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

January 29, 2010

Mr. Paul Harden
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4, Route 168
Shippingport, PA 15077-0004

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2009005 AND 05000412/2009005

Dear Mr. Harden:

On December 31, 2009, the U. S. 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 results, which were discussed on January 12, 2010, with you and other members of your staff.

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

Based on the results of this inspection, this report documents two (2) self-revealing findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues have been entered in the corrective action program, the NRC is treating the findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the findings in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Beaver Valley. In addition, if you disagree with the characterization of the cross-cutting aspect of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region 1 and the NRC Senior Resident Inspector at the Beaver Valley Power Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). We appreciate your cooperation. Please contact me at 610-337-5200 if you have any questions regarding this letter.

Sincerely,

/RA/

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

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

Enclosures: Inspection Report 05000334/2009005; 05000412/2009005
w/ Attachment: Supplemental Information

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

Distribution w/encl:
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C. Newport, DRP
J. Greives, DRP
D. Werkheiser, SRI
D. Spindler, RI
P. Garrett, Resident OA
L. Trocine, RI OEDO
RidsNRRPMBeaver Valley Resource
RidsNRRDorlLpl1-1 Resource
ROPreportsResource@nrc.gov

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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.	05000334/2009005 and 05000412/2009005
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	October 1, 2009 through December 31, 2009
Inspectors:	D. Werkheiser, Senior Resident Inspector D. Spindler, Resident Inspector E. Burket, Reactor Inspector J. D'Antonio, Operations Engineer P. Kaufman, Senior Reactor Inspector T. Moslak, Health Physicist T. O'Hara, Reactor Inspector J. Tomlinson, Operations Engineer
Approved by:	R. Bellamy, Ph.D., Chief Reactor Projects Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000334/2009005, IR 05000412/2009005; 10/01/2009 - 12/31/2009; Beaver Valley Power Station, Units 1 & 2; Event Followup

The report covered a 3-month period of inspection by resident inspectors, regional reactor inspectors, and a regional health physics inspector. Two Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. Cross-cutting aspects associated with findings are determined using IMC 0305, "Operating Reactor Assessment Program," dated January 2009. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Initiating Events

- <u>Green</u>. A self-revealing NCV of TS 5.4.1, "Procedures", was identified in that operators failed to properly align and check the position of the "B" reactor coolant system (RCS) loop bypass valve [2RCS*45], as required by procedure. This deficiency caused an incorrect lineup of the required vent path and resulted in the over-pressurization of the isolated "B" RCS loop while filling. The estimated pressure exceeded the pressure/ temperature limit for an isolated RCS loop on November 11.

  The inspectors determined that the finding was not similar to the examples for minor deficiencies contained in IMC 0612, Appendix E, "Examples of Minor Issues". The finding was more than minor because if left uncorrected could have the potential to lead to a more significant safety concern. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

  The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, CHECKLIST 4 "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer." Because the loop was isolated from the reactor vessel and pressurizer, the required reactor coolant inventory and the decay heat removal system were not affected. There were no conditions indicating a loss of control as listed in Appendix G, Table 1 "Losses of Control." Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). Because this finding is of very low safety significance and has been entered into FENOC's corrective action program, the violation is being treated as a non-cited violation.

  The cause of this finding is related to the cross-cutting area of human performance, work practices, in that FENOC's failed to follow station procedures resulting in an over-pressurization of the isolated "B" RCS loop. [H.4.(b)]. (Section 4OA3.1)

- <u>Green</u>. A self-revealing NCV of TS 5.4.1, "Procedures", was identified in that procedures for securing Residual Heat Removal System (RHS) were not adequately maintained and did not contain relevant operating restrictions resulting in the inadvertent lifting of the "A" RHS pump suction relief [2RHS-RV721A] during normal operation,

excessive identified leakage of reactor coolant to the Pressurizer Relief Tank, and a declaration of an Unusual Event.
The inspectors determined that the finding was not similar to the examples for minor deficiencies contained in IMC 0612, Appendix E, “Examples of Minor Issues”. The finding was more than minor because if left uncorrected could have the potential to lead to a more significant safety concern. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC’s regulatory function, and was not the result of any willful violation of NRC requirements.

The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G. There were no conditions indicating a loss of control as listed in Table 1 “Losses of Control.” Attachment 1, Checklist 1 “PWR Hot Shutdown Operation: Time to Core Boiling <2 Hours” guidelines were used to evaluate the event. All mitigating capabilities were available, therefore a Phase 2 quantitative assessment was not required. The issue screens to Green (very low safety significance). Because this finding is of very low safety significance and has been entered into FENOC’s corrective action program (CR 09-68214), the violation is being treated as a non-cited violation.

The cause of this finding is related to the cross-cutting area of human performance, resources, in that procedures for RHS system shutdown were not complete and up to date. [H.2(c)]. (Section 4OA3.1)

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period at 100 percent power. The unit remained at 100 percent power for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power. On October 10, the unit reduced power to 97 percent for planned main feedwater regulating valve testing and reduced power to 60 percent the next day for planned main steam safety valve setpoint checks. On October 12, the unit was shut down to commence refueling outage 2R14. The unit returned to full power on December 7 and remained at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity [R]

1R01 Adverse Weather Protection (71111.01)

Seasonal Susceptibility

a. Inspection Scope (1 sample)

The inspectors reviewed the Beaver Valley Power Station (BVPS) design features and FENOC's implementation of procedures to protect risk significant mitigating systems from cold weather conditions and high winds. The inspectors walked down risk significant plant areas for several days in November and December 2009 to assess FENOC's protection of these systems for cold weather conditions. The inspectors were sensitive to outside instrument line conditions and the potential for unheated ventilation. Walkdowns included the emergency diesel generator rooms, low head safety injection and service/river water systems. The inspectors also reviewed 1OST-45.11, "Cold Weather Protection Verification," Rev. 18 & 19 and 2OST-45.11, "Cold Weather Protection Verification," Rev. 19. Other documents that were reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns (71111.04Q)

a. Inspection Scope (3 samples)

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

- Unit 2, on October 9, 2-2 Emergency Diesel Generator (EDG) during maintenance on the 2-1 EDG;
- Unit 2, on October 18, RCS loops during pressurizer draindown; and
- Unit 1, on October 29, offsite power to 138kV Bus #2 while breaker OCB-83 was open for planned maintenance.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope (1 sample)

On December 1, the inspectors completed a detailed review of the alignment and condition of the Unit 2 Recirculation Spray System (RSS) following 2R14 refueling outage. The inspectors conducted a walkdown of the system to verify that the critical portions, such as valve positions, switches, and breakers, were correctly aligned in accordance with procedures, and to identify any discrepancies that may have had an effect on operability.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Sample Review (71111.05Q)

a. Inspection Scope (4 samples)

The inspectors reviewed the conditions of the fire areas listed below, to verify compliance with criteria delineated in Administrative Procedure 1/2-ADM-1900, "Fire Protection," Rev. 21. This review included FENOC's control of transient combustibles and ignition sources, material condition of fire protection equipment including fire detection systems, water-based fire suppression systems, gaseous fire suppression systems, manual firefighting equipment and capability, passive fire protection features, and the adequacy of compensatory measures for any fire protection impairments. Documents reviewed are listed in the Attachment:

- Unit 2, Reactor containment building (Fire Area RC-1);

- Unit 1, Storeroom (Fire Area WH-1);
- Unit 1, Storeroom (Fire Area WH-2); and
- Unit 2, West cable vault (Fire Area CV-1).

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08P)

a. Inspection Scope (1 sample)

From October 19-29, the inspectors conducted a review of FENOC's implementation of risk-informed in-service inspection (ISI) program activities for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries for Beaver Valley Unit 2 using the criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The inspectors also conducted a review of TI 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds for Beaver Valley Unit 2. The inspectors reviewed documentation, observed in-process non-destructive examinations (NDE) and interviewed inspection personnel to verify that the activities were performed in accordance with the ASME Boiler and Pressure Vessel Code Section XI requirements.

Non-Destructive Examination (NDE) Activities

The following inspection activities and examination records were reviewed by the inspectors: reactor pressure vessel (RPV) lower head bare metal visual (BMI) inspection video (sampled a selection of the 50 penetrations that were examined), reactor vessel upper head visual inspection, automated ultrasonic testing (UT) examination of reactor pressure vessel head penetration control rod drive mechanism (CRDM) nozzles, manual UT examination records of several materials reliability program (MRP) MRP-146 welds, manual UT of weld overlay to pressurizer surge line nozzle dissimilar metal weld, manual UT of weld overlay to pressurizer spray line nozzle dissimilar metal weld, manual UT of weld overlay to pressurizer power operated relief valve (PORV) nozzle dissimilar metal weld, eddy current testing (ECT) examinations of three steam generators, and dye penetrant (PT) examinations of the Unit 2 reactor vessel head (RVH) penetration weld overlays on penetrations #57 and #49 installed during 2R14, and RVH weld overlay PT on penetration #51 installed during 2R13.

