January 29, 2004

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

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2003005 AND 05000412/2003005

Dear Mr. Pearce:

On December 31, 2003, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Beaver Valley Power Station Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 22, 2004 with Mr. Pearce and other members of your staff.

The inspection(s) 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.

This report documents three self-revealing findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. If you contest any NCV 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.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the orders. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year 2002, and the remaining inspection activities for Beaver Valley were completed in calendar year 2003. The NRC will continue to monitor overall safeguards and security controls at Beaver Valley.

In accordance with 10 CFR 2.790 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).

Sincerely,

/RA/

Peter W. Eselgroth, Chief
Reactor Projects Branch 7
Division of Reactor Projects

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

Enclosures: Inspection Report 05000334/2003005 and 05000412/2003005
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-334, 50-412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2003005 and 05000412/2003005
Licensee:	First Energy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	September 28, 2003 - December 31, 2003
Inspectors:	P. Cataldo, Senior Resident Inspector R. Barkley, Senior Project Engineer G. Smith, Resident Inspector J. McFadden, Health Physicist T. Fish, Senior Operations Engineer J. Laughlin, Operations Engineer
Approved by:	Peter W. Eselgroth, Chief Reactor Projects Branch 7 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000334/2003005 and IR 05000412/2003005; 09/28/2003 - 12/31/2003; Beaver Valley Power Station, Units 1 & 2; Maintenance Risk Assessment and Emergent Work Control, Refueling and Outage Activities, Event Follow-up, and Cross-Cutting Areas.

The report covered a three-month period of inspection by resident inspectors, an announced inspection by a regional health physics inspector, and an announced inspection by regional operations engineers. Four Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for the failure to take adequate and timely corrective actions associated with the spurious trip of Unit 1 480V circuit breaker 9P7. This failure rendered an Emergency Diesel Generator (EDG) out of service for more than seven hours. The licensee is currently in the planning stages to resolve the spurious trip issue associated with the affected 480 volt circuit breakers.

  The finding was considered a performance deficiency since there were existing generic communications, i.e., NRC IN 93-75, as well as similar failures at Beaver Valley that occurred in 1997, that were not adequately addressed. The finding is more than minor because it affected the Mitigating System cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because the EDG was out of service for less than the technical specification allowed outage time of 72 hours. (Section 1R13)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, for failure to have adequate work instructions that led to the failure of a cable clamp associated with the Unit 2 spent fuel pool upender frame. The licensee affected repairs and performed an extent of condition on the containment side upender as well as the Unit 1 upender equipment.

  This issue was determined to be more than minor, because if left uncorrected, could become a more significant safety concern involving the potential damage to fuel assemblies. Because this issue involves SFP handling and storage issues, it cannot be evaluated under the NRC Significance Determination Process. Therefore, this finding was reviewed by NRC management and determined to be of low safety significance (Green), because the event did not result in damage to a fuel assembly and was identified while the upender was empty. (Section 1R20)

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of Technical Specification (TS) 6.8.1, for failure to follow a procedure associated with safety-related equipment on Unit 1. This failure involved human performance errors and resulted in an automatic reactor trip. The corrective actions for this event included procedural improvements and increased management oversight for risk significant activities.

  This finding is greater than minor because it affected the Initiating events cornerstone in that the probability of a reactor trip was increased. The finding is of very low safety significance because the event did not increase the likelihood that mitigation equipment or functions would not be available. (Section 4OA3)

B. Licensee Identified Violations

Violations of very low significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

- 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that procedures shall include appropriate quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, calibration and engineering test procedures involving RCS flow did not contain requirements to ensure correct low flow setpoints, resulting in non-compliance with Technical Specification 3.3.1, Reactor Trip System Instrumentation, Table 3.3-1. However, because this small setpoint error was isolated to only one of nine RCS flow instruments, and did not result in a loss of any RPS safety function, this violation is of very low safety significance and is being treated as an NCV. (Section4OA3) This event was entered into FENOC's corrective action system as CR 03-08007.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated essentially at 100 percent power throughout the inspection period. However, on 11/13/2003, the unit experienced an automatic reactor trip during surveillance testing. After stabilization and evaluation of the cause, the unit returned to 100 percent power on 11/16/2003. Unit 2 began this inspection period in the midst of the tenth refueling outage (2R10), with the unit having been taken offline on 9/13/03. Following the completion of 2R10, the Unit 2 startup commenced on 10/11/03 attained 65 percent power on 10/13/03, followed by power reduction for repairs to secondary equipment. However, on 10/14/03, the unit experienced an automatic reactor trip due to equipment failure. The unit commenced startup activities on 10/16/03, and achieved 100 percent power on 10/20/03. The unit operated essentially at 100 percent power for the remaining of the inspection period, with the exception of minor load reductions on 11/05/03 (2 percent load reduction) and on 11/30/03 (10 percent load reduction) to affect repairs on a feedwater flow measuring instrument, and to support the calibration of a protection circuit loop, respectively.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors reviewed the station’s preparations for adverse weather, relative to the overall protection of safety-related systems, structures, and components (SSCs) from low temperatures. This review included a selective verification of attributes contained within the surveillance procedures listed below, which included operating checks of heat trace circuits, space heaters, and the cooling tower deicing systems, to ensure that equipment is adequately protected from cold weather. The inspectors verified that deficiencies identified through performance of the cold weather protection surveillance were of minor significance, i.e., did not impact operability and were properly captured in the corrective action program for resolution. The following were reviewed to support this inspection:

- 1OST-45.11, Rev. 15/16 — Cold Weather Protection Verification
- 2OST-45.11, Rev. 15 — Cold Weather Protection Verification
- CR-03-11870 — Cold Weather Protection Verification Deficiencies

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

Partial System Walkdowns. (3 samples)

The inspectors performed three partial system walkdowns during this inspection period. The inspectors evaluated the operability of the selected train or system when the redundant train or system was inoperable or unavailable, by verifying correct valve positions and breaker alignments in accordance with the applicable procedures, as well as applicable chapters of the Updated Final Safety Analysis Report (UFSAR).

- On November 17, 2003, the inspectors walked down the Unit 1 'B' quench spray (QS) train while the 'A' train was out of service during performance of maintenance surveillance procedure (MSP), 1-MSP-13.07, Rev. 13, "L-QS100C, Refueling Water Storage Tank Level Channel I Test."

- On November 21, 2003, the inspectors walked down the Unit 1 'B' recirculation spray (RS) train while the '2A' inside RS pump was in pull-to-lock and unavailable during maintenance.

- On November 28, 2003, the inspectors walked down the Unit 2 'B' QS train while the 'A' QS train was unavailable due to the performance of operations surveillance test (OST), 2OST-13.1, Rev. 19, "Quench Spray Pump {2QSS*P-21A} Test."

Complete System Walkdown. (1 sample)

The inspectors conducted a detailed review of the alignment and condition of the Unit 2 High Head Safety Injection (HHSI) System. This system was selected based on its risk significance and the results of previous inspections. The inspectors reviewed plant drawings, abnormal operating procedures, and emergency operating procedures to determine proper equipment alignment. The inspectors reviewed and evaluated the impact on the HHSI system operation due to open work requests and work orders based on existing deficiencies. The condition reports (CRs) associated with the HHSI system were analyzed to verify that the licensee was adequately identifying and correcting system deficiencies. In addition, the inspectors performed a detailed review of the charging/HHSI system latent issues review report and the system health report in order to gain insights on any longstanding issues.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 8 samples)

a. Inspection Scope

The inspectors reviewed the Unit 1 Updated Fire Protection Appendix ‘R’ Review, Rev. 16 and the Unit 2 Fire Protection Safe Shutdown Report, Addendum 18, and identified the following eight risk significant areas for inspection:

- Unit 1 Diesel Generator Room (Fire Area DG-1)
- Unit 1 Diesel Generator Room (Fire Area DG-2)
- Unit 1 Cable Spreading Room (Fire Area CS-1)
- Unit 1 Emergency Switchgear Room (Fire Area ES-2)
- Unit 2 Diesel Generator Building - Orange (Fire Area DG-1)
- Unit 2 Diesel Generator Building - Purple (Fire Area DG-2)
- Unit 2 Cable Tunnel (Fire Area CT-1)
- Unit 2 Service Building Normal Switchgear (Fire Area SB-4)

The inspectors reviewed the fire protection conditions of the areas listed above in accordance with the criteria delineated in 1/2-ADM-1900, Rev. 7, “Fire Protection.” This review evaluated the licensee’s control of transient combustibles, material condition of fire protection equipment, and the adequacy of any compensatory measures for existing fire protection impairments. The inspection also focused on the adequacy of pre-fire plans in accordance with 1& 2OM-56B.3.B, “Pre-Fire Plan Strategies.”

