff Nuclear Engineer
R. Dulee	Project Manager, SGRP
D. Dwulit	Supervisor, I & C Maintenance
R. Feden	Regulatory Compliance
J. Fontaine	Supervisor, ALARA
K. Frederick	Senior Consultant
J. Freund	Supervisor, Rad Operations Support
D. Girdwood	Supervisor, Technical Training
D. Grabski	ISI Coordinator
J. Habuda	U-1 Auxiliary Feedwater and Main Steam System Engineer
K. Halliday	SGRP
R. Hansen	Manager, Nuclear Oversight
A. Hartner	U-1 Shift Manager
T. Heimel	NDE Level III
M. Hernandez	Nuclear Unit Supervisor
S. Hovanec	Supervisor, System Engineering
C. Hrelec	Senior Radiation Protection Technician
L. Hunt	Supervisor, I & C Maintenance
S. Janes	Senior Radiation Protection Technician
R. Jenkins	Primary Simulator Instructor
H. Kahl	Design Engineering
G. Kammerdeiner	Principal Consultant
J. Kasunick	Superintendent, Nuclear
C. Keller	Supervisor, PRA and Safety Analysis
T. King	Reactor Control System Engineer
J. Kramer	Senior Nuclear Specialist, QA
J. Lash	Site Vice President
R. Lieb	Manager, Plant Engineering
T. Lonnett	Senior Nuclear Specialist

G. Loose	Unit 2 Shift Manager
T. Mahoney	Senior Radiation Protection Technician
C. Mancuso	Supervisor, Nuclear Mechanics
M. Manoleras	Manager, Design Engineering
J. Maracek	Regulatory Affairs
J. Mauck	Senior Nuclear Specialist
D. McKee	Supervisor, RadWaste Operations
R. Mende	Director, Site Operations
L. Meyers	Executive Vice President, Projects
D. Mickinac	Senior Nuclear Specialist
C. Miller	Senior Radiation Protection Technician
J. Miller	Fire Protection Engineer
N. Morrison	SGRP
M. Mouser	Unit 1 Shift Manager
J. Patterson	Unit 1 Containment System Engineer
B. Paul	Senior Nuclear Specialist
P. Pauvlinch	Rapid Response Supervisor
R. Pucci	Senior Nuclear Specialist, ALARA Coordinator
J. Rinckl	Fleet Oversight
G. Ritz	Nuclear Engineer
J. Saunders	Supervisor, Dosimetry
K. Schweikart	Staff Nuclear Engineer
D. Schwer	Shift Manager
J. Scott	Supervisor, I & C Maintenance
P. Sena	Director Engineering
B. Sepelak	Supervisor, Regulatory Compliance
M. Shaw	Advanced Nuclear Instructor, HP/Radwaste/Shipping
J. Sipp	Manager, Chemistry
T. Sockaci	Supervisor, Nuclear Engineering
G. Storolis	Unit 2 Shift Manager
H. Szklinski	Nuclear Quality Assessor
W. Williams	BACC Program Owner
J. Witter	Shift Manager
K. Wolfson	Superintendent, Nuclear Maintenance

<u>Other Personnel</u>

L. Boynton	Westinghouse RRVCH Lead
P. Dadlani	Bechtel
G. Dee	Westinghouse RRVCH Lead
T. Dorris	Bechtel
F. Gigele	
C. Holmes	WesDyne Project Manager
D. Miller	Bechtel
W. Olson	Bechtel

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000412/2004002-01	URI	Potential for Spurious Opening of PORVs Due to Fire Induced Hot Shorts (Section 4OA5)

Open/Closed

05000412/2005001-01	LER	Containment Isolation Valve Relay Failure Unknowingly Leads to Technical Specification Noncompliance. (Section 4OA3)
05000334/2006001-00	LER	Main Steam Valve Relief Tests Exceeded Technical Specification Required Setpoint Tolerance (Section 4OA3)
05000412/2006001-00	LER	Turbine-Generator Trip Due to Loss of Generator Excitation Power Results in Reactor Trip (Section 4OA3)
05000412/2006002-00	LER	Entry into Technical Specification 3.0.3 Due to Inoperability of Both Trains of the Supplemental Leak Collection and Release System (Section 4OA3)
05000412/2006003-01	URI	Beaver Valley Unit 2 NOED for Two Trains of SLCRS OOS (06-1-01). (Section 1R14)
05000334/2006003-02	NCV	Failure to Perform an Adequate Operability Determination For Current Leakage

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures
2-ADM-2033, “Risk Management Program”
2OST-7.6, “Centrifugal Charging Pump [2CHS*P21C]”
UFSAR-2 Section 6.3.4.2, “Reliability Tests and Inspection Description of Test Planned”
UFSAR-2 Section 9.3.4.4, “Inspection and Testing Requirements”
Technical Specification BCPS-2 3.6.2.3, “Chemical Addition System”
2OST-36.7, “Offsite to Onsite Power Distribution System Breaker Alignment Verification,” Rev. 8
1OM-11.3.B.1, “Valve List - 1SI,” Rev. 14
1OM-11.3.C, “Power Supply and Control Switch List,” Rev. 8
2OM-13.3.B.1, “Valve List - 2QSS,” Rev. 9
2OM-13.3.C, “Power Supply and Control Switch List,” Rev. 7

Drawings
10080-RM-407-1A, Rev. 14, Sheet 1, “Valve Diagram Chemical and Volume Control System”
10080-RM-407-1A, Rev. 4, Sheet 2, “Valve Diagram Chemical and Volume Control System”
10080-RE-1E, Unit 2 “4160V One Line Diagram, Sh. 2,” Rev. 9
10080-RE-1F, Unit 2 “4160V One Line Diagram, Sh. 3,” Rev. 19
8700-RM-411-1, “Safety Injection System,” Rev. 18
10080-RM-413-2, “Quench Spray System,” Rev. 13

Condition Reports (IR)
05-02708, 05-03192, 05-07736, 06-01210, 06-02639

Technical Specifications - Unit 2 (through Amendments dated 02/10/06)
3/4.8.1, A. C. Sources
3/4.8.2, Onsite Power Distribution System

BVPS UFSAR Unit 2, Rev. 15
Section 8.2, Offsite Power System
Section 8.3.1, AC Power Systems

Section 1R05: Fire Protection

Procedures
1/2-ADM-1900, “Fire Protection,” Rev.13

Other
BVPS Unit 1 Updated Fire Protection Appendix R Review, Rev. 26
BVPS Unit 2 Fire Protection Safe Shutdown Report, Addendum 28

BVPS UFSAR Unit 1, Rev. 22
Section 7.4, Systems Required for Safe Shutdown

BVPS UFSAR Unit 2, Rev. 15
Section 7.4, Systems Required for Safe Shutdown
Section 8.3.3, Fire Protection for Cable Systems

Section 1R06: Flood Protection Methods

Procedures

2OM-13.4.AAC, Rev. 0	Recirc Spray Instrument Pit Level High
2OM-13.1.E, Rev. 3	Containment Depressurization System - Specific Instrumentation And Control
2OM-13.4.AAI, Rev. 9	Recirc Spray System Trouble
2OM-13.2.B, Rev. 3	Containment Depressurization System - Precautions, Limitations and Setpoints

Section 1R08: Inservice Inspection

Procedures

NDE-VT-510	NDE-VT-502	NDE-VT-513	NDE-VT-500

CR's

06-00838	06-01987	04-07768	971793	04-08309

Other
RCP 1C Seal Supply Drain Mode Hold Resolution Forms (for 2004 and 2006)
1CH-327 Boric Acid Corrosion Control Leak Inspection
'A' RHR Pump Discharge Check Valve Mode Hold Resolution Form
1R17 Boric Acid Team Process
1R17 Mode Hold List
Boric Acid Corrosion Control Program
Liquid Penetrant Examination, Summary No. 058700
UT Erosion/Corrosion Examination, Summary No. CA 04-08309-02
Beaver Valley Unit No. 1 Sixteenth Refueling Outage Inservice Inspection Report
Visual Examination for Leakage (VT-2) Summary No. Press. Nozzles

Certifications
Memorandum, Review of NDE Certifications – Integrated Technologies Incorporated Personnel
Various Performance Demonstration Initiative program documents
FITS Qualification Matrices – Boric Acid Corrosion Control Inspectors