The inspectors performed direct visual inspection of the Unit 2 containment liner and reviewed visual inspection records of the Unit 2 containment liner, including condition reports issued as a result of the licensee walkdown conducted per work order 200338759. The inspection results were also discussed with the FENOC containment liner program owner to very the condition of the containment liner and coating and ensure FENOC met the corrective action program which addresses License Renewal Application request for additional information RAI B.2.3-4 and RAI B.2.3-5 to complete a 100% visual examination of accessible containment liner plate area for Beaver Valley Unit 2 during the 2R14 in October/November 2009.

Repair/Replacement Consisting of Welding Activities

The Unit 2 repair/replacement activity associated with replacement of 2RCS-45, a Kerotest valve, and the weld overlay of Unit 2 reactor vessel head penetration #57 and #49 J-groove weld, were reviewed by the inspectors to ensure that welding and applicable NDE activities were performed in accordance with ASME Code requirements.

Reactor Vessel Upper Head Penetration Inspection Activities

The inspectors directly observed a sample of in-process Unit 2 reactor pressure vessel head and vessel head penetration control rod drive mechanism (CRDM) nozzle weld ultrasonic testing (UT), supplementary eddy current testing (ECT) examinations and portions of the repair activities during the 2R14 refueling outage. The inspectors reviewed the UT examination records and evaluated the automated UT data scans of the circumferential indication 0.75" long and 0.402" in depth that was identified on the outside diameter (OD) of CRDM penetration tube #57 at the toe of the J-groove weld and circumferential indication 0.45" long and 0.27" in depth that was identified on the OD of CRDM penetration tube #49 at the toe of the J-groove weld. The inspectors reviewed the weld repair activities to ensure that the indications in penetrations #57 and #49 were mitigated by repair, and weld overlays of the J-groove weld area were conducted in accordance BVPS-2, Relief Request No. 2-TYP-3-RV-01.

The inspectors reviewed the certifications of the NDE technicians performing the weld overlay PT examinations. The inspectors also reviewed a sample of the remote bare metal visual (VT-2) examination of the Unit 2 RPV head surface and 360 degrees around each of the 65 CRDM penetration nozzle welds and verified that no boric acid leakage had been observed on the upper reactor head surface.

Reactor Pressure Vessel Lower Head Penetration Nozzle Inspection Activities

The inspectors verified the adequacy of VT-2 visual inspection results of the bare metal inspection (BMI) of the Unit 2 reactor pressure vessel lower head penetration nozzle welds that were conducted by FENOC personnel during 2R14. The inspectors reviewed a sample of photos and visual inspection documentation records of the BMI inspection.

Boric Acid Corrosion Control (BACC) Inspection Activities

The inspectors discussed the boric acid control program with the boric acid corrosion control program owner and sampled photographic inspections of boric acid identified on safety significant piping and components inside Unit 2 containment during the mode 3 walkdowns conducted by FENOC personnel, and directly observed by the resident inspectors, to verify that the visual inspections were performed in accordance with the procedure and checklists which emphasized the areas and locations where boric acid leaks could cause degradation of safety significant components and that deficient conditions were identified and documented.

A sample of engineering evaluations/corrective actions associated with these boric acid deficiencies were reviewed by the inspectors. The inspectors confirmed that condition reports were assigned corrective actions consistent with the requirements of the ASME Code and 10 CFR 50 Appendix B Criterion XVI.

Steam Generator (SG) Tube Inspection Activities

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

Eddy current testing of all tubes in the three steam generators was conducted during the 2R14 outage. The inspectors reviewed plant specific steam generator information, tube inspection criteria, integrity assessments, degradation modes and tube plugging criteria. The inspectors discussed the in-process ECT inspection activities with the data management and the data acquisition personnel and the resolution analysts and observed a sample of the tubes being examined from each of the steam generators. Examination data records for selected tubes from each of the steam generators and the characterization and disposition of the identified flaws were reviewed by the inspectors to verify the steam generator inspection program was implemented in accordance with the rules and regulations of the steam generator examination program.

The inspectors participated in an outage conference call between NRC and FENOC on October 23, to discuss Unit 2 steam generator examination results obtained and the status of eddy current inspections up to that time. The inspectors reviewed the outside diameter stress corrosion cracking (ODSCC) circumferential indications that were detected within free-span dings at two locations (Row 43 Col 36 and Row 13 Col 13) in the "C" steam generator (SG) and at one location (Row 41 Col 36) in the "A" SG. The inspectors remotely observed a sample of the 92 SG tubes that required plugging and also observed the in-situ pressure testing of one of the tubes in the "C" SG that had the circumferential indications within the free-span dings and no leakage was identified. The inspectors confirmed the steam generator eddy current inspections, testing and documentation activities were conducted in accordance with Beaver Valley Unit 2 steam generator examination guidelines, station and vendor procedures, and EPRI guidelines.

Problem Identification and Resolution

The inspectors reviewed a sample of condition reports related to ISI activities to assess FENOC's effectiveness in problem identification and resolution and determined that deficiencies are being appropriately identified and that deficiencies are being adequately entered into and evaluated by the corrective action program.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope (1 sample)

The inspectors observed a sample of Unit 2 licensed operator simulator just-in-time training on October 29 reviewing aspects of unit reactor and plant startup. The inspectors evaluated licensed operator performance regarding command and control, implementation of normal, annunciator response, abnormal, and emergency operating procedures, communications, technical specification review and compliance, and emergency plan implementation. The inspectors evaluated the licensee staff training personnel to verify that deficiencies in operator performance were identified, and that conditions adverse to quality were entered into the licensee's corrective action program for resolution. The inspectors reviewed simulator physical fidelity to assure the simulator appropriately modeled the plant control room. The inspectors verified that the training evaluators adequately addressed that the applicable training objectives had been achieved.

b. Findings

No findings of significance were identified.

.2 Biennial Review by Regional Specialist (71111.11B)

a. Inspection Scope

The inspection activities were performed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Rev. 9, supplement 1, Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program," and 10 CFR Part 55, "Operators' Licenses."

A review was conducted of two years of operating history documentation found in inspection reports, licensee event reports, the licensee's corrective action program, and the most recent NRC plant issues matrix. The inspectors also reviewed specific events from the licensee's corrective action program for possible training deficiencies or appropriate training corrective actions. The resident inspectors were also consulted for insights regarding licensed operators' performance.

Observations were made of the dynamic simulator exams and job performance measures (JPMs) administered during the week of June 22, 2009. These observations included facility evaluations of crew and individual performance during the dynamic simulator exams and individual performance of simulator and in-plant JPMs. Four additional weeks of operating examination material and two weeks of written examinations were reviewed for compliance with the criteria of the examiner's standards.

The remediation plans for a crew/individual's failure and a written exam failure were reviewed to assess the effectiveness of the remedial training.

The inspectors reviewed the licensee's program to implement the guidance of ANSI/ANS-3.4-1983, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power." The inspectors emphasized the licensee's method for conducting tactile testing of the operators. Also, twenty-two medical examinations were reviewed for compliance with license conditions.

Simulator testing records were reviewed to verify that scheduled tests were performed and deficiencies addressed.

A review was conducted of licensee requalification exam results for the complete testing cycle. The inspection assessed whether pass rates were consistent with the guidance of the examination standards and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

Upon completion of all scheduled examination activities, the inspector reviewed examination results and verified that:

For Unit 1:

- Crew pass rate was greater than or equal to 80% (Pass rate was 100%);
- Individual pass rate on the dynamic simulator test was greater than 80% (Individual pass rate was 100%);
- Individual pass rate on the walkthrough (JPMs) was greater than 80% (Pass rate was 100%);
- Individual pass rate on the comprehensive written exam was greater than 80% (No written examination was administered at Unit 1 this year); and
- More than 80% of the individuals passed all portions of the exam (100% of the individuals passed all portions of the exam).

For Unit 2:

- Crew pass rate was greater than or equal to 80% (Pass rate was 100%);
- Individual pass rate on the dynamic simulator test was greater than 80% (Individual pass rate was 100%);
- Individual pass rate on the walkthrough (JPMs) was greater than 80% (Pass rate was 100%);
- Individual pass rate on the comprehensive written exam was greater than 80% (Pass rate was 95%);
- More than 80% of the individuals passed all portions of the exam (95% of the individuals passed all portions of the exam); and
- All licensed operators had been administered an examination (1 SRO had not completed the examination due to being on medical leave).

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope (2 samples)

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

- On November 5, Unit 1 Loop 1A ΔT/Tave MR a(1) action plan goal schedule issues as documented in CR 09-65505; and
- On December 7, Unit 2 Emergency Lighting power supply 2RB-EL-052 (2EP-ELT-003, 2EP-ELT-004, and 2RB-ELT-052) failed. These emergency lights did operate properly with the test switch as documented in CR 09-66546.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (3 samples)

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

- Week of October 17, Unit 2 'Yellow' risk assessment during RCS draindown;
- Week of November 2, Unit 2 'Yellow' risk contingency #2R14-10 to reconcile RHS heat exchange restoration sequence; and
- On November 9, Unit 2 'Yellow' risk associated with emergent work activity to repair the 'A' train RHS heat exchanger bypass flow control valve, [2RHS-FCV605A].