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 2 samples)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report and the Individual Plant Examination, to evaluate the design basis and risk significance of two areas, the Unit 1 emergency diesel generator cubicles, DG-1 and DG-2, relative to internal flooding. The inspectors also reviewed Technical Specifications and operating logs to verify procedures and operator actions for coping with floods were appropriate. The inspector performed walkdowns of the areas to evaluate the material condition of potential sources of internal flooding, and verified various floor drains, sump pumps, and level alarm circuits were operable. The inspector reviewed the following documents in support of this inspection:

- 1/2OM-53C.4A.75.2, Rev. 18 Acts of Nature - Flood
- CR-03-10974 Unit 1 EDG Floor Drains Partially Plugged

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 - 2 samples)

a. Inspection Scope

The inspectors reviewed the licensee's surveillance and control of heat exchanger performance by reviewing the results of a Unit 2 heat exchanger inspection, as well as reviewing the execution of bio-fouling controls for Unit 2 as detailed below:

- The inspectors reviewed the inspection results obtained for the Unit 2 "C" component cooling water heat exchanger, 2CCP-E21C. The review included an assessment of condition report 03-10666, and the heat exchanger inspection report performed in accordance with 1/2-ADM-2106, Rev. 0, "River/Service Water System Control And Monitoring Program." The inspector reviewed the results and evaluated against applicable acceptance criteria, and verified the inspection was consistent with GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

- The inspectors reviewed the licensee's bio-fouling treatment program by reviewing the performance of 2OM-30.4.M, Rev. 22, "BV-2 Asiatic Clam And Zebra Mussel Chemical Treatment Program," and 2BVT-1.30.3, Rev. 8, "Service Water Heat Exchanger Performance Program," conducted during the week of 11/3/2003. The inspector verified results against applicable acceptance criteria, and that deficiencies were captured, as appropriate, in the corrective action program for resolution.

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification (71111.11)

1. Resident Inspector Quarterly Review of Requalification Training (71111.11Q - 1 sample)

a. Inspection Scope

The inspectors observed the conduct of Unit 2 licensed operator requalification training examinations conducted in the facility's simulator on December 11, 2003. The inspectors observed licensed operator performance relative to the following activities: effective communications, implementation of normal, abnormal and emergency operating procedures, command and control, technical specification compliance, and emergency plan implementation. The inspectors evaluated simulator fidelity to ensure major plant configurations or changes were captured in the simulator to ensure adequate training was provided. Inspectors evaluated the operating crew simulator examination for identified deficiencies in operator performance, as well as performance of the staff evaluator's, and verified that any identified conditions adverse to quality were appropriately entered into the licensee's corrective action program for resolution.

b. Findings

No findings of significance were identified.

2. Biennial Licensed Operator Requalification Program Inspection (71111.11 - 1 sample)

a. Inspection Scope

The inspectors reviewed documentation of operating history since the last requalification program inspection. The inspectors also discussed facility operating events with the resident staff. Documents reviewed included NRC inspection reports and Condition Reports (02-04516, 02-09091, 03-01916, and 03-03723) that involved human performance issues. The purpose of this review was to determine whether any plant events were indicative of training deficiencies.

The inspectors reviewed three examples of the Unit 2 2003 comprehensive written exams, and one example of the Unit 2 2003 annual operating test (consisting of three scenarios and five job performance measures). The inspectors also observed the administration of the annual operating test to one operating crew. The purpose of these reviews and assessments was to determine whether exam quality and exam administration met the criteria of NUREG-1021, Rev. 8, "Operator Licensing Examination Standards for Power Reactors," and 10 CFR 55.59, "Requalification."

The inspectors interviewed two instructors, two evaluators, two training supervisors, four ROs, and four SROs for feedback regarding the implementation of the Unit 2 licensed operator requalification program to determine whether training staff modified the program, when appropriate. In addition, six plant and industry events or changes were reviewed to verify that these items were adequately addressed in the Requalification Training Program.

Inspectors reviewed remedial training packages for six individuals that failed an evaluation during the current two-year (January 1, 2002 to December 31, 2003) cycle.

Inspectors reviewed the following documents and records to verify operators were complying with license conditions:

- A sample of Unit 2 attendance records (ten records) for the current two-year training cycle.
- A sample of medical records (five from Unit 1; five from Unit 2) were reviewed to verify that 1) restrictions noted by the doctor were reflected on the individual's license; and 2) the physical exams were given within the last 24 months.
- A sample of Unit 2 license renewals (five records), proficiency watch-standing (five records), and license reactivations (three records).

The inspectors observed Unit 2 simulator performance during the conduct of the examinations, and reviewed Unit 2 simulator performance tests and discrepancy reports to verify compliance with the requirements of 10 CFR 55.46, "Simulator Rule." The following simulator tests were reviewed:

Normal operations tests:
- Steady State Drift Test - Full Power
- Steady State Drift Test - Interim Power
- Plant Startup From Zero Power To Full Power
- OST 2.2.2 Nuclear Intermediate Range Channel Functional Test
- OST 2.24.2 Motor-Driven Auxiliary Feed Pump 2FWE*P23A Test

Transient tests:
- Reactor Trip Test
- Complete Loss of Reactor Coolant Flow Test
- Partial Loss of Reactor Coolant Flow Test
- Turbine Trip Without Direct Reactor Trip Test

Malfunction tests:
- Dropped Rod(s) Test
- Rod Position Step Counter Failure Test
- Reactor Coolant Pump No. 3 Seal Failure Test`

Core Performance tests:
- Reactor Startup

On December 15, 2003, inspectors conducted an in-office review of Unit 1 and Unit 2 test results. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspectors verified that:

- Crew failure rate was less than 20%. (Unit 1 and 2 crew failure rate was 0%)

- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Unit 1 individual failure rate was 0%; Unit 2 individual failure rate was 2%)

- Individual failure rate on the walk-through test was less than or equal to 20%. (Unit 1 and 2 individual failure rate was 0%)

- Individual failure rate on the comprehensive written exam was less than or equal to 20%. (Unit 1 did not administer the comprehensive written exam this year; Unit 2 individual failure rate was 3.9%)

- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Unit 1 overall pass rate was 100%; Unit 2 was 94%)

b. Findings

Introduction. An unresolved item (URI) related to Unit 2 simulator testing was identified. Licensee staff compares simulator transient test data to UFSAR accident analyses data, which appears to be inconsistent with the testing methods described in Appendix A and B of ANSI/ANS 3.5-1985, “Nuclear Power Plant Simulators for Use in Operator Training.”

Description. Beaver Valley training staff determines acceptability of Unit 2 simulator transient test data by comparing these data to UFSAR accident analysis data. This test methodology may not be acceptable for two reasons:

First, ANSI/ANS 3.5-1985, Appendix A, Section A3.3(2), states, in part, that “UFSAR transients are based on ‘worst case’ situations and as such may be inappropriate for real-time dynamic simulation comparisons.” The inspectors reviewed three simulator tests in detail, (summarized in Appendix 1), and identified numerous instances where assumptions used in simulator tests are not necessarily the same as those used in the UFSAR. For example, the turbine trip w/o direct reactor trip test on the simulator assumes the test is initiated from 45% power. However, the facility evaluates data from this test by comparing them against data from the UFSAR test, which assumes a turbine trip is initiated from 100% power.

Second, ANSI/ANS 3.5-1985, Appendix B, Section B2.2.1, lists 11 key parameters that should be trended during the turbine trip w/o direct reactor trip test. However, the UFSAR turbine trip analysis does not include six of those parameters, specifically: pressurizer level, pressurizer temperature, total steam flow, total feed flow, hot leg temperature, and steam generator level are not trended.

Facility staff generated Condition Report CR 03-1167 to address the finding related to simulator testing.

Analysis. The inspector noted one potential performance deficiency (PD). Licensee staff compares simulator transient test data to UFSAR accident analysis data, which appears to be inconsistent with the methodology described in Appendix A and B of ANSI/ANS 3.5-1985. Regulatory Guide 1.149, Revision 1, states that Appendix A and B should be considered integral parts of the Standard. The licensee has committed to this ANSI standard and Regulatory Guide, without exception.

The apparent PD is more than minor because it may affect the ability of Unit 2 simulator transient tests to detect replication problems. Also, it is more than minor because it affects the Human Performance (Human Error) attribute of the Initiating Events and Mitigating Systems cornerstones and the Procedure Adherence attribute for the Barrier Integrity cornerstone. Furthermore, the PD is an important issue because it could mask problems that under certain conditions (e.g., operator conditioning, procedure non-adherence) could lead to a safety significant operational event.