Section 1R11: Licensed Operator Requalification Program

Procedures
1OM-53A.1.E-0(ISS1C), "Reactor Trip or Safety Injection," Issue 1C, Rev. 9
1OM-53A.1.E-2(ISS1C), "Faulted Steam Generator Isolation," Issue 1C, Rev. 4
1OM-53A.1.ES-0.1(ISS1C), "Reactor Trip Response," Issue 1C, Rev. 6
1OM-53C.4.1.6.7, "Excessive Primary Plant Leakage," Rev. 1
1/2OM-48.1.I, "Technical Specifications Compliance," Rev.19

Other
1LRTS-2006M4, "Operations at Power with Malfunctions," Rev. 0

Section 1R12: Maintenance Rule Implementation

Drawings
8700-RM-421-1, "VALVE OPER NO DIAGRAM MAIN STEAM," Rev. 17

Calculations
8700-UR(B)-219, "Site Boundary and Control Room Doses following a Steam Generator Tube Rupture Based on Core Uprate and Alternative Source Term Methodology," Rev. 2
8700-UR(B)-219, "Site Boundary and Control Room Doses following a Steam Generator Tube Rupture Based on Core Uprate and Alternative Source Term," Rev. 2, Addendum 3

Procedures
NOP-LP-2001, "Corrective Action Program," Rev 13
1/2-ADM-2113, "Operability Determination & Basis For Continuous Operation," Rev. 2
1OM-53A.1.E-3(ISS1C), "Steam Generator Tube Rupture," Issue 1C, Rev. 8

Other
BVPS, Unit 1 Licensing Requirements Manual, LR 6.1, Atmospheric Steam Release (ASR) Valves, Rev. 11
BVPS, Unit 1 Licensing Requirements Manual Bases, B.6.1, Atmospheric Steam Release Valves, Rev. 11
Maintenance Rule System Basis Document, Unit 1, System 21, Rev. 7
NRC Generic Letter 91-18, "Resolution of Degraded and Nonconforming Conditions," Rev 1
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 273 to Facility Operating License No. DPR-66, First Energy Operating Company, First Energy Nuclear Generation Corp., Beaver Valley Power Station, Unit No. I (BVPS-1), Docket No. 50-334
Unit 1 Operations Log dated June 6, 2006
Unit 1 Operations Log dated June 7, 2006

Condition Reports
CR-06-03664, Steam Leak-by on 1MS-26 (Residual Heat Release Isolation Valve)
CR-06-03726, HCV-1MS-104 and 1MS-26 Leak-by
CR-06-03904, Additional SGTR Dose Consequences of HCV-1MS-104/1MS-26 Leak

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures
1/2-ADM-2033, "Risk Management Program," Rev. 3
1OM-7.4.Q, "Makeup to the Refueling Water Storage Tank," Rev. 8
2MSP-1.04-I, "Solid State Protection System Train A Bi-Monthly Test," Issue 4, Rev. 30
2OM-4.4.AAC, "PRI/SEC Process Rack Power Supply Failure," Rev. 7
2OM-53A.1.ES-1.2(ISS1C), "Post LOCA Cooldown and Depressurization," Issue 1C, Rev. 8
NOP-OP-1005, "Shutdown Safety," Rev. 09
NOP-WM-2001, "Work Management Process," Rev. 4
1/2-ADM-0804, "On-Line Work Management and Risk Assessment," Rev. 4
1/2-ADM-2114, "Maintenance Rule Program Administrative Procedure," Rev. 2
Conduct of Operations Procedure 1/2OM-48.1.I, "Technical Specification Compliance," Rev. 18.

Work Orders
200205488, Replace Universal logic cards in Train A

200208344, Troubleshoot problem with operation of BV-2SIS-HC868A operation.

Condition Reports
CR-06-02717, Yellow Electrical Shutdown Risk Not Planned in the Shutdown Safety Report

Other
ECP No. 02-0269, “Piping Changes to 1EE-P-2A and 2B Lube Oil Circulation Pump Discharge Lines,” Rev. 1
BVPS Weekly Maintenance Risk Summary for the Week of May 29, 2006, Rev. 0
BVPS Unit 2 Weekly Maintenance Risk Summary for the Week of June 5, 2006, Rev. 0, 1, and 2
Unit 2 Operations Logs dated May 4, 2006

Technical Specifications - Unit 2 (through Amendments dated 02/10//06)
3/4.7.8, Supplemental Leak Collection and Release System (SLCRS)

Section 1R14: Personnel Performance During Non-routine Plant Evolutions

Procedures
2OM-51.4.L, “Station Shutdown from 40% Power to Mode 5,” Rev. 14
2OM-52.4.B, “Load Following,” Rev. 43

Technical Specifications - Unit 2 (through Amendments dated 02/10//06)
3.0.3, Limiting Condition for Operation
3/4.7.8, Supplemental Leak Collection and Release System (SLCRS)

Condition Reports
CR-06-02770, Unit 2 Fire protection System Deluge Valve Actuations

Other
Event Notification Worksheet 42491 dated April 11, 2006
Unit 2 Operations Logs dated April 11-12, 2006

Section 1R15: Operability Evaluations

Drawings
10080-RM-513-1, “Flow Diagram - Containment Depressurization System,” Rev. 12

Calculations
8700-US(B)-263, “Assessment of Beaver Valley Unit 1 for Design Basis Accidents for Containment Atmospheric Conversion Project,” Rev. 2

Procedures
2BVT 1.16,6, “SLCRS Train A Filter Efficiency and Flow Test,” Rev. 6
2BVT 1.16,6, “SLCRS Train A Filter Efficiency and Flow Test,” Rev. 7

2BVT 1.16,7, “SLCRS Train B Filter Efficiency and Flow Test,” Rev. 7
2BVT 1.16,8, “SLCRS Train A Charcoal Sample,” Rev. 5
2BVT 1.16,8, “SLCRS Train A Charcoal Sample,” Rev. 6
2BVT 1.16,11, “SLCRS Train B Charcoal Sample,” Rev. 5
1/2-ADM-2115, “Fatigue Cycle Monitoring Program,” Rev. 0

Technical Specifications - Unit 1 (through Amendments dated 04/03/06)
3/4.5.2, ECCS Subsystems - Tavg > or = 350 degrees F
3/4.6.2.1, Containment Quench Spray System
3/4.6.2.2, Containment Recirculation Spray System
BASES 3/4.6.2.1 and 3/4.6.2.2, Containment Quench and Recirculation Spray Systems

Technical Specifications - Unit 2 (through Amendments dated 02/10/06)
3/4.7.8, Supplemental Leak Collection and Release System (SLCRS)

BVPS UFSAR Unit 1, Rev. 22
Section 6.4, Containment Depressurization System
Section 14.3, Loss-of-Coolant Accident

Condition Reports (* denotes a CR generated as a result of this inspection)
06-02985*, 10 CFR 50.59 Screen Not Performed in a Timely Manner
06-03127, LIR - Certain Quench Spray Flow Lines Not Periodically Tested or Inspected
06-03134, 2HVS-FLTA205A Charcoal Sample Fails Radioactive Iodine Removal Efficiency Test
06-03199, Charcoal Samples for SLCRS Filter Banks for 2HVS-FLTA205A, 205B, 208B
06-03239, Provide an Engineering Analysis of the reported 1FW-P-4 discharge pressure of 1450 psig during the performance of 1OST-24.7 on 5/9/06
06-03630, Pin Hole Leak on Piping to 2SWS-486 (Vacuum Break for 2SWS-P21A)

Vendor Manual
2503.180-44A-005, Vendor Manual for Refurbished 1747 KIP Hydraulic Snubbers, Rev B.