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (4 samples)

The inspectors evaluated the technical adequacy of selected immediate operability determinations (IOD), prompt operability determinations (POD), or functionality assessments (FA), to verify that determinations of operability were justified. In addition, the inspectors verified that technical specification (TS) limiting conditions for operation (LCO) requirements and UFSAR design basis requirements were properly addressed. In addition, the inspectors reviewed compensatory measures implemented to ensure the measures worked and were adequately controlled. Documents reviewed are listed in the Attachment.

- On October 20, Unit 2, Transfer of control issues for the steam-driven auxiliary feedwater pump solenoid-operated valves as documented in CR 09-66281;
- On October 20, Unit 2, Recirculation spray system motor operated valve, 2RSS-155D, higher running current documented in CR 09-66325;

- On October 22, Unit 1, Power range nuclear instrument (N-44) positive rate comparator as-found failed, and extent of condition, as documented in CR 09-66484; and
- On November 3, Unit 1, Turbine first outlet pressure transmitter [1PT-MS-447] instrument line steam leak documented in CR 09-67076.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

Temporary Plant Modifications

a. Inspection Scope (1 sample)

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

- On October 13, TMOD ECP 09-397, Temporary Service Water Piping between flanged connections at 2SWC-980 and 2SWC-983;

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (6 samples)

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

- On October 20, Unit 2, 2OM-36.4.AE for 2-1 emergency diesel generator after governor replacement;
- On November 1, Unit 1, Bistable card replacements for power range neutron monitors;
- On November 2, Unit 2, ECP 02-0290 and WO200337514 for replacement of Kerotest valve 2RCS*621 during deep draindown;
- On November 9, Unit 2, Train 'A' RHS heat exchanger bypass flow control valve [2RHS-FCV605A] actuator overhaul and repack;

- On November 10, Unit 2, ECP 08-0504, WO 200338922, and 2OST-1.10 after torque setting modification to service water valve 2SWS-103B; and
- On November 16, Unit 2, Reactor vessel head repair to penetrations 49 and 57, ECP 09-0673-000.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

Unit 2 Refueling Outage (2R14)

a. Inspection Scope (1 sample)

The inspectors observed selected Unit 2 outage activities from October 12 - November 27 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 2 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. Related events are documented in section 40A3, Event Followup. The inspectors also evaluated the following activities:

- Pre-Outage Shutdown Safety Review;
- Reactor plant shutdown and cooldown, including evaluation of cooldown rates;
- Initial containment and containment sump walkdown, including containment liner inspection;
- Coordination of electrical bus work, emergency diesel generator auto-load tests;
- Monitoring of decay heat removal processes;
- Installation of containment sodium tetra-borate and retirement of sodium hydroxide systems;
- Refueling activities; fuel handling and inspection;
- Licensee inspection of all fuel-handling cables and devices;
- Reactor coolant system draindown and vessel head lift;
- Reactor vessel deep-draindown and associated valve replacement and removals;
- Restoration of reactor coolant loops (also see section 4OA3);
- 2C15 core map / fuel assembly verification;
- Reactor head repairs and inspections;
- Final containment walkdown and closeout inspection;
- Reactor start-up and low power physics testing;
- Control rod drop measurement and testing;
- Reactor and plant start-up and heat-up (also see section 4OA3); and
- Balance-of-plant walkdown during power ascension.

The inspectors also observed selected management review activities associated with restart readiness of Unit 2, following completion of the 2R14 refueling activities.

The restart readiness review meeting was accomplished as required by procedure NOBP-OM-4010, "Restart Readiness for Plant Outages" Rev. 5, on November 8-10. The purpose of the review, in part, was to assure that the plant's material condition, programs/processes, and personnel were ready for startup and safe, reliable operation after completion of outage activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (4 samples: 1 isolation valve, 1 leak rate, 1 in-service testing and 1 routine)

The inspectors witnessed the performance of or reviewed test data for the four following 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. The inspectors also verified that the licensee established proper test conditions, that no equipment pre-conditioning activities occurred, and that acceptance criteria were met.

- On October 18, Unit 1, 1OST-47.3P, Rev. 12, "Containment System Operating Surveillance Test Containment Isolation and ASME Test-Work Week 12";
- On November 10, Unit 2, 2BVT-1.47.5, Rev. 21, "Type C Leak Test" for 2CWS-93;
- On November 18, Unit 2, 2OST-11.14B, rev. 27, "HHSI Full Flow Test."; and
- On December 16, Unit 2, 2OST-6.2A, Rev. 27, "Computer Generated Reactor Coolant System Water Inventory Balance".

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed Unit 1 licensed-operator simulator evaluations conducted on October 1 & 2, and a portion was counted as input into the NRC's emergency response drill and exercise performance indicator (PI). The inspectors observed FENOC's critique of the training activity to verify that weaknesses and deficiencies were adequately identified. The inspectors focused on ensuring FENOC identified operator performance issues associated with event classification, notification, and protective action recommendations.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (10 samples)

During the period October 26 - 29, the inspectors conducted the following activities to verify that the licensee was properly implementing physical, administrative, and engineering controls for access to locked high radiation areas, and other radiological controlled areas during the Unit 2 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant TSs, and the licensee's procedures. This inspection activity represents the completion of ten (10) samples relative to this inspection area.

Plant Walkdown and Radiation Work Permits (RWP) Reviews

- The inspector toured accessible radiologically controlled areas in the Unit 2 reactor building containment (RBC), primary auxiliary building, condensate polishing facility, and radwaste building and with the assistance of a radiation protection technician, performed independent radiation surveys of selected areas to confirm the accuracy of survey data, and the adequacy of postings. Radiation protection technicians were questioned regarding their knowledge of plant radiological conditions for selected jobs, and the associated controls.

- The inspector identified radiological significant jobs being performed in the Unit 2 RBC. The inspector reviewed the applicable RWPs, ALARA Plans (AP), and the electronic dosimeter dose/dose rate set points, for the associated tasks, to determine if the radiological controls were acceptable and if the set points were consistent with plant policy. Jobs reviewed included steam generator primary side demobilization (RWP 209-5016), GSI-191 insulation removal/replacement (RWP 209-5048), overhaul PCV-1RC-455A&B (RWP 209-5035), remove/replace core exit thermocouples (RWP 209-5019), reactor head welding repairs (RWP 209-5071), and Kerotest valve replacement (RWP 209-5062).

- For the jobs reviewed, the inspector determined that dosimetry was appropriately relocated to the portion of the body receiving the highest dose rate, for significant dose gradients; e.g., under reactor head repairs. The inspector determined that tele-dosimetry was extensively used to monitor and control worker exposure for dose intensive jobs.

- There were no current radiation work permits for airborne radioactivity areas with the potential for individual worker internal exposures to exceed 50 mrem during the 2R14 outage. The inspector reviewed air sampling records for on going jobs to confirm that airborne contamination was insignificant.

- The inspector evaluated the effectiveness of contamination controls by reviewing personnel contamination event reports (and related condition reports), and observing practices at various work locations in the RBC and at the step off pad.

High Radiation Area and Very High Radiation Area Controls

- The inspector reviewed procedures related to the control of high dose rate, high radiation area and very high radiation areas. The inspector discussed these procedures with the Radiation Protection Supervision to determine that any changes made to these procedures did not reduce safety measures.

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

- The inspector reviewed the preparations made for various potentially high dose rate jobs including reactor head repairs, and insulation modifications made to various systems in the RBC. Included in this review was the evaluation of the effectiveness of contamination control measures, source term controls, and use of temporary shielding.

Radiation Worker and Radiation Protection Technician Performance

- During tours of radiologically controlled areas in the Unit 2 RBC, the inspector questioned radiation workers and radiation protection technicians regarding the radiological conditions at the work site and the radiological controls that applied to their task. Additionally, radiologically-related condition reports, including dose/dose rate alarm reports, were reviewed to evaluate if the incidents were caused by repetitive radiation worker or technician errors and to determine if an observable pattern traceable to a similar cause was evident.

- The inspector attended the pre-job ALARA briefing for refueling activiites to determine if workers were properly informed including discussions of past operating experiences, identification of the radiological conditions associated with their tasks, electronic dosimetry dose/dose rate set points, and dose mitigation measures.

Problem Identification and Resolution

- The inspectors evaluated the licensee's program for assuring that access controls to radiologically significant areas were effective and properly implemented by reviewing various Nuclear Oversight audits and field observation reports, and relevant condition reports. The inspector determined if problems were identified in a timely manner, that an extent of condition and cause evaluation were performed when appropriate, previous radiation surveys remained valid, and corrective actions were appropriate to preclude repetitive problems.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (9 samples)

During the period October 26 - 29, the inspectors conducted the following activities to verify that the licensee was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for activities performed during the 2R14 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, and the licensee's procedures. This inspection activity represents the completion of nine (9) samples relative to this inspection area.

Radiological Work Planning

- The inspector reviewed pertinent information regarding site cumulative exposure history, current exposure trends, and the ongoing exposure challenges for the Unit 2 outage. The inspector reviewed the 2 R14 Outage ALARA Plan.