Licensee representatives reported that their best-estimate data for simulated transients are, in some cases, actual plant data, and in other cases, data derived from UFSAR accident and transient analysis. Furthermore, they stated the best-estimate data has been maintained by comparison of year-to-year data, including analysis and resolution of year to year deviations. Although the facility performs comparisons of year to year data, the inspector questioned the validity of using UFSAR data as the original baseline engineering evaluation for the two reasons noted earlier: 1) Assumptions used in the three plant-specific simulator tests reviewed were not always the same as the assumptions used in the UFSAR; and 2) Key plant parameters that should be trended in accordance with the ANSI standard are not always the same parameters trended in the UFSAR accident data.

The issue of whether this PD is a finding, as well as its appropriate significance is unresolved pending further review by NRC staff. (URI 05000412/2003005-01: Acceptability of licensee's simulator testing methodology.)

Enforcement. 10 CFR 55.46(c)(1) requires, in part, that the simulator must demonstrate expected plant response to transient and accident conditions. Although the facility staff appears to inappropriately compare simulator test data to UFSAR data, the inspectors did not identify any examples where the simulator had failed to demonstrate expected plant response to transient and accident conditions. Consequently, no violation of regulatory requirements was identified to date. However, this determination could change pending resolution of the adequacy of how the licensee conducts simulator testing.

1R12 Maintenance Rule Implementation (71111.12 - 3 samples)

a. Inspection Scope

The inspectors evaluated Maintenance Rule (MR) implementation for the three issues listed below. The inspector evaluated specific attributes, such as, MR scoping, characterization of failed structures, systems, and components (SSCs), MR risk categorization of SSCs, SSC performance criteria or goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," and 1/2-ADM-2114, "Maintenance Rule (MR) Program Administration," Revision 0. For selected systems, the inspectors evaluated whether system performance was properly dispositioned for MR category (a)(1) or (a)(2) performance monitoring. MR System Basis Documents were also reviewed, as appropriate during the review. The following conditions were evaluated:

- CR-03-10432 2SWS-MOV102C failure to stroke electrically
- CR-03-11187 Maintenance Rule criteria exceeded requires (a)(1) evaluation for the Unit 1 System 30, River Water
- CR-03-08668 2CHS-P21B high gearbox vibrations

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the scheduling and control of seven maintenance activities to evaluate the effect on overall plant risk. This review was against criteria contained in NOP-OP-1005, Rev. 5, “Shutdown Safety;” 1/2-ADM-2033, Rev. 2, “Risk Management Program;” NOP-WM-2001, Rev. 2, “Work Management Process;” 1/2-ADM-0804, Rev. 3, “On-Line Work Management and Risk Assessment;” 1/2-ADM-2114, Rev. 0, “Maintenance Rule (MR) Program Administrative Procedure;” and Conduct of Operations Procedure 1/2OM-48.1.I, Rev. 13, “Technical Specification Compliance.” The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following conditions:

- Emergent work activity associated with the Unit 1 No. 2 EDG on October 13, 2003. Specifically, the inspector evaluated the unexpected trip of feeder breaker, 9P7, associated with the E-8 motor control center (MCC). This MCC supplies the No. 2 EDG auxiliaries and rendered the No. 2 EDG inoperable and unavailable for 7.17 hours.

- The failure of Unit 2 Loop ‘A’ hot leg resistance temperature detector (RTD) 2RCS-TE412B1, which provides input into the reactor protection system, that occurred on October 24, 2003.

- Planned maintenance associated with 2SIS-MOV841, a Unit 2 hot leg injection isolation valve, conducted on November 24, 2003.

- Planned Unit 2 surveillance test 2OST-13.1, “Quench Spray Pump [2QSS*P21A] Test, conducted on November 28, 2003.

- Emergent analog rod position indication discrepancy with demand position related to Shutdown Bank Rod N-9, and subsequent entry into Abnormal Operating Procedure 1.1.7, “Rod Position Indication Malfunction,” that occurred on December 8, 2003.

- Planned Yellow Plant risk during Unit 1 ‘A’ Train Solid State Protection System testing, conducted on December 18, 2003.

- Emergent failure associated with the Unit 2 No. 1 EDG fuel rack return spring that occurred on December 23, 2003.

b. Findings

Introduction. The inspectors identified a self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, for the failure to take adequate and timely corrective actions associated with the spurious trip of Unit 1 480V circuit breaker 9P7. This finding was determined to be of very low safety significance (Green) because the loss of safety function occurred for less than the technical specification allowed outage time.

Description. At 0625, on October 13, 2003, Unit 1 experienced a spurious trip of 480V breaker, 9P7, which supplies power to the vital E-8 MCC, and ultimately to the No. 2 EDG auxiliary loads. The licensee subsequently declared the No. 2 EDG inoperable and Technical Specification 3.8.1 was entered. Subsequent troubleshooting revealed that a nonsafety-related room heater powered from the E-8 bus had experienced a ground and resulted in the spurious trip of the 9P7 feeder breaker. No other faults were detected, the heater was removed from service, the E-8 bus was reenergized, and the No. 2 EDG was declared operable at 1335, totaling seven hours and 10 minutes of unavailability.

Follow-up discussions with the licensee revealed that a fault on a single phase should not have caused a ground fault condition, as evidenced by the lack of ground fault flags that were noted for the 9P7 breaker following the breaker trip. Inspectors' review of the breaker coordination curves associated with the 9P7 breaker and the room heater breaker indicated no coordination issues.

The 9P7 breaker is equipped with a General Electric (GE) RMS-9 trip unit. NRC Information Notice (IN) 93-75, "Spurious Tripping of Low-Voltage Power Circuit Breakers with GE RMS-9 Digital Trip Units," described a potential for the loss of safety-related buses or individual loads because of spurious tripping of low-voltage circuit breakers fitted with GE RMS-9 digital trip units. GE's testing evidence suggested that ungrounded systems were the most susceptible to the spurious tripping problem. The licensee's initial response to the IN in 1994 was that no problems with these style breakers had been observed and that they were awaiting more information from GE. However, in September 1997, Beaver Valley experienced two separate spurious trips of breakers equipped with the RMS-9 style breakers, failures that were similar to the most recent failure. In all cases, further testing could not duplicate the failures, and based on an existing PM program for these trip units, no further action was taken.

Analysis. The issue was determined to be a performance deficiency because the licensee had previous opportunities to address and correct the spurious tripping issue due to failures that occurred in 1997, as well as with the information contained in the NRC IN 93-75. The licensee is currently planning to either replace the entire population of affected breakers (40) or, at a minimum, replace the faulty RMS-9 trip units that were the subject of the previously mentioned IN. The inspectors determined that the issue was more than minor because it affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of an EDG, as well as the safety-related 480V electrical system to respond to initiating events and prevent undesirable consequences (i.e., core damage).

Subsequently, the inspectors determined the finding was of very low safety significance (Green) through performance of a Phase 1 SDP in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Specifically, the inspectors determined that: the finding affected the Mitigating Systems cornerstone; did not represent an actual loss of a safety function of a system; did not represent a loss safety function of a single train for greater than its allowable outage time; did not represent a loss of risk significant non-technical specification systems; and did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. This finding was related to the Problem Identification and Resolution Cross-Cutting area, in that inadequate problem resolution led to the recurrence of spurious trips of safety-related breakers.

Enforcement. 10 CFR50 Appendix B, Criterion XVI, “Corrective Action,” requires that significant conditions adverse to quality are determined and corrective actions shall be taken to preclude repetition. Contrary to these requirements, the licensee failed to prevent recurrence of spurious tripping of breakers containing RMS-9 trip units, given the recent breaker trip that affected the operability and availability of the #2 emergency diesel generator at Unit 1, previous failures of a similar nature experienced in 1997, and the information provided in NRC Information Notice 93-75. Because the finding was determined to be of very low safety significance and was entered into the licensee’s corrective action program as CR 03-10888, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-334/03-05-02: Failure to perform adequate and timely corrective actions for spurious 480 volt circuit breaker trips).

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

a. Inspection Scope

The inspectors reviewed human performance during the following two non-routine plant evolutions to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety.

- The inspectors evaluated the licensee’s response to a high water level condition on the Ohio River. On November 19, 2003, the licensee entered Abnormal Operating Procedure (AOP) 1/2OM-53.C4A.75.2, “Acts of Nature - Flood,” Rev. 18. The inspectors monitored the licensee’s actions as directed by this AOP, walked down potential flood-affected areas, and interviewed operators to assess the ability of the site to cope with the elevated river levels. The river ultimately crested at 680.5 feet and receded below 670 feet on November 25; at that point, the flooding AOP was exited.