Miscellaneous
Letter from L. William Pierce, Site Vice President, Beaver Valley Power Station to NRC, Subject: “Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 License Amendment Request Nos. 317 and 190,” dated June 2, 2004
Letter from NRC to Mr. James Lash, Vice President, Beaver Valley Power Station, Subject: “Beaver Valley power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) - Issuance of Amendments Re: Containment Conversion from Subatmospheric Operating Conditions (TAC Nos. M3394 and MC3395),” dated February 6, 2006

Section 1R17: Permanent Plant Modifications

Other
ECP No. 02-0269, “Piping Changes to 1EE-P-2A and 2B Lube Oil Circulation Pump

Discharge Lines," Rev. 1
Unit 1 Operations Log dated April 10 - 12, 2006

Condition reports
CR-01-0394, Lube Oil Circulation Pump

Section 1R19: Post-Maintenance Testing

Procedures
1MSP-6.14-I, "P-457, Pressurizer Pressure Channel III Test," Issue 4, Rev. 9
1MSP-21.24-I, "P-1MS486 Loop 2 Steamline Pressure Protection Channel IV Calibration," Issue 4, Rev. 4
1OM-53A.1.E-3(ISS1C), "Steam Generator Tube Rupture," Issue 1C, Rev. 8
1OST-24.7, Dedicated Auxiliary Feed Pump [1-FW-P-4] Test, Rev. 13
2MSP-1.04-I, "Solid State Protection System Train A Bi-Monthly Test," Issue 4, Rev. 30
2OST-7.6, "Operating Surveillance Test Centrifugal Charging Pump [2CHS*P21C]," Rev. 27
1/2CMP-7CH-P-1A-B-C-2M, "Charging/High Head Safety Injection Pump Bearing Inspection," Issue 4, Rev. 10
1/2PMP-7CH-P-GEARCASE-1M, "Charging/High Head Safety Injection Pump Gearcase Overhaul," Issue 4, Rev. 10
2OM-54.2.S1, Unit 2 Operations Log, dated April 2 - 4, 2006
2OST-36.2, Emergency Diesel Generator (2EGS*EG2-2) Monthly Test, Rev 48, dated April 4, 2006
CR 06-02525, Control Failure Alarm Received during Shutdown of EDG 2-2

Technical Specifications - Unit 1 (through Amendments dated 04/03/06)
3/4.3.1, Reactor Trip System Instrumentation
3/4.3.2, Engineered Safety Feature Actuation System Instrumentation

Technical Specifications - Unit 2 (through Amendments dated 02/10/06)
3/4.3.1, Reactor Trip System Instrumentation
3/4.3.2, Engineered Safety Feature Actuation System Instrumentation
3/4.5.2, ECCS Subsystems - Tavg > or = 350 degrees F

Work Orders
200129774, Obtain Oil Sample from Each Motor Lube Oil reservoir [1-FW-P-4 Motor]
200129775, Sample Both Pump Oil Reservoirs [1-FW-P-4]
200132842, Change Inboard/Outboard Pump Bearing Reservoir Oil [1-FW-P-4]
200133090, Perform Bearing Inspections High Head Safety Injection Pump
200133455, Replace Relays K1613, K1004, K506, and K510 during Performance of 1MSP-6.14
200149498, Verify Operability of RX Protection Logic System Train A Test
200210462, ECP 04-0295-07; Revise lead/lag constants
200205488, Replace Universal Logic Cards in Train A
200208986, Corrective Maintenance to valve positioner for Residual Heat Release Valve (HCV-1MS-104)

200204018, Troubleshoot to determine cause of control failure on Unit 2 EDG 2EGS-EG2-2

Condition Reports
CR-06-03696, Positioner Adjustments Found Loose
CR-06-03902, PM-1MS-486D As Found Time Constants Found Out of Tolerance
CR-06-03310, Thermal Overload Trip of MCC2-E08-5F

Other
BVPS, Unit 1 Licensing Requirements Manual, LR 6.1, Atmospheric Steam Release (ASR) Valves, Rev. 11
BVPS, Unit 1 Licensing Requirements Manual Bases, B.6.1, Atmospheric Steam Release Valves, Rev. 11
Unit 2 Operations Log dated April 26, 2006
Unit 1 Operations Log dated May 10, 2006
Unit 1 Operations Log dated June 6, 2006
Unit 1 Operations Log dated June 7, 2006

Section 1R20: Refueling and Outage Activities

Procedures
1RST-2.2, “Core Design Check Test,” Rev 8
2OM-50.4.D, “Reactor Startup From Mode 3 to Mode 2,” Rev. 45
2OM-50.4F, “Performing An Estimated Critical Position Calculation,” Rev. 5
2OM-52.4.A, “Increasing Power from 5% Reactor Power and Turbine on Turning Gear to Full Load Operation,” Rev. 53
NOBP-OM-4010, “Restart Readiness for Plant Outages” Rev. 3
1BVT-1.47.1, “Containment Structural Integrity Test,” Rev. 6
1BVT 1.47.2, “Containment Type A Leak Test,” Rev.5
1BVT 2.1.1, “Control Rod plant Exercise and Data Collection,” Issue 1, Rev. 0
1RST-2.1, “Initial Approach to Criticality after Refueling,” Rev. 9
1OM-50.4L, “Plant Heatup From Mode 6 to Mode 3,” Rev. 12
1OM-50.4.D, “Reactor Startup From Mode 3 to Mode 2,” Rev. 46
NOP-OP-1005, “Shutdown Safety,” Rev. 8
2OM-50.4D, “Reactor Startup From Mode 3 to Mode 2,” Rev. 45
1OM-20.4.E, “Draining The Refueling Cavity,” Rev.29
1OST-11.14B, “HHSI Full Flow Test,” Rev. 15
1OST-24.8A, “3A Motor Driven Auxiliary Feed Pump Check Valves and Flow Test,” Rev. 3
1OM-50.4L, “Plant Heatup From Mode 6 to Mode 3, Data Sheet 2: RCS Heatup/Cooldown Determination,” Rev.12
1MSP-9.04-M, “Containment Sump Inspection,” Rev. 7
1OST-47.2B, “Containment Closeout Inspection,” Rev. 4

Condition Reports
CR-06-02797, Rod Drop During BVT 2.1.1
CR-06-02802, Rod Bottom Lights Issue During Performance of 1BVT 2.1.1
CR-06-02643, U2 ECP Calculation/B-10 Correction Review

Technical Specifications - Unit 1 (through Amendments dated 04/03/06)
4.6.1.2.a, Containment Systems, Containment Leakage, Surveillance Requirements
6.17, Containment Leakage Rate Testing Program

Other
FENOC Test Results Report for 1BVT 1.47.2, “Containment Type A Leak Test,” Rev. 5
FENOC Test Results Report for 1BVT-2.1.1, “Control Rod Cold Plant Exercise and Data Collection,” Issue 1, Rev. 0

Miscellaneous
Generic Letter 88-17, “Loss of Decay Heat Removal”
Narrative Logs
Shutdown Safety Risk Status Sheets
NRC Information Notice 2005-16, “Outage Planning and Scheduling - Impacts on Risk”

Section 1R22: Surveillance Testing

Procedures
1/2OM-48.3.D, “Administrative Control of Valves and Equipment,” Rev. 4
1OST-7.4, “Centrifugal Charging Pump Test [1CH-P-1A],” Rev. 34
1OST-24.10, “Operating Surveillance Test, Auxiliary Feedwater System Monthly Verification,” Rev. 7
1OST-36.1, “Diesel Generator No. 1 Monthly Test,” Rev. 44
2OST-24.4, “Steam Driven Auxiliary Feed Pump [2FWE*P22] Quarterly Test,” Rev. 56

Technical Specifications - Unit 1 (through Amendments dated 04/03/06)
4.0.5, Surveillance Requirements for Inservice Inspection and Testing
3/4.5.2, ECCS Subsystems - Tavg > or = 350 degrees F
3/4.8.1, A.C. Power Systems

Section 1R23: Temporary Plant Modifications

Condition Reports
05-06165
06-02832
06-03033, ERF Ventilation Damper D-3 Failed to Close During Performance of 1/2OST-58.8

Regulatory Applicability Determination and 10 CFR 50.59 Screens
05-04788
06-02073

Procedures
NOP-LP-4003, Evaluation of Changes, Tests, and Experiments, Rev 3.
1/2-ADM-2028, “Temporary Modifications,” Rev 5
1/2OST-58.8, “ERF Emergency Ventilation Recirculation Test,”, Rev. 6

Drawings
10080-RM-444D-2, Valve Oper No Diagram Aux Bldg Ventilation System, Rev 10.

Calculations
10080-MT-271, BVPS-2 Charging Pump Cubicle Heatup Following a Loss of HVAC, Rev 0, Addendum 3.
12241, Primary Aux Building and Waste Handling Building Air Conditioning Loads and Air Flow Rates, Rev B-2A.
10080-MT-266, Evaluation of Charging Pump Cubicle Temperatures Following a DBA, Rev 0, Addendum 3.