- The inspector reviewed the exposure status for tasks performed during the Unit 2 outage and compared actual exposure with forecasted estimates contained in various project ALARA Plans (AP). The inspector reviewed the Work-In-Progress ALARA reviews for those jobs whose actual dose approached 75% of the forecasted estimate. Outage jobs reviewed included scaffolding installation (AP 09-2-41/28), insulation modifications (AP 09-2-34/35/36), reactor disassembly/reassembly (AP 09-2-30), reactor head inspections (AP 09-2-48) and radiation protection support activities (AP 09-2-61).

- The inspector evaluated the departmental interfaces between radiation protection, operations, maintenance crafts, and engineering to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing site staff, reviewing outage Work-in-Progress reviews, attending Station ALARA Managers Committee (AMC) meetings, and reviewing recent AMC meeting minutes. The AMC meeting agendas, which the inspector attended, included planning for scaffolding installation/removal dose, steam generator demobilization dose, outage operations dose, Kerotest valve replacements, GSI-191 insulation replacements, and reviewing outage RWPs whose actual dose exceeded 50% of stretch goals.

Verification of Dose Estimates

- The inspector reviewed the assumptions and basis for the 2R14 outage ALARA plan. The inspector also reviewed the revisions made to various outage project dose estimates due to emergent work; e.g. insulation modifications, scaffolding activities, and reactor head repairs, authorized by the Station ALARA Committee.

- The inspector reviewed the licensee's procedures associated with monitoring and re-evaluating dose estimates when the forecasted cumulative exposure for tasks was approached and the implementation of these procedures during the outage. The inspector reviewed the exposures for the ten (10) workers who received the highest doses for 2009 to confirm that no individual exceeded the regulatory annual limit.

Job Site Inspections

- The inspector reviewed the ALARA controls specified in ALARA Plans and RWPs, for reactor head repairs, pressurizer spray valve overhaul, GSI-191 insulation replacement, refueling activities, and Kerotest valve replacements.

- During tours of the RBC, the inspector observed workers performing steam generator demobilization from eddy current testing, Kerotest valve replacements, and pressurizer spray valve testing. Workers were questioned regarding their knowledge of job site radiological conditions and ALARA measures applied to their tasks.

Source Term Reduction and Control

- The inspector reviewed the status and historical trends for the Unit 2 source term. Through review of survey maps and interviews with the Senior Nuclear Specialist-ALARA, the inspector evaluated recent source term measurements and control strategies. Specific strategies being employed included use of macro-porous clean up resin, increased filtration flow, enhanced chemistry controls, system flushes, and temporary shielding.

Declared Pregnant Workers

- The inspector reviewed the procedural controls, and associated records, for managing declared pregnant workers (DPW) and determined that one DPW was employed during the Unit 2 outage.

Problem Identification and Resolution

- The inspector reviewed elements of the licensee's corrective action program related to implementing the ALARA program to determine if problems were being entered into the program for timely resolution. Condition reports related to programmatic dose challenges, personnel contaminations, dose/dose rate alarms, and the effectiveness in predicting and controlling worker exposure were reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (6 samples total)

The inspectors sampled licensee submittals for Performance Indicators (PI) listed below for both Unit 1 and Unit 2 to verify accuracy of the data recorded. The inspectors reviewed Licensee Event Reports, condition reports, portions of various plant operating logs and reports, and PI data developed from monthly operating reports. Methods for compiling and reporting the PIs were discussed with cognizant technicians, 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 5, were used for each data element.

.1 Cornerstone: Mitigating Systems

a. Inspection Scope (4 samples)

The inspectors verified the accuracy and completeness of the data reported from October 1, 2008 through September 30, 2009 for the two following PIs:

- Unit 1 and Unit 2 Emergency AC power systems [MS06] - Emergency Diesel Generators;
- Unit 1 and Unit 2 High pressure safety injection systems [MS07] -High Head Safety Injection;

b. Findings

No findings of significance were identified.

.2 Cornerstone: Occupational Exposure Radiation Safety

Occupational Exposure Control Effectiveness

a. Inspection Scope (1 sample)

The inspector reviewed implementation of the licensee's Occupational Exposure Control Effectiveness PI Program. Specifically, the inspector reviewed reports, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in NEI 99-02.

b. Findings

No findings of significance were identified.

.3 Cornerstone: Public Radiation Safety

RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope (1 sample)

The inspector reviewed relevant effluent release reports for the period September 1, 2008 through October 1, 2009, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Daily Review of Problem Identification and Resolution

a Inspection Scope

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

b. Findings

No findings of significance were identified.

.2 Annual Sample: Review of Condition Report 09-62705, 8/24/09, NRC FINDING/NCV 2009003-01, INADEQUATE POST MAINTENANCE TESTING ON 1RW-57

a. Inspection Scope (1 sample)

The inspectors reviewed FENOC's actions taken to resolve the condition reported via Condition Report (CR) 09-59866. This CR identified a condition where, on May 28, 2009, First Energy returned valve 1RW-57 to service without completing the requisite In-Service Test (IST) required by the ASME Code.

Upon discovery of this reported condition, FENOC performed an apparent cause evaluation which determined that the governing scheduling and maintenance procedures did not contain sufficient administrative controls to ensure that the correct IST requirements are specified and completed before the affected components are returned to service.

Further research by FENOC determined that the condition reported by CR 09-59866 was an isolated occurrence. Additionally, FENOC reviewed all work instructions to be used during future maintenance for the necessary constraints to ensure that the requisite IST requirements were specified and will be completed before the maintenance is completed and the component returned to service.

b. Findings, Assessment, and Observations

No findings of significance were identified. The inspectors determined that FENOC had performed a complete and accurate identification of the problem in a timely manner commensurate with the issue's significance and ease of discovery. The inspectors also determined that FENOC had, upon determination of the apparent cause, determined that reportability and operability of the issue was properly completed.

The inspectors determined that FENOC had identified and implemented appropriate corrective actions to address the apparent cause of the issue and that those corrective actions had been completed in a timely manner.

.3 Annual Sample: Review of the Operator Workaround Program

a. Inspection Scope (1 sample)

The inspectors reviewed the cumulative effects of the existing operator work-arounds (OWAs), the list of operator burdens, existing operator aids and disabled alarms, and the list of open main control room deficiencies. This review was performed to identify any

impact on emergency operating procedures, operator actions, and any impact on possible operator actions related to initiating events and mitigating systems. The inspectors evaluated whether station personnel were identifying, assessing, and reviewing OWAs as specified in administrative procedure NOBP-OP-0012, "Operator Work-Arounds, Burdens and Control Room Deficiencies" Rev.1.
The inspectors reviewed FENOC's process to identify, prioritize and resolve main control room distractions to minimize operator burden. The inspectors reviewed the system used to track these operator work-arounds and burdens. The inspectors performed a control room walkdown to determine if the existing control room deficiencies were identified and included on the current tracking list. The inspectors discussed the open items with the operators to ensure the items were being addressed on schedule consistent with their safety significance.

The inspectors reviewed CRs and Snapshot Self-Assessments related to compliance with NOBP-OP-0012, including assessments performed in the fourth quarter of 2008 and 2009. The inspectors interviewed FENOC staff to determine their knowledge and implementation of the NOBP-OP-0012 process and the OWA tracking system.

b. Findings, Assessment, and Observations

No findings of significance were identified. At the time of the inspection, FENOC had two issues classified as OWAs and three operator burdens. The OWAs and burdens were determined to have a minimal impact on the operators' ability to promptly and appropriately respond to an event.

Two tracking systems are in place; one for input into the process (Lotus Notes) and another to manage the work process (SAP) that corrects the issue. These systems were effective most of the time to ensure operators and management are aware of OWAs and burdens and ensure the items are addressed in a timely fashion. Some items in Lotus Notes do not have, or require, a SAP entry. The inspector identified minor examples where actions have not been timely. Also, the inspectors noted a recurrence of a failure to perform quarterly assessments of the aggregate effects of OWAs and burdens in accordance with NOBP-OP-0012. The assessments were not performed in the second and third quarters of 2009. The licensee identified the failure to perform these quarterly assessments prior to the inspectors arriving onsite, generated CRs documenting this failure and performed an aggregate assessment and two Snapshot Self-Assessments to identify the cause for the failure to perform the second and third quarter assessments. In the fourth quarter of 2008, resident inspectors identified the failure to perform 2008 second and third quarter quarterly assessments. Although the failure to perform the quarterly assessments in 2009 is a recurrence of a previously identified issue, the safety significance is not more than minor. The inspectors observed a less than expected level of program ownership which may have contributed to the recurrence. The inspectors discussed the proposed corrective actions to prevent recurrence with FENOC and concluded that proposed corrective actions were adequate to address the issue.

4OA3 Event Followup (71153)

The inspectors performed six event followup inspection activities (4 plant events and 2 LER reviews). Documents reviewed for this inspection activity are listed in the attachment to this report.

.1 Plant Event Review

a. Inspection Scope (4 samples)

For the plant events below, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems. The inspectors communicated the plant events to regional personnel and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of additional reactive inspection activities. The inspectors reviewed FENOC's follow-up actions related to the events to assure that appropriate corrective actions were implemented commensurate with their safety significance. Documents reviewed during the inspection are listed in the Attachment.