- The inspectors evaluated the licensee’s response to a Unit 2 Reactor Trip that occurred on October 14, 2003, due to low steam generator water level resulting from a circuit card failure associated with the ‘B’ feedwater flow control loop. The inspectors reviewed sequence of events logs, shift narrative logs, and instrument traces; interviewed control room personnel; reviewed abnormal and emergency operating procedures to ensure appropriate actions were taken; and reviewed findings from the Event Review Team.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 samples)

a. Inspection Scope

The inspectors reviewed the following six conditions to determine whether proper operability justifications were performed. In addition, where applicable, the inspectors verified that TS limiting conditions for operation (LCO) requirements were properly addressed.

- On October 18, 2003, the 480 volt feeder breaker, 9P7, associated with the E-8 MCC spuriously tripped. This caused a loss of power to the MCC that powers various EDG loads. The breaker was ultimately reset and the EDG was returned to operable status. A Basis for Continued Operation (BCO) 1-03-008 was prepared to address the impact of spurious operation on the 480 volt emergency power distribution system. The BCO concluded no impact on probabilistic risk assessment (PRA) core damage frequency and large early release fraction since the breaker failure rate was calculated to be 7.13E-7, consistent with the PRA model's value of 7.95E-7.

- The inspectors reviewed BCO 2-03-004, which described four process measurement factors that potentially affected uncertainty measurements associated with setpoints for steam generator water level, which are inputs into the reactor protection system.

- The inspectors reviewed an operability determination associated with CR 03-10864, which described potential two-phase flow considerations that were not addressed in calculation 8700-DMC-3157. Specifically, the assumed temperature utilized in the calculation for recirculation spray heat exchanger river water outlet temperature was 120°F, versus the expected 175°F. With the use of the higher, more appropriate outlet temperature, coupled with the potential for sub-atmospheric pressures in the downstream river water piping, two-phase river water flow may result, leading to increased stress on piping and supports and challenge system functions during accident conditions.

- The inspectors reviewed CR-03-10279, which described the October 1, 2003, identification of four holes, approximately 3 square inches in total area, located on the containment sump cover. The inspectors evaluated the licensee's analysis of the condition, since the potential for debris to bypass containment sump screens could have an adverse impact on the ability of safety-related components from performing safety functions during design basis accidents. The inspector evaluated the adequacy of the licensee's conclusion that the sump remained operable, which included that hole locations on the sump cover were confined to one-half of the sump, thereby reducing the potential impact to only one train of safety-related equipment during post-accident conditions.

Additionally, the inspectors reviewed licensee analyses that assessed, in part: (1) risk significance, (2) containment flood-up and debris transport considerations, and (3) inherent design characteristics of the containment sump, safety-related pumps and equipment utilized during design basis accidents to mitigate the impact of any debris that would potentially bypass the screens.

- The inspectors reviewed the licensee's operability assessment following a September 30, 2003, 10CFR21 notification from Framatome regarding a defect affecting Cutler-Hammer thermal overload heaters installed in Unit 2 MCC cubicles. The inspector reviewed condition report 03-10263, an assessment of the population of affected breakers, the circumstances surrounding the defect, and the licensee's conclusion regarding the operability of these breakers given the potential defect in the thermal overload heater portion of the cubicles.

- The inspectors reviewed CRs 03-12232 and 03-12509, which details the identification and subsequent actions involving minor leakage past power-operated relief valve (PORV) 2RCS-455D. The inspector evaluated the licensee's assessment of continued operability with leak-by of approximately 0.05 gpm, as well as operability and compliance with TSs following the isolation of the PORV through closure of its associated block valve, 2RCS-MOV537.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17 - 1 sample)

a. Inspection Scope

The inspectors evaluated the adequacy and design basis impact of a permanent modification installed following the identification of bypass holes on the Unit 2 containment sump (See Section 1R15). The modification, implemented via Order 200062089, was performed in accordance with Engineering Change Package 03-0523, and installed plates to cover potential bypass holes identified in the sump cover.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 3 samples)

a. Inspection Scope

The inspectors reviewed and/or observed three post-maintenance tests (PMTs) to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component;

and 3) the PMT was performed in accordance with procedures. The following PMTs were observed:

- Operational surveillance test 2OST-7.6, Rev. 22, "Centrifugal Charging Pump [2CHS*P21C]," performed on October 23, 2003, following the replacement of the rotating assembly.

- Partial valve stroke surveillance in accordance with 2OST-21.7, Rev. 9, "Main Steam Trip Valves [2MSS*AOV101A, B and C]," performed on October 11, 2003, following an overhaul of 2MSS*AOV101C.

- Valve stroke surveillance performed in accordance with 2OST-47.3M, Rev. 3, "Containment Penetration and ASME Section XI Valve Test - Work Week 8," performed on November 24, 2003, following breaker pan replacement for 2SIS-MOV841.

b. Findings

No findings of significance were identified.

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

1. Outage Activities

a. Inspection Scope

The inspectors observed selected refueling, outage maintenance, and reactor startup activities to determine whether shutdown safety functions (e.g., reactor decay heat removal and reactivity control) were properly maintained as required by technical specifications, and in accordance with 1/2-ADM-1800, "Shutdown Safety," Rev. 0. The inspectors evaluated specific performance attributes, including configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors closely evaluated risk management activities and plant configuration and control during periods of reduced reactor coolant system inventory due to the associated increase in shutdown risk. Other specific activities evaluated included:

- 2RP-3.16, Rev. 1 Refueling Procedure Core Unload
- 1/2RP-3.23, Rev. 3 Refueling Procedure Core Reload
- 2OST-36.3, Rev. 14 Emergency Diesel Generator [2EGS*EG2-1] Automatic Test
- 2OST-36.4, Rev. 14 Emergency Diesel Generator [2EGS*EG2-2] Automatic Test
- 2OM-20.4.H, Rev. 11 Draining the Refueling Cavity to the RWST
- 2OM-50.4.D, Rev. 36 Reactor Startup from Mode 3 to Mode 2
- 2RST-2.1, Rev. 6 Initial Approach to Criticality after Refueling

- 2OM-52.4.A, Rev. 42 Increasing Power from 5% Reactor Power and Turbine on Turning Gear to Full Load Operation

Additionally, the inspectors reviewed the station's commitments to NRC Generic Letter 88-17, "Loss of Decay Heat Removal," contained in the following procedures: 1) 2OST-6.11, Rev. 1, "Prerequisites for Entering A Reduced RCS Inventory or Midloop Condition;" 2) 2OM-6.4.U, Rev. 8, "Draining the RCS to Reduced Inventory or Midloop Condition," and 3) 2OM-6.4.V, Rev. 1, "Reduced RCS Inventory Operation Checklist." The inspectors observed the 2R10 RCS draindown and verified that the reduced RCS inventory level as defined in GL 88-17 was not reached.

b. Findings

No findings of significance were identified.

2. Spent Fuel Pool Upender Cable Clamp Failure

The inspectors evaluated the licensee's response to a cable clamp failure that occurred on the Unit 2 spent fuel pool (SFP) upender on September 28, 2003. The inspectors reviewed the root cause report, evaluated the adequacy of short term corrective actions, and verified appropriate measures were implemented to prevent recurrence.

a. Inspection Scope

A self-revealing violation of 10 CFR 50, Appendix B, Criterion V, was identified for failure to have adequate work instructions that led to the failure of a cable clamp associated with the Unit 2 SFP upender frame.

b. Findings

Introduction

Prior to the Unit 2 refueling outage, identified wear on the SFP upender cable necessitated replacement of the cable. During the outage on September 28, 2003, after the successful transfer of 157 spent fuel assemblies from the reactor vessel to the SFP, and the subsequent transfer of 40 assemblies from the SFP to the vessel, a cable clamp attached to the SFP upender frame loosened and resulted in the upender falling back to the horizontal position.

The licensee formed an Event Review Team to evaluate the cause of the event, initiated CR 03-10148 to capture the issue in the corrective action program, and performed a root cause evaluation to determine the root and contributing causes of the event. The licensee determined that inadequate work management practices, inadequate work instructions, and lack of contractor oversight resulted in the improper installation of the cable clamp and ultimately, the failure of the clamp. Specifically, inadequate work order instructions led to a contractor installing 1/2 inch Flexloc nuts versus the required 7/16 inch nuts that would appropriately secure to the 7/16 inch U-bolt of the Crosby clamp.