Other
TM No. 1-06-06, Temporary Modification to ERF Damper D-3 dated April 26, 2006
Regulatory Applicability Determination No.06-02287 for Temporary Modification to ERF Damper D-3. Rev. 0
TM No. 2-05-14, UPS-VITBUS2-3 Regulator MOV Termination Disconnection, Rev. 0

Section 1EP6: Emergency Action Level (EAL) Revision Review
Beaver Valley Power Station Emergency Preparedness Plan:
- Section 1, Definitions, Rev 16
- Section 2, Scope and Applicability, Rev 15
- Section 3, Summary, Rev 15
- Section 4, Emergency Conditions, Rev 19
- Section 5, Emergency Organization, Rev 20
- Section 6, Emergency Measures, Rev 23
- Section 7, Emergency Facilities and Equipment, Rev 21
- Section 8, Maintaining Preparedness, Rev 20

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

Procedures

1/2-ADM-1601, Rev 13	Radiation Protection Standards
1/2-ADM-1611, Rev 8	Radiation Protection Administrative Guide
1/2-ADM-1621, Rev 3	ALARA Program
1/2-ADM-1630, Rev 9	Radiation Worker Practices
1/2-ADM-1631, Rev 5	Exposure Control
1/2-HPP-3.02.003, Rev 8	Decontamination Control
1/2-HPP-3.02.004, Rev 4	Area Posting
1/2-HPP-3.04.002, Rev 5	Bioassay Administration
1/2-HPP-3.05.001, Rev 4	Exposure Authorization
1/2-HPP-3.07.002, Rev 4	Radiation Survey Methods
1/2-HPP-3.07.013, Rev 3	Barrier Checks
1/2-HPP-3.08.001, Rev 8	Radiological Work Permit
1/2-HPP-3.08.003, Rev 9	Radiation Barrier Key Control
1/2-HPP-3.08.005, Rev 4	ALARA Review Program

BVBP-RP-0003, Rev 3	Dosimetry Practices
BVBP-RP-0013, Rev 2	Radiation Protection Risk Assessment Process
NOP-WM-7001, Rev 0	ALARA Program
NOP-WM-7002, Rev 0	Operational ALARA Program
NOP-WM-7003, Rev 0	Radiation Work Permit
NOP-WM-7017, Rev 0	Contamination Control Program
NOP-WM-7021, Rev 0	Radiological Postings, Labeling, and Markings

Condition Reports

06-01457, 06-01458, 06-01949, 06-01958, 06-01639, 06-01707, 06-01685, 06-01629, 06-02268, 06-02211, 06-01632, 06-01633, 06-01634, 06-01683, 06-01688, 06-01692, 06-01723, 06-02292, 06-02290, 06-02291, 06-02252, 06-01572, 06-01554, 06-01563, 06-01566

ALARA Post-Job Reviews

RWP#106-4019, ALARA Plan #06-1-11, Reactor Disassembly/Reassembly
RWP#106-4022, ALARA Plan #06-1-42, Refueling Operations
RWP#106-4023, ALARA Plan #06-1-14, Upper Internals Lift
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RWP#106-4033, ALARA Plan #06-1-54, AOV Testing on PCV-RC-455A &B
RWP#106-4040, ALARA Plan #06-1-43, Replace 1CH-1
RWP#106-4044, ALARA Plan #06-1-47, Repair RC-261
RWP#106-8006, ALARA Plan #06-1-28, Welding of New Steam Generators
RWP#106-8007, ALARA Plan #06-1-60, SGRP Pipe End Decon
RWP#106-8014, ALARA Plan #06-1-36, Transport of old SG's & Reactor Head to Storage
RWP#106-8016, ALARA Plan #06-1-38, Disassembly Old Reactor Head
RWP#106-4031, ALARA Plan #06-1-18, Pressurizer Inspections

ALARA Committee Meeting Minutes

Meeting Nos:1R17-03, R17-04,1R17-07, 1R17-08, 1R17-09,1R17-11, 1R17-12, 1R17-13, 1R17-14,1R17-15, 1R17-16, 1R17-17, 1R17-18, 1R17-19

Miscellaneous Reports

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

Section 2PS2: Radioactive Material Processing and Transportation

Procedures

NOP-OP-2002, Rev 5	Shipment of Radioactive Material/Waste
1/2-PCP-1.01, Rev 1	Process Control Program (PCP)
1/2-HPP-3.02.003, Rev 3	Decontamination Control
1/2-HPP-3.03.004, Rev 3	Handling Radioactive Material
1/2-HPP-3.03.005, Rev 2	Removing Material from an RCA
1/2-HPP-3.03.042, Rev 4	Radioactive Waste/Material/Container Accountability

1/2-HPP-3.07.014, Rev 1	Sampling of Volumetric Materials and Miscellaneous Media for Radioactivity Evaluation
BVBP-RP-0022, Rev 0	Temporary On-Site Storage of Radioactive Waste (Interim Waste Storage Facility)
FO-AD-002, Rev 32	Operating Guidelines for Use of Polyethylene High Integrity Containers

Manifest Reviews
Shipment No. 3247, Mixed Filter Media, LSA II
Shipment No. 3260, Reactor Vessel Surveillance Specimen, Type A
Shipment No. 3290, Dewatered Resin, Type B
Shipment No. 3300, Dewatered Resin, Type B,
Shipment No. 3307, Dewatered Resin, LSA II
Shipment No. 3308, Dewatered Resin, Type B

Nuclear Quality Assessment Field Observations & Assessment Report
BV120062616, BV320052229, BV320052220, BV120052152, BV120062608
Assessment Report Third Quarter 2005 (BV-C-05-03)

Training Related Materials
Course Materials for MISC-3105, Packaging, Transport, and Disposal of Radioactive Waste
Course Materials for GEN-USDOT-FEN-01, USDOT Regulations General Awareness Training
Training Certificates for Individuals Responsible for Classifying radioactive materials and preparing radioactive material for shipment

Condition Reports
06-02628, 06-00792, 06-02372, 05-05552, 05-05147, 05-04100, 05-03133, 05-03098, 05-03034, 05-00694, 05-00038, 05-06858, 05-05135, 04-06569, 05-05952, 04-04165

Section 4OA2: Identification and Resolution of Problems

Calculations

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

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

Miscellaneous
2OM-56B Safe Shutdown Procedures Manual Operator Actions Timeline Study, dated January 11, 2004
2OM-56C Safe Shutdown Procedures Manual Operator Actions Timeline Study, dated January 11, 2004
Work Order 200013371, Addendum Nos. 1 and 3
Material & Revision Request No. 13140
Pump Vendor Drawing 300-B51106
Lube Oil Cooler Vendor Manual Drawing 5-162-08-042-00213
Beaver Valley Unit 2 Dwg. 10080-2806-259-920-048-SH
Lube Oil System Drawing 450-B51106
Charging Pump Outline No. 300-B51106

Section 4OA3: Event Response

Condition Reports
CR-06-02520, "Unit 2 Reactor Trip"

Procedures
2OM-53A.1.ES-0.1 (ISS1C), Rev. 4, Reactor Trip Response
2OM-53A.1.E-0 (ISS1C), Rev. 6, Reactor Trip Or Safety Injection
2OM-36.1.C, Rev 3, 4KV Station Service System

Section 4OA5: Other Activities
American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and Section III, 1968 Edition through the Winter 1968 Addenda
Westinghouse General Equipment Specification 676413, Revision 1 with Addendum Equipment Specification 678801, Rev. 2
Westinghouse Procedure MRS-SSP-1824, Beaver Valley Unit 1 - Installation of CRDMs, Guide Cones, and CETNA Head Port Adapters, Rev. 0
FENOC Specification No. 8700-DGS-0035, Rev. 2, Procurement Specification for Replacement Reactor Vessel Closure Head at Beaver Valley Unit 1
FENOC Specification No. 8700-DGS-0038, Rev. 1, Procurement Specification for Simplified Head Assembly and Installation Services for Replacement Reactor Vessel Closure Head for Beaver Valley Power Station Unit 1
FENOC Specification No. 8700-DGS-0051, Rev. 1, General Design Specification (G-677470) For Procurement of Control Rod Drive Mechanisms (1RC-R-1) Endorsement of Westinghouse Specification G-677470, Rev. 5
FENOC Procedure 1/2-ADM-0819, Handling of NUREG 0612 Heavy Loads, Rev. 4
Bechtel Construction Procedure CP-C-9, Rigging, Rev. 2
Bechtel Work Plan and Inspection Record, C-RCR-147, Handling RVCHs Outside the Containment, Rev. 1
Bechtel Work Plan and Inspection Record, C-RCR-148, Rigging and Handling of the RVCHs Inside the Containment, Rev. 3