- Unit 2: On October 14, an unplanned reduction in letdown flow occurred while the reactor was shutdown and in a solid condition requiring operator action to maintain primary plant pressure in band. The reduction in letdown flow was caused by the normal letdown flow orifices closing, as designed, during planned maintenance. The RHS letdown flowpath was unaffected. The operator quickly recognized the change in plant configuration and took appropriate timely actions to stabilize plant parameters. The licensee documented this issue in CR 09-65941;

- Unit 2: On October 23, during refueling outage 2R14 a planned ultrasonic examination of the reactor vessel head identified two penetrations that required repair. The licensee reported this issue to the NRC (EN # 45463) and documented it in CR 09-66489. The penetrations were repaired to ASME code requirements (also see section 1R08) and the reactor vessel head returned to service;

- Unit 2: On November 11, during refueling outage 2R14, with the reactor vessel head removed and cavity flooded, the isolated "B" reactor coolant loop was inadvertently over-pressurized during its filling evolution due to a valve out of position [2RCS*45] in the over-pressure vent path to the reactor side. Once identified, operators responded appropriately to correct the configuration issue and reduce isolated loop pressure. The loop was drained, evaluated, and inspected. After replacing the RCP seals, the loop was filled and subsequently returned to service. The licensee documented this issue in CR 09-67705 and CR 09-67767; and

- Unit 2: On November 24, at 3:05 a.m., an Unusual Event (UE) was declared (EN #45517) in response to Identified Leakage greater than 25 gpm (EAL 2.6). During refueling outage 2R14, in hot shutdown, while securing the Residual Heat Removal Systems (RHS) to prepare for plant heatup, the "A" RHS suction relief valve lifted, relieving reactor coolant to the Pressurizer Relief Tank (PRT). The cause of the relief valve lift was the RHS shutdown procedure placed the plant in a configuration (one RHS train shutdown and one RHS train operating with the cross-connect valve open) and allowed a system pressure (~350 psig) that would challenge the relief valve setpoint (~450 psig). Operators took appropriate actions to correct the condition and the relief valve reseated close, terminating the event. The inspectors responded to the control room and reviewed licensee actions and response. The UE was terminated at 4:04 a.m. The licensee documented this issue in CR 09-68214.

b. Findings

Unit 2 Isolated 'B' Loop Over-pressurization during Refueling Outage 2R14

<u>Introduction</u>. A Green self-revealing NCV of TS 5.4.1, "Procedures", was identified in that operators failed to properly align and check the position of the "B" reactor coolant system (RCS) loop bypass valve [2RCS*45], as required by procedure. This deficiency caused an incorrect lineup of the required vent path and resulted in the over-pressurization of the isolated "B" RCS loop while filling. The estimated pressure exceeded the pressure/temperature limit for an isolated RCS loop on November 11. <u>Description</u>. While filling the "B" RCS loop, the required vent path for the loop [2RCS*45] was discovered isolated by field operators. The loop was full, and as a result, it was exposed to full charging water pressure. The licensee bounded this estimate to be 2715 psig. This resulted in a challenge to the RCS pressure boundary and reactor coolant pump seals. The maximum allowable pressure for an isolated loop at 60F (current charging water temperature) was approximately 800 psig as derived from Figure 5.2-7 (Isolated Loop Pressure – Temperature Limit Curve) in the Pressure and Temperature Limits Report. The operators entered LCO 3.4.3 which requires parameters restored within limits in 30 minutes and determine if the RCS is acceptable for continued operation prior to entry into MODE 4. Both actions were completed within the required time frames. The licensee replaced the "B" RCP seals as recommended by the vendor.

On November 11, 2009, operators were filling the isolated "B" RCS loop in accordance with 2OM-6.4.N, Revision 22, "Reactor Coolant Loop Recovery with Fuel in the Vessel." This procedure requires both loop isolation valves closed and uses the reactor coolant loop bypass line to relieve pressure back to the reactor side of the loop isolation valves once the loop is full. A vacuum is drawn on the loop bypass high point vent and charging water is injected to the isolated loop through the seal water injection line. A precaution for the procedure is as follows:

"An overpressure relief path shall be maintained for an isolated loop at all times to prevent an inadvertent over-pressurization of the loop."

This precaution is reiterated as a note in the procedure. Also, the isolation valve [2RCS*45], "Loop B Bypass Flow Isolation Valve" is to be verified open and independently checked.

While lining up the system for filling, the operators performing the step to ensure the vent path was open, noticed the valve [2RCS*45] was locked with a chain. At the time, the operators failed to bring the correct key to unlock the valve for verification. The operators attempted to check the valve open with the chain installed. They obtained minimal movement of the valve handle. The operators incorrectly concluded that the valve was open. The valve was actually locked in the closed position and had been locked closed since November 2, 2009 as recorded in the shift narrative logs.

On November 2, 2009, two operators were sent into the containment to restore the lock and chain for [2RCS*45]. The chain and operator were removed to allow maintenance personnel to install piping insulation on the RCS. The operators received instructions to close and lock [2RCS*45]. The Normal System Alignment (NSA) for this valve is locked open. 1/2OM-48.3.C, Rev. 16 "Padlocks/Locking Devices Administrative Requirement" states:

"Whenever manipulating a locked component out of NSA, the locking device should be locked and secured locally. If a chain and lock are used, the chain and lock should be placed around the pipe or otherwise secured or located in such a way that it is obvious the chain and lock are not securing the component, but are available for subsequent return to NSA. (Exclusion – during procedure performance periods that will not exceed shift turnover)."

Contrary to this requirement, a component was locked in a position other than its NSA position. In this instance, the exclusion was not applicable. On November 2 and 11, operators failed to follow specified procedural guidance resulting in [2RCS*45] out of its NSA position since November 2. The verifier also failed to identify this discrepancy. An immediate stand-down and training was conducted to reinforce valve check and verifications expectations. Also, loop fill procedures were revised to include additional pressure indications during loop fill.

Analysis. The failure to properly align and check the position of the "B" reactor coolant system (RCS) loop bypass valve [2RCS*45], as required by procedure is considered a performance deficiency. The inspectors determined that the finding was not similar to the examples for minor deficiencies contained in IMC 0612, Appendix E, "Examples of Minor Issues". The finding was more than minor because if left uncorrected could have the potential to lead to a more significant safety concern. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, Checklist 4 "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer." There were no conditions indicating a loss of control as listed in Appendix G, Table 1 "Losses of Control." Because the loop was isolated from the reactor vessel and pressurizer, the required reactor coolant inventory and the decay heat removal system was not affected. All mitigating capabilities were available, therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). Because this finding is of very low safety significance and has been entered into FENOC's corrective action program, the violation is being treated as a non-cited violation.

The cause of this finding is related to the cross-cutting area of human performance, work practices, in that FENOC's failed to follow station procedures resulting in an over-pressurization of the isolated "B" RCS loop. [H.4.(b)].

Enforcement. TS 5.4.1, "Procedures", requires that written procedures be implemented as recommended in Appendix A of RG 1.33, including Instructions for Filling, Venting, and Draining of the Reactor Coolant System (RCS). Contrary to this requirement, on November 11, 2009, FENOC failed to ensure a required vent path was open which resulted in an over-pressurization of the isolated "B" RCS loop. Because this deficiency is considered to be of very low significance (Green), and was entered into the corrective action program (CR 09-67705 and 09-67767) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 0500412/2009005-1, Failure to Properly Verify Valve Line-up Results in an Over-pressurization of the Isolated "B" Reactor Coolant Loop)**

Unit 2 RHS Pump Relief Valve Lift Causes RCS Identified Leakage >25 gpm

Introduction. A Green self-revealing NCV of TS 5.4.1, "Procedures", was identified in that procedures for securing Residual Heat Removal System (RHS) were not adequately maintained and did not contain relevant operating restrictions resulting in the inadvertent lifting of the “A” RHS pump suction relief [2RHS-RV721A] during normal operation, excessive identified leakage of reactor coolant to the Pressurizer Relief Tank (PRT), and a declaration of an Unusual Event.

Description. On November 24, 2009 while in hot shutdown (MODE 4), bubble in the pressurizer, preparations were being made to commence plant heat-up to MODE 3. Both trains of RHS were in service and the “A” and “C” reactor coolant pumps were running. Preparations to start the “B” RCP were being made. During past outages, the sequence of events and the progression of plant startup would have both RHS pumps in standby to commence plant heat up, with subsequent isolation and cool down of the RHS trains. But during this outage, partially due to monitoring the seal injection flow characteristics for the “B” RCP, the crew continued to operate the “B” RHS pump for more efficient RCS temperature control and the possibility to return to MODE 5 for additional RCP maintenance. The crew began to shutdown the “A” train RHS after the “B” RCP was started. Reactor pressure was 350 psig to support the RCP start.