Analysis

The inspectors determined that this finding was a performance deficiency, in that the failure to have oversight of contractors and inadequate work instructions was reasonably within its ability to foresee and detect and could have been prevented. This issue was determined to be more than minor, because if left uncorrected, could become a more significant safety concern involving the potential damage to fuel assemblies. Because this issue involves fuel assembly handling and storage issues, it is not suitable for evaluation under the NRC Significance Determination Process. Therefore, this finding was reviewed by NRC management and determined to be of low safety significance (Green) because the event did not result in damage to a fuel assembly, and occurred while the upender was empty. This finding involved the cross-cutting area of Human Performance due to the inadequate work instructions and installation of an incorrectly-sized retaining nut.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings," requires in part, that activities affecting quality shall be prescribed by documented instructions, and shall be accomplished in accordance with these instructions. Contrary to this requirement, the licensee did not ensure that the replacement of the Unit 2 spent fuel pool upender cable clamp was performed in accordance with appropriate instructions, and resulted in the failure of the cable clamp.

This violation is considered to be of very low safety significance (Green) and is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. This issue is in the licensee's corrective action program as CR-03-10148. (NCV 05000412/2003005-03; Inadequate Work Instructions Results in Spent Fuel Pool Upender Cable Clamp Failure)

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors observed and/or reviewed the following five OSTs and MSPs. This review was conducted to verify that the equipment or systems were capable of performing their intended safety functions and to ensure compliance with related TS, UFSAR, and that procedural requirements:

- 2OST-11.14B, Rev. 17 — HHSI Full Flow Test
- 1OST-11.1, Rev. 16 — Safety Injection Pump Test - [1SI-P-1A]
- 1OST-36.1, Rev. 43 — Diesel Generator No. 1 Monthly Test
- 1OST-24.3, Rev. 24 — Motor Driven Auxiliary Feedwater Pump Test [1FW-P-3B]

- 2OST-7.5, Rev. 25 Centrifugal Charging Pump [2CHS*P21B]

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 3 samples)

a. Inspection Scope

The inspectors reviewed three temporary modifications (TMs) and associated implementing documents to verify that plant design bases and the system or component operability were maintained. Nuclear Power Division Administrative Procedure 7.4, Rev. 8, “Temporary Modifications,” specified requirements for development and installation of TMs. The inspectors reviewed TMs associated with the following items:

- The inspectors reviewed the Unit 2 TM 2-03-10, Rev. 0, “Service Water Header Bypass to Chillers.” This TM installed an alternate polyethene piping line on the non-safety related portion of the Unit 2 service water system that supplies the chilled water condensers. The primary reason for the installation of this alternate line was to ensure a reliable source of water should the normal supply line, which is buried pipe, needed to be isolated for extended periods of time to affect repairs due to leaks.

- The inspectors reviewed the Unit 2 TM 2-03-12, Rev. 0, “Processing Equipment for RCS Loop ‘A’ Hot Leg Temperature #1 Defeat Function Due to Failed RTD 2RCS-TE412B1 and Rescale T-Hot Average Summator, 2RCS-TU412U.” Due to a recent failure of the ‘A’ channel reactor coolant system (RCS) hot leg RTD, one of three RTDs that provides input to various protection and control schemes, the licensee bypassed the failed RTD, removing its signal from the averaging circuit. With appropriate bias applied, the inspectors verified that this condition was consistent with requirements contained in TSs. TS Table 3.3-1 of TS 3.3.1.1 details that an operable channel includes two operable RTDs with the third RTD inoperable, provided the failed RTD is disconnected and the proper bias applied. This TM documents this effort and references the Westinghouse analysis WCAP-12478, which supports this process.

- The inspectors reviewed the Unit 2 TM 2-03-16, Rev. 0, “MSS Isolation Valve for Temporary Gauge Utilized in 2BVT-11.26.06,” Rev. 0. This TM documents the addition of a 3/8" Whitey valve in an instrument line off the main steam system. The new valve is located downstream of an existing isolation valve, TMS-275, that exhibited seat leakage that necessitated the modification. A previous design change had a pressure instrument, PI-215, removed and the existing line capped. The new 3/8" tubing will also be capped downstream of the newly added valve. This section of piping/tubing is only used for the purpose of gathering data during thermal testing of the high pressure turbine.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed an annual simulator evaluation, (See Section 1R11) and evaluated operator performance regarding event classification and notification. The simulator evaluation involved multiple safety-related component failures and plant conditions warranting simulated Alert and Site Area Emergency event declarations. The licensee counted this evolution toward Emergency Preparedness Drill/Exercise Performance (DEP) Indicators, therefore the inspectors reviewed the classifications to determine whether they were appropriately credited. Additionally, the inspectors verified the DEP performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, Rev. 2, "Regulatory Assessment Performance Indicator Guideline." Other documents utilized in this inspection include the following:

- 1/2-ADM-1111, Rev. 1 — NRC EPP Performance Indicator Instructions
- EPP-IP-1.1, Rev. 31 — Notifications
- EPP/I-1a/b, Rev. 8 — Recognition and Classification of Emergency Conditions

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 3 samples)

a. Inspection Scope

The inspector reviewed radiological work activities, practices, and procedural implementation during observations and tours of the facilities. The review included procedures, records, and other program documents to evaluate the effectiveness of the access controls to radiologically significant areas. This inspection activity represents the completion of three (3) samples relative to this inspection area (i.e., inspection procedure sections 02.05.a, b, and c).

On October 23, the inspector toured various locations within the Unit 1 and Unit 2 radiologically-controlled areas (RCA), which included auxiliary and fuel handling buildings of Unit 1, the auxiliary, fuel handling, decontamination, and waste handling buildings of Unit 2, and the common health physics access control point. At the common control point, the inspector observed RCA entry and exit practices for various radiation workers. The inspector evaluated the use of personnel dosimetry and the radiological briefings for ingoing radiation workers. During these walkdowns, the inspector also observed and verified the appropriateness of the posting, labeling, and barricading of radioactive material, radiation, contamination, high radiation, and locked high radiation areas. The inspector observed work activities by both radiation workers and radiation protection technicians for compliance with the radiation work permit (RWP) requirements and radiological protection procedures. The inspector reviewed work activities and/or work documentation related to RWPs 203-2210 and 303-3302. (See also the List of Documents Reviewed section).

On October 21, the inspector witnessed a pre-job briefing for a containment entry, while at 100 percent power, to inspect and troubleshoot a containment sump pump. The inspector verified that the briefing covered the appropriate industrial and radiological safety issues and controls involved with the work, as well as the scope of work activities.

On October 20 and 22, the inspector met with a radiation protection supervisor and a senior radiation protection specialist. The discussion topics included high radiation areas and very high radiation areas and their associated controls, as well as areas that have the potential to become such during certain plant operations.

The inspector also reviewed selected documents (as listed in the List of Documents Reviewed section) to evaluate the adequacy of radiological controls against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts B, C, D, F through J, L, and M), site Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 1 sample)

a. Inspection Scope

The inspector reviewed the effectiveness of the licensee’s program to maintain occupational radiation exposure as low as reasonably achievable (ALARA). This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.02.c).

The inspector reviewed the cumulative collective exposure for the recently completed Unit 2 refueling outage (2R10). The inspector also reviewed the 2R10 Daily RWP Exposure Summaries for September 25 and October 21, 2003, and a draft ALARA report for 2R10. Based on discussions with the senior ALARA specialist, and the documents discussed previously, the inspector compared the actual dose results (i.e., dose rate reductions, person-rem used) with the projected dose listed in the licensee's ALARA radiological work planning for these work activities. The inspector discussed any differences between the intended and actual work activity doses with the senior ALARA specialist.

The inspector reviewed selected documents (as listed in the List of Documents Reviewed section) for regulatory compliance and for adequacy of control of radiation exposure, against criteria contained in 10 CFR 20.1101, "Radiation protection programs," 10 CFR 20.1701, "Use of process or other engineering controls," and site procedures.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 1 sample)

a. Inspection Scope

The inspector reviewed the program for installed radiation monitoring instrumentation to determine the accuracy and operability of the instrumentation. This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.03).

During the plant tour described in Section 2OS1, the inspector verified the overall condition, operability, and calibration status of selected, installed area and process radiation monitors, and any accessible local indication information for those monitors. The inspector also subsequently reviewed associated calibration documents for various instruments evaluated during the inspection. (See also the List of Documents Reviewed section).

The inspector reviewed selected documents (as listed in the List of Documents Reviewed section) for regulatory compliance and adequacy, against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

1. RETS/ODCM Radiological Effluent Occurrences (RETS/ODCM) (1 sample)

a. Inspection Scope

On October 21, the inspector selectively examined records involving Radiological Effluent Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM) radiological effluent occurrences. This examination covered the time period from the second quarter of 2002 through the third quarter of 2003. This review was conducted against the applicable criteria specified in Nuclear Energy Institute's (NEI) Regulatory Assessment Performance Indicator Guideline No. 99-02, Revision 2. The inspector verified that all conditions that met the NEI criteria and impacted the Performance Indicators were identified by the licensee. The inspector performed a selective examination of records (See List of Documents Reviewed section) for regulatory compliance and adequacy. The inspector also discussed the reviewed records with the two radiation specialists responsible for the effluent dose program.

b. Findings

No findings of significance were identified.