Bechtel Order # 24928-400-HC4-MJKP-000001, Beaver Valley Unit 1 Steam Generator Replacement Project Polar Crane Inspection Report, completion dated 2/13/2006
Preventive Maintenance - Work Order 200124437, Take Oil Samples: 1CR-1 Reactor Containment Crane, completed 2/14/2006
Preventive Maintenance - Work Order 200124436, 1CR-1 Mechanical Inspect & Test Reactor Containment Crane, completed 2/16/2006
1/2-PMP-60-001, Reactor Containment Polar Crane Mechanical Inspection and Lubrication, Rev. 3
1PMP-60CR-1-2E, Unit 1, Polar Crane Electrical Maintenance, Rev. 9
RVLIS & Head Vent Piping & Components, Certification of Compliance, 2/24/2006
CRDMs ASME Code Data Form –2, 1/4/2006
CRDMs Certification of Compliance, 1/4/2006
Liquid Penetrant Inspection Reports for CRDM Pressure Housings (S/N 254, 255, 256, 257, 261, 268, 275, & 280)
Westinghouse Significant Quality Problem - SQP 05-006, M504 CRDM Life Evaluation Test Mis-stepping, 12/22/2005
FENOC Supplier Action Request 10403-S039-01, March 16, 2006
ENSA -1MB1 GS 101, Post Weld Stress Relief Heat Treatment, Rev.1
ENSA - Beaver Valley Unit 1 RRVCH, Certificate of Compliance, August 31, 2005

Condition Reports

05-04552	05-03107	05-02716	06-00227	06-00409
06-00414	06-00383	06-00415	06-02383	06-00436
06-02386	06-00437	06-02550	06-01363	06-02605
06-02634	06-02688			

Plant Modifications
ECP 02-0212, Rev 0, Replacement Steam Generator (RSG) Master ECP BVPS-1
ECP 03-0193-01, Revision 0, 9/2/05; SGRP Replacement Steam Generator Evaluation
ECP 03-0195-01, Revision 0, 4/18/05; Steam Generator Supports - Steam Generator A
ECP 03-0195-02, Revision 0, 4/18/05; Steam Generator Supports - Steam Generator B
ECP 03-0195-03, Revision 0, 4/18/05; Steam Generator Supports - Steam Generator C
ECP 03-0194-01, thru 05, Revision 0, 3/31/05; SGRP Primary Nozzles and RC Piping
ECP 03-0195, TOC, Revision 0; Steam Generator Supports
ECP 03-0196-01, Revision 1, 5/20/05; SGRP Steam Generator and Piping Insulation - Loop A
ECP 03-197-04, Revision 0, 3/25/06; Permanently Seal Old Sgs and Remove Temp FW Instruments
ECP 03-0199-01, Revision 1; SGRP Containment Construction Opening
ECP 03-0199-02, Revision 0; SGRP Containment Construction Opening - Reinstall Liner Plate
ECP 03-0199-03, Revision 1; SGRP Containment Construction Opening - Reinstall Liner Plate
ECP 03-0199-04, Revision 1; SGRP Containment Construction Opening - Reinstall Liner Plate
ECP 03-0201-02, Revision 0; Temp Removal of Containment Temperature Detectors TRB-1LM100-9, 10, 11
ECP 03-0201-04, Revision 0; Remove CRDM Cable Trays (Trays replaced per ECP-03-0295)
ECP 03-0294-06, Revision 1; RSG Water Level NR Transmitter Replacement & Associated Loop Inst. Rescaling

ECP 03-0294-10, Revision 0; SG Water Level NR Transmitter BV-LC-1FW-478B Repl & Associated Loop Inst. Rescaling
ECP 03-0294-11, Revision 0; SG Water Level NR Transmitter BV-LC-1FW-488B Repl & Associated Loop Inst. Rescaling
ECP 03-0294-12, Revision 0; SG Water Level NR Transmitter BV-LC-1FW-498B Repl & Associated Loop Inst. Rescaling
ECP 03-0294-13, Revision 0; Document Only Update for Replacement SG Baseline Design Data
ECP 05-0017-02, Revision 0; Eliminate Snubber BV-CC-HSS-1A
ECP 05-0017-03, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG A)
ECP 05-0017-04, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG A)
ECP 05-0017-05, Revision 0; Modify Pipe Supports WGCB-2716-PS-1A & 1B
ECP 05-0017-06, Revision 0; Modify Pipe Clamp for snubber BV-CC-HSS-1B
ECP 05-0017-07, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG B)
ECP 05-0017-08, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG C)
ECP 05-0017-09, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG B)
ECP 05-0017-10, Revision 0; SGRP Piping, Adjust Whip Restraint Shim Packs - away from SGs (SG C)
05-06604, Revision 0, 1/9/06; Pressure Relief Modification for the Unit 1 East and West Cable Vaults

<u>10 CFR 50.59 Safety Evaluations</u>
05-02827, Revision 0, 9/2/05; SGRP Replacement Steam Generators - Compound Changeout
04-03449, Revision 1, 1/6/06; SGRP Containment Construction Opening
05-06604, Revision 0, 1/9/06; Pressure Relief Modification for the Unit 1 East and West Cable Vaults

<u>10 CFR 50.59 Screened-out Evaluations</u>
04-02375, 3/16/05; Steam Generator Primary Piping Cutting and Rewelding
05-02827, 8/24/05
04-03449, Revision 2, 1/6/06; SGRP Containment Construction Opening
04-03445, Revision 0; Steam Generator Supports
04-04349, Revision 0; Large Bore Secondary Piping Removal and Reinstallation
04-03077, Revision 1; Steam Generator Replacement Project Small Bore Secondary Piping
04-03448, Revision 2; SGRP Rigging & Transport
04-03444, Revision 0; SGRP Miscellaneous Containment Modifications

<u>Regulatory Applicability Determination</u>
06-00604, Revision 0, SGRP Steam Generator and Piping Insulation
04-03447, Revision 0; SGRP Temporary Power

Audits and Self-Assessments
Stone & Webster Review Of Rigging International Procedures For The Inside Lift System, Steam Generator Replacement Project - BVPS Unit 1

Calculations
11700-GA-037, Revision 0, Addendum 2; ULR adequacy
24928-001, 4/8/05; Design Loads For Replacement Steam Generator Temporary Upper Lateral Supports
24928-C-907, 4/6/05; Hydraulic Pedestal Assembly at Steam Generator Lower Supports
8700-DMC-1585, Revision 0; New Foot Bolts
11700-GA-37, Revision 0, Addendum 1; 3/26/03; Stress Analysis of the Steam Generator Upper Restraint
Duquesne Light Company Calculation 8700-DSC-156W, 2/26/91; Liner Minimum Wall Thickness S&W Calculation 11700-EA-41, 11/3/71; Duquesne - Beaver Valley Unit 1 - Reactor Containment Liner Stress Analysis
S&W Calculation 11700-EA-50, 11/6/71; Duquesne - Beaver Valley Unit 1 - Reactor Containment Liner Buckling Analysis
S&W Calculation 11700-EA-49, 11/15/71; Duquesne - Beaver Valley Unit 1 - Reactor Containment Liner; Stress and Buckling Analyses of the Liner Wall Mat Junction and the Liner Skirt
Calculation 8700-DSC-0231, Revision 0, ½0/05; Permanent Stress Changes in the Containment Structure due to Creation and Restoration of the Temporary Construction Opening