Per station start-up procedure 2OM-50.4.M, Rev. 11, “Station Startup-Mode 5 To Mode 3,” the crew was beginning to secure the first train of RHS. The station startup procedure directs the use of 2OM-10.4.C, “Residual Heat Removal System Shutdown” to shutdown, isolate, and cool down the RHS trains. The train “A” RHS valve interlocks were first restored. The “A” RHS pump was already stopped and in standby with the “A” train cross-connect valve opened and discharge valve shut. The “B” train cross-connect valve was already open to support “B” train RHS operation. When the operator shut the “A” RHS pump suction valve [2RHS-MOV701A], suction relief valve [2RHS-RV721A, 450 psig setpoint] lifted as designed due to over-pressure applied to the “A” RHS loop from the combination of “B” pump discharge and system pressure through the two open cross-connect valves and “A” pump mini-flow line. The relief valve lift resulted in the transfer of water from the reactor coolant system to the PRT. The flow rate was sufficient to exceed the emergency action level for Identified Leakage (EAL 2.6). An Unusual Event (UE) was declared at 3:05 a.m. The crew responded to indications of lowering pressurizer level and rising PRT level and took appropriate actions and stabilized the plant. The event was terminated in nine minutes by closing the “A” train cross-connect valve [2RHS-MOV750A]. The relief valve reseated. The UE was terminated at 4:04 a.m.

The procedure for securing RHS is blind to the interaction of the other train. There are no precautions, limitations, or procedural restraints that would have alerted or prevented the plant condition and configuration that existed during this event even though similar operating experience exists (i.e. LER 87-008, Farley Unit 2). The licensee has revised station start-up and RHS system running and shutdown procedures to prevent operating with both cross-connect valves while one RHS pump is running.

Analysis. The failure to adequately maintain RHS shutdown procedure by not containing relevant operating restrictions to prevent lifting of the system relief valve during normal operation is considered a performance deficiency. The inspectors determined that the finding was not similar to the examples for minor deficiencies contained in IMC 0612, Appendix E, “Examples of Minor Issues”. The finding was more than minor because if left uncorrected could have the potential to lead to a more significant safety concern. Traditional enforcement does not apply because the issue did not have an actual safety consequence or the potential for impacting NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G. There were no conditions indicating a loss of control as listed in Table 1 "Losses of Control." Attachment 1, Checklist 1 "PWR Hot Shutdown Operation: Time to Core Boiling <2 Hours" guidelines were used to evaluate the event. All mitigating capabilities were available, therefore a Phase 2 quantitative assessment was not required. The issue screens to Green (very low safety significance). Because this finding is of very low safety significance and has been entered into FENOC's corrective action program, the violation is being treated as a non-cited violation. The cause of this finding is related to the cross-cutting area of human performance, resources, in that procedures for RHS system shutdown were not complete and up to date. [H.2(c)].

Enforcement. TS 5.4.1, "Procedures", requires that procedures be maintained as recommended in Appendix A of RG 1.33, including Instructions for Shutdown Cooling Systems; Residual Heat Removal System. Contrary to this requirement, FENOC failed to adequately maintain the RHS shutdown procedure by not containing relevant operating restrictions to prevent lifting of the system relief valve during normal operations. Because this deficiency is considered to be of very low significance (Green), and was entered into the corrective action program (CR 09-68214) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 0500412/2009005-2, Inadequate RHS Shutdown Procedure Results in Unusual Event Declaration)**

.2 Review of Licensee Event Reports (LERs) (2 samples)

(Closed) LER 05000412/2009-001-00: Equipment Operability for Steam Generator Tube Rupture Safety Analysis Not Met.

The LER discusses the licensee's identification that a combination of two inoperable same-train components could invalidate the design basis accident safety analysis for a postulated Steam Generator Tube Rupture (SGTR), even though Technical Specification requirements would be met. The licensee identified that if the "A" Emergency Diesel Generator (EDG) is not available and one of the four automatic steam dump valves (ADV) is not available, a condition may exist where the steam generator could overfill during a postulated SGTR event. Corrective actions include an operations order, revising Technical Specification Basis for clarity and revising site procedures to prohibit removing one EDG and one ADV from service simultaneously. The inspectors reviewed the LER, verified the appropriateness of corrective actions, and extent of condition reviews. No findings of significance were identified and no violation of NRC requirements occurred. This LER is closed.

(Closed) LER 05000334/2009-004-01: Two Ultrasonic Indications Found in Reactor Coolant System Drain Pipe.

The LER discussed the most probable cause of the two indications, identified on April 26, 2009 during planned MRP-146 inspections, based on the completed metallurgical analysis of the affected pipe segment. No new issues were identified. The inspectors reviewed the LER and no findings of significance were identified and no violation of NRC requirements occurred. This LER is closed.

4OA5 Other

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with site security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours. These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515/172, RCS Dissimilar Metal Butt Welds (DMBW) (Unit 2)

a. Inspection Scope

Temporary Instruction (TI) 2515/172 provides for confirmation that owners of pressurized-water reactors (PWRs) have implemented the industry guidelines of the MRP-139 regarding nondestructive examination and evaluation of certain dissimilar metal welds in reactor coolant systems containing nickel based Alloys 600/82/182.

Beaver Valley Power Station Unit 2 has MRP-139 applicable Alloy 600/82/182 RCS welds. Unit 2 has three 29" reactor vessel outlet hot nozzle-to-safe-end welds (2RCS*REV21-N-24, N-26, N-28) and three 27.5" reactor vessel inlet cold leg nozzle-to-safe end welds (2RCS*REV21-N-23, N-25, N-27) DMBW connections that were examined from the inside volumetrically by ultrasonic testing (UT) and on the inside diameter (ID) surface by eddy current during 2R13, which were previously reviewed and inspected by the NRC inspectors. In addition, six pressurizer nozzle dissimilar metal welds were preemptively mitigated by full structural weld overlay during 2R12. During 2R14 FENOC performed visual inspections of the three hot leg nozzle-to-safe-end DMBW connections and one cold leg (N-23) nozzle-to-safe-end DMBW connection.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

.1 Licensed Operator Requalification

The inspectors presented the inspection results to members of licensee management at the conclusion of the onsite inspection on June 26, 2009. Full requalification examination results were reviewed in a telephone call between the lead inspector and Mr. B. Rudolph, Superintendent of Operations Training, on August 19, 2009.

.2 Inservice Inspection

The inspectors presented the Unit 2 ISI and TI 2515/172 inspection results to Mr. Paul Harden, Site Vice President, and other members of the FENOC staff at the conclusion of the inspection on October 29, 2009. The licensee acknowledged the conclusions and observations presented. Some proprietary information was reviewed during this inspection and was either returned to the licensee or properly destroyed.

.3 Access Control / ALARA Planning and Control

The inspector presented the inspection results of 2S01 and 2S02 to Mr. Ray Lieb, Director of Site Operations, and other members of FENOC staff, at the conclusion of the inspection on October 29, 2009.

.4 Quarterly Inspection Report Exit

On January 12, 2010, the inspectors presented the normal baseline inspection results to Mr. Paul Harden, Site Vice President, and other members of the FENOC staff. The inspectors confirmed that proprietary information was not retained at the conclusion of the inspection period. No proprietary information is presented in this report.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Alberti	Steam Generator Program Owner
S. Baker	Site, Radiation Protection Manager
J. Bowden	Superintendent, Operations
S. Checketts	Operations Manager
T. Crella	Senior Radiation Protection Technician
E. Crosby	Supervisor, ALARA
B. Goff	Supervisor, Nuclear Work Planning
D. Grabski	ISI Coordinator
G. Hackett	Supervisor, Rad Operations Support
P. Harden	Site Vice President
T. Heimel	NDE Level III
D. Jones	Staff Nuclear Engineer
R. Lieb	Director, Site Operations
J. Lutz	Senior Reactor Operator, Operations
T. Metler.	Senior Radiation Protection Technician
N. Morrison	Superintendent, Nuclear Work Planning
J. Patterson	Containment Liner Program Owner
R. Pucci	Senior Nuclear Specialist – ALARA
J. Saunders	Supervisor, Radwaste/Shipping
P. Sena	Site Vice President
B. Sepelak	Supervisor, Regulatory Compliance
J. Severyn	Supervisor, Nuclear Engineering Programs
B. Tuite	Site Training Manager
W. Williams	Alloy 600 Program Owner

Other Personnel

L. Ryan	Inspector, Pennsylvania Department of Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open/Closed

05000412/2009005-01	NCV	Failure to Properly Verify Valve Line-up Results in an Over-pressurization of the Isolated "B" Reactor Coolant Loop. (Section 4OA3.1)
05000412/2009005-02	NCV	Inadequate RHS Shutdown Procedure Results in Unusual Event Declaration. (Section 4OA3.1)

Closed

05000412/2009001-00	LER	Equipment Operability for Steam Generator Tube Rupture Safety Analysis Not Met. (Section 4OA3.2)
05000334/2009004-01	LER	Two Ultrasonic Indications Found in Reactor Coolant System Drain Pipe. (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures
1OST-36.7, Rev. 15, Offsite to Onsite Power System Distribution Breaker Alignment Verification

Drawings
8700-RE-21G, Rev. 20, Three Line Current Diagram Generator and Transformer
8700-RE-21Q, Rev. 9, Three Line Synchronizing Diag 345kV Switchyard
8700-RE021R, Rev. 6, Three Line Synchronizing Diag 138kV Switchyard
10080-RM-85A, Rev. 22, Flow Diagram-Containment Depressurization Piping, Sh. 1