2. RCS Identified Leak Rate (2 samples)

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 Performance Indicators (PI) for unidentified RCS leak rate for the period December 2002 through November 2003. The accuracy of reported data was verified by reviewing selected monthly operating reports, shift operating logs, Licensee Event Reports (LERs), and surveillance tests. The inspectors reviewed detailed records to determine whether the reported RCS leak rate data was consistent with NRC approved guidance, provided in NEI 99-02, Rev. 2, "Regulatory Assessment PI Guideline."

b. Findings

No findings of significance were identified.

3. RCS Specific Activity (2 samples)

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 PIs for RCS specific activity, for the period December 2002 through November 2003. The accuracy of reported data was verified by reviewing the results from TS sampling, other chemistry samples of the RCS, and supporting calculations and calculation methodology. The inspectors verified the reported RCS specific activity data was consistent with NRC approved guidance provided in NEI 99-02, Rev. 2, “Regulatory Assessment PI Guideline.”

b. Findings

No findings of significance were identified.

4. Residual Heat Removal Safety System Unavailability (2 samples)

a. Inspection Scope

The inspector reviewed the Unit 1 and 2 performance indicators for the systems that provide post-accident recirculation and shutdown cooling functions. The specific systems reviewed included the Unit 1 low head safety injection, recirculation spray, residual heat removal systems and the Unit 2 recirculation spray and residual heat removal systems. Due to the plant specific design, Nuclear Energy Institute 99-02, Rev. 2, Appendix D, "Plant Specific Design Issues," was used to determine the scope of the data collected. The inspector verified the accuracy of the reported data for the time period of December 2002 through November 2003, through reviews of shift narrative log entries. In addition, the following documents were reviewed to evaluate the adequacy of the licensee’s determination of availability:

- BVBP-RAS-0005, Rev. 7 NRC Performance Indicators
- 2OST-30.20A, Rev. 2 Train A RSS HXs And SWS Supply Header Dry Layup Check
- 1/2OM-48.1.I, Rev. 13 Technical Specification Compliance

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Annual Sample Review (1 sample)

a. Inspection Scope and Observations

The inspectors selected one CR for detailed review, CR 02-04594. The CR details a concern regarding the general health of relief valves at BVPS. The CR requested an evaluation of the relief valves in each system at Unit 1 and 2, to determine if all relief valves are being adequately tested to ensure they continue to perform their design function.

The CR documents evaluations of each relief valve on all systems, including whether or not the valve was safety related, supported a safety system or function, was located in the vicinity of safety-related equipment, and was subject to the current testing methodology. The evaluation identified a total of 1163 relief valves at BVPS (530 on Unit 1 and 633 on Unit 2). The evaluation concluded that all relief valves required to be tested in accordance with the inservice testing (IST) program were identified and being properly tested.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope and Observations

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

b. Findings

No findings of significance were identified.

3. Daily Condition Report Review

a. Inspection Scope and Observations

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

b. Findings

No findings of significance were identified.

4. Occupational Radiation Safety

a. Inspection Scope

The inspector selected eight issues identified in the Corrective Action Program (CAP) for detailed review (Condition Report Nos. 02-05579, 02-07135, 03-09460, 03-09659, 03-

09865, 03-10019, 03-10553, and 03-10778). The issues were associated with the following: an investigation of methods for reducing gaseous tritium effluent; a review of the viability of the site radiation monitoring program; an accounting for a Unit 2 reactor building key; a collective significance review of condition reports in the radiation protection area generated during 2R10; ALARA lessons-learned from the 2R10 refueling outage; results from radiation badges processed by vendors showing exposure of unknown origin; respiratory protection ALARA evaluations; and an evaluation of the need for a liquid radioactive waste discharge authorization associated with the activation of a Terry turbine steam-driven auxiliary feed pump. The documented reports for the issues were reviewed to ensure that the full extent of the issues were identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized.

b. Findings

No findings of significance were identified.

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

Section 1R13 describes a finding that involved inadequate problem resolution for failure to correct a condition adverse to quality that should have been identified by the licensee based on internal and external operating experience and NRC Generic Communications.

4OA3 Event Follow-up

1. Unit 1 Reactor Trip Caused by Procedural Error

a. Inspection Scope

On November 13, 2003, at 11:35 a.m., Unit 1 automatically tripped from 100 percent power. A procedural error caused a turbine trip, which resulted in an automatic reactor trip (CR 03-11523). The inspectors evaluated the response of plant equipment and mitigating systems following the trip, as well as operator actions including communications, use of emergency operating procedures, and their efforts to stabilize the plant to a safe shutdown condition. The inspectors also reviewed various instrument responses and sequence of events reports, and interviewed various plant personnel. The inspectors verified the reactor trip was properly reported (Event Notification #40320) in accordance with 10 CFR 50.72.

The inspectors assessed the licensee's restart preparation to determine whether station personnel properly evaluated plant readiness for safe restart in accordance with 1/2-ADM-0703, Rev. 0, "Event Review." The Event Review Team (ERT) concluded that the reactor trip was caused by the failure of plant personnel to utilize self-checking and peer checks that led to a failure to follow a surveillance procedure, and subsequently resulted in the unit trip. The inspectors verified that the licensee's decision to restart was consistent with the measures taken and in accordance with attributes contained in 1/2-ADM-0703.

b. Findings

Introduction. A self-revealing finding was identified of very low safety significance (Green) regarding the failure to properly implement a maintenance surveillance procedure that resulted in an unplanned Unit 1 reactor trip. This finding represented a human performance error in the execution of the surveillance. This finding was of very low safety significance because of its short duration and the issue did not impact the operation of mitigating equipment.

Description. On November 13, with the plant at 100 percent power, technicians were performing Unit 1 reactor trip breaker testing in accordance with 1MSP-1.05-I, Rev. 20, “Solid State Protection System Train ‘B’ Bi-Monthly Test.” The procedure step immediately prior to the event involved the measurement of resistance across the reactor trip bypass breaker contact. While attempting to measure this resistance, the technicians mistakenly placed the digital voltmeter across the reactor trip breaker contact. Since the meter was set to the ohms scale, a current flow path across the normally open contact was created and completed a circuit which tripped the main turbine. The normal function of this circuit is to ensure a turbine trip following a reactor trip. Since reactor power was above 49 percent, an automatic reactor trip signal was generated.

Analysis. A surveillance test on the reactor trip switchgear was in progress at the time of the trip and the inspectors determined that failure to properly execute this surveillance procedure caused the event. The issue was determined to be a performance deficiency because the licensee failed to meet a procedural requirement and it was within the licensee’s ability to foresee and control. The issue was more than minor because it is associated with the human performance attribute under the Initiating Events cornerstone, and affected the cornerstone objective since it increased the likelihood of an initiating event, e.g., an actual reactor trip did occur.

The inspectors determined the finding was of very low safety significance (Green) through performance of a Phase 1 SDP in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” Specifically, the inspectors determined that the finding impacted the Initiating Events Cornerstone, did not contribute to the likelihood of a primary or secondary loss of coolant initiator, did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment or functions, and did not increase the likelihood of a fire or internal/external flood. This finding was related to the Human Performance Cross-Cutting area, in that human performance errors resulted in a reactor trip.

Enforcement. Technical Specification 6.8.1 requires that written procedures are properly implemented covering the activities referenced in Appendix “A” of Regulatory Guide 1.33, Rev. 2, February 1978. Appendix “A” of Regulatory Guide 1.33 specifies, in part, that written procedures are followed for operation of safety-related systems. Contrary to these requirements, technicians attempted to measure resistance across the wrong set of contacts as required by the plant procedure 1MSP-1.05-I. This issue was entered into the licensee’s corrective action program as CR 03-11523. This violation of TS 6.8.1 is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: (NCV 50-334/03-05-04: Reactor Trip due to Personnel Error during Solid State Protection System Testing).

2. (Closed) LER 05000334/2003003-00: Automatic Actuation of Emergency Diesel Generator Following Loss of Emergency Bus Offsite Source

   This event was discussed in NRC Inspection Report No. 50-334/412/03-02. No new issues were revealed by the LER. This LER is closed.

3. (Closed) LER 05000334/2003004-00: Safety Related River Water Pump Not Declared Inoperable When Vibration Levels Exceeded Allowable Limit

   This event was discussed in NRC Inspection Report No. 50-334/412/03-02. No new issues were revealed by the LER. This LER is closed.