Licensee Corrective Action Reports CR

03-04487	04-08333	05-03813	05-07761	05-07985
06-00270	06-01114	06-01118	06-01122	06-01195
06-01229	06-01722	06-01768	06-01750	06-01744
06-01742	06-01741	06-01726	06-01820	05-07735
06-01331	06-01499	06-00744	06-00844	06-01637
06-01349	06-01352	06-01047	06-01012	06-01061
06-01122	06-01148	06-01519	06-01527	06-00744
06-00643	06-01519	06-01624	06-01629	06-01636
06-01638	06-01639	06-01640	06-01643	06-01646
06-01647	06-01658	06-01659	06-01662	06-01668
06-01669	06-01678	06-01685	06-01690	06-01694
06-01696	06-01643	06-01635	06-01638	06-01639
06-01640	06-01647	06-01626	06-01691	06-01944
06-01945	06-01947	06-01953	06-02323	06-02735
06-00922	06-00562	06-00915	04-04929	04-09643
04-09566	04-07851	06-00168	06-00195	04-08189
06-01113	06-01114	06-01126	06-01154	06-01169
06-01191	06-01820	04-08339	04-09047	04-07851
04-09046				

Bechtel HPCR's, NCR's, SDDR's and CR's

NCR 104	NCR 001	NCR-055	NCR-066	NCR-067
NCR-068	NCR-065	NCR-063	NCR-062	NCR-060
NCR-061	NCR-064	NCR-074	NCR-081	NCR-093
NCR-001	NCR-002	NCR-003	NCR-004	NCR-005
NCR-006	NCR-007	NCR-008	NCR-009	NCR-010
NCR-011	NCR-012	NCR-013	NCR-014	NCR-015
NCR-016	NCR-017	NCR-018	NCR-019	NCR-036
NCR-040	NCR-044			

HPCR-05-18	HPCR-06-0008	HPCR-06-0014	HPCR-05-0024
HPCR-04-0019	HPCR-06-0027	HPCR-06-0026	HPCR-06-0025
HPCR-06-000924928-400		SDDR-UA30-00001-014	
24928-400-SDDR-UA30-00001-015		24928-400-SDDR-UA30-00001-013	

Drawings

Westinghouse Drawing 6653E94, Sheet 1 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 2 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 3 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 4 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 5 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 6 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E94, Sheet 7 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Outline

Westinghouse Drawing 6653E95, Sheet 1 of 2, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator General Arrangement

Westinghouse Drawing 6653E95, Sheet 2 of 2, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator General Arrangement

Westinghouse Drawing 6654E08, Sheet 1 of 3, Revision 0; Beaver Valley 1 Model 54F Replacement Steam Generator Elliptical Head Forging

Westinghouse Drawing 6654E08, Sheet 2 of 3, Revision 0; Beaver Valley 1 Model 54F Replacement Steam Generator Elliptical Head Forging

Westinghouse Drawing 6654E08, Sheet 3 of 3, Revision 0; Beaver Valley 1 Model 54F Replacement Steam Generator Elliptical Head Forging

Westinghouse Drawing 6654E16, Sheet 1 of 4, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Plate And Lower Shell Assembly

Westinghouse Drawing 6654E16, Sheet 2 of 4, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Plate And Lower Shell Assembly

Westinghouse Drawing 6654E16, Sheet 3 of 4, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Plate And Lower Shell Assembly

Westinghouse Drawing 6654E16, Sheet 4 of 4, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Plate And Lower Shell Assembly
Westinghouse Drawing 6654E18, Sheet 1 of 3, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Lane Block And Blowdown Weld Assembly
Westinghouse Drawing 6654E18, Sheet 2 of 3, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Lane Block And Blowdown Weld Assembly
Westinghouse Drawing 6654E18, Sheet 3 of 3, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Tube Lane Block And Blowdown Weld Assembly
Westinghouse Drawing 6654E21, Sheet 1 of 2, Revision 2; Beaver Valley 1 Model 54F RSG Feedwater Forging Welding And Machining
Westinghouse Drawing 6654E21, Sheet 2 of 2, Revision 2; Beaver Valley 1 Model 54F RSG Feedwater Forging Welding And Machining
Westinghouse Drawing 6654E23, Sheet 1 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 2 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 3 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 4 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 5 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 6 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6654E23, Sheet 7 of 7, Revision 2; Beaver Valley 1 Model 54F Replacement Steam Generator Upper Shell Detail Assembly
Westinghouse Drawing 6656E03, Rev 0, Weld End Preparation, Transition Welds (Sh 1-4)
Westinghouse Drawing 1C83205, Sheet 1 of 1, Revision 1; Beaver Valley 1 Model 54F Replacement Steam Generator Auxiliary Nozzle and Water Level Boss
8700-RV-47B, Revision 4, 7/19/91; Steam Generator Lower Support Assembly
8700-RV-47F, Revision 4, 7/19/91; Steam Generator Lower Support Weldment Details Sheet 4
8700-RV-47G, Revision 7, 4/27/76; Steam Generator Lower Support Foot Attachment Details
8700-RV-47L, Revision 3, 7/11/72; RCS Component Support Details
8700-RV-47M, Revision 3, 11/5/71; Steam Generator Upper Restraint Assembly
8700-RV-47N, Revision 6, 2/5/70; Steam Generator Upper Restraint Details, Sheet 1
8700-RV-47D, Revision 5, 9/24/74; Steam Generator Upper Restraint Details, Sheet 2
8700-RV-47Q, Revision 8, ½7/70; Steam Generator Upper Restraint Shim Assembly Procedure
8700-RV-40F, Revision 9, 4/15/05; Reactor Coolant Pump Support Details, Sheet 2
4162-SH1A, Revision 2, 9/18/05; Upper Lateral Restraint RC-E-1A, Schedules
4162-SH2A, Revision 2, 9/18/05; Upper Lateral Restraint RC-E-1A, Assembly Plans and Sections
4162-SH3A, Revision 2, 9/18/05; Upper Lateral Restraint RC-E-1A, Top Half Ring Fabrication Details
4162-SH4A, Revision 2, 9/18/05; Upper Lateral Restraint RC-E-1A, Bottom Half Ring Fabrication Details
4162-SH5A, Revision 2, 9/18/05; Upper Lateral Restraint RC-E-1A, Ring Fabrication Details

Westinghouse WM750, Sheet 1 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Upper Shell Platform Interfaces
Westinghouse WM750, Sheet 2 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Enlarged View & Section
Westinghouse WM750, Sheet 3 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Enlarged View & Section
Westinghouse WM750, Sheet 4 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Platform Mounting Plate Location
Westinghouse WM750, Sheet 5 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Installation Guidelines
Westinghouse WM750, Sheet 6 of 6, Revision 8, 10/25/05; Beaver Valley 1 Model 54F Replacement Steam Generator-SG-1A, Tap Guard Arrg't
Westinghouse 1098J65, Sheet 1 of 4, Revision 3, Steam Generator Lower Support Outline
Westinghouse 1098J67, Sheet 5 of 5, Revision 4, Steam Generator Lower Support - Lower Support Fabrication
Westinghouse 1098J70, Sheet 1 of 2, Revision 4, Steam Generator Lower Support - Final Machining
Westinghouse 1098J70, Sheet 2 of 2, Revision 4, Steam Generator Lower Support - Final Machining
1-03.018-0092-E03-0195-01, Unit 1, Revision D, 4/15/05
1-03.018-0094-E03-0195-01, Unit 1, Revision D, 4/15/05
Westinghouse 1098J71, Sheet 1 of 2, Revision 4, Steam Generator Lower Support - Final Assembly
Westinghouse 1098J71, Sheet 2 of 2, Revision 4, Steam Generator Lower Support - Final Assembly
1-03.018-0145-E03-0195-01, Unit 1, Revision B, 4/15/05
1-03.018-0146-E03-0195-01, Unit 1, Revision B, 4/15/05
1-03.018-0152-E03-0195-01, Unit 1, Revision C, 4/15/05
Westinghouse 1098J82, Sheet 3 of 3, Revision 6, Steam Generator Upper Restraint - Final Assembly & Machining
Westinghouse 1098J82, Sheet 2 of 3, Revision 6, Steam Generator Upper Restraint - Final Assembly & Machining
Westinghouse 1098J82, Sheet 1 of 3, Revision 6, Steam Generator Upper Restraint - Final Assembly & Machining
1-03.018-0178-E03-0195-01, Unit 1, Revision B, 4/15/05
1-03.018-0180-E03-0195-01, Unit 1, Revision A, 4/15/05
6653 E94 Rev 2, Sh 1,2,3,4,5,6,7
6654 E18 Rev 2, Sh 1,2,3
6654 E21 Rev 2, Sh 1,2
6654 E23 Rev 2, Sh 1,2,3,4,5,6,7
6653 E95 Rev 1, Sh 1,2
6654 E08 Rev 0, Sh 1,2,3
6654 E16 Rev 3, Sh 1,2,3,4
1C83205 Rev 1