Other
2DBD-13, Rev.11, Design Basis Document for Containment Depressurization System
Unit 1 Shift Operating Logs, October 29, 2009

Section 1R05: Fire Protection

Procedures
2-PFP-RCBX-692, Rev. 1, Reactor Containment Building (Fire Area RC-1)
2-PFP-RCBX-718, Rev. 1, Reactor Containment Building (Fire Area RC-1)
2-PFP-RCBX-738, Rev. 1, Reactor Containment Building (Fire Area RC-1)
2-PFP-RCBX-767, Rev. 1, Reactor Containment Building (Fire Area RC-1)
1-PFP-STOR-725, Rev. 1, Storeroom (Fire Area WH-1 and WH-2)
2-PFP-MSCV-735, Rev. 4, West cable Vault (Fire Area CV-1)

Section 1R08: Inservice Inspection

Other
Ultrasonic Report Data Sheet Penetration No. 25, dated 10/21/2009
Ultrasonic Report Data Sheet Penetration No. 48, dated 10/21/2009
Ultrasonic Report Data Sheet Penetration No. 49, dated 10/22/2009
Ultrasonic Report Data Sheet Penetration No. 57, dated 10/21/2009
Ultrasonic Report No. UT-09-1085, 2SIS-006-12-1, dated 10/18/2009
Ultrasonic Report No. UT-09-1086, 2SIS-006-26-1, dated 10/18/2009
Ultrasonic Report No. UT-09-1080, 2SIS-006-24-1, dated 10/17/2009
Ultrasonic Report No. UT-09-1081, 2SIS-006-25-1, dated 10/17/2009
Ultrasonic Report No. UT-09-1076, 2RCS-002-65-1, dated 10/16/2009

Ultrasonic Report No. VEN-09-1001 and VEN-09-1002, dated 10/25/2009
Ultrasonic Report No. VEN-09-1003 and VEN-09-1004, dated 10/25/2009
Ultrasonic Report No. VEN-09-1011 and VEN-09-1012, dated 10/25/2009
Liquid Penetrant Examination Report No. BOP-PT-09-122, dated 10/28/2009
Visual Examination for Leakage (VT-2) Report No. VT-09-1325, RV Bottom Mounted Instrumentation Nozzles, dated 10/19/2009
Visual Examination for Leakage (VT-2) Report No. VT-09-1327, BMI Nozzle to Tube Welds, dated 10/19/2009
BVPS Unit 1/2 ISIE-ECP-2 Steam Generator Examination Program, Rev. 21
BVPS Unit 1/2 ISIE1-8, Unit 2 Steam Generator Examination Guidelines, Rev. 11
Westinghouse Procedure MRS 2.3.2 GEN-13, Mechanical Ribbed Plugging of Steam Generator Tubes, Rev. 25
BVPS Unit 1/2 ADM-2099, Primary Containment ISI Program, Rev. 0
BVPS Unit 1/2 NDE-GP-106, Reactor Vessel head Inspection, Rev. 1
WesDyne Procedure WDI-PJF-1304060-EPP-001, UT Programs, Rev. 0
WesDyne Procedure WDI-ET-002, IntraSpect Eddy Current Inspection of Vessel Head Penetration J-Welds and Tube OD Surfaces, Rev. 13
WesDyne Procedure WDI-STD-1040, Procedure for Ultrasonic Examination of Reactor Vessel Head Penetrations, Rev. 3
WesDyne Procedure WDI-STD-1041, Reactor Vessel Head Penetration Ultrasonic Examination Analysis, Rev. 2
WesDyne Procedure WDI-STD-1042, Procedure for Eddy Current Examination of Reactor Vessel Head Penetrations, Rev. 0
WesDyne Procedure WDI-UT-013, IntraSpect UT Analysis Guidelines, Rev. 13
ASME Section XI, Code Case N-729-1, Alternative Examination Requirements for PWR Reactor Vessel Upper Head with Nozzles Having Pressure-Retaining Partial-Penetration Welds
MRP-146, Implementation for Thermal Fatigue Management of RCS-Attached Nonisolable Piping
Westinghouse Procedure STD-011, Liquid Penetrant Examination, Rev. 5
Westinghouse Procedure DLW-SG-001, Standard In Situ Pressure Test Using the Computerized Data Acquestion System, Rev. 00
Beaver Valley Power Station Unit 2 - Relief Request No. 2-TYP-3-RV-01 Regarding Alternative Repair Methods For Reactor Vessel Head Penetrations & J-Groove Welds, dated 10/6/2009
SG-CDME-08-33, Beaver Valley Unit 2 Steam Generator Cycle 14 Operational Assessment, August 7, 2008
SG-SGMP-09-15, Beaver Valley Unit 2 2R14 Steam Generator Degradation Assessment, October 15, 2009

Certifications
WesDyne UT examiner certifications

CRs

09-51612	09-55146	09-65313	09-65756	09-65777	09-66202
09-66246	09-66489	09-66557	09-66685		

Section 1R11: Licensed Operator Requalification Program

Lesson Plans:
3LRT-09C2.OER 2009 Cycle 2 Operating Experience Review

2LRT-2R13.SD Cycle 2008 Module 2 2R13 Pre Outage Training
3LRT-09C3.OER 2009 Cycle 3 Operating Experience Review
Procedures:
Licensed Operator Requalification Exam Development and Administration, Rev. 6
1/2- ADM-1362 Security Provisions for Licensed Operator Examinations
1/2 ADM-1360 Licensed Operator Tracking
1/2 ADM-1351 Licensed Operator Continuing Training Program
1/2 ADM-1357 Conduct of Simulator Training
1/2-ADM-0713 Time-Critical Operator Action Standard
BVPS 2 Fire Protection Safe Shutdown Report Table 2 - Time Restrictions for Manual Actions Following a Fire.
BVPS 1 Appendix R Review Table 5.1-3 Time critical Manual Actions.

Condition Reports:
08-40825 08-44957 08-39835

Simulator Testing:
BVBP-TR-0017 Simulator Configuration Control

Plant Data Comparison:
Simulator Evaluation of Down Power Transient for Water Box Maintenance 20 Feb 2009
Simulator Evaluation of Up Power Transient from Water Box Maintenance 22 Feb 2009

Transient Tests:
SQT-5.1 Manual Reactor Trip 3/6/2008
SQT-5.2 Complete Loss of All Feedwater 3/6/2008
SQT-5.3 Main Steam Isolation Valves Closure 3/6/2008
SQT-5.4 Complete Loss of Reactor Coolant Flow 3/6/2008
SQT-5.7 Maximum Rate Power Ramp 3/6/2008

Core Performance Tests:
SQT-3.1 Core Cycle Installation Test 8/26/2008

Simulator Steady State Testing:
SQT-6.1, Steady State Drift Test – Full Power 11/25/08
SQT-6.2, Steady State Drift Test – Mid Power 3/27/09
SQT-6.3, Steady State Drift Test – Low Power 12/2/08

Simulator Transient Testing:
SQT-5.3, Main Steam Isolation Valve closure 3/6/08
SQT-5.5, Partial Loss of Reactor Coolant Flow 3/6/08
SQT-5.7, Maximum Rate Power Ramp 3/6/08

Simulator Malfunction Testing:
SQT-14.1.5.4.21.03, Inadvertent Safety Injection Signal Test 3/13/08
SQT-14.1.5.4.12.07, Inadvertent Turbine Trip Test 3/10/08
SQT-14.1.5.4.08.08, Stuck Rod(s) Test 2/11/08
SQT-14.1.5.4.06.03, Condenser Tube leak Test 2/4/09

Section 1R12: Maintenance Rule Implementation

Notifications
600575987 600575988 600575989 200380116
Procedures
NOP-ER-3004, Rev. 1, FENOC Maintenance Rule Program

Other
Unit 2 System Health Report 2009-2, System 33-Unit 2 Fire Protection System

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Work Orders
200393093

Condition Reports
09-67509 09-67510 09-67454

Section 1R15: Operability Evaluations

Calculations
10080-E-222, Rev. 0, Addendum A1

Procedures
1OM-24.4.IF, Rev. 9, Instrument Failure Procedure

Condition Reports
08-38908 09-67166

Miscellaneous
2DBD-37, Rev. 6, Design Basis Document for 480v Distribution System
WO 200392582

Section 1R18: Plant Modifications

Drawings
10080-RM-0047D, Rev. 0
10080-RM-0047A, Rev. 0
10080-RM-0430-004, Rev. 0
10080-2806.263-920-360, Rev. 0
2808.262-920-605, Rev. 0

Section 1R19: Post-Maintenance Testing

Procedures
2OST-36.2, Rev. 58, Emergency Diesel Generator [2EGS*EG2-2] Monthly Test
2OM-36.40AE, Rev. 25, Diesel Generator 2 Automatic Test
2MSP-36.18-E, Rev.15, No. 2 Emergency Diesel Generator Electrical Inspection
2MSP-36.20-M Rev. 6, #2 Emergency Diesel Generator Inspection
2MSP-36.30-M, Rev. 18, #2 Emergency Diesel Generator, Filter, Strainer, Heat Exchanger, and Woodward Governor Maintenance
2MSP-36.0018-E, Rev. 11, Load Shedding and Auto sequencing of 'B' Train Emergency Bus Breaker Cubicles

1MSP-2.12-I, Issue 4, Rev. 16, Power Range Neutron Flux Channel N42 Channel Operational Test
1MSP-2.14-I, Issue 4, Rev. 16, Power Range Neutron Flux Channel N43 Channel Operational Test
1MSP-2.14-I, Issue 4, Rev. 16, Power Range Neutron Flux Channel N44 Channel Operational Test
2OST-10.3, Rev. 23, Residual Heat Removal System Train 'A' Valve Exercise

Work Orders

200334047	200334045	200334014	200334844	200291016	200392119
200392120	200392121	200392122	200392123	200392124	200392125
200392130	200392121	200392132	200392133	200060019	200247098
200139151					

Condition Reports
09-66827 09-66829

Other
ECP 08-216
ODMI "Nuclear Instrumentation Bistable Card Failure Issue," Rev. 0

Section 1R20: Refueling and Outage Activities

Procedures and Surveillances
2OM-6.4.I, "Draining the RCS for Refueling"
2OM-47.4.B, "Personnel Air Lock Operations"
2OM-49.4.H, "Movement of Spent Fuel Pool Crane Checklist"
2OM-51.4.I, "Station Shutdown-Preparation for Entering Refueling (Mode 6)"
2OST-6.2A, "Computer Generated RCS Water Inventory Balance"
2OST-47.3.E, Rev. 5, "Verification of Administrative Closure Controls for Containment / Fuel Building during Refueling"
2OST-49.3, "Refueling Operations Prerequisites"
2RP-2.6, "Remove Reactor Vessel Studs/Clean"
2RST-2.1, "Initial Approach to Criticality After Refueling"
AOP-2.6.5, Shutdown LOCA
AOP-2.10.1, RHR System Loss
AOP-2.36.1, Loss of All AC while Shutdown
IPTE - Draining Down the RCS for Refueling
RWP 309-3002
2OST-49.2, "Shutdown Margin Calculation", performed on October 14, 2009
2OST-11.18, "Low Head Safety Injection Pump Boric Acid Flowpath Verification"

Condition Reports

09-68436	09-68096	09-66279	09-65798
09-68288	09-68046	09-66247	09-65714
09-68232	09-68043	09-66243	09-65702
09-68217	09-68026	09-66241	09-64619
09-68142	09-68001	09-66211	09-64078
09-68124	09-67799	09-66182	09-64039
09-68118	09-66352	09-65883	
09-68116	09-66325	09-65866	

Other
2R14 Outage Handbook
2OM-50.4.L, RCS and Pressurizer Spray Heatup Data and Plots
Unit 2 Plant Computer Cooldown Data tables and plots, dated October 12-13, 2009

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

Procedures:
Access Control to Radiologically Significant Areas/ALARA Planning & Controls
1/2ADM-1601, Rev 20 Radiation Protection Standards
1/2ADM-1611, Rev 10 Radiation Protection Administrative Guide
1/2HPP-3.02.004, Rev 4 Area Posting
1/2HPP-3.05.001, Rev 8 Exposure Authorization
1/2HPP-3.07.002, Rev 7 Radiation Survey Methods
1/2HPP-3.07.013, Rev 7 Barrier Checks
1/2HPP-3.08.003, Rev 4 Radiation Barrier Key Control
1/2HPP-3.08.006, Rev 2 Shielding
BVBP-RP-0003, Rev 8 Dosimetry Practices
BVBP-RP-0013, Rev 3 Radiation Protection Risk Assessment Process
BVBP-RP-0020, Rev 15 RP Job Coverage General Guidance
NOP-OP-4206, Rev 1 Bioassay Administration
NOP-OP4005, Rev 1 ALARA Program
NOP-OP-4005, Rev 1 Operational ALARA Program
NOP-OP-4107, Rev 4 Radiation Work Permit
NOP-WM-7017, Rev 1 Contamination Control Program
NOP-OP-4102, Rev 4 Radiological Postings, Labeling, and Markings

Nuclear Oversight Field Audits/Observation Reports
Quarterly Reports Fleet Oversight for 2nd, 3rd, and 4th quarters 2009

BV320093800	BV320093807	BV320093778	BV120093770
BV120093772	BV220093754	BV320093752	

Condition Reports

09-66849	09-66795	09-66798	09-66811	09-66682	09-66669
09-65483	09-65694	09-65476	09-66583	09-62421	09-61973
09-62049	09-63423	09-63408	09-64836	09-65538	09-65816
09-66056	09-65816	09-65980	09-66073	09-66600	09-66258
09-66270	09-66150	09-65701			

ALARA Plans & related Work-in-Progress /Post-Job Reviews
09-2-30, Reactor Disassembly/Reassembly
09-2-41, Scaffolding for GSI-191 Activities
09-2-48, Under reactor head inspections
09-2-57, Scaffold Installation in Containment Building
09-2-60, Kerotest Valve Replacement
09-2-61, Radiation Protection Insulation Support

ALARA Committee Meeting Minutes

Attended Meeting Nos.2R14-08m, 2R14-10m, 2R14-11m,
Reviewed meeting minutes for: 2R14-01m 2R14-02m 2R14-03m 9-22m
9-21m 09-08 m 09-09 m (m-manager's, s-subcommittee)

Miscellaneous ALARA Reports
2R14 Outage ALARA Plan
EPRI Standard Radiation Monitoring Program - Unit 2 Source Term Measurements
High Dose Individuals for 2009
Dose and Dose Rate Alarm Reports for 2009

Section 4OA2: Identification and Resolution of Problems

Condition Reports
09-62705 09-59866 09-68992 09-69032 08-49448 09-68967
08-47144

Procedures
Procedure 1/2 ADM-0801 ASME Section XI Repair/Replacement Program, Rev. 8, 9/30/09
BVPS-SITE-0053, Post-Maintenance Test Requirements, Rev. 3, 4/17/09
NOP-WM-1001; Order Planning Process, Rev. 12, 8/12/09
NOP-WM-1005; Work Management Order Testing Process, Rev. 2, 8/29/08
NOBP-OP-0012, "Operator Work-Arounds, Burdens and Control room Deficiencies", Rev. 01

Notifications
0600343638 0600055702 0600505843

Reports
Snapshot Self-Assessment Plan, BV-SA-09-042, "Unit 2 Operator Work Arounds, Burden and Control Room Deficiencies"
Snapshot Self-Assessment Plan, BV-SA-09-043, "Unit 1 Operator Work-Arounds, Burdens and Control Room Deficiencies"

Section 4OA3: Event Response

Condition Reports
09-65941 09-66489 09-67606 09-67463

Procedures
2MSP-6.90-I, "Calibration of Various In-Containment Protection Transmitters During Shutdown", Issue 4 Rev. 8
2OM-6.2.B, "Reactor Coolant System Operations-Setpoints", Rev.12
2OM-10.4.C, "Residual Heat Removal System Shutdown", Rev. 33

MISC
Operations Dept Assessment of Operator Performance, Reduction of Letdown Flow Event (Oct 14, 2009)
BV2 Operations Shift Logs dated October 14-15, 2009
BV2 Plant Computer Parameter Printouts, October 14, 2009
L-09-309, "10 CFR 50.55a Request for Alternative Weld Repair Method for Reactor Vessel Head Penetration J-Groove Welds", dated November 14, 2009
Unit 2 Reactor Vessel Repair Plans dated November 5-14, 2009
Vendor Assessment of Unit 2 'C' RCP Seal #1 delta Pressure, October 16, 2009

LIST OF ACRONYMS

ADM	Administrative Procedure
ALARA	As Low As is Reasonably Achievable
AMC	ALARA Managers Committee
AP	ALARA Plan
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
BCO	Basis for Continued Operations
BMI	Bare Metal Inspection
BVPS	Beaver Valley Power Station
CFR	Code of Federal Regulations
CR	Condition Report(s)
CRDM	Control Rod Drive Mechanism
DMBW	Dissimilar Metal Butt Welds
DPW	Declared Pregnant Workers
ECT	Eddy Current Testing
FA	Functionality Assessments
FENOC	First Energy Nuclear Operating Company
ID	Inside Diameter
IMC	Inspection Manual Chapter
IOD	Immediate Operability Determination
IP	Inspection Procedure
ISI	Inservice Inspection
IST	In Service Test
JPM	Job Performance Measures
LCO	Limiting Conditions for Operations
LER	Licensee Event Report
MR	Maintenance Rule
MRP	Materials Reliability Program
MSP	Maintenance Surveillance Package
NDE	Non Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	Operability Determinations
OD	Outside Diameter
ODSCC	Outside Diameter Stress Corrosion Cracking
OST	Operations Surveillance Test
OWA	Operator Work Around
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Testing
POD	Prompt Operability Determination
PORV	Power Operated Relief Valve
PT	Penetrant Testing
PWR	Pressurized-Water Reactors
RBC	Reactor Building Containment
RPV	Reactor Pressure Vessel
RSS	Recirculation Spray System
RWP	Radiation Work Permit

SSC	Structures, Systems, and Components
SG	Steam Generator
TMOD	Temporary Modifications
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VT	Visual Testing