4. (Closed) Licensee Event Report (LER) 50-334/2003-005-00: Non-Conservative Reactor Coolant System Low Flow Reactor Trip Setpoint.

a. Inspection Scope

This LER described a situation where one of the nine (9) channels of RCS flow (i.e., three per loop) was non-conservative with respect to TS requirements. The inspector performed an in-office review of the LER and FENOC's related corrective actions to verify that the event was accurately reported and to assess FENOC's causal assessment and corrective actions. This LER is closed.

b. Findings

Introduction. FENOC determined that the root cause of the non-conservative setpoint was an inadequate process to verify RCS flow setpoints were within TS limits prior to reaching ten-percent reactor power. The safety significance of this event was very low because the setpoint error was small and all other RCS flow instruments remained operable.

Description. The non-conservative setpoint change occurred during the refueling outage in April 2003, largely because one of eight (8) RCS flow transmitters replaced during that outage provided a small, unanticipated increase in delta pressure (i.e., ~1% higher), resulting in a slightly higher indicated RCS flow. Since the RCS low flow setpoint is a relative, normalized setpoint and the increased RCS flow indication resulted in a decreased RCS flow setpoint, the low flow setpoint generated (i.e., 89.2% actual versus 89.8% per TSs) was inadequate for this channel. The error was not noted by FENOC engineering until a post-outage review of an RCS flow test in July 2003; it was promptly corrected at that time.

Analysis. The inspectors determined the safety significance of this finding was very low (Green) using Phase One of the NRC Significance Determination Process. Specifically, the RCS low flow setpoint deviation was small (<1% low) and all eight other flow instrument setpoints remained operable. This licensee-identified finding screened to Green because the error did not result in a loss of any RPS safety function based on FENOC engineering analysis. (The Beaver Valley Unit 1 PRA does not explicitly model these instruments; therefore, the slightly low setting on one channel has no direct impact on the PRA core damage frequency.)

Enforcement. The inspectors determined that the event involved a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," in that procedures did not require verification of the RCS low flow setpoint prior to exceeding ten-percent power. This licensee identified violation is documented in section 4OA7 of this report.

4OA4 Crosscutting Issues

Human Performance Problems

Section 1R20 describes a finding that involved the cross-cutting area of Human Performance, due to inadequate work instructions and installation of an incorrectly-sized retaining nut.

Section 40A3 describes a finding that was related to the Human Performance Cross-Cutting area, in that human performance errors resulted in a reactor trip.

4OA5 Other

1. NRC Temporary Instruction (TI) 2515/153, Reactor Containment Sump Blockage (NRC Bulletin 2003-01)

a. Inspection Scope

NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated June 9, 2003, requested licensee's to either (1) perform a plant-specific evaluation to confirm compliance with 10 CFR 50.46(b)(5), or (2) implement interim compensatory measures to reduce the potential risk due to post-accident containment sump debris blockage pending completion of a plant specific evaluation. FENOC chose the second option and described their interim compensatory measures in their response to NRC Bulletin 2003-01, dated August 8, 2003. The inspectors interviewed station personnel, reviewed records, and inspected various areas in containment to verify the implementation of compensatory measures as committed to in their bulletin response. Additionally, the inspectors reviewed the adequacy and results of surveillance, 2BVT 2.47.11, Rev. 0, "Containment Walkdown For Potential Sump Screen Debris Sources," and 2MSP-9.04-M, Rev. 2, "Containment Sump (2DAS-TK204) Inspection."

b. Findings

In accordance with instructions contained within TI-2515/153, the inspector reviewed the following:

- During performance of 2BVT 2.47.11 listed above, which was conducted during the Unit 2 refueling outage in the current inspection period, no debris of significance was identified. However, the licensee identified holes in the containment sump cover (See Section 1R15) and completed a modification that removed the bypass potential.

- The inspectors performed an inspection of the Unit 2 containment sump to verify sump screen integrity, determine if potential sump screen bypass flowpaths existed, evaluated overall cleanliness, and assessed whether sump screen mesh size was consistent with the current design basis. No issues were identified.

- The Unit 1 containment sump was inspected, in accordance with 1MSP-9.04M, during the Unit 1 refueling outage conducted between 3/8/03 and 4/29/03. In addition, inspectors evaluated the sump and no significant issues were identified as detailed in NRC Inspection Report 05000334/2003003, dated 7/29/03.

- The Unit 1 containment was inspected for debris prior to final closeout toward the end of the refueling outage conducted between 3/8/03 and 4/29/03, in accordance with 1OST-47.2, "Containment Integrity Verification." In addition, inspectors evaluated the containment for potential debris sources that could affect long-term decay heat removal; no significant issues were identified as detailed in NRC Inspection Report 05000334/2003003, dated 7/29/03.

4OA6 Management Meetings

1. Exit Meeting Summary

The inspectors presented the inspection results to Mr. William Pearce and members of licensee management following the conclusion of the inspection on January 22, 2003. The licensee acknowledged the findings presented.

Additionally, inspectors from Division of Reactor Safety, Region 1, performed interim exits on October 31, 2003, regarding the results of the biennial licensed operator requalification inspection, and on October 23, 2003, regarding an occupational radiation safety inspection.

The licensee did not indicate that any of the information presented at the exit meeting was proprietary.

2. Site Management Visit

From December 14 - 16, 2003, Mr. Peter Eselgroth, Chief, Reactor Projects Branch 7, toured Beaver Valley Power Station and met with station personnel to review plant performance.

4OA7 Licensee-Identified Violations

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

- 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” requires in part that procedures shall include appropriate quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, calibration and engineering test procedures involving RCS flow did not contain requirements to ensure correct low flow setpoints, resulting in non-compliance with Technical Specification 3.3.1, Reactor Trip System Instrumentation, Table 3.3-1. However, because this small setpoint error was isolated to only one of nine RCS flow instruments and did not result in a loss of any RPS safety function, this violation is of very low safety significance and is being treated as an NCV. (Section 4OA3). This event was entered into FENOC’s corrective action system as CR 03-08007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

A. Castagnacc, Supervisor RP Services-Rad Waste/Shipping/Environmental
T. Cosgrove, Director, Plant Engineering
J. Dobo, Senior RP Technician
R. Ferrie, Plant Engineer
L. Freeland, Manager, Nuclear Regulatory Affairs & Corrective Actions
R. Freund, Supervisor RP Services-Technical Support
D. Gallagher, RP Supervisor-Procedures
M. Helms, RP Specialist-RMS/DRMS
V. Kaminskas, Director, Maintenance
J. Lash, Plant General Manager
J. Lebda, Supervisor, Radiological Engineering and Health
A. Lonnett, RP Specialist-Effluents
R. Mende, Director, Work Management
R. Moore, RP Specialist-Effluents
W. Pearce, Vice President
P. Sena, Manager, Nuclear Operations
J. Sipp, Manager, Nuclear Radiation Protection, Rad Ops, Units 1 and 2
D. Weitz, Senior RP Specialist-RWP/ALARA

NRC Personnel

P. Cataldo, Senior Resident Inspector
G. Smith, Resident Inspector

LIST OF ITEMS, OPENED, CLOSED, AND DISCUSSED

Opened

50-412/03-05-01	URI	Acceptance of Licensee's Simulator Testing Methodology

Opened/Closed

05-334/03-05-02	NCV	Failure to Perform Adequate and Timely Corrective Actions for Spurious 480 volt Circuit Breaker Trips
05-334/03-05-03	NCV	Inadequate Work Instructions Results in Spent Fuel Pool Upender Cable Clamp Failure

05-334/03-05-04	NCV	Reactor Trip due to Personnel Error during Solid State Protection System Testing

Closed

50-334/03-03	LER	Automatic Actuation of Emergency Diesel Generator Following Loss of Emergency Bus Offsite Source (Section 4OA3)
50-334/03-04	LER	Safety Related River Water Pump Not Declared Inoperable When Vibration Levels Exceeded Allowable Limit (Section 4OA3)
50-334/03-05	LER	Non-Conservative Reactor Coolant System Low Flow Reactor Trip Setpoint (Section 40A3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignments

Drawings

-Unit 1 Operating Manual (OM) Figure Number 13-1, "Containment Depressurization System," Rev. 16
-Unit 1 OM Figure Number 13-2, "Containment Depressurization System," Rev. 8
-Unit 2 OM Figure Number 13-2, "QS System," Rev. 11
-Unit 2 OM Figure Number 7-1A, "Chemical and Volume Control, Sh. 1," Rev. 12

Procedures

-1OM-13.3.B.1, "Valve List - 1QS," Rev. 12
-1OM-13.3.B.2, "Valve List - 1RS," Rev. 7
-1OM-13.3.C, "Power Supply and Control Switch List," Rev. 4
-2OM-13.3.B.1, "2QSS Valve List," Rev. 7
-2OM-13.3.C, "Power Supply and Control Switch List," Rev. 7
-2OM-7.3.B.1, "Valve List - 2CHS," Rev. 15
-2OM-7.3.C, "Power Supply and Control Switch List," Rev. 13