Evaluations
ECP-03-0193 Rev 0, Design Interface Evaluation-Evaluates West Model 54F SG

Licensing Document Change Notice
104-029 UFSAR Change - Restoration of Containment Liner Changes in Detail
04-01756 UFSAR Change - Inspection of Structural Steel per ANSI N45.2.5

FENOC Supplier Surveillances
10403-S035, 7/1/05; Westinghouse & Subcontractor ENSA
10403-S036, 8/1/05; Westinghouse & Subcontractor ENSA

Procedures
1/2-ADM-0804, On-Line Risk Assessment and Management, Rev. 6
1/2OM-35.4A.A, Voltage Schedule Guidance, Rev. 2
1/2OM-53C.4.35.1, Degraded Grid, Rev. 3
BVBP-WMI-006, Emergency Load Conservation Guidelines, Rev. 1
NOP-LP-2001, Revision 12, 9/26/05; Corrective Action Program
1-SPT-25-30193-1, Revision 1; Beaver Valley Power Station Unit 1, SGRP Warranty Test Blowdown System Flow Capacity (OSG)
1-SPT-25-30193-1, Revision 2; Beaver Valley Power Station Unit 1, SGRP Warranty Test Blowdown System Flow Capacity (RSG)
10M-54.4.C1-3, Revision 17; Beaver Valley Unit 1, Daily Heat Balance Step-by-Step
Rigging International, P-2696-48; Upend, Lift & Set RSGs
Rigging International, P-2696-47; Handling Replacement Steam Generators Outside Containment
Rigging International, P-2696-46; Load & Transport RSGs To Containment
Rigging International, P-2696-45; Transport OSGs To OSGF And Offload
Rigging International, P-2696-44; Handling Old Steam Generators Outside Containment
Rigging International, P-2696-42; Lift & Old Steam Generators
Westinghouse Electric Company, Quality Man Down end agement System, Revision 5, 10/1/02
DLW-PQP-005, Revision 1, 1/4/06; Project Quality Plan for Beaver Valley Unit 1 Replacement Reactor Vessel Head Installation
Westinghouse Electric Company, Field Service - Control Of Activities, Revision 10, 3/31/04
Westinghouse Electric Company, Westinghouse Corrective Action Process, Revision 4, 3/31/04
BVBP-SGRP-0013, Revision 0, 2/16/06; SGRP Oversight For Vendor Condition Reporting Program
BVBP-SGRP-0016 Rev 2, Steam Generator Replacement Project Receiving Inspection Plan Development Guidelines
ENSA Procedure, OMB2CS601, Revision 2, 6/19/05; Hydrostatic Pressure Tests Of The Steam Generator - Primary And Secondary Sides
Beaver Valley Unit 1 Steam Generator Replacement Project, Construction Procedure CP-20, Revision 1, 8/10/05; Human Performance Condition Monitoring
NOP-LP-2001, Revision 12, 9/26/05; Corrective Action Process
Beaver Valley Unit 1 Steam Generator Replacement Project, Construction Procedure CP-7, Revision 1, 5/11/05; Deviation Control
1 BVT-01.74.02, Issue 2, Revision 4,5/29/03; Containment Type A Leak Test
1-BVT 1.47.2, Revision 5; Containment Type A Leak Test
1/2CMP-75-SNUBBER-2M, Issue 4 Revision 5, 1/7/03; Removal and Installation of Bergen-Patterson 12 Inch X 6.25 Inch Stroke Hydraulic Snubber
1BVT 1.3.1, Issue 4, Revision 6; Narrow Range RTD Cross Calibration
1BVT 1.6.1, Issue 2, Revision 17; Reactor Coolant System Flow Measurement
1-SPT-21-30193-2, Revision 0; SGRP Warranty Test: Moisture Carryover
1/2-ADM-2205, Revision 0; Flow-Accelerated Corrosion (FAC) Program
1BVT 01.47.01, Issue 1, Revision 6, Completed 4/21/03; Containment Structural Integrity Test

1BVT 01.47.01, Issue 1, Revision 6, Completed 4/5/00; Containment Structural Integrity Test
1BVT 01.47.01, Revision 7, 3/8/06; Containment Structural Integrity Test
1-BVT 1.47.2, Revision 5; Containment Type A Leak Test Results, 4/26/06

NDT Procedures

UT-TM Rev 0	Ultrasonic Thickness Measurement
MT-ASME/USAS Rev 0	Magnetic Particle Examination
PT(SR)-ASME/USAS Rev 0	Liquid Penetrant Examination
VT-ASME III CL B 1968 Rev 0	Visual Examination Piping and Structural
VT-USAS/ANSI B31.1.0 Rev 1	Visual Examination Piping and Structural
VT-AWS D 1.1 Rev 0	Visual Examination Structural to Structural Steel
VT-AWS D 1.3 Rev 0	Visual Examination Sheet Metal to Structural Steel
VT-AWS D 1.4 Rev 0	Visual Examination Butt Splice and Repair
RT-USAS/ANSI B31.1.0 Rev 1	Radiographic Examination
RT-ASME III CL B 1968 Rev 1	Radiographic Examination
RT-ASME IX Rev 2	Radiographic Examination
LT(VB)-Liner Plate Rev 0	Leak Testing (Vacuum Box)
UT-323 Rev 1	Ultrasonic Examination of Welds Joining Cast Austenitic Piping Components
NDE-UT-320 Rev 4	Ultrasonic Examination of Ferritic Pipe Welds
NDE-LP-101 Rev 18	Solvent Removable Visible Dye
NDE-MT-201 Rev 15	Magnetic Particle Examination
1RPSI-July/Aug 2005	Eddy Current Examination Report Pre-Service Inspection Unit 1 Replacement Steam Generator
MRS 2.4.2 GEN-35 Rev 11	Eddy Current Inspection and Inservice Heat Exchanger Tubing
PSI RSG-1 Rev 0	Pre-Service Inspection Guidelines for the Unit 1 Replacement Steam Generators

Steam Generator Tube Inspections

SG-SGDA-05-28	Steam Generator Degradation Assessment for BV 1 Pre-Service Inspection for Model 54F RSG's
DLW-01-05 Rev 0	SG Eddy Current Inspection Examination Technique Spec Sheet Bobbin 40 IPS (ETSS) Full/Low Row .720
DLW-02-05 Rev 0	ETSS Bobbin 24 IPS Full and Partial Length .720
DLW-03-05 Rev 0	ETSS Bobbin 24 IPS Restricted or U-Bend Low Row .700
DLW-04-05 Rev 0	ETSS Bobbin 24 IPS Restricted or U-Bend Low Row .680
DLW-05-05 Rev 0	ETSS 3 Coil +PT Straight Section Special Interest
DLW-06-05 Rev 0	ETSS Low RowU-B +PT Row1&2 U-Bend/Special Interest
DLW-07-05 Rev 1	ETSS U-B MR +PT U-Bend Special Interest
DLW-08-05 Rev 0	ETSS 3 Coil +PT Straight Section Special Interest

MRS 2.4.2 GEN - 35, Rev 11 — Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing

1RPSI-July/August 2005 Eddy Current Examination Report Unit 1 Replacement Steam Generator Preservice Inspection

BVBP-SGRP-0016 Rev 2 Steam Generator Replacement Project Receiving Inspection Plan Development Guidelines
PSI RSG-1 Preservice Inspection Guidelines for the Unit 1 Replacement Steam Generators
DLW-01-05 thru 08-05 Steam Generator Eddy Current Inspection Examination Technique Specification Sheets (ETSS)

Vendor Information
Westinghouse Technical Manual No.: TM 1440-C388, Volume 1, Revision 1, 8/1/05; BV1 Model 54F Vertical Replacement Steam Generator Instructions
Westinghouse Technical Manual No.: TM 1440-C388, Volume 2, Revision 1, 8/1/05; BV1 Model 54F Vertical Replacement Steam Generator Instructions For Receipt Inspection, Storage And Assembly

Work Orders & WIPR's

M-ULA-186	200110778	200136709	200136665
C-RCA-126	C-RCA-123	C-JTD-110	C-HYD-109
C-LPR-111	C-LPR-112, Rev 1	C-OLS-118	C-OLS-117
P-FWC-200	P-RCC-224	P-RCC-227	P-MSC-215
P-RCC-224	P-RCA-222	M-LSA-178	M-ULA-189

Specifications

214928-C-304(Q) Rev 1	Installation of Reinforcing Steel and Accessories (SMAW)
24928-C-310(Q) Rev 0	Installation of Cadweld Rebar Splices

Welding Procedure Specifications (WPS) and Procedure Qualification Records(PQR)

P1-A-Lh Rev 0	Manual Shielded Metal Arc of P1-P1, All Groups PQR 695, 892
P1-T-o	Machine Gas Tungsten Arc Welding of P1 in the Post Weld Heat Treated Condition, PQR 1259
P1-A-Lh (CVN -20) Rev 1	Manual Shielded Metal Arc of P1-P1, Group 1 & 2, PQR 1343
P1-AT-Lh Rev 0	Manual Gas Tungsten Arc & Manual Shielded Metal Arc of Carbon Steel P1-P1 All Groups, PQR 695, 892, 1259 and 1350
P1-A-Lh Rev 0 (Structural)	AWS D 1.1 Pre-qualified with GWS-Structural
GWS-Rebar	General Standard Arc Welding of Reinforcing Steel
P1-Rebar (0.87CE) Rev 0	Manual Shielded Metal Arc Welding of Rebar (max CE of 0.87) Used with GWS-Rebar
24928-C-310(Q) Rev 0	Technical Specification for Install Cadweld Rebar Splices
P1-STUD (AWS D 1.1) Rev 0	Semi Automatic Stud Welding of Carbon Steel to Carbon Steel, Studs to be ASTM A 108 Gr 1010 thru 1020
P3(G3)-P1(G2)-T-o CVN +70	Machine Gas Tungsten Arc Welding of P3 to P1 with CVN and Post Weld Heat Treatment, ASME Section III PQR 1282
P3(G3)-P1(G2)-T(CVN+70)R0	Manual Gas Tungsten Arc Welding of P3 to P1 with CVN and Post Weld Heat Treatment, ASME Section III PQR 1243 and 1282
P3(G3)-P1(G1/2)-T CVN +10	Manual Gas Tungsten Arc of P3(G3) to P1(G1&2) PQR 1282
P1-T Rev 2	Manual Gas Tungsten Arc Welding of P1, PQR695

P8-T(RA) Rev 1	Machine Gas Tungsten Arc Welding of Austenitic Stainless Steel (Reduced Angle Groove)
PHT-1 Rev 0	Post Weld Heat Treatment

Performance Qualification Test Records for Welders —153, —044 and —167

Miscellaneous

ASME Code Case –416-2
Bechtel Ltr. B-FENOC-06-067, 2/21/06; Beaver Valley Unit 1 Steam Generator Replacement Project, Response To Steam Generator Rigging Review
Beaver Valley Unit 1 UFSAR, Section 7.4 Systems Required For Safe Shutdown
Beaver Valley Unit 1 UFSAR, Section 5.0; Containment Design
First Energy Ltr. L-05-163, 10/31/05; Beaver Valley Power Station, Unit Nos. 1 and 2 BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPf-73, Supplement to License Amendment Request Nos. 327/197 (Unit No. 1 TAC No. MC4649/Unit No. 2 TAC No. 4650), 317/190 (Unit No. 1 TAC No. MC3394/Unit No. 2 TAC No. MC3395) and 320 (Unit No. 1 TAC No. MC6725)
Splicer 747 Rebar Qualification Test Record, Horizontal, Vertical and Diagonal
Splicer 761 Rebar Qualification Test Record, Horizontal, Vertical and Diagonal
Splicer 306 Rebar Qualification Test Record, Horizontal, Vertical and Diagonal
900078-ABS01-Unit 1 As Built Survey Data for Reactor Coolant System Piping
Field Welding Checklist for Weld WFPD-22-83-F-1C - Feedwater “C”
Field Welding Checklist for Weld WFPD-22-83-F-9DB - Feedwater “C”
Field Welding Checklist for Weld WFPD-24-6A-F-13 - Feedwater “C”
Field Welding Checklist for Weld DLW-Loop 3-2-F-28A RCS “C” Hot Leg
Field Welding Checklist for Weld DLW-Loop 3-3-F-29A RCS “C” Cold Leg
Field Welding Checklist for Weld DLW-Loop 1-2-F-4A RCS “A” Hot Leg
Field Welding Checklist for Weld DLW-Loop 1-3-F-5A RCS “A” Cold Leg
Field Welding Checklist for Weld SHP-56-1-F-1A & TAC Blocks “A” Main Steam
Field Welding Checklist for Weld SHP-56-1A-F-14 & TAC Blocks “A” Main Steam
Field Welding Checklist for Weld SHP-58-1-F-1A Main Steam “C”
Field Welding Checklist for Weld SHP-58-1A-F-12 Main Steam “C”
Field Welding Checklist for Field Welds 1 thru 38, Containment Liner Restoration
RT Report for Weld #DLW-LOOP-3-2-F-28A “C” Hot Leg, Elbow to Nozzle
RT Report for Weld # DLW-LOOP-3-3-F-29A “C” Cold Leg Nozzle to Elbow
RT Reports for Field Welds 2 and 3 for Restoration of Containment Liner Plate
PI-900364-05 Project Instruction - Reactor Coolant System Existing Piping/Elbow Work
NDE Report BOP-VT-06-041 Visual Examination of Containment Liner Plate Corrosion
Primary Containment Inservice Inspection Program Plan, Initial Inspection Interval, Revision 0
BVBP-WMI-006, Emergency Load Conservation Guidelines, Rev. 1
1/2-ADM-0804, On-Line Risk Assessment and Management, Rev. 6
1/2OM-35.4A.A, Voltage Schedule Guidance, Rev. 2
1/2OM-53C.4A.35.1, Degraded Grid, Rev. 3

LIST OF ACRONYMS

ADM	Administrative Procedure
AFW	Auxiliary Feedwater
ASME	American Society Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BCO	Basis for Continued Operation
BMI	Bottom Mounted Instrumentation
BOP	Balance of Plant
BVPS	Beaver Valley Power Station
CAQ	Condition Adverse to Quality
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CR	Condition Report(s)
CRDM	Control Rod Drive Mechanism
DAFW	Dedicated Auxiliary Feed Pump
DAW	Dry Active Waste
DEP	Drill/Exercise Performance
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ENSA	Equipos Nucleares, S.A.
EQ	Environmental Qualification
ER	Engineering Request
ERF	Emergency Response Facility
ET	Eddy Current Testing
FENOC	First Energy Nuclear Operating Company
GDC	General Design Criteria
HIC	High Integrity Container
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LR	Licensing Requirement
LRM	Licensing Requirements Manual
MR	Maintenance Rule
MRB	Management Review Board
MSP	Maintenance Surveillance Package
MSSV	Main Steam Safety Valve
NCV	Non-cited Violation
NDE	Non Destructive Examination
NOED	Notification of Enforcement Discretion

NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OD	Operability Determination
ODMI	Operational Decision Making Issue
OE	Operating Experience
ORVCH	Old Reactor Vessel Closure Head
OSG	Old Steam Generator
OST	Operations Surveillance Test
PCE	Personnel Contamination Event Report
PCP	Process Control Program
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PORV	Power Operated Relief Valve
PMT	Post-maintenance Testing
PQR	Procedure Qualification Record (Welding Procedures)
PRA	Probability Risk Analysis
psig	Pounds per Square Inch Gage
PT	Liquid Dye Penetrant Testing
PWSCC	Primary Water Stress Corrosion Cracking
QS	Quench Spray
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RRVCH	Replacement Reactor Vessel Closure Head
RSG	Replacement Steam Generator
RSS	Recirculation Spray System
RT	Radiographic Test (Radiography)
RVHVS	Reactor Vessel Head Vent System
RVLIS	Reactor Vessel Level Instrumentation System
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SE	Safety Evaluation
SG	Steam Generator(s)
SGRP	Steam Generator Replacement Project
SGTR	Steam Generator Tube Rupture
SLCRS	Supplemental Leak Collection and Release System
SSC	Structure, System, and Component
SSPS	Solid State Protection System
TI	Temporary Instruction
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VHRA	Very High Radiation Area
VT	Visual Test
WO	Work Order
WPIR	Work Plan and Inspection Records

WPS Weld Procedure Specification