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures

1/2OM-48.1.I, "Conduct of Operations," Rev 13

Documents

CR 03-12556
CR 03-10888
WO 200065050
WO 200018973
WO 200073382
Engineering Change Package 02-0837-18
Unit 1 and Unit 2 control room logs

Section 2OS1: Access Control to Radiologically Significant Areas (71121.01)

Documents

-RWP 203-2210, Rev. 0, Reactor building containment-leak search/troubleshooting
-RWP 303-3302, Rev. 1, NRC inspection/surveillance
-PCM alarm decision chart from procedure 1/2-HPP-4.04.023, Rev. 1, Eberline personnel contamination monitor (PCM-2)

Procedures

1/2-ADM-1630, Rev. 6, Radiation worker practices
1/2-ADM-1601, Rev. 8, Radiation protection standards
1/2-HPP-3.02.003, Rev. 3, Decontamination control
1/2-HPP-3.02.004, Rev. 2, Area Posting
1/2-HPP-3.07.013, Rev. 2, Barrier checks
1/2-HPP-4.04.023, Rev. 1, Eberline personnel contamination monitor (PCM-2)

Section 2OS2: ALARA Planning and Controls (71121.02)

Documents

-2R10 Daily RWP Exposure Summary through September 25, 2003
-2R10 Daily RWP Exposure Summary through October 21, 2003
-ALARA report for Unit 2's tenth refueling outage (2R10), Draft as of October 17, 2003
-ALARA committee meeting minutes for meeting 03-15 on September 3, 2003

Procedures

-1/2-HPP-3.08.011, Rev. 0, Respiratory protection ALARA evaluation

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

Documents

-DRMS detector check/calibration worksheet, 2RMR-RQ 303A (Unit 2), performed on October 17, 2003
-DRMS detector check/calibration worksheet, 2RMR-RQ 303B (Unit 2), performed on October 17, 2003
-Containment airborne radiation monitor calibration procedure 2MSP-43.19-I, 2RMR-RQI 303 (Unit 2), performed on May 31, 2002
-In-containment high range area radiation monitor calibration procedure 2MSP-43.40I, 2RMR-DAU 206 (Unit 2), performed on September 23, 2003
-Containment purge radiation monitor calibration procedure 2MSP-43.01-I, 2HVR-DAU 104A and 104B (Unit 2), performed on September 24 and 22, 2003, respectively
-Reactor coolant letdown high/low range radiation monitor calibration procedure 2MSP-43.16-I, 2CHS-RQI 101 (Unit 2), performed on May 9, 2003
-Fuel building fuel pool bridge area radiation monitor calibration procedure 1MSP-43.48-I, RM-RM 207 (Unit 1), performed on December 11, 2002 and on June 4, 2003
-CCW heat exchanger supply manifold radiation monitor calibration procedure 1MSP-43.30-I, RM-CC100 (Unit 1), performed on June 21, 2002
-Containment purge exhaust gross activity radiation monitor calibration procedure 1MSP-43.17-I, RM-VS 104A (Unit 1) and RM-VS 104b (Unit 1), performed on March 10 and 12, 2003, respectively

Section 4OA1: Performance Indicator Verification (71151)

Documents

-Performance indicator documentation and data review forms for RETS/ODCM radiological effluent occurrences March 2002 through September 2003
-RWDA-L summary listings for March 2002 through September 2003
-RWDA-G summary listings for March 2002 through September 2003
-Condition Reports (CRs) for 2003 involving effluent control program issues

Procedures

-Procedure ½-HPP-3.06.005, Rev. 2, Radioactive waste discharge authorization-liquid (computer calculation method)
-Procedure ½-HPP-3.06.006, Rev. 2, Batch radioactive waste discharge authorization-gas (computer calculation method)

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOP	Abnormal Operating Procedure
BCO	Basis for Continued Operation
BVPS	Beaver Valley Power Station
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
DEP	Drill/Exercise Performance
EDG	Emergency Diesel Generator
FENOC	First Energy Nuclear Operating Company
GE	General Electric
HHSI	High Head Safety Injection
IST	Inservice Testing
LER	Licensee Event Report
LCO	Limiting Condition for Operation
MCC	Motor Control Center
MR	Maintenance Rule
MSP	Maintenance Surveillance Procedure
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
OS	Occupational Radiation Safety
OST	Operations Surveillance Test
OM	Operating Manual
PD	Performance Deficiency
PI	Performance Indicator
PMT	Post-Maintenance Test
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
QS	Quench Spray
RCA	Radiologically-Controlled Area
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specification
RS	Recirculation Spray
RTD	Resistance Temperature Detector
RWP	Radiological Work Permit
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SSC	System, Structure, and Component
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	Work Order

APPENDIX 1 (Relates to Section 1R11)

"Simultaneous trip of all reactor coolant pumps" ***(ANSI/ANS 3.5-1985 App B, Section B2.2(4) Transient*)**		
INITIAL CONDITIONS / ASSUMPTIONS		
	FSAR Section 15.3.2, "Complete Loss of Forced Reactor Coolant Flow"	*Sim Test SQT-14.1.5.8.3.07 "Complete Loss of Reactor Coolant Flow Test"*
Initial Average RCS Temp	*Assumed at 4.7°F above nominal value*	*Temperature on program for current power level*
Initial RCS Pressure	*Assumed at 7.5 psi below nominal value*	*At normal operating pressure*
Event Initiation	*Assumes a frequency decay of 5 hz/sec on power to reactor coolant pumps.*	*Event initiated by simultaneous trip of all reactor coolant pumps. Pumps begin coastdown from normal operating speed.*
Reactor Trip Signal	*RCP underfrequency and undervoltage trips are not available. Trip eventually occurs on low reactor coolant loop flow.*	*All trips are available. Reactor trips immediately upon loss of all RCPs.*

"Trip of any single reactor coolant pump" ***(ANSI/ANS 3.5-1985 App B, Section B2.2(5) Transient)***		
INITIAL CONDITIONS / ASSUMPTIONS		
	FSAR Section 15.3.1, "Partial Loss of Forced Reactor Coolant Flow"	*Sim Test SQT-14.1.5.8.3.08 "Partial Loss of Reactor Coolant Flow Test"*
Initial Average RCS temp	*Assumed at 4.7°F above nominal value*	*Temperature on program for current power level*
Initial RCS Pressure	*Assumed at 7.5 psi below nominal value*	*At normal operating pressure*
Flow restrictions/ Asymmetry	*Assumes steam generator tube plugging levels up to 30% with a maximum loop flow asymmetry of 5%*	*RCS models built to initial system design specifications*
Reactivity Coefficients	*Uses most negative Doppler power coefficient and moderator temperature coefficient of zero*	*Time in core life not specified, simulator test uses realistic coefficients for time in core life associated with selected initial condition set*

"Main turbine trip (maximum power level which does not result in immediate reactor trip" ***(ANSI/ANS 3.5-1985 App B, Section B2.2(6) Transient)***		
INITIAL CONDITIONS / ASSUMPTIONS		
	FSAR Section 15.2.3, "Turbine Trip"	***Sim Test SQT-14.1.5.8.3.09 "Turbine Trip Without Direct Reactor Trip Test"***
Initial Power Level	*Max power level, incl. allowances for calibration and instrument errors*	*45% Reactor Power*
Initial RCS Temperature	*Max temperature, including allowances for cal and inst errors*	*Temperature on program for current power level*
Initial RCS Pressure	*Assumes nominal full power value minus a pressure bias*	*Normal operating pressure*
Reactivity Coefficients	*Two cases evaluated; with least neg Moderator (MTC) and Doppler power coefficient (DPC), and with large neg MTC / DPC*	*Time in core life not specified, simulator test uses realistic coefficients for time in core life associated with selected initial condition set*
Rod Control	*In MANUAL*	*In AUTOMATIC*
Steam Dumps and SG PORVs	*Assumed not available*	*Functions normally*
Pressurizer Spray Valves and Primary PORVs	*Assumed not available*	*Functions normally*
Main Feedwater	*Assumed lost at time of turbine trip*	*Available, feed flow continues to maintain SG level*
Auxiliary Feedwater	*No credit taken for aux feed flow*	*System available for use*
Reactor Trip Signal	*Assumes reactor does not trip on turbine trip above 49% power, allows reactor to trip on other signals (high presssurizer pressure, OTΔT, high pressurizer level)*	*By design, simulator test is intended to record plant response to a turbine trip from a power level below that which would result in an automatic reactor trip. It is expected that the steam load will shift to the condenser and no reactor trip set point will be reached